UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

#6
ASB

MAY 13 1985

Docket No. 50-458

MEMORANDUM FOR: Dennis Crutchfield, Assistant Director for Safety Assessment, Division of Licensing

FROM: L. S. Rubenstein, Assistant Director for Core and Plant Systems, Division of Systems Integration

Some problems

SUBJECT: RIVER BEND STATION - REVIEW OF FINAL DRAFT COPY OF TECHNICAL SPECIFICATIONS

The Auxiliary Systems Branch has reviewed those portions of the Final Draft Copy of the River Bend Technical Specifications which are in ASB's area of primary responsibility. Our review included a comparison of the River Bend Technical Specifications with the proposed draft standard specifications, the River Bend Final Safety Analysis Report, and our Safety Evaluation Report for the following sections

Technical Specification	Subject
3/4.1.3.1	Control Rods
3/4.1.3.3*	Control Rod Scram Accumulators
3/4.1.5*	Standby Liquid Control System
3/4.7.4*	Remote Shutdown Monitoring Instrumentation and Controls
3/4.4.3*	Reactor Pressure Boundary Leakage Detection Systems
3/4.4.7*	Main Steam Line Isolation Valves
3/4.4.10*	Main Steam Line Shutoff Valves
3/4.6.1.5*	MSIV Leakage Control System
3/4.6.1.10*	Penetration Valve Leakage Control System
3/4.6.5.5*	Shield Building Annulus Mixing System
3/4.6.5.6*	Fuel Building Ventilation
3/4.7.1.1*	Standby Service Water System
3/4.7.1.2*	Ultimate Heat Sink
3/4.7.2*	Main Control Room Air Conditioning System
3/4.7.8*	Area Temperature Monitoring
3/4.7.11*	Spent Fuel Storage Pool Temperature
3/4.9.6*	Refueling and Fuel Handling Platform
3/4.9.7*	Crane Travel - Spent and New Fuel Storage, Transfer and Upper Containment Pools

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Technical Specification	Subject
3/4.9.8*	Water Level - Reactor Vessel
3/4.9.9*	Water Level - Spent Fuel Storage and Upper Containment Fuel Pools
3/4.9.12*	Inclined Fuel Transfer System
B3/3.1.3	Bases - Control Rods
B3/4.1.5*	Bases - Standby Liquid Control System
B3/4.3.7.4*	Bases - Remote Shutdown Monitoring Instrumentation
B3/4.4.3.1*	Bases - Leakage Detection Systems
B3/4.4.7	Bases - Main Steam Line Isolation Valves
B3/4.4.10*	Bases - Main Steam Shutoff Valves
B3/4.6.1.5	Bases - MSIV Positive Leakage Control System
B3/4.6.1.10	Bases - Penetration Valve Control System
B3/4.7.1	Bases - Standby Service Water System
B3/4.7.2	Bases - Main Control Room Air Conditioning System
B3/4.7.8	Bases - Area Temperature Monitoring
B3/4.7.11*	Bases - Spent Fuel Storage Pool Temperature
B3/4.9.6	Bases - Refueling Platform
B3/4.9.7*	Bases - Crane Travel - Spent and New Fuel Storage, Transfer and Upper Containment Fuel Pools
B3/4.9.8* and B3/4.9.9	Water Level - Reactor Vessel and Water Level - Spent Fuel Storage and Upper Containment Fuel Pools
B3/4.9.12*	Inclined Fuel Transfer System
5.6*	Fuel Storage
6.8.4*	(No Title)
6.9.2.b*	Special Reports

The * indicates those sections with which we have comments. Our comments include the re-writing of some for the Technical Specifications in accordance with your instructions which were provided at the 1:00 p.m., April 23, 1985 meeting. Therefore, if the applicant provides the enclosed Technical Specifications as the River Bend Technical Specifications, they will be acceptable.

Technical Specification 3/4.7.1.2, Ultimate Heat Sink, includes a surveillance of the cooling tower basin water temperature. This aspect of this technical specification is an interim compensatory measure until the applicant has installed a permanent continuous monitoring system by the first refueling outage. A license condition covering this monitoring system will be included in our next SSER input to DL.

L S Rubenstein

L. S. Rubenstein, Assistant Director
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CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

1. With one control rod scram accumulator inoperable, within 8 hours:

   a) Restore the inoperable accumulator to OPERABLE status, or

   b) Declare the control rod associated with the inoperable accumulator inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

   a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch or place the reactor mode switch in the Shutdown position.

   b) Insert the inoperable control rods and disarm the associated directional control valves either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within one hour, either:

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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ACTION: (Continued)

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

2. With more than one withdrawn control rod with the associated scram accumlator inoperable and with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

c. The provisions of Specification 3.0.4 are not applicable.

d

4.1.3.3 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to ~~1520~~ 955 psig unless the control rod is inserted and disarmed or scrammed.

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of ~~1000~~ 970 ± 15* psig on decreasing pressure.

2. Verifying that each individual accumulator check valve maintains the associated accumulator pressure above the alarm set point for at least ~~greater than or equal to~~ 10 minutes, ~~starting at normal system operating pressure,~~ with no control rod drive pump operating.

d. IN OPERATIONAL CONDITIONS 2, 3, 4, and 5*:

1. with more than one accumulator inoperable, the reactor pressure less than 500 psig, and more than one control rod withdrawn:

a. station an operator at the redundant CRD pump and monitor the CRD ~~pressure~~ discharge header pressure.

b. If the discharge header pressure drops below 970 PSIG, immediately start the redundant CRD pump and re-align all necessary valves for the redundant pump to deliver water to the HCU's.

