UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 254
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee), dated October 1, 1997, as supplemented October 14, 1997, March 16 and 20, April 1 and 28, May 1, 20 and 22, June 12, 17 and 26, and July 17, 24 and 31, and September 1, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 2.C.(1) and 2.(C). 2 of Facility Operating License No. DPR-52 are hereby amended to read as follows:

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PDR ADOCK 05000260
P PDR
9809170203

(1) Maximum Power level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 254, are hereby incorporated into this license. Tennessee Valley Authority shall operate the facility in accordance with the Additional Conditions.

4. This license amendment is effective as of its date of issuance and shall be implemented before Cycle 11 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment 1: Page 3 of License DPR-52
Attachment 2: Appendix B
Attachment 3: Changes to the Technical Specifications

Date of Issuance: September 8, 1998

(2) Pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use at any time source and special nuclear material as reactor fuel in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 253 to Final Operating License DPR-52, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment. For SRs that existed prior to Amendment 253, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment.

APPENDIX B

ADDITIONAL CONDITIONS

Amend. Number	Additional Conditions	Implementation Date
253	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's application dated September 6, 1996, as supplemented May 1, August 14, November 5 and 14, December 3, 4, 11, 22, 23, 29 and 30, 1997, January 23, March 12, April 16, 20 and 28, May 7, 14, 19 and 27, and June 2, 5, 10 and 19, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.
254	TVA will perform an analysis of the design basis loss-of-coolant accident to confirm compliance with General Design Criterion (GDC)-19 and offsite limits considering main steam isolation valve leakage and emergency core cooling system leakage. The results of this analysis will be submitted to the NRC for its review and approval by March 31, 1999. Following NRC approval, any required modifications will be implemented during the refueling outages scheduled for Spring 2000 for Unit 3 and Spring 2001 for Unit 2. TVA will maintain the ability to monitor radiological conditions during emergencies and administer potassium-iodide to control room operators to maintain doses within GDC-19 guidelines. This ability will be maintained until the required modifications, if any, are complete.	This amendment is effective immediately.

APPENDIX B

ADDITIONAL CONDITIONS

254	Classroom and simulator training on all power uprate related changes that affect operator performance will be conducted prior to operating at uprated conditions. Simulator changes that are consistent with power uprate conditions will be made and simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985. Training and the plant simulator will be modified, as necessary, to incorporate changes identified during startup testing.	This amendment is effective immediately.

ATTACHMENT TO LICENSE AMENDMENT NO. 254

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

Remove	Insert
1.1-6	1.1-6
3.1-25	3.1-25
3.3-7	3.3-7
3.3-8	3.3-8
3.3-35	3.3-35
3.3-61	3.3-61
3.4-4	3.4-4
3.4-8	3.4-8
3.4-30	3.4-30
3.4-31	3.4-31
3.5-6	3.5-6
3.5-13	3.5-13
3.7-1	3.7-1
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-5	3.7-5
3.7-6	3.7-6
3.7-7	3.7-7
3.7-8	3.7-8
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
3.7-12	3.7-12
3.7-13	3.7-13
3.7-14	3.7-14
3.7-15	3.7-15
3.7-16	3.7-16
3.7-17	3.7-17
3.7-18	3.7-18
3.7-19	3.7-19
--------	3.7-20
5.0-20	5.0-20

Remove	Insert
B3.1-54	B3.1-54
B3.4-5	B3.4-5
B3.4-67	B3.4-67
B3.4-68	B3.4-68
B3.4-69	B3.4-69
B3.5-5	B3.5-5
B3.5-16	B3.5-16
B3.5-31	B3.5-31
B3.5-35	B3.5-35
B3.6-3	B3.6-3
B3.6-8	B3.6-8
B3.6-37	B3.6-37
B3.7-1	B3.7-1
B3.7-2	B3.7-2
B3.7-3	B3.7-3
B3.7-4	B3.7-4
B3.7-5	B3.7-5
B3.7-6	B3.7-6
B3.7-7	B3.7-7
B3.7-8	B3.7-8
B3.7-9	B3.7-9
B3.7-1 0	B3.7-1 0
B3.7-13	B3.7-13

1.1 Definitions (continued)

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Section 13.10, Refueling Test Program, of the FSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

a. The reactor is xenon free;

b. The moderator temperature is 68°F; and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify the SLC conditions satisfy the following equation: $$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days AND Once within 24 hours after water or boron is added to the solution
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig.	18 months
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS
SR 3.1.7.8	Verify all piping between storage tank and pump suction is unblocked.	18 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 66% RTP and ≤ 120% RTP
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3[b]	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

SURVEILLANCE REQUIREMENTS

NOTE

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel Water Level - Low Low, Level 2 Function.	24 hours
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level - Low Low, Level 2: ≥ 471.52 inches above vessel zero; and b. Reactor Steam Dome Pressure - High: ≤ 1175 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Main Steam Valve Vault Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 188°F
b. Pipe Trench Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 135°F
c. Pump Room A Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
d. Pump Room B Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
e. Heat Exchanger Room Area (West Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 143°F
f. Heat Exchanger Room Area (East Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
g. SLC System Initiation	1,2	1[(a)]	H	SR 3.3.6.1.6	NA
h. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 115 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2[(b)]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
c. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig

(a) One SLC System Initiation signal provides logic input to close both RWCU valves.

