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	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
	OR		
	C.2.1	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	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 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
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
	<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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
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
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
	OR		
	D.2.1	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	D.2.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	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 $\geq$ 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

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 2 ITS (Revised pages marked *R1)

NOTE: Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace pages 3.7-4 through 3.7-15 with pages 3.7-6 R1 through 3.7-18 R1.

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	Verify the average water temperature of UHS is $\leq$ 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
	OR		
	C.2.1	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	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 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
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
	<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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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
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
	OR	
	D.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	D.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	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 $\geq$ 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

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 3 ITS (Revised pages marked *R1)

NOTE: Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace pages 3.7-4 through 3.7-15 with pages 3.7-6 R1 through 3.7-18 R1.

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 AND B.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	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
	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
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

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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. AND C.2 Be in MODE 4.	12 hours 36 hours
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. OR D.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment. AND D.2.2 Suspend CORE ALTERATIONS. AND D.2.3 Initiate action to suspend OPDRVs.	 Immediately Immediately Immediately 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 $\geq$ 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 $\geq$ 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

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 1 ITS BASES (Revised pages marked *R1)

NOTE: Bases Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace pages B 3.7-7 through B 3.7-30 with pages B 3.7-9 R1 through 3.7-32 R1.

B 3.7 PLANT SYSTEMS

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

BASES

BACKGROUND

The EECW System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), residual heat removal (RHR) pump coolers, and room coolers for other Emergency Core Cooling System equipment, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The EECW System also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, the essential loads are provided cooling water by automatically starting RHRSW pumps aligned to EECW headers.

The EECW System, which is common to the three BFN units, consists of the UHS and two independent and redundant loops with each loop consisting of a header, two 4500 gpm pumps, a suction source, valves, piping and associated instrumentation. Two EECW pumps (one per loop or both on one loop) are capable of providing the required cooling capacity to support the required systems. The two loops are separated from each other so failure of one loop will not affect the OPERABILITY of the other. The EECW System is described in the FSAR, Section 10.10 (Ref. 3)

Cooling water is pumped from the Wheeler Reservoir by the EECW pumps to the essential components through the two main headers. After removing heat from the components, the water is discharged back to the Wheeler Reservoir.

APPLICABLE SAFETY ANALYSES

Sufficient water inventory is available for all EECW System post LOCA cooling requirements for a 30 day period with no additional makeup water source available. The ability of the EECW System to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the FSAR, Chapters 5 and 14 (Refs. 1 and 2, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ability of the EECW System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 1 and 2. The ability to provide onsite emergency AC power is dependent on the ability of the EECW System to cool the DGs. The long term cooling capability of the RHR and core spray pumps is also dependent on the cooling provided by the EECW System.

The EECW System, together with the UHS, satisfy Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

The EECW loops are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, two EECW pumps are required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, three EECW pumps must be OPERABLE. At least two pumps will operate if the worst single active failure occurs coincident with the loss of offsite power.

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 is based on having a maximum water temperature of 95°F.

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.

(continued)

BASES (continued)

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.

The 7 day Completion Time is based on the redundant EECW System capabilities afforded by the remaining OPERABLE pumps, the low probability of an accident occurring during this time period and is consistent with the allowed Completion Time for restoring an inoperable DG.

B.1 and B.2

If the required EECW pump cannot be restored to OPERABLE status within the associated Completion Time, or two or more EECW pumps are inoperable 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 3.7.2.1

Verification of the UHS temperature ensures that the heat removal capability of the EECW System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

Verifying the correct alignment for each manual and power operated valve in the EECW System flow paths provide assurance that the proper flow paths will exist for EECW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were 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 automatically realigned to its accident position within the required time. 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.

This SR is modified by a Note indicating that 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. As such, when required EECW pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the EECW System is still OPERABLE.

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

SR 3.7.2.3

This SR verifies that the EECW System pumps will automatically start to provide cooling water to the required safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. This SR includes a functional test of the initiation logic and a functional test and calibration of the EECW pump timers (both normal power and diesel power).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.3 (continued)

Operating experience has shown that these components will usually pass the SR when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Chapter 5.
2. FSAR, Chapter 14.
3. FSAR, Section 10.10.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Ventilation (CREV) System

BASES

BACKGROUND

The CREV System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the CREV System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air. The system has a high efficiency particulate air (HEPA) filter bank in the portion of the inlet piping common to both subsystems. Each subsystem consists of a motor-driven fan, an electric duct air heater, an activated charcoal adsorber section, an electric charcoal heater, and the associated ductwork and dampers. The HEPA filter bank removes particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREV System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room. Outside air is taken in through the CREV System ventilation intake and is passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles.

The CREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. A single CREV subsystem will pressurize the control room to about 0.125 inches water gauge to prevent infiltration of air from surrounding buildings and the outdoors. CREV System operation in maintaining control room habitability is discussed in the FSAR, Section 10.12 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the CREV System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, Chapters 10 and 14 (Refs. 2 and 3, respectively). The pressurization mode of the CREV System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the FSAR, Section 14.6 (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 3. No single active failure will cause the loss of filtered outside air from the control room.

The CREV System satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

Two redundant subsystems of the CREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

The CREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. The electric duct heater, ductwork, and dampers are OPERABLE.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

(continued)

0

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the CREV System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one CREV subsystem inoperable, the inoperable CREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREV subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREV System capability. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. 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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

C.1, C.2.1, C.2.2, and C.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREV subsystem may be placed in the pressurization mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES

ACTIONS (continued)

D.1

If both CREV subsystems are inoperable in MODE 1, 2, or 3, the CREV System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDVRs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture that has accumulated in the charcoal as a result of humidity in the ambient air. The CREV System must be operated for ≥ 10 continuous hours with the heaters energized to dry out any moisture and to demonstrate the function of the system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.3.2

This SR verifies that the required CREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.3.3

This SR verifies that on an actual or simulated initiation signal, each CREV subsystem starts and operates. This SR includes verification that dampers necessary for proper CREV operation function as required. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 and SR 3.3.7.1.6 overlaps this SR to provide complete testing of the safety function.

SR 3.7.3.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to outdoors is periodically tested to verify proper function of the CREV System. During the emergency mode of operation, the CREV System is designed to slightly pressurize the control room ≥ 0.125 inches water gauge positive pressure

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.4 (continued)

with respect to the outdoors to prevent unfiltered inleakage. The CREV System is designed to maintain this positive pressure at a flow rate of ≥ 2700 cfm and ≤ 3300 cfm to the control room in the pressurization mode. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration systems SRs.

REFERENCES

1. FSAR, Section 10.12.
2. FSAR, Chapter 10.
3. FSAR, Chapter 14.
4. FSAR, Section 14.6.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Unit 1 and 2 Control Room AC System provides temperature control for the Unit 1 and 2 Control Room following isolation of the control room. The Unit 1 and 2 Control Room AC System consists of two redundant subsystems that provide cooling and heating of recirculated control room air. A subsystem consists of an air handling unit, a chilled water pump, a water chiller, ductwork, dampers, piping, and instrumentation and controls to provide for control room temperature control.

The Unit 1 and 2 Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. A single subsystem provides the required temperature control to maintain the control room temperature within acceptable limits for operation of equipment and for uninterrupted safe occupancy under all plant conditions. The design conditions for the control room environment are 76°F and 50% relative humidity. Each subsystem is capable of maintaining the control room temperature at or below 104°F during abnormal or accident conditions. Alternate methods of cooling the Unit 1 and 2 Control Room are available. These include, but are not limited to, the use of the emergency chiller, the Unit 3 Control Room AC System and the Relay Room AC Systems. The Control Room AC System operation in maintaining the control room temperature is discussed in the FSAR, Section 10.12 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for uninterrupted safe occupancy under normal and accident conditions.

The Control Room AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single failure of a component of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The Control Room AC System satisfies Criterion 3 of the NRC Policy Statement (Ref. 2).

LCO

Two redundant subsystems of the Unit 1 and 2 Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Unit 1 and 2 Control Room AC System is considered OPERABLE when the components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the air handling units, chilled water pumps, water chillers, ductwork, dampers, piping, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

(continued)

BASES

APPLICABILITY (continued)

a. During operations with a potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one Unit 1 and 2 control room AC subsystem inoperable, the inoperable Unit 1 and 2 control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE Unit 1 and 2 control room AC subsystem is adequate to perform the Unit 1 and 2 control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the Unit 1 and 2 control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

B.1, B.2.1, B.2.2

With both Unit 1 and 2 control room AC subsystems inoperable, cooling by a Unit 1 and 2 control AC subsystem must be restored without delay.