REACTIVITY CONTROL SYSTEMS

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3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 Two standby liquid control subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

   1. With one subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

   2. With both subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

   1. With one subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

   2. With both subsystems inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 Each standby liquid control subsystem shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

   1. The temperature of the sodium pentaborate solution ~~in the storage tank is greater than or equal to 70°F.~~ is within the limits of Figure 3.1.5-1.

   This is FSAR figure 9.3-15

   2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-2 for the percent weight concentration determined once per 31 days per Specification 4.1.5.b.2.

   3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping up to the first storage tank outlet valve to be greater than or equal to 70°F.

b. At least once per 31 days by;

   1. Verifying the continuity of the explosive charge.

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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SURVEILLANCE REQUIREMENTS (Continued)

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2. Determining that the available weight of sodium pentaborate is greater than or equal to 4246 lbs and the percent weight concentration of sodium pentaborate in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

3. Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to ~~1220~~ 1190 psig is met.

d. At least once per 18 months during shutdown by;

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 36 months.

2. **Demonstrating that all heat traced piping ~~between the storage tank and the reactor vessel~~ is unblocked by pumping from the storage tank to the test tank

   ~~a) Isolating the pump suction manual Maintenance valves and the demineralized water supply line, and~~

   ~~b) Opening each motor operated pump suction isolation valve independently and verifying flow to the collection shipping drum,~~ and then draining and flushing the piping ~~used for the test~~ with demineralized water. ~~after closing both motor-operated pump suction isolation valves~~.

3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized.

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F.