(b) Only one channel per trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

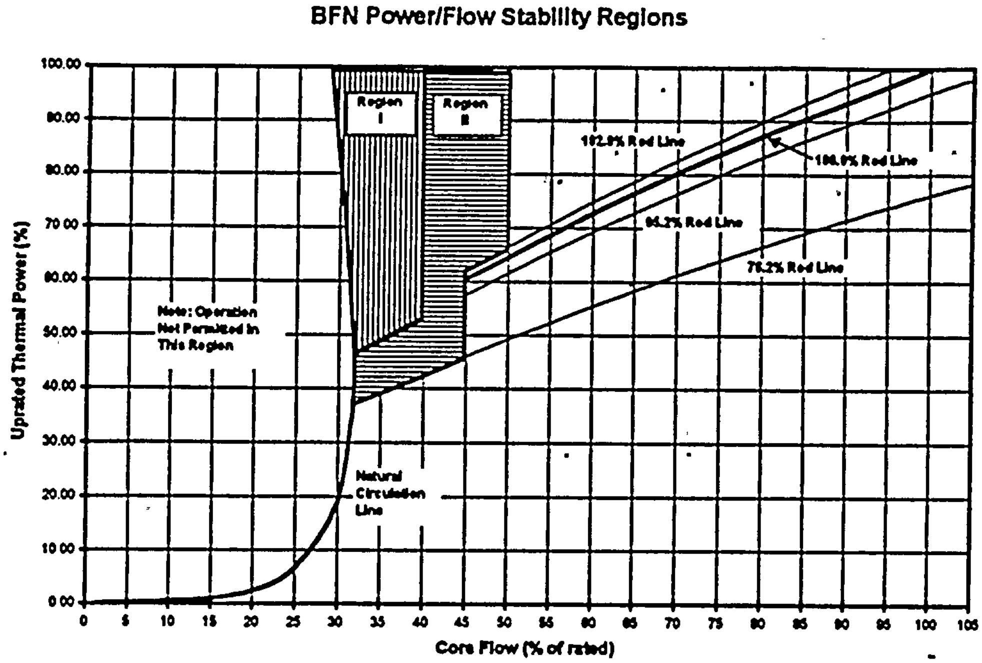


Figure 3.4.1-1
THERMAL POWER VERSUS CORE FLOW STABILITY REGIONS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift settings of the required 12 S/RVs are within ± 3% of the setpoint as follows: Number of S/RVs — Setpoint (psig) 4 — 1135 4 — 1145 5 — 1155 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each required S/RV opens when manually actuated.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1050 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1050 psig.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.7	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1040 and ≥ 950 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.1.8	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	18 months
SR 3.5.1.9	-----NOTE----- Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1040 psig and ≥ 950 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	18 months

(continued)

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS)

LCO 3.7.1

---NOTE---
The number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for ≥ 24 hours.

Four RHRSW subsystems and UHS shall be OPERABLE with the number of OPERABLE pumps as listed below:

1. 1 unit fueled - four OPERABLE RHRSW pumps.
2. 2 units fueled - six OPERABLE RHRSW pumps.
3. 3 units fueled - eight OPERABLE RHRSW pumps.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required RHRSW pump inoperable.	A.1	------------NOTES------------ 1. Only applicable for the 2 units fueled condition. 2. Only four RHRSW pumps powered from a separate 4 kV shutdown board are required to be OPERABLE if the other fueled unit has been in MODE 4 or 5 for ≥ 24 hours. ----------------------------------- Verify five RHRSW pumps powered from separate 4 kV shutdown boards are OPERABLE.	Immediately
	<u>OR</u>		
	A.2	Restore required RHRSW pump to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One RHRSW subsystem inoperable.	B.1	-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling - Hot Shutdown," for RHR shutdown cooling made inoperable by the RHRSW system. ---------- Restore RHRSW subsystem to OPERABLE status.	30 days
C. Two required RHRSW pumps inoperable.	C.1	Restore one inoperable RHRSW pump to OPERABLE status.	7 days
D. Two RHRSW subsystems inoperable..	D.1	-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.4.7, for RHR shutdown cooling made inoperable by the RHRSW System. ---------- Restore one RHRSW subsystem to OPERABLE status.	7 days

(continued)

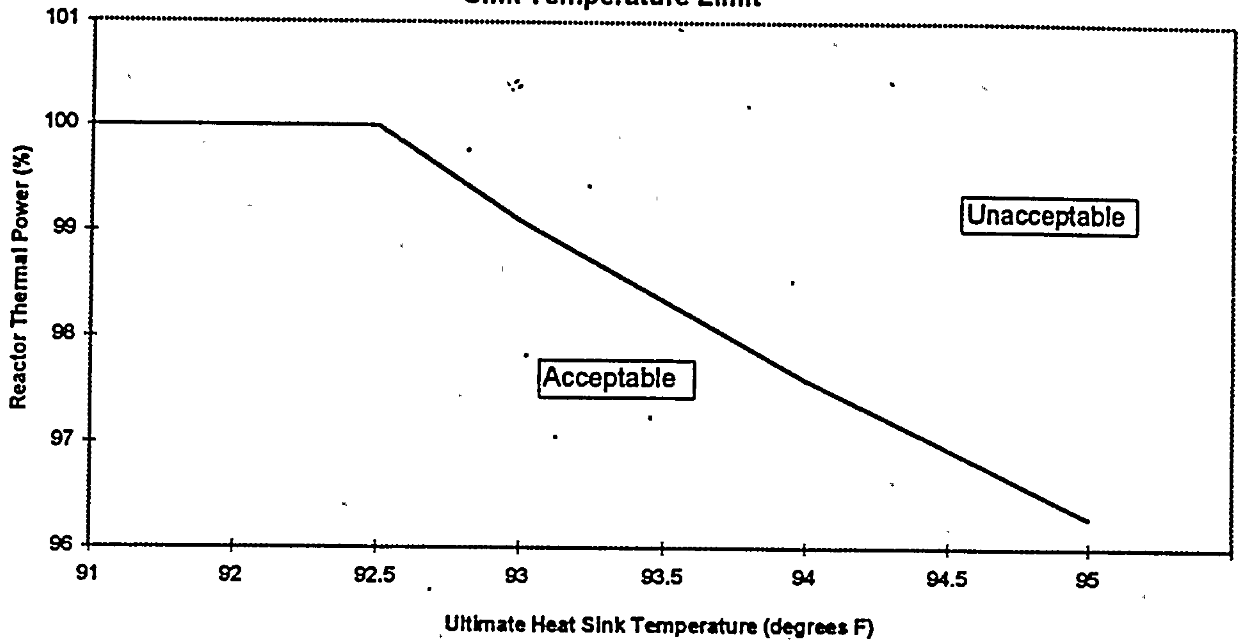
ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Three or more required RHRSW pumps inoperable.	E.1	Restore one RHRSW pump to OPERABLE status.	8 hours
F. Three or more RHRSW subsystems inoperable.	F.1	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by the RHRSW System. ------------------------------ Restore one RHRSW subsystem to OPERABLE status.	8 hours
G. Required Action and associated Completion Time not met. OR UHS inoperable	G.1	Be in MODE 3.	12 hours
	AND		
	G.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual and power operated valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.7.1.2	Verify the average water temperature of UHS is within the limits specified in Figure 3.7.1-1.	24 hours UHS temperature ≤ 91°F <u>AND</u> 1 hour UHS temperature > 91°F