Until Unit 1 and 2 control room AC OPERABILITY is re-established, an alternate method of control room cooling must be placed in service within 24 hours. Alternate means should be taken as necessary to maintain the Unit 1 and 2 control room temperature during this Condition. These include, but are not limited to, the use of the emergency chiller, the Unit 3 Control Room AC System and the Relay Room AC System. A Completion Time of 7 days (Required Action B.2.2) is provided to restore at least one Unit 1 and 2 control room AC subsystem to OPERABLE status. A 7 day time period is allowed to restore the function based on the

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2 (continued)

low probability of an event occurring that requires control room isolation, the alternate method of cooling, and the potential for decreased safety if the unit operator's attention is diverted from the actions necessary to restore control room AC to the actions associated with taking the unit to shutdown within this time limit.

C.1 and C.2

In MODE 1, 2, or 3, if the inoperable Unit 1 and 2 control room AC subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. 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.

D.1, D.2.1, D.2.2, and D.2.3

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE Unit 1 and 2 control room AC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

(continued)

BASES

ACTIONS

D.1, D.2.1, D.2.2, and D.2.3 (continued)

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 18 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

REFERENCES

1. FSAR, Section 10.12.

2. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11.3.3 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during abnormal operational transients (e.g., the feedwater controller failure-maximum demand event), as discussed in the FSAR, Section 14.5.1.1 (Ref. 2). Opening the bypass valves during the event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR and MCPR penalties.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR and MCPR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR and MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during abnormal operational transients. The 4 hour Completion Time is 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 3.7.5.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.5.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the cycle specific transient analyses performed to support the preparation of FSAR, Appendix N, Supplemental Reload Licensing Report (Ref. 4). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 7.11.3.3.
2. FSAR, Section 14.5.1.1.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
4. FSAR, Appendix N.

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the FSAR, Section 14.6.4.5 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inabiljty to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 14.6.4.
3. NUREG-0800, Section 15.7.4, Revision 1, July 1981.
4. 10 CFR 100.
5. Regulatory Guide 1.25, March 1972.
6. FSAR, Section 14.6.4.5.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 2 ITS BASES (Revised pages marked *R1)

NOTE: Bases Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace pages B 3.7-7 through B 3.7-30 with pages B 3.7-9 R1 through 3.7-32 R1.

B 3.7 PLANT SYSTEMS

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

BASES

BACKGROUND

The EECW System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), residual heat removal (RHR) pump coolers, and room coolers for other Emergency Core Cooling System equipment, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The EECW System also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, the essential loads are provided cooling water by automatically starting RHRSW pumps aligned to EECW headers.

The EECW System, which is common to the three BFN units, consists of the UHS and two independent and redundant loops with each loop consisting of a header, two 4500 gpm pumps, a suction source, valves, piping and associated instrumentation. Two EECW pumps (one per loop or both on one loop) are capable of providing the required cooling capacity to support the required systems. The two loops are separated from each other so failure of one loop will not affect the OPERABILITY of the other. The EECW System is described in the FSAR, Section 10.10 (Ref. 3)

Cooling water is pumped from the Wheeler Reservoir by the EECW pumps to the essential components through the two main headers. After removing heat from the components, the water is discharged back to the Wheeler Reservoir.

APPLICABLE SAFETY ANALYSES

Sufficient water inventory is available for all EECW System post LOCA cooling requirements for a 30 day period with no additional makeup water source available. The ability of the EECW System to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the FSAR, Chapters 5 and 14 (Refs. 1 and 2, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ability of the EECW System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 1 and 2. The ability to provide onsite emergency AC power is dependent on the ability of the EECW System to cool the DGs. The long term cooling capability of the RHR and core spray pumps is also dependent on the cooling provided by the EECW System.

The EECW System, together with the UHS, satisfy Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

The EECW loops are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, two EECW pumps are required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, three EECW pumps must be OPERABLE. At least two pumps will operate if the worst single active failure occurs coincident with the loss of offsite power.

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 is based on having a maximum water temperature of 95°F.

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.

(continued)

BASES (continued)

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.

The 7 day Completion Time is based on the redundant EECW System capabilities afforded by the remaining OPERABLE pumps, the low probability of an accident occurring during this time period and is consistent with the allowed Completion Time for restoring an inoperable DG.