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

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Figure 3.1.5-2

This does not agree with FSAR figure 9.3-14

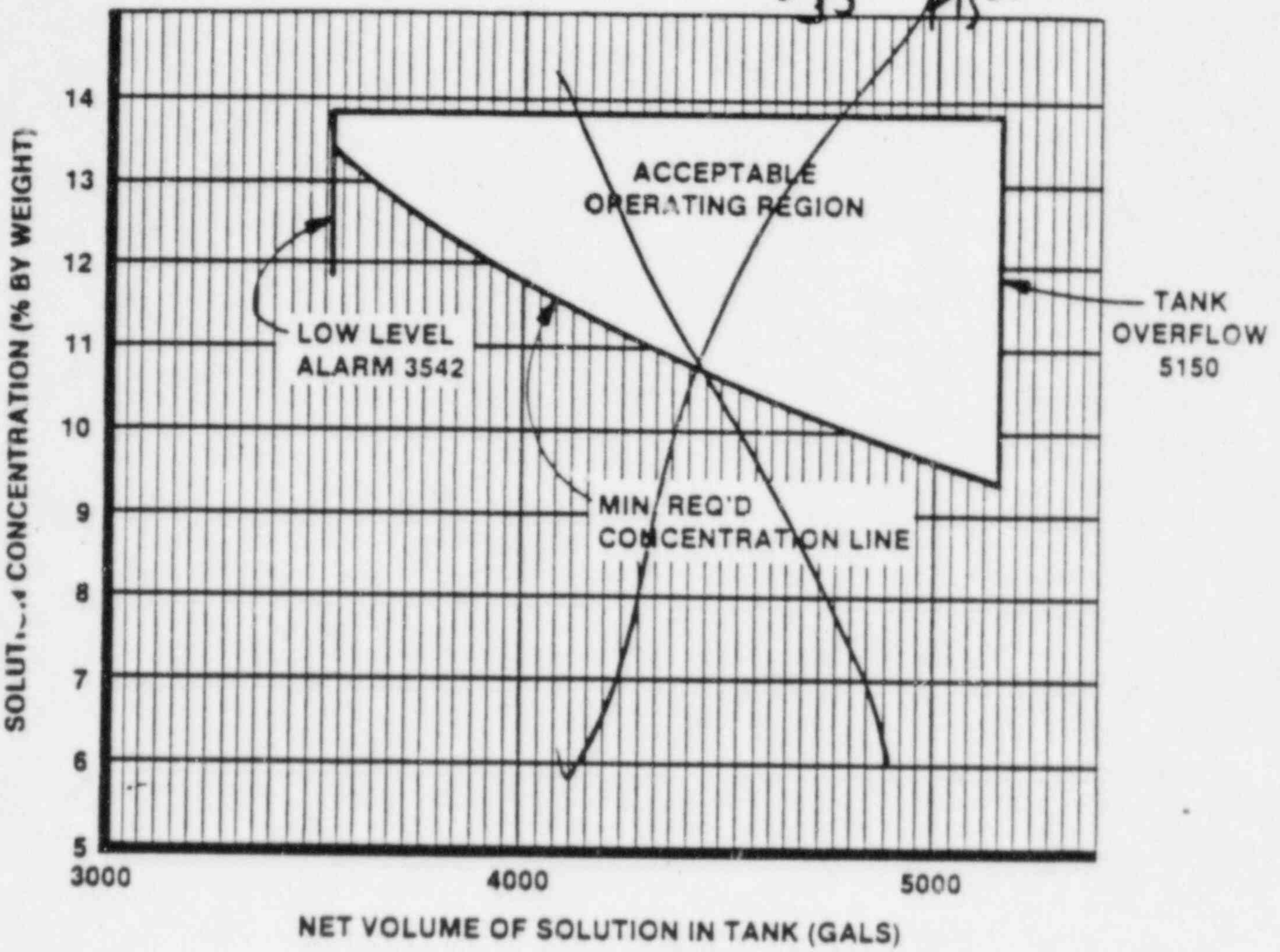

FIGURE 3.1.5-2
SODIUM PENTABORATE SOLUTION VOLUME/CONCENTRATION REQUIREMENTS

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION*	MINIMUM CHANNELS OPERABLE PER PANEL
1. Reactor Vessel Pressure	RSP1, RSP2	1
2. Reactor Vessel Water Level	RSP1, RSP2	1
3. Safety/Relief Valve Demand Position, (3) valves	RSP1, RSP2	1/valve
4. Suppression Pool Water Level	RSP1, RSP2	1
5. Suppression Pool Water Temperature	RSP1, RSP2	1
6. Drywell Pressure	RSP1, PRS2	1
7. Drywell Temperature	RSP1, RSP2	1
8. RHR System Flow: Loop A	RSP1	1
Loop B	RSP2	1
Loop C	RSP2	1
9. RHR Hx Cooling Water System Flow: Loop A	RSP1	1
Loop B	RSP2	1
~~10~~ 11. RCIC System Flow	RSP1	1
~~11~~ 12. RCIC Turbine Speed	RSP1	1
13. Standby Service Water Pump Discharge Pressure	RSP1 RSP2	1 1
14. Condensate Storage Water Tank Level	RSP1 RSP2	1 1

*RSP1 - Remote Shutdown Panel Division I
RSP2 - Remote Shutdown Panel Division II

10 RHR Hx Cooling Water Outlet Pressure	RSP1 RSP2 RSP2	1 1 1

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TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

	MINIMUM CHANNELS OPERABLE DIV. 1	DIV. 2
22. RHR Outboard Shutdown Isolation MOV (1E12*MOVF008)	1	NA
23. RHR Inboard Shutdown Isolation MOV (1E12*MOVF009)	1	NA
24. RHR Hx Flow to Supp. Pool MOV (1E12*MOVF011A, B)	1	1
25. RHR Reactor Head Spray MOV (1E12*MOVF023)	1	NA
26. RHR Test Line MOV (1E12*MOVF024A, B)	1	1
27. RHR Hx Flow to RCIC MOV (1E12*MOVF026A)	1	NA
28. RHR Injection Shutoff MOV (1E12*MOVF027A, B)	1	1
29. RHR Upper Pool Cooling Shutoff MOV (1E12*MOVF37A, B)	1	1
30. RHR Injection MOV (1E12*MOVF042A, B, C)	1	2(a)
31. RHR Hx Shell Side Inlet MOV (1E12*MOVF047A, B)	1	1
32. RHR Hx Shell Side Bypass MOV (1E12*MOVF048A, B)	1	1
33. RHR Discharge to Radwaste MOV (1E12*MOVF040)	1	NA
34. RHR Steam Isolation MOV (1E12*MOVF052A, B)	1	1
35. RHR Injection MOV (1E12*MOVF053A, B)	1	1
36. RHR Pump Minimum Flow MOV (1E12*MOVF064A, B, C)	1	2(a)
37. RHR Hx Water Discharge MOV (1E12*MOVF068A, B)	1	1
38. Safety Relief Valves (1B21*RVF051, C, G, D)	3(a)	3(a)
39. SSW Pump (1SWP*P2A, 2C, 2B, 2D)	2(a)	2(a)
40. Normal Service Water Isolation MOV (1SWP*MOV96A, B)	1	1
41. SSW Cooling Tower Inlet MOV (1SWP*MOV55A, B)	1	1

42. Diesel Generators + controls
43. Switchgear
Breakers
transfer switches

(a) - One per control equipment

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure		M	R
2. Reactor Vessel Water Level		M	R
3. Safety/Relief Valve Position		M	NA
4. Suppression Pool Water Level		M	R
5. Suppression Pool Water Temperature		M	R
6. Drywell Pressure		M	R
7. Drywell Temperature		M	R
8. RHR System Flow:	Loop A	M	R
	Loop B	M	R
	Loop C	M	R
9. RHR Hx Cooling Water System Flow:	Loop A	M	R
	Loop B	M	R
10 —		M	R
~~10~~ 11. RCIC System Flow		M	R
~~11~~ 12. RCIC Turbine Speed		M	R
13 —		M	R
14 —		M	R

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3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell atmosphere particulate radioactivity monitoring system,

b. The drywell and pedestal floor sump drain flow monitoring systems, ~~and~~

c. The containment floor drain and equipment drain sump flow monitoring systems, and

d. Either the drywell air coolers condensate flow rate monitoring system or the drywell atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With only ~~two~~ a, b, and c three of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. The ~~Drywell sump and pedestal floor~~ sump drain flow monitoring systems-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Drywell air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage ~~(~~of averaged over any 24-hour period~~)~~.

d. 1 gpm leakage at a reactor coolant system pressure of ~~1025 ± 15~~ 950 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed manual, deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. with one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.7.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm set point at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

* Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate radioactivity at least once per 12 hours,

b. Monitoring the various ~~drywell and pedestal floor drain~~ sump flow rates at least once per 12 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, ~~and~~

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate,

c. Prior to entering HOT SHUTDOWN whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months, and

d. within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

!!!!