Figure 3.7.1-1
Reactor Thermal Power Versus Ultimate Heat Sink Temperature Limit



3.7 PLANT SYSTEMS

3.7.2 Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 The EECW System with three pumps and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required EECW pump inoperable.	A.1 Restore the required EECW pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Two or more required EECW pumps inoperable. OR UHS inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	------------------------------NOTE------------------------------ Refer to SR 3.7.1.2 for additional UHS requirements. -- Verify the average water temperature of UHS is ≤ 95°F.	 24 hours
SR 3.7.2.2	------------------------------NOTE------------------------------ Isolation of flow to individual components does not render EECW System inoperable. -- Verify each EECW system manual and power operated valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	 31 days
SR 3.7.2.3	Verify each required EECW pump actuates on an actual or simulated initiation signal.	18 months

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Ventilation (CREV) System

LCO 3.7.3 Two CREV subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREV subsystem inoperable.	A.1 Restore CREV subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 ------------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------------ Place OPERABLE CREV subsystem in pressurization mode.	Immediately
	<u>OR</u>	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two CREV subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two CREV subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	---NOTE--- LCO 3.0.3 is not applicable. Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CREV subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.3.2	Perform required CREV filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.3.3	Verify each CREV subsystem actuates on an actual or simulated initiation signal.	18 months
SR 3.7.3.4	Verify each CREV subsystem can maintain a positive pressure of ≥ 0.125 inches water gauge relative to the outdoors during the pressurization mode of operation at a flow rate of ≥ 2700 cfm and ≤ 3300 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two Unit 1 and 2 control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Unit 1 and 2 control room AC subsystem inoperable.	A.1	Restore Unit 1 and 2 control room AC subsystem to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two Unit 1 and 2 control room AC subsystems inoperable.	B.1	Initiate action to restore one Unit 1 and 2 control room AC subsystem to OPERABLE status.	Immediately
	AND		
	B.2	Place an alternate method of cooling in operation.	24 hours
	AND		
	B.3	Restore one control room AC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	D.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ------------------------------ Place OPERABLE control room AC subsystem in operation.	Immediately
	<u>OR</u>	
	D.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	D.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	D.2.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	18 months

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.5.2	Perform a system functional test.	18 months
SR 3.7.5.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	18 months

3.7 PLANT SYSTEMS

3.7.6 Spent Fuel Storage Pool Water Level

LCO 3.7.6 The spent fuel storage pool water level shall be ≥ 21.5 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 ------------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------------ Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the spent fuel storage pool water level is ≥ 21.5 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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:

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

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

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

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.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 50.6 psig. The maximum allowable primary containment leakage rate, L_a, shall be 2% of primary containment air weight per day at P_a.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration and enrichment requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. The 18 month Frequency is acceptable since inservice testing of the pumps, performed every 92 days, will detect any adverse trends in pump performance.

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. Additionally, replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	determined to be bounded by the 76.2% rod line and the 50% core flow line. BFN conservatively implements this generic region with the "Operation Not Permitted" Region and Regions I and II of Figure 3.4.1-1. This conforms to Reference 3 recommendations. Operation is permitted in Region II provided neutron flux noise levels are verified to be within limits. The reactor mode switch must be placed in the shutdown position (an immediate scram is required) if Region I is entered. Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).
LCO	Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. In addition, the core flow expressed as a function of THERMAL POWER must be outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.
APPLICABILITY	In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur. In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of design basis accidents and transients.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of ≤ 1055 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"). Since the design basis accident and the transient analyses are performed at nominal operating pressures or above (1035 psig

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	and 1040 psig, respectively), a reactor steam dome pressure limit is chosen at 1050 psig, to ensure the plant is operated within the bounds of the uncertainties of the design basis accident and transient analyses. Reactor steam dome pressure satisfies the requirements of Criterion 2 of the NRC Policy Statement (Ref. 3).
LCO	The specified reactor steam dome pressure limit of ≤ 1050 psig ensures the plant is operated within the assumptions of the transient analysis. Operation above the limit may result in a transient response more severe than analyzed.
APPLICABILITY	In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these MODES, the reactor may be generating significant steam and the design basis accidents and transients are bounding. In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1

If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Verification that reactor steam dome pressure is ≤ 1050 psig ensures that the initial conditions of the design basis accidents and transients are met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.

(continued)

BASES

BACKGROUND (continued)

water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1174 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open (for CS and RHR they are already open) to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using the pressure suppression chamber head tank or condensate head tank. The HPCI System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of References 13 and 15. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established by testing or analysis or during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 950 psig to perform SR 3.5.1.7 and ≥ 150 psig to perform SR 3.5.1.8. Adequate steam flow is represented by reactor power ≥ 2.5% for SR 3.5.1.7 and at least two turbine bypass valves open for SR 3.5.1.8. Therefore, sufficient time is allowed after adequate

(continued)

BASES

BACKGROUND (continued)

The RCIC System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1174 psig). Upon receipt of an initiation signal, the RCIC turbine accelerates until a 600 gpm flow rate (design flow) is achieved. As the RCIC turbine flow varies, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge piping is kept full of water. The RCIC System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for RCIC is such that the water in the feedwater lines keeps the remaining portion of the RCIC discharge line full of water.

APPLICABLE SAFETY ANALYSES

The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system satisfies Criterion 4 of the NRC Policy Statement (Ref. 4).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

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

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≥ 950 psig to perform SR 3.5.3.3 and ≥ 150 psig to perform SR 3.5.3.4. Adequate steam flow is represented by reactor criticality for SR 3.5.3.3 and at least one turbine bypass valve > 50% open for SR 3.5.3.4. Therefore, sufficient time is allowed

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum allowable leakage rate for the primary containment (L_a) is 2.0% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 50.6 psig (Ref. 1).

Primary containment satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

(continued)

BASES

BACKGROUND (continued)	the air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions allowed by this LCO, the primary containment may be accessed through the air lock, when the interlock mechanism has failed, by manually performing the interlock function. The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.
APPLICABLE SAFETY ANALYSES	The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 2.0% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 50.6 psig (Ref. 3). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment. The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak drywell temperature does not exceed the maximum calculated temperature of 336°F (Ref. 2). Exceeding this temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.

Drywell air temperature satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.1 Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System.