B.1 and B.2

If the required EECW pump cannot be restored to OPERABLE status within the associated Completion Time, or two or more EECW pumps are inoperable 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 3.7.2.1

Verification of the UHS temperature ensures that the heat removal capability of the EECW System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

Verifying the correct alignment for each manual and power operated valve in the EECW System flow paths provide assurance that the proper flow paths will exist for EECW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were 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 automatically realigned to its accident position within the required time. 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.

This SR is modified by a Note indicating that 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. As such, when required EECW pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the EECW System is still OPERABLE.

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

SR 3.7.2.3

This SR verifies that the EECW System pumps will automatically start to provide cooling water to the required safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. This SR includes a functional test of the initiation logic and a functional test and calibration of the EECW pump timers (both normal power and diesel power).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.3 (continued)

Operating experience has shown that these components will usually pass the SR when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Chapter 5.

2. FSAR, Chapter 14.

3. FSAR, Section 10.10.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Ventilation (CREV) System

BASES

BACKGROUND

The CREV System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the CREV System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air. The system has a high efficiency particulate air (HEPA) filter bank in the portion of the inlet piping common to both subsystems. Each subsystem consists of a motor-driven fan, an electric duct air heater, an activated charcoal adsorber section, an electric charcoal heater, and the associated ductwork and dampers. The HEPA filter bank removes particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREV System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room. Outside air is taken in through the CREV System ventilation intake and is passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles.

The CREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. A single CREV subsystem will pressurize the control room to about 0.125 inches water gauge to prevent infiltration of air from surrounding buildings and the outdoors. CREV System operation in maintaining control room habitability is discussed in the FSAR, Section 10.12 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the CREV System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, Chapters 10 and 14 (Refs. 2 and 3, respectively). The pressurization mode of the CREV System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the FSAR, Section 14.6 (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 3. No single active failure will cause the loss of filtered outside air from the control room.

The CREV System satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

Two redundant subsystems of the CREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

The CREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. The electric duct heater, ductwork, and dampers are OPERABLE.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the CREV System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one CREV subsystem inoperable, the inoperable CREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREV subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREV System capability. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. 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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

C.1, C.2.1, C.2.2, and C.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREV subsystem may be placed in the pressurization mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES

ACTIONS (continued)

D.1

If both CREV subsystems are inoperable in MODE 1, 2, or 3, the CREV System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDVRs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture that has accumulated in the charcoal as a result of humidity in the ambient air. The CREV System must be operated for ≥ 10 continuous hours with the heaters energized to dry out any moisture and to demonstrate the function of the system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.3.2

This SR verifies that the required CREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.3.3

This SR verifies that on an actual or simulated initiation signal, each CREV subsystem starts and operates. This SR includes verification that dampers necessary for proper CREV operation function as required. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 and SR 3.3.7.1.6 overlaps this SR to provide complete testing of the safety function.

SR 3.7.3.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to outdoors is periodically tested to verify proper function of the CREV System. During the emergency mode of operation, the CREV System is designed to slightly pressurize the control room ≥ 0.125 inches water gauge positive pressure

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.4 (continued)

with respect to the outdoors to prevent unfiltered inleakage. The CREV System is designed to maintain this positive pressure at a flow rate of ≥ 2700 cfm and ≤ 3300 cfm to the control room in the pressurization mode. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration systems SRs.

REFERENCES

1. FSAR, Section 10.12.
2. FSAR, Chapter 10.
3. FSAR, Chapter 14.
4. FSAR, Section 14.6.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Unit 1 and 2 Control Room AC System provides temperature control for the Unit 1 and 2 Control Room following isolation of the control room. The Unit 1 and 2 Control Room AC System consists of two redundant subsystems that provide cooling and heating of recirculated control room air. A subsystem consists of an air handling unit, a chilled water pump, a water chiller, ductwork, dampers, piping, and instrumentation and controls to provide for control room temperature control.

The Unit 1 and 2 Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. A single subsystem provides the required temperature control to maintain the control room temperature within acceptable limits for operation of equipment and for uninterrupted safe occupancy under all plant conditions. The design conditions for the control room environment are 76°F and 50% relative humidity. Each subsystem is capable of maintaining the control room temperature at or below 104°F during abnormal or accident conditions. Alternate methods of cooling the Unit 1 and 2 Control Room are available. These include, but are not limited to, the use of the emergency chiller, the Unit 3 Control Room AC System and the Relay Room AC Systems. The Control Room AC System operation in maintaining the control room temperature is discussed in the FSAR, Section 10.12 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for uninterrupted safe occupancy under normal and accident conditions.