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be ~~[illegible]~~ demonstrated OPERABLE with alarm set points per Table 3.4.3.2-1 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

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TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SYSTEM	VALVE NUMBER	ALARM SET POINT PSIG	FUNCTION
a) LPCS	1E21*AOVF006	≤ 455	LPCS Injection
	1E21*MOVF005		LPCS Injection
b) HPCS	1E22*AOVF005	≤ 475	HPCS Injection
	1E22*MOVF004		HPCS Injection
c) RCIC	1E51*AOVF065	≤ 475	RCIC Head Spray
	1E51*MOVF013		RCIC Head Spray
d) RHR	1E12*MOVF023	≤ 455	RHR Head Spray
	1E12*AOVF041A		LPCI A Injection
	1E12*MOVF042A		LPCI A Injection
	1E12*AOVF041B		LPCI B Injection
	1E12*MOVF042B		LPCI B Injection
	1E12*AOVF041C		LPCI C Injection
	1E12*MOVF042C		LPCI C Injection
	1E12*MOVF009		Shutdown Cooling A & B Suction
	1E12*MOVF008	≤ 455	Shutdown Cooling A & B Suction
	1RHS*V240		Shutdown Cooling A & B Suction

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3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

   a) Restore the inoperable valve(s) to OPERABLE status, or

   b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~b. The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

at least once per 18 months by:

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE ~~by~~ verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5. ~~The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITIONS 2 or 3 provided the surveillance is performed within 12 hours after reaching a reactor steam pressure of 600 psig and prior to entry into OPERATIONAL CONDITION 1.~~

~~[illegible] per [illegible] months~~

b. performance of a MSIV leakage rate test per valve and demonstrating that each valve ~~[illegible]~~ has a maximum leakage rate of 30.0 SCFH per MSIV.

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3/4.4.10 MAIN STEAM LINE ~~ISOLATION~~ Shutoff VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 ~~Two~~ One main steam line ~~isolation~~ shutoff valve~~s~~ ~~MSIVs~~ (MSSV) per main steam line shall be OPERABLE~~, with closing times greater than or equal to 3 and less than or equal to 5 seconds~~.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSSVs inoperable:

1. ~~Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours,~~ Either:

   a) Restore the inoperable valve(s) to OPERABLE status within ~~8 hours~~ 30 days, or

   b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~b. The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSSVs shall be demonstrated OPERABLE by verifying full closure ~~between 3 and 5 seconds~~ when tested pursuant to Specification 4.0.5. ~~The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITIONS 2 or 3 provided the surveillance is performed within 12 hours after reaching a reactor steam pressure of 600 psig and prior to entry into OPERATIONAL CONDITION 1.~~

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MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.5 Two independent main steam positive leakage control system (MS-PLCS) divisions shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MS-PLCS division inoperable, restore the inoperable division to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 Each MS-PLCS division shall be demonstrated OPERABLE:

a. By performing Surveillance Requirement 4.6.1.10.a.

b. At least once per 31 days by ~~verifying compresser OPERABILITY by~~ starting the compressor from the control room and operating the compresser loaded for at least 15 minutes.

c. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each remote, manual and automatic motor operated valve through at least one complete cycle of full travel.

d. At least once per 18 months by performance of a functional test which includes simulated actuation of the division throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and that 8.5 ± 3 psid sealing pressure is established in each steam line.

e. By verifying the operating instrumentation to be OPERABLE by performance of a:

1. channel check at least once per 24 hours,
2. channel Functional TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months.

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PENETRATION VALVE LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.10 Two independent penetration valve leakage control system (PVLCS) divisions shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one PVLCS division inoperable, restore the inoperable division to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.10 Each PVLCS division shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying division PVLCS accumulator pressure greater than or equal to 101 psig.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each remote, manual and automatic motor operated valve through at least one complete cycle of full travel.

c. At least once per 18 months by:

1. Performance of a functional test which includes simulated actuation of the system throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and that a sealing pressure greater than or equal to 21 psig is established in each sealing valve, and

2. Leakage from valves equipped with the PVLCS will be included in computation of 0.6 La. undefined term.

d. By verifying the operating instrumentation to be OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months.

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CONTAINMENT SYSTEMS

SHIELD BUILDING ANNULUS MIXING SYSTEM

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LIMITING CONDITION FOR OPERATION

3.6.5.5 Two independent Shield Building Annulus Mixing subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one Shield Building Annulus Mixing subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. with both Shield Building Annulus Mixing subsystems INOPERABLE, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.5 Each Shield Building Annulus Mixing subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, ~~the verification that the~~ flow through subsystem operates for at least 15 minutes ~~and~~ at rated conditions.

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FUEL BUILDING VENTILATION

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent Fuel Building Ventilation Charcoal Filtration subsystems shall be OPERABLE, and in OPERATIONAL CONDITION *, one operating in the emergency mode when

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

a. With one Fuel Building Ventilation Charcoal Filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both Fuel Building Ventilation Charcoal Filtration subsystems inoperable or with one not operating in the emergency mode in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each Fuel Building Ventilation Charcoal Filtration subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours in OPERATIONAL CONDITION *, by verifying one Fuel Building Ventilation Charcoal Filtration System operation.

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

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3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

STANDBY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least two independent standby service water (SSW) system ~~subsystems~~ loops, with each ~~subsystem~~ loop comprised of:

a. Two OPERABLE SSW pumps, and

b. An OPERABLE flow path capable of taking suction from the standby cooling tower basin and transferring the water through the RHR heat exchangers, ECCS pump room seal coolers, & diesel generator coolers, and associated coolers and pump heat exchangers,

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two subsystems.

b. In OPERATIONAL CONDITION 4, 5 and*, the subsystem(s) associated with systems and components required OPERABLE by Specifications 3.4.9.2, 3.5.2, 3.9.11.1, 3.9.11.2 and 3.8.1.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:

insert (A) from next page

2. With one SSW ~~subsystem~~ loop inoperable, align all the diesel generators to the available loop, declare all equipment aligned to the inoperable loop inoperable, ~~and~~ R restore the inoperable loop ~~subsystem~~ to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With both SSW ~~subsystems~~ loops inoperable, or 3 out of 4 SSW pumps inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

INSERT (B) from next page

~~b. In OPERATIONAL CONDITION 3 or 4 with the SSW subsystem inoperable which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, as applicable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.~~

*When handling irradiated fuel in primary or secondary containment.