The RHRSW System is common to the three BFN units and consists of four independent and redundant loops, each of which feeds one RHR heat exchanger in each unit. Each loop is made up of a header, two 4500 gpm pumps, a suction source, valves, piping, and associated instrumentation. One loop with one pump operating is capable of providing 50% of the required cooling capacity to maintain safe shutdown conditions for one unit. As such, a subsystem consists of a loop with one or two OPERABLE pumps, a heat exchanger, a suction source, and associated valves, piping and instrumentation. The RHRSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RHRSW System is described in the FSAR, Section 10.9 (Ref. 1).

Cooling water is pumped by the RHRSW pumps from the Wheeler Reservoir through the tube side of the RHR heat exchangers, and discharged back to the Wheeler Reservoir.

(continued)

BASES

BACKGROUND (continued)	The system is initiated manually from each of the three units control rooms. If operating during a loss of coolant accident (LOCA), the system is automatically tripped on degraded bus voltage to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time the degraded bus voltage signal clears, and is assumed to be manually started within 10 minutes after the LOCA.
APPLICABLE SAFETY ANALYSES	The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RHRSW System to support long term cooling of the reactor or primary containment is discussed in the FSAR, Chapters 5 and 14 (Refs. 2 and 3, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.
	The safety analyses for long term cooling were performed for various combinations of RHR System failures and considers the number of units fueled. With one unit fueled, the worst case single failure that would affect the performance of the RHRSW System is any failure that would disable two subsystems or pumps of the RHRSW System (e.g., the failure of an RHR Suppression Pool Cooling/Spray return line valve which

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

effectively disables two RHRSW subsystems or pumps). With two and three units fueled, a worst case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The RHRSW flow assumed in the analyses is 4000 gpm per pump with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are 177°F (as reported in Reference 3) and 50.6 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

The RHRSW System together with the UHS satisfies Criterion 3 of the NRC Policy Statement (Ref 5).

LCO

Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled.

An OPERABLE RHRSW subsystem consists of an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the required RHR heat exchangers at the assumed flow rate with at least one OPERABLE RHRSW pump in the flow path.

(continued)

BASES

LCO (continued)

In addition to the required number of OPERABLE subsystems, there must be an adequate number of pumps OPERABLE to provide cooling for the fueled non-accident units.

The number of required OPERABLE RHRSW pumps required is modified by a Note which specifies that the number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for over 24 hours. This Note acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in MODE 4 or 5 for over 24 hours.

The OPERABILITY of the UHS for RHRSW is based on having a maximum water temperature within the limits specified in Figure 3.7.1-1.

APPLICABILITY

In MODES 1, 2, and 3, the RHRSW System and UHS are required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling (LCO 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray") and decay heat removal (LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown"). The Applicability is therefore consistent with the requirements of these systems.

In MODES 4 and 5, the OPERABILITY requirements of the RHRSW System and UHS are determined by the systems they support.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

Required Action A.1 requires immediate verification that five RHRSW pumps powered from separate 4 kV shutdown boards are OPERABLE. The Required Action is modified by two notes. Note 1 indicates that the required action is applicable only when two units are fueled. In the two unit fueled condition, a single failure (loss of a 4 kV shutdown board) could result in inadequate RHRSW pumps if two pumps are powered from the same power supply. This corresponds to the LCO requirement of six OPERABLE pumps when two units are fueled, which still provides the minimum required four RHRSW pumps with the worst case single failure. If five RHRSW.pumps are powered from separate 4 kV shutdown boards, then no postulated single active failure could occur to prevent the RHRSW system from performing its design function. This is equivalent to any six RHRSW pumps OPERABLE with a maximum of two sets of two pumps allowed to be powered from the same power supply. Operation can continue indefinitely if Required Action A.1 is met.

Note 2 requires only four RHRSW pumps powered from separate 4 kV shutdown boards to be OPERABLE if the other fueled unit has been in MODE 4 or 5 greater than 24 hours. This acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in MODE 4 or 5 for greater than 24 hours.

If Required Action A.1 cannot be met, then Required Action A.2 must be complied with. With one RHRSW pump inoperable, the inoperable RHRSW pump must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW pumps are adequate to perform

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

B.1

With one RHRSW subsystem inoperable, the inoperable RHRSW subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

C.1

With two required RHRSW pumps inoperable, the remaining RHRSW pumps are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure of the OPERABLE RHRSW pumps could result in a loss of RHRSW function. The seven day Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE RHRSW pumps and the low probability of an event occurring during this period.

D.1

With two RHRSW subsystems inoperable, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The seven day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

E.1

With three or more required RHRSW pumps inoperable, the RHRSW System is not capable of performing its intended function. The requisite number of pumps must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

F.1

With three or more required RHRSW subsystems inoperable, the RHRSW System is not capable of performing its intended function. The requisite number of subsystems must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

G.1 and G.2

If the RHRSW subsystems cannot be restored to OPERABLE status within the associated Completion Times or the UHS is determined inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual and power operated valve in each RHRSW subsystem flow path provides assurance that the proper flow paths will exist for RHRSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be realigned to its accident position. This is acceptable because the RHRSW System is a manually initiated system.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.2

Verification of the UHS temperature ensures the heat removal capability of the RHRSW System is within the assumptions of the DBA analysis. The 24 hour and 1 hour Frequencies are based on operating experience relating to trends of the parameter variations during the applicable MODES.

REFERENCES

1. FSAR, Section 10.9.
2. FSAR, Chapter 5.
3. FSAR, Chapter 14.
4. FSAR, Section 14.6.3.3.2.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
6. Deleted.

BASES

LCO (continued)

The EECW System is considered OPERABLE when it has an OPERABLE UHS, three OPERABLE pumps, and two OPERABLE flow paths capable of taking suction from the intake structure and transferring the water to the appropriate equipment. .

The OPERABILITY of the UHS for EECW is based on having a maximum water temperature of 95°F. Additional requirements for UHS temperature are provided in SR 3.7.1.2

The isolation of the EECW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the EECW System.

APPLICABILITY

In MODES 1, 2, and 3, the EECW System and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the EECW System. Therefore, the EECW System and UHS are required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the EECW System and UHS are determined by the systems they support.

ACTIONS

<u>A.1</u>

With one required EECW pump inoperable, the required EECW pump must be restored to OPERABLE status within 7 days. With the system in this condition, the remaining OPERABLE EECW pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the EECW System could result in loss of EECW function.