The Control Room AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single failure of a component of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The Control Room AC System satisfies Criterion 3 of the NRC Policy Statement (Ref. 2).

LCO

Two redundant subsystems of the Unit 1 and 2 Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Unit 1 and 2 Control Room AC System is considered OPERABLE when the components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the air handling units, chilled water pumps, water chillers, ductwork, dampers, piping, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

(continued)

BASES

APPLICABILITY (continued)

a. During operations with a potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one Unit 1 and 2 control room AC subsystem inoperable, the inoperable Unit 1 and 2 control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE Unit 1 and 2 control room AC subsystem is adequate to perform the Unit 1 and 2 control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the Unit 1 and 2 control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

B.1, B.2.1, B.2.2

With both Unit 1 and 2 control room AC subsystems inoperable, cooling by a Unit 1 and 2 control AC subsystem must be restored without delay.

Until Unit 1 and 2 control room AC OPERABILITY is re-established, an alternate method of control room cooling must be placed in service within 24 hours. Alternate means should be taken as necessary to maintain the Unit 1 and 2 control room temperature during this Condition. These include, but are not limited to, the use of the emergency chiller, the Unit 3 Control Room AC System and the Relay Room AC System. A Completion Time of 7 days (Required Action B.2.2) is provided to restore at least one Unit 1 and 2 control room AC subsystem to OPERABLE status. A 7 day time period is allowed to restore the function based on the

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2 (continued)

low probability of an event occurring that requires control room isolation, the alternate method of cooling, and the potential for decreased safety if the unit operator's attention is diverted from the actions necessary to restore control room AC to the actions associated with taking the unit to shutdown within this time limit.

C.1 and C.2

In MODE 1, 2, or 3, if the inoperable Unit 1 and 2 control room AC subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. 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.

D.1, D.2.1, D.2.2, and D.2.3

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE Unit 1 and 2 control room AC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

(continued)

BASES

ACTIONS

D.1, D.2.1, D.2.2, and D.2.3 (continued)

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 18 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

REFERENCES

1. FSAR, Section 10.12.
2. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11.3.3 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during abnormal operational transients (e.g., the feedwater controller failure-maximum demand event), as discussed in the FSAR, Section 14.5.1.1 (Ref. 2). Opening the bypass valves during the event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR and MCPR penalties.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR and MCPR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR and MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during abnormal operational transients. The 4 hour Completion Time is 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 3.7.5.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.5.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the cycle specific transient analyses performed to support the preparation of FSAR, Appendix N, Supplemental Reload Licensing Report (Ref. 4). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 7.11.3.3.
2. FSAR, Section 14.5.1.1.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
4. FSAR, Appendix N.

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the FSAR, Section 14.6.4.5 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 14.6.4.
3. NUREG-0800, Section 15.7.4, Revision 1, July 1981.
4. 10 CFR 100.
5. Regulatory Guide 1.25, March 1972.
6. FSAR, Section 14.6.4.5.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 3 ITS BASES (Revised pages marked *R1)

NOTE: Bases Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace pages B 3.7-7 through B 3.7-30 with pages B 3.7-9 R1 through 3.7-32 R1.

B 3.7 PLANT SYSTEMS

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

BASES

BACKGROUND

The EECW System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), residual heat removal (RHR) pump coolers, and room coolers for other Emergency Core Cooling System equipment, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The EECW System also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, the essential loads are provided cooling water by automatically starting RHRSW pumps aligned to EECW headers.

The EECW System, which is common to the three BFN units, consists of the UHS and two independent and redundant loops with each loop consisting of a header, two 4500 gpm pumps, a suction source, valves, piping and associated instrumentation. Two EECW pumps (one per loop or both on one loop) are capable of providing the required cooling capacity to support the required systems. The two loops are separated from each other so failure of one loop will not affect the OPERABILITY of the other. The EECW System is described in the FSAR, Section 10.10 (Ref. 3)

Cooling water is pumped from the Wheeler Reservoir by the EECW pumps to the essential components through the two main headers. After removing heat from the components, the water is discharged back to the Wheeler Reservoir.

APPLICABLE SAFETY ANALYSES

Sufficient water inventory is available for all EECW System post LOCA cooling requirements for a 30 day period with no additional makeup water source available. The ability of the EECW System to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the FSAR, Chapters 5 and 14 (Refs. 1