~~**Whenever both RHR shutdown cooling mode loops are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~

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Inserts for Page 3/4 7-1

(A)

1. With one SSW pump inoperable, restore the inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(B)

b. With only one SSW pump powered from either Div. 1 or Div 2 diesel generator and its associated flow path operable, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or:

1. In OPERATIONAL CONDITION 4 or 5, declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2, 3.8.1.2, 3.9.11.1, and 3.9.11.2.

2. In OPERATIONAL CONDITION *, verify adequate cooling remains available for the diesel generators required to be OPERABLE or declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

~~c. In OPERATIONAL CONDITION 4 or 5 with the SSW subsystem inoperable which is associated with an ECCS pump required OPERABLE by Specification 3.5.2, declare the associated ECCS pump inoperable and take the ACTION required by Specification 3.5.2.~~

~~d. In OPERATIONAL CONDITION 5 with the SSW subsystem inoperable which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2.~~

~~e. In Operational Condition *, with the SSW subsystem inoperable which is associated with a diesel generator required OPERABLE by Specification 3.8.1.2, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.2.~~

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required standby service water system loop ~~subsystem~~(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by verifying that:

1. Each automatic valve ~~servicing safety related equipment or isolating nonsafety related equipment~~ actuates to the correct position on a normal service water low pressure signal,

2. Each ~~One~~ pump in each loop ~~subsystem~~ starts on a normal service water low pressure signal, and

3. Each pump in each loop ~~subsystem~~ starts on a manual control signal from the main control room.

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ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.2 The standby cooling water storage basin shall be OPERABLE with:

a. A minimum basin water level at or above elevation 108'6" Mean Sea Level, USGS datum, and

b. An average basin water temperature of less than or equal to 82°F.

c. Two OPERABLE automatically initiated cooling tower fan cells (5 fans per cell) per division.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

With the requirements of the above specification not satisfied:

a. With the basin water level less than 108'6" MSL or the water temperature greater than 82°F, then declare the SSW system inoperable and take the Action required by Specification 3.7.1.1.

b. With any one fan cell inoperable, restore the inoperable fan cell to OPERABLE status within ~~30 days~~ 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the next 24 hours.

c. With more than one fan cell ~~per division~~ inoperable, ~~restore at least one to OPERABLE status within 7 days or~~ be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

~~d. In OPERATIONAL CONDITION 1, 2 or 3 with only one fan cell OPERABLE, declare one SSW subsystem inoperable and take the ACTION required by Specification 3.7.1.1.~~

~~e. In OPERATIONAL CONDITION 4, 5 with no fan cells OPERABLE, declare the SSW system inoperable and take the ACTION required by Specification 3.7.1.1.~~

~~f. In Operational Condition * with no fan cells OPERABLE, declare the SSW system inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.~~

*When handling irradiated fuel in primary or secondary containment.

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SURVEILLANCE REQUIREMENTS

4.7.1.2 The standby cooling tower and water storage basin shall be determined OPERABLE:

~~a. At least once per 24 hours by verifying the basin water temperature and water level to be within their limits.~~

a. At least one per 31 days by starting the cooling tower fans in each cell from the control room and operating ~~the~~ each fan in each cell for at least 15 minutes.

b. By verifying the basin water level to be ~~within~~ at atleast elevation 108'6" MSL at least once per 24 hours.

c. By verifying the average water temperature to be less than or equal to 82°F:

1. at least once every four hours when the basin water temperature is greater than or equal to 75°F and less than 80°F;

2. at least once every two hours when the basin water temperature is greater than or equal to 80°F, and

3. at least once per 24 hours* when the basin water temperature is greater than 32°F.

* The water temperature shall be verified when the temperature is likely to be the highest.

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3/4.7.2 MAIN CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The main control room air conditioning system with two independent air handling unit/filter train subsystems shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one main control room air conditioning subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

   1. With one main control room air conditioning air handling/filter train subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the emergency mode of operation.

   2. With both main control room air conditioning air handling/filter train subsystems inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary, ~~and~~ secondary containment, and fuel building and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each main control room air conditioning subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 104°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

*When irradiated fuel is being handled in the primary containment or Fuel Building.

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

3. Verifying a subsystem flow rate of 4000 cfm + 10% during subsystem operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7 inches Water Gauge while operating the subsystem at a flow rate of 4000 cfm ± 10%; verifying that the prefilter pressure drop is less than .6 inch water gauge and that the pressure drop across each HEPA filter is less than 2 inches water gauge.

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, the isolation valves close within 5 seconds,

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that on each of the below emergency mode actuation test signals, the subsystem automatically switches to the emergency mode of operation and the control room is maintained at a positive pressure of ≥ 1/8 inch ~~W~~water ~~G~~gauge relative to the outside atmosphere during subsystem operation at a flow rate less than or equal to 4,000 cfm:

   a) LOCA, ~~and~~

   b) Local air intake radiation monitors - High, and

   c) manual initiation from the control room.

3. Verifying that the heaters dissipate 23 ± 2.3 Kw when tested in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks ~~satisfies the inplace penetration~~ removes greater than or equal to 99.95% ~~and bypass leakage testing acceptance criteria of less than 0.05% in~~ of the DOP when they are tested inplace in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber banks ~~satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05%~~ removes 99.95% of a in accordance with ANSI N510-1975 ~~for~~ a halogenated hydrocarbon refrigerant test gas when they are tested inplace while operating the system at a flow rate of 4000 cfm ± 10%.