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. DPR-68

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 1, 1997, as supplemented October 14, 1997, March 16 and 20, April 1 and 28, May 1, 20 and 22, June 12, 17 and 26, and July 17, 24, and 31, and September 1, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 2.C.(1) and 2.(C). 2 of Facility Operating License No. DPR-68 are hereby amended to read as follows:

(1) Maximum Power level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 214, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 214, are hereby incorporated into this license. Tennessee Valley Authority shall operate the facility in accordance with the Additional Conditions.

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment 1: Page 3 of License DPR-68
Attachment 2: Appendix B
Attachment 3: Changes to the Technical Specifications

Date of Issuance: September 8, 1998

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 214 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 212 to Final Operating License DPR-68, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment. For SRs that existed prior to Amendment 212, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment.

(3) Deleted.

APPENDIX B

ADDITIONAL CONDITIONS

Amend. Number	Additional Conditions	Implementation Date
212	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's application dated September 6, 1996, as supplemented May 1, August 14, November 5 and 14, December 3, 4, 11, 22, 23, 29 and 30, 1997, January 23, March 12, April 16, 20 and 28, May 7, 14, 19 and 27, and June 2, 5, 10 and 19, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.
214	TVA will perform an analysis of the design basis loss-of-coolant accident to confirm compliance with General Design Criterion (GDC)-19 and offsite limits considering main steam isolation valve leakage and emergency core cooling system leakage. The results of this analysis will be submitted to the NRC for its review and approval by March 31, 1999. Following NRC approval, any required modifications will be implemented during the refueling outages scheduled for Spring 2000 for Unit 3 and Spring 2001 for Unit 2. TVA will maintain the ability to monitor radiological conditions during emergencies and administer potassium-iodide to control room operators to maintain doses within GDC-19 guidelines. This ability will be maintained until the required modifications, if any, are complete.	This amendment is effective immediately.

APPENDIX B

ADDITIONAL CONDITIONS

Amendment Number	Additional Condition	Implementation Date
214	Classroom and simulator training on all power uprate related changes that affect operator performance will be conducted prior to operating at uprated conditions. Simulator changes that are consistent with power uprate conditions will be made and simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985. Training and the plant simulator will be modified, as necessary, to incorporate changes identified during startup testing.	This amendment is effective immediately.

ATTACHMENT TO LICENSE AMENDMENT NO. 214

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

Remove	Insert
1.1-6	1.1-6
3.1-25	3.1-25
3.3-7	3.3-7
3.3-8	3.3-8
3.3-35	3.3-35
3.4-4	3.4-4
3.4-8	3.4-8
3.4-30	3.4-30
3.4-31	3.4-31
3.5-6	3.5-6
3.5-13	3.5-13
3.7-1	3.7-1
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-5	3.7-5
3.7-6	3.7-6
3.7-7	3.7-7
3.7-8	3.7-8
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
3.7-12	3.7-12
3.7-13	3.7-13
3.7-14	3.7-14
3.7-15	3.7-15
3.7-16	3.7-16
3.7-17	3.7-17
3.7-18	3.7-18
3.7-19	3.7-19
--------	3.7-20
5.0-20	5.0-20

Remove	Insert
B3.1-54	B3.1-54
B3.4-5	B3.4-5
B3.4-67	B3.4-67
B3.4-68	B3.4-68
B3.4-69	B3.4-69
B3.5-5	B3.5-5
B3.5-16	B3.5-16
B3.5-31	B3.5-31
B3.5-35	B3.5-35
B3.6-3	B3.6-3
B3.6-8	B3.6-8
B3.6-37	B3.6-37
B3.7-1	B3.7-1
B3.7-2	B3.7-2
B3.7-3	B3.7-3
B3.7-4	B3.7-4
B3.7-5	B3.7-5
B3.7-6	B3.7-6
B3.7-7	B3.7-7
B3.7-8	B3.7-8
B3.7-9	B3.7-9
B3.7-1 0	B3.7-1 0
B3.7-13	B3.7-13

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Section 13.10, Refueling Test Program; of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.1.7.5	Verify the SLC conditions satisfy the following equation: $$\frac{(C)(Q)(E)}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days **AND** Once within 24 hours after water or boron is added to the solution
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig.	18 months
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS
SR 3.1.7.8	Verify all piping between storage tank and pump suction is unblocked.	18 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 66% RTP and ≤ 120% RTP
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

SURVEILLANCE REQUIREMENTS

---NOTE---

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel Water Level - Low Low, Level 2 Function.	24 hours
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level - Low Low, Level 2: ≥ 471.52 inches above vessel zero; and b. Reactor Steam Dome Pressure - High: ≤ 1175 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