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3/4.7.8 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.8 The temperature of each area shown in Table 3.7.8-1 shall be maintained within the limits indicated in Table 3.7.8-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.8-1:

a. For more than eight hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 16°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.8 The temperature in each of the areas shown in Table 3.7.8-1 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1. Auxiliary Building	
a. LPCS area	122
b. RHR A pump room	122
c. RCIC pump room	122
d. RHR B pump room	122
e. RHR C pump room	122
f. HPCS pump room	122
g. MCC area (West)	122
h. MCC area (East)	122
i. Main steam tunnel (north)	122
j. Standby gas treatment rooms	114
k. Annulus mixing fan area	122
2. Diesel Generator Control Rooms	
a. Diesel Generator 1A	104
b. Diesel Generator 1B	104
c. Diesel Generator 1C	104
3. Control Building	
a. Standby switchgear room 1A	104
b. Standby switchgear room 1B	104
c. Division I battery room	70
d. Division II battery room	70
e. Division III battery room	70
f. Inverter 1A room	104
g. Inverter 1B room	104
h. Inverter 1C room	104
4. Standby Service Water Pump House	
a. SSW pump rooms	104

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3/4.7.11 SPENT FUEL STORAGE POOL TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.7.11 The spent fuel storage pool temperature shall be maintained at less than or equal to 140°F.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel storage pool.

ACTION: With the spent fuel storage pool water temperature greater than 140°F:

a. Isolate the spent fuel pool cleanup system within one hour.

b. Initiate additional spent fuel pool cooling prior to the pool water temperature reaching 160°F.

c. For longer than 72 hours on a pool water temperature of 160°F or higher, prepare and submit a Special Report pursuant to Specification 6.9.2 within the next 7 days outlining the cause of the high temperature condition, and the plans for restoring the spent fuel storage pool temperature to less than 140°F.

d. Concurrent with fuel handling operations, suspend fuel handling or if concurrent with refueling, return freshly discharged fuel bundles to the reactor until the pool water temperature is less than 140°F.

SURVEILLANCE REQUIREMENTS

4.7.11.1 The spent fuel storage pool water temperature shall be verified to be less than or equal to 140°F at least once per 12 hours (once per 4 hours during fuel handling operations). Too frequent.

4.7.11.2 Start each fuel pool cooling and cleanup ~~pu~~ system loop, which is not already operating, at least once per 31 days and maintain operation for at least 15 minutes.

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3/4.9.6 REFUELING AND FUEL HANDLING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling and fuel handling platform shall be OPERABLE and used for handling fuel ~~assemblies~~ bundles or control rods.

APPLICABILITY: During handling of fuel ~~assemblies~~ bundles or control rods.

ACTION:

With the requirements for refueling and fuel handling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel ~~assemblies~~ bundles after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6 Each refueling and fuel handling platform hoist used for handling of control rods or fuel ~~assemblies~~ bundles shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist before the load exceeds 1200 pounds.

b. Demonstrating operation of the overload cutoff on the frame mounted and monorail mounted auxiliary hoists when the load exceeds 500 ± 50 pounds.

c. Demonstrating operation of the uptravel interlock ~~mechanical stop on the frame mounted and monorail hoists~~ when uptravel brings the top of an active fuel assembly to 8 feet, 6 inches below the water level.

d. Demonstrating operation of the downtravel interlock ~~mechanical cutoff~~ on the main hoist when grapple hook down travel reaches 4 inches below fuel assembly handle.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds 485 ± 50 pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds ~~[illegible]~~ 485 ± 50 pounds.

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3/4.9.7 CRANE TRAVEL-SPENT AND NEW FUEL STORAGE, TRANSFER AND UPPER CONTAINMENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel ~~assemblies~~ bundles in the spent or new fuel storage, transfer or upper containment fuel pool racks.

APPLICABILITY: With fuel ~~assemblies~~ bundles in the spent or new fuel storage, transfer or upper containment fuel pools.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 The fuel building crane loads shall be verified to weigh less than or equal to 1200 pounds before travel over fuel ~~assemblies~~ bundles in the spent or new fuel storage pools and the lower transfer pools.

4.9.7.2 The reactor building polar crane loads shall be verified to weigh less than or equal to 1200 pounds before travel over fuel ~~assemblies~~ bundles in the upper transfer and containment fuel pools.

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3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel ~~assemblies~~ bundles or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel ~~assemblies~~ bundles being handled are irradiated or the fuel ~~assemblies~~ bundles seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel ~~assemblies~~ bundles or control rods within the reactor pressure vessel after placing all fuel ~~assemblies~~ bundles and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel ~~assemblies~~ bundles or control rods within the reactor pressure vessel.

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3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel ~~assemblies seated~~ bundles in the spent fuel storage and upper containment fuel pool racks.

APPLICABILITY: Whenever irradiated fuel ~~assemblies~~ bundles are in the spent fuel storage or upper containment fuel pools.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel ~~assemblies~~ bundles and crane operations with loads in the spent fuel storage or upper containment fuel pool areas, as applicable after placing the fuel ~~assemblies~~ bundles and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage and upper containment fuel pools shall be determined to be at least at its minimum required depth at least once per 7 days.

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3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The access door and floor plugs of all rooms through which the transfer system penetrates are closed and locked.

b. All access interlocks and palm switches are OPERABLE.

c. The blocking valve located in the fuel building IFTS hydraulic power unit is OPERABLE.

d. ~~At least one~~ All IFTS primary and secondary carriage position and liquid level indicators ~~at each carriage position~~ ~~is~~ are OPERABLE ~~and at least one liquid level sensor is OPERABLE~~.

e. All keylock switches which provide IFTS access control-transfer system lockout are OPERABLE.

f. ~~The~~ All flashing warning lights outside of the access doors are OPERABLE.

APPLICABILITY: When the IFTS containment blank flange is removed.

ACTION:

With the requirements of the above specification not satisfied, suspend IFTS operation with the IFTS at either terminal point. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12.1 Within 1 hour prior to the startup of the IFTS, verify that no personnel are in areas immediately adjacent to the IFTS tube and that the access door and floor plugs to rooms through which the IFTS tube penetrates are closed and locked.

4.9.12.2 Within 4 hours prior to the operation of IFTS and at least once per 12 hours thereafter, verify that:

a. ~~at least one~~ All IFTS primary and secondary carriage position and water level indicators ~~at each carriage position~~ ~~is~~ are OPERABLE ~~and at least one level sensor is OPERABLE~~.

b. ~~The~~ All flashing warning lights outside of the access doors are OPERABLE.

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SURVEILLANCE REQUIREMENTS

~~4.9.12.3 Within 4 hours prior to the operation of IFTS and at least once per 7 days thereafter, verify that:~~

c All access interlocks and palm switches are OPERABLE.

d The blocking valve in the Fuel Building IFTS hydraulic power unit is OPERABLE.

e All keylock switches which provide IFTS access control-transfer system lockout are OPERABLE.

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ROD PATTERN CONTROL SYSTEM (Continued)

The RPCS provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section (15.__) of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RPCS is also designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during higher power operation.

A dual channel system is provided that, above the low power setpoint, restricts the withdrawal distances of all non-peripheral control rods. This restriction is greatest at highest power levels.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately (90 to 120) minutes. A minimum available quantity of 3542 gallons of sodium pentaborate solution
5500 containing a minimum of ~~4246~~ lbs. of sodium pentaborate is required to meet a shutdown requirement of 3% Δk/k. There is an additional allowance of (150) ppm in the reactor core to account for imperfect mixing and the filling of other piping systems connected to the reactor vessel. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972
2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972
3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

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MONITORING INSTRUMENTATION (Continued)

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit to go to and maintaining COLD ~~shutdown and maintenance of HOT SHUTDOWN of the unit from~~ from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. (This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980).

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

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3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973. In conformance with Regulatory Guide 1.45, the channel calibration tests will verify the ability to detect a 1 gpm leak in less than 1 hour and an atmospheric gaseous radioactivity system sensitivity of 10^{-6} μCi/cc.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

3/4.4.10 Main Steam Shutoff Valves

The main steam shutoff valves are provided on each of the main steam lines to assure isolation of the main steam lines for operability of the main steam line leakage control system in the event of a LOCA. While these valves are safety-related as are the main steam isolation valves, the shutoff valves need the same surveillance requirements except for the closure time. The shutoff valves are slow closing valves and need only be assumed to be closed prior to initiation of the leakage control system after a LOCA.

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3/4.7.8 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY.

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis in FSAR Chapter 15.

3/4 7.10 STRUCTURAL SETTLEMENT

Structural settlement limitations are imposed and required to be verified so as to preserve the assumptions made in the static design of the major safety related structures.

3/4.7.11 SPENT FUEL Storage POOL Temperature

The monitoring of the spent fuel storage pool water temperatures ensures that the cleanup subsystem can be isolated prior to system degradation and that additional cooling can be provided in a timely manner, as necessary.

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3/4.9.7 CRANE TRAVEL - SPENT AND NEW FUEL STORAGE, TRANSFER AND UPPER CONTAINMENT FUEL POOLS

bundle bundles

The restriction on movement of loads in excess of the nominal weight of a fuel ~~assembly~~ over other fuel ~~assemblies~~ in the pools ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a sin fuel bundle ~~in 123 fuel rods~~, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL -SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel ~~assembly~~ bundle. This minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE and in operation or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

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3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

The purpose of the inclined fuel transfer system specification is to control personnel access to those potentially high radiation areas immediately adjacent to the system and to assure safe operation of the system.

, to ensure the operability of the system,

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1 of the FSAR.

b. A fuel assembly minimum center to center storage spacing of 7 in. within rows and 12.25 in. between rows in the Low Density Storage Racks in the dryer containment pool.

c. A fuel assembly minimum center to center storage spacing of 6.2~~8~~5 in. with a neutron poison material between stored spaces in the High Density Storage Racks in the spent fuel storage facility in the fuel building

The storage of ~~spent~~ fuel in the upper containment fuel storage pool is prohibited during ~~normal operation~~ OPERATIONAL MODES 1, 2, 3, and 4.

5.6.1.2 The K_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall be administratively controlled to not exceed 0.98 when optimum moderation (foam, spray, fogging, or small droplets) is assumed.

5.6.1.3 Provisions shall be taken to avoid the entry of sources of optimum moderation (foam, spray, fogging, or small droplets) to preclude that K_{eff} for new fuel, stored in the new fuel storage facility, could exceed 0.98.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 95'.

CAPACITY

2680 no change

5.6.3 The spent fuel storage pool in the fuel building is designed and shall be maintained with a storage capacity limited to no more than ~~[illegible]~~ fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

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PROCEDURES AND PROGRAMS (Continued)

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

d. Biofouling Prevention and Detection

A program, approved by the NRC Staff prior to introduction of river water to the systems, which will ensure the procedures to prevent biofouling of safety-related equipment, assure detection of Corbicula in the intake embayment and the Mississippi River at the River Bend Station site, and monitor and survey safety-related equipment to detect biofouling. Changes to this program will be submitted to and approved by the NRC (both the Region and NRR) prior to implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to

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SPECIAL REPORTS

6.9.2 Special reports shall be submitted in the following manner:

a. Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

upon identification of infestation to the NRC which describes the level of infestation, affected systems, and measures taken to prevent further infestation and

b. Special reports in regard to Corbicula will be submitted in accord- dance with the settlement agreement dated October 10, 1984.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

i. Records of analyses required by the radiological environmental monitoring program.

j. Records of emergency drills and exercises.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

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3/4.10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
May 13, 1985

MEMORANDUM FOR: Dennis M. Crutchfield, Assistant Director
for Safety Assessment
Division of Licensing

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

No Problems

FROM: Don H. Beckham, Acting Deputy Director
Division of Human Factors Safety

SUBJECT: REVIEW OF RIVER BEND TECHNICAL SPECIFICATIONS

DHFS has reviewed Section 3/4.10 and Sections 6.1 through 6.8.3 of the final draft Technical Specifications for River Bend Unit 1. Our comments are provided in Enclosure 1. Enclosure 2 is a copy of the pages from the draft Technical Specifications that we recommend should be changed, with the changes noted.