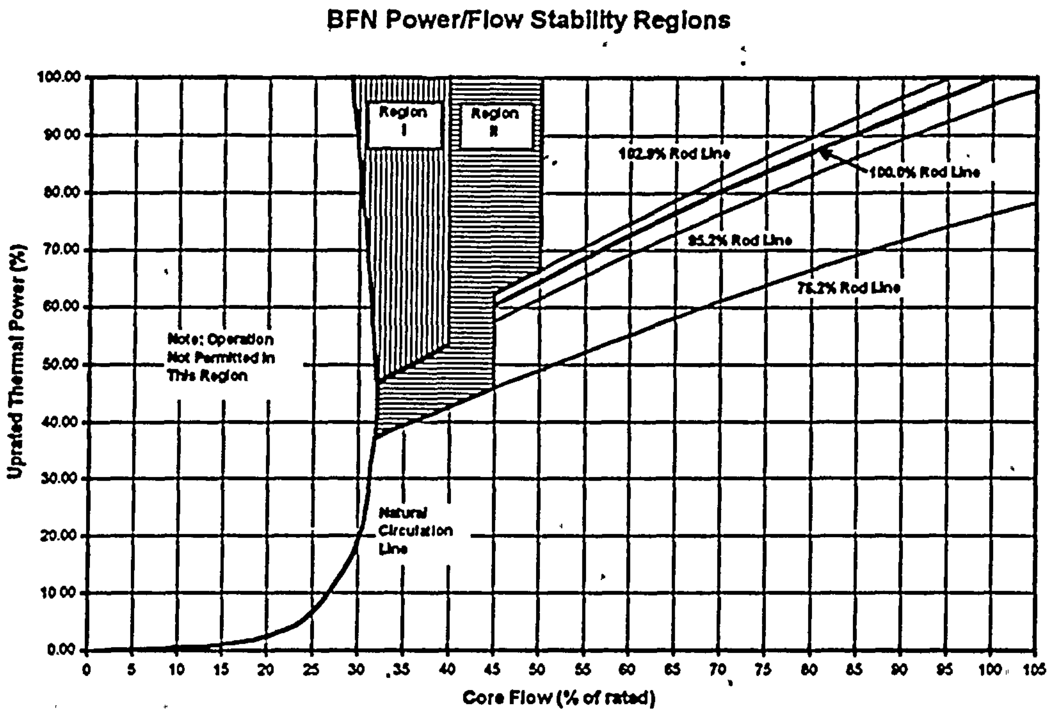


Figure 3.4.1-1
THERMAL POWER VERSUS CORE FLOW STABILITY REGIONS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift settings of the required 12 S/RVs are within ± 3% of the setpoint as follows: Number of S/RVs — Setpoint (psig) 4 — 1135 4 — 1145 5 — 1155 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	---NOTE--- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ------ Verify each required S/RV opens when manually actuated.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1050 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1050 psig.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.7	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify, with reactor pressure ≤ 1040 and ≥ 950 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.1.8	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	18 months
SR 3.5.1.9	-----NOTE----- Vessel injection/spray may be excluded. ---------- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	---NOTE--- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. Verify, with reactor pressure ≤ 1040 psig and ≥ 950 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	---NOTE--- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	18 months

(continued)

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS)

LCO 3.7.1

-----NOTE-----
The number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for ≥ 24 hours.

Four RHRSW subsystems and UHS shall be OPERABLE with the number of OPERABLE pumps as listed below:

1. 1 unit fueled - four OPERABLE RHRSW pumps.
2. 2 units fueled - six OPERABLE RHRSW pumps.
3. 3 units fueled - eight OPERABLE RHRSW pumps.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required RHRSW pump inoperable.	A.1	------------NOTES------------ 1. Only applicable for the 2 units fueled condition. 2. Only four RHRSW pumps powered from a separate 4 kV shutdown board are required to be OPERABLE if the other fueled unit has been in MODE 4 or 5 for $\geq$ 24 hours. ----------------------------------- Verify five RHRSW pumps powered from separate 4 kV shutdown boards are OPERABLE.	Immediately
	<u>OR</u>		
	A.2	Restore required RHRSW pump to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One RHRSW subsystem inoperable.	B.1	---NOTE--- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling - Hot Shutdown," for RHR shutdown cooling made inoperable by the RHRSW system. ------ Restore RHRSW subsystem to OPERABLE status.	30 days
C. Two required RHRSW pumps inoperable.	C.1	Restore one inoperable RHRSW pump to OPERABLE status.	7 days
D. Two RHRSW subsystems inoperable..	D.1	---NOTE--- Enter applicable Conditions and Required Actions of LCO 3.4.7, for RHR shutdown cooling made inoperable by the RHRSW System. ------ Restore one RHRSW subsystem to OPERABLE status.	7 days

(continued)

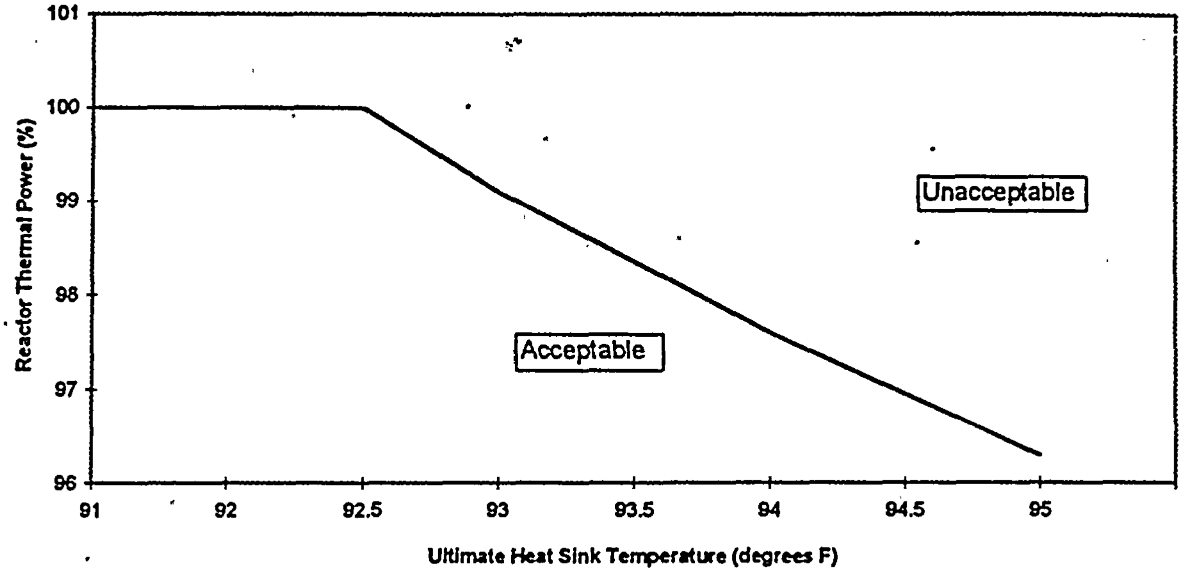
ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Three or more required RHRSW pumps inoperable.	E.1	Restore one RHRSW pump to OPERABLE status.	8 hours
F. Three or more RHRSW subsystems inoperable.	F.1	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by the RHRSW System. --------------------------------- Restore one RHRSW subsystem to OPERABLE status.	8 hours
G. Required Action and associated Completion Time not met. <u>OR</u> UHS inoperable	G.1 <u>AND</u> G.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual and power operated valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.7.1.2	Verify the average water temperature of UHS is within the limits specified in Figure 3.7.1-1.	24 hours UHS temperature ≤ 91°F <u>AND</u> 1 hour UHS temperature > 91°F

Figure 3.7.1-1
Reactor Thermal Power Versus Ultimate Heat Sink Temperature Limit



3.7 PLANT SYSTEMS

3.7.2 Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 The EECW System with three pumps and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required EECW pump inoperable.	A.1 Restore the required EECW pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR Two or more required EECW pumps inoperable. OR UHS inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	-----NOTE----- Refer to SR 3.7.1.2 for additional UHS requirements. --------------- Verify the average water temperature of UHS is ≤ 95°F.	24 hours
SR 3.7.2.2	-----NOTE----- Isolation of flow to individual components does not render EECW System inoperable. --------------- Verify each EECW system manual and power operated valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.2.3	Verify each required EECW pump actuates on an actual or simulated initiation signal.	18 months

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Ventilation (CREV) System

LCO 3.7.3 Two CREV subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREV subsystem inoperable.	A.1	Restore CREV subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1	------NOTE------ LCO 3.0.3 is not applicable. ------------------ Place OPERABLE CREV subsystem in pressurization mode.	Immediately
	<u>OR</u>		
	C.2.1	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.2.3	Initiate action to suspend OPDRVs.	Immediately
D. Two CREV subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two CREV subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	-----NOTE----- LCO 3.0.3 is not applicable. ------------- Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CREV subsystem for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.3.2	Perform required CREV filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.3.3	Verify each CREV subsystem actuates on an actual or simulated initiation signal.	18 months
SR 3.7.3.4	Verify each CREV subsystem can maintain a positive pressure of ≥ 0.125 inches water gauge relative to the outdoors during the pressurization mode of operation at a flow rate of ≥ 2700 cfm and ≤ 3300 cfm.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two Unit 3 control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One Unit 3 control room AC subsystem inoperable.	A.1	Restore Unit 3 control room AC subsystem to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two Unit 3 control room AC subsystems inoperable.	B.1 Initiate action to restore one Unit 3 control room AC subsystem to OPERABLE status.	Immediately
	<u>AND</u>	
	B.2 Place an alternate method of cooling in operation.	24 hours
	<u>AND</u>	
	B.3 Restore one control room AC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	C.2 Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	D.1	----------NOTE---------- LCO 3.0.3 is not applicable. ----------------------------- Place OPERABLE control room AC subsystem in operation.	Immediately
	<u>OR</u>		
	D.2.1	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	D.2.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	D.2.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	18 months

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.5.2	Perform a system functional test.	18 months
SR 3.7.5.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	18 months

3.7 PLANT SYSTEMS

3.7.6 Spent Fuel Storage Pool Water Level

LCO 3.7.6 The spent fuel storage pool water level shall be ≥ 21.5 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1	------------NOTE------------ LCO 3.0.3 is not applicable. --------------------------------	
		Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the spent fuel storage pool water level is $\geq$ 21.5 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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:

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

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

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

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.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 50.6 psig. The maximum allowable primary containment leakage rate, L_a, shall be 2% of primary containment air weight per day at P_a.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.6

Demonstrating that each SLC System pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration and enrichment requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. The 18 month Frequency is acceptable since inservice testing of the pumps, performed every 92 days, will detect any adverse trends in pump performance.

SR 3.1.7.7 and SR 3.1.7.8

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. Additionally, replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturer's assembly date. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	determined to be bounded by the 76.2% rod line and the 45% core flow line. BFN conservatively implements this generic region with the "Operation Not Permitted" Region and Regions I and II of Figure 3.4.1-1. This conforms to Reference 3 recommendations. Operation is permitted in Region II provided neutron flux noise levels are verified to be within limits. The reactor mode switch must be placed in the shutdown position (an immediate scram is required) if Region I is entered. Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).
LCO	Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. In addition, the core flow expressed as a function of THERMAL POWER must be outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.
APPLICABILITY	In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur. In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of design basis accidents and transients.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of ≤ 1055 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"). Since the design basis accident and the transient analyses are performed at nominal operating pressures or above (1035 psig

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	and 1040 psig, respectively) a reactor steam dome pressure limit is chosen at 1050 psig, to ensure the plant is operated within the bounds of the uncertainties of the design basis accident and transient analyses. Reactor steam dome pressure satisfies the requirements of Criterion 2 of the NRC Policy Statement (Ref. 3).
LCO	The specified reactor steam dome pressure limit of ≤ 1050 psig ensures the plant is operated within the assumptions of the transient analysis. Operation above the limit may result in a transient response more severe than analyzed.
APPLICABILITY	In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these MODES, the reactor may be generating significant steam and the design basis accidents and transients are bounding. In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1

If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Verification that reactor steam dome pressure is ≤ 1050 psig ensures that the initial conditions of the design basis accidents and transients are met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.

(continued)

BASES

BACKGROUND (continued)

water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1174 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open (for CS and RHR they are already open) to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using the pressure suppression chamber head tank or condensate head tank. The HPCI System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.6, SR 3.5.1.7, and SR 3.5.1.8

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of References 13 and 15. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established by testing or analysis or during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 950 psig to perform SR 3.5.1.7 and ≥ 150 psig to perform SR 3.5.1.8. Adequate steam flow is represented by reactor power ≥ 2.5% for SR 3.5.1.7 and at least two turbine bypass valves open for SR 3.5.1.8. Therefore, sufficient time is allowed after adequate

(continued)

BASES

BACKGROUND (continued)	The RCIC System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1174 psig). Upon receipt of an initiation signal, the RCIC turbine accelerates until a 600 gpm flow rate (design flow) is achieved. As the RCIC turbine flow varies, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV. The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge piping is kept full of water. The RCIC System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for RCIC is such that the water in the feedwater lines keeps the remaining portion of the RCIC discharge line full of water.
APPLICABLE SAFETY ANALYSES	The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system satisfies Criterion 4 of the NRC Policy Statement (Ref. 4).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

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

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≥ 950 psig to perform SR 3.5.3.3 and ≥ 150 psig to perform SR 3.5.3.4. Adequate steam flow is represented by reactor criticality for SR 3.5.3.3 and at least one turbine bypass valve > 50% open for SR 3.5.3.4. Therefore, sufficient time is allowed

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum allowable leakage rate for the primary containment (L_a) is 2.0% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 50.6 psig (Ref. 1).

Primary containment satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

(continued)

BASES

BACKGROUND (continued)

the air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions allowed by this LCO, the primary containment may be accessed through the air lock, when the interlock mechanism has failed, by manually performing the interlock function.

The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 2.0% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 50.6 psig (Ref. 3). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak drywell temperature does not exceed the maximum calculated temperature of 336°F (Ref. 2). Exceeding this temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident. Drywell air temperature satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.1 Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System.

The RHRSW System is common to the three BFN units and consists of the UHS and four independent and redundant loops, each of which feeds one RHR heat exchanger in each unit. Each loop is made up of a header, two 4500 gpm pumps, a suction source, valves, piping, and associated instrumentation. One loop with one pump operating is capable of providing 50% of the required cooling capacity to maintain safe shutdown conditions for one unit. As such, a subsystem consists of a loop with one or two OPERABLE pumps, a heat exchanger, a suction source, and associated valves, piping and instrumentation. The RHRSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RHRSW System is described in the FSAR, Section 10.9 (Ref. 1).

Cooling water is pumped by the RHRSW pumps from the Wheeler Reservoir through the tube side of the RHR heat exchangers, and discharged back to the Wheeler Reservoir.

(continued)

BASES

BACKGROUND (continued)

The system is initiated manually from each of the three units control rooms. If operating during a loss of coolant accident (LOCA), the system is automatically tripped on degraded bus voltage to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time the degraded bus voltage signal clears, and is assumed to be manually started within 10 minutes after the LOCA.

APPLICABLE SAFETY ANALYSES

The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RHRSW System to support long term cooling of the reactor or primary containment is discussed in the FSAR, Chapters 5 and 14 (Refs. 2 and 3, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The safety analyses for long term cooling were performed for various combinations of RHR System failures and considers the number of units fueled. With one unit fueled, the worst case single failure that would affect the performance of the RHRSW System is any failure that would disable two subsystems or pumps of the RHRSW System (e.g., the failure of an RHR Suppression Pool Cooling/Spray return line valve which

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

effectively disables two RHRSW subsystems or pumps). With two and three units fueled, a worst case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The RHRSW flow assumed in the analyses is 4000 gpm per pump with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are 177°F (as reported in Reference 3) and 50.6 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

The RHRSW System together with the UHS satisfies Criterion 3 of the NRC Policy Statement (Ref 5).

LCO

Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled.

An OPERABLE RHRSW subsystem consists of an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the required RHR heat exchangers at the assumed flow rate with at least one OPERABLE RHRSW pump in the flow path.

(continued)

BASES

LCO (continued)	In addition to the required number of OPERABLE subsystems, there must be an adequate number of pumps OPERABLE to provide cooling for the fueled non-accident units. The number of required OPERABLE RHRSW pumps required is modified by a Note which specifies that the number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for over 24 hours. This Note acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in MODE 4 or 5 for over 24 hours. The OPERABILITY of the UHS for RHRSW is based on having a maximum water temperature within the limits specified in Figure 3.7.1-1.
APPLICABILITY	In MODES 1, 2, and 3, the RHRSW System and UHS are required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling (LCO 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray") and decay heat removal (LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown"). The Applicability is therefore consistent with the requirements of these systems. In MODES 4 and 5, the OPERABILITY requirements of the RHRSW System and UHS are determined by the systems they support.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

Required Action A.1 requires immediate verification that five RHRSW pumps powered from separate 4 kV shutdown boards are OPERABLE. The Required Action is modified by two notes. Note 1 indicates that the required action is applicable only when two units are fueled. In the two unit fueled condition, a single failure (loss of a 4 kV shutdown board) could result in inadequate RHRSW pumps if two pumps are powered from the same power supply. This corresponds to the LCO requirement of six OPERABLE pumps when two units are fueled, which still provides the minimum required four RHRSW pumps with the worst case single failure. If five RHRSW pumps are powered from separate 4 kV shutdown boards, then no postulated single active failure could occur to prevent the RHRSW system from performing its design function. This is equivalent to any six RHRSW pumps OPERABLE with a maximum of two sets of two pumps allowed to be powered from the same power supply. Operation can continue indefinitely if Required Action A.1 is met.

Note 2 requires only four RHRSW pumps powered from separate 4 kV shutdown boards to be OPERABLE if the other fueled unit has been in MODE 4 or 5 greater than 24 hours. This acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in MODE 4 or 5 for greater than 24 hours.

If Required Action A.1 cannot be met, then Required Action A.2 must be complied with. With one RHRSW pump inoperable, the inoperable RHRSW pump must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW pumps are adequate to perform

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

B.1

With one RHRSW subsystem inoperable, the inoperable RHRSW subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

C.1

With two required RHRSW pumps inoperable, the remaining RHRSW pumps are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure of the OPERABLE RHRSW pumps could result in a loss of RHRSW function. The seven day Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE RHRSW pumps and the low probability of an event occurring during this period.

D.1

With two RHRSW subsystems inoperable, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The seven day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

E.1

With three or more required RHRSW pumps inoperable, the RHRSW System is not capable of performing its intended function. The requisite number of pumps must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

F.1

With three or more required RHRSW subsystems inoperable, the RHRSW System is not capable of performing its intended function. The requisite number of subsystems must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

G.1 and G.2

If the RHRSW subsystems cannot be restored to OPERABLE status within the associated Completion Times or the UHS is determined inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual and power operated valve in each RHRSW subsystem flow path provides assurance that the proper flow paths will exist for RHRSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be realigned to its accident position. This is acceptable because the RHRSW System is a manually initiated system.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.7.1.2 Verification of the UHS temperature ensures that the heat removal capability of the RHRSW System is within the assumptions of the DBA analysis. The 24 hour and 1 hour Frequencies are based on operating experience relating to trending of the parameter variations during the applicable MODES.
REFERENCES	1. FSAR, Section 10.9. 2. FSAR, Chapter 5. 3. FSAR, Chapter 14. 4. FSAR, Section 14.6.3.3.2. 5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993. 6. Deleted.

BASES

LCO (continued)

The EECW System is considered OPERABLE when it has an OPERABLE UHS, three OPERABLE pumps, and two OPERABLE flow paths capable of taking suction from the intake structure and transferring the water to the appropriate equipment.

The OPERABILITY of the UHS for EECW is based on having a maximum water temperature of 95°F. Additional requirements for UHS temperature are provided in SR 3.7.1.2.

The isolation of the EECW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the EECW System.

APPLICABILITY

In MODES 1, 2, and 3, the EECW System and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the EECW System. Therefore, the EECW System and UHS are required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the EECW System and UHS are determined by the systems they support.

ACTIONS

A.1

With one required EECW pump inoperable, the required EECW pump must be restored to OPERABLE status within 7 days. With the system in this condition, the remaining OPERABLE EECW pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the EECW System could result in loss of EECW function.

(continued)