Subject to inclusion of the corrections noted, DHFS concurs with issuance of the Technical Specifications for River Bend Unit 1.

W. Russell for.
Don H. Beckham, Acting Deputy Director
Division of Human Factors Safety

cc: E. Butcher

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Enclosure 1

1. Section 6.1.2 - The words in lines 1 and 2 of this section which read "or during his absence from the control room, a designated individual" should be enclosed in parentheses.

   Reason: To be consistent with the Standard Technical Specifications.

2. Section 6.2.2.a - The words "on duty" in the first line should be hyphenated.

   Reason: To be grammatically correct and to be consistent with the Standard Technical Specifications.

3. Section 6.2.2.f - The words "health physicists" in lines 3 and 4 should be deleted and replaced with the words "radiation protection technicians."

   Reason: To be consistent with the applicants' title for these individuals, as used in Section 6.2.2.c and in Figure 6.2.2.-1.

4. Section 6.2.3.2 - Change the final clause in this section to read, "at least 1 year <u>of which</u> experience <u>shall be</u> in the nuclear field."

   Reason: To improve the grammar and to be consistent with the wording of the Standard Technical Specifications.

5. Section 6.4.1 - Insert a comma in the second line after the words Manager-Administration.

   Reason: To be grammatically correct and to be consistent with the Standard Technical Specifications.

6. Section 6.5.1.6.e - Insert the words "Vice President - RBNG and the" in the last line between the words "the" and "Nuclear."

   Reason: We have customarily required these FRC investigation reports to be furnished to the utility individual at the level of the Vice President - RBNG. Such a change also would make this section consistent with the wording of the Standard Technical Specifications.

7. Section 6.3.5.7 - Change the lead words of this section to read, "The NRB shall be responsible for the review of:"

   Reason: The NRB need not itself perform the reviews. The revised wording is consistent with the Standard Technical Specifications.

8. Section 6.6 - Change the title of this section to read "Reportable Event Action."

   Reason: To be consistent with the Standard Technical Specifications.

Enclosure 2

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6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Senior Vice President - River Bend Nuclear Group shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Operator shall be in the control room;

c. A Radiation Protection Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site fire brigade of at least five members shall be maintained on site at all times*. The fire brigade shall not include the Shift Supervisor, the Shift Technical Advisor, the Control Operating Foreman, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Radiation Protection Technician and fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

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UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, ~~health physicists~~ radiation protection technicians, auxiliary operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

3. A break of at least eight hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or either one of the Assistant Plant Managers or the Supervisor-Radiological Programs, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Vice President - Safety and Environment.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time
of which engineers located onsite. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his shall be
field, at least 1 year experience in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Vice President - Safety and Environment.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

OK For the dual role position shown in Table 6.2.2-1, the Shift Technical Advisor shall have a bachelor's degree or shall have completed all technical courses required for the degree in a scientific or engineering discipline and shall have received all of the training for the normal STA position described above.

* ot responsible for sign-off function.

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QUORUM

6.5.1.5 The quorum of the FRC necessary for the performance of the FRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and [four] members including no more than two alternates.

This could be reduced to three members if desired.

RESPONSIBILITIES

6.5.1.6 The FRC shall be responsible for:

a. Review of all plant general administrative procedures and changes thereto;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix A Technical Specifications;

d. Review of all proposed changes or modifications to structures, components, systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Vice President - RBNG and the Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of unit operations to detect potential hazards to nuclear safety, items that may be included in this review are NRC inspection reports, QA audits/surveillance reports of operating and maintenance activities, NRB audit results, and American Nuclear Insurer (ANI) inspection results;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager or the Nuclear Review Board; and

i. Review of initial start-up testing phase start-up procedures and revisions.

6.5.1.7 The FRC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6.a. through d. prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6.a. through e. constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President - RBNG and the Nuclear Review Board of disagreement between the FRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

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MEETING FREQUENCY

6.5.3.5 The NRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.3.6 The quorum of the NRB necessary for the performance of the NRB review and audit functions of these Technical Specifications shall consist of the Chairman or the Vice Chairman and at least six NRB members including no more than two alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.3.7 The NRB shall ~~review~~ be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS:

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the FRC.

AUDITS

6.5.3.8 Audits of unit activities shall be performed under the cognizance of the NRB. These audits shall encompass:

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RECORDS (Continued)

a. Minutes of each NRB meeting shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.3.7 shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.3.8 shall be forwarded to the Senior Vice President - RBNG and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73 and

b. Each REPORTABLE EVENT shall be reviewed by the FRC and the results of this review shall be submitted to the NRB and the Plant Manager.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President - RBNG and the NRB chairman (or personnel acting for their function) shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NRB, and the Senior Vice President - RBNG within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

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STARTUP REPORT (Continued)

Subsection 14.2.12.2

the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. Documentation of all challenges to safety/relief valves.

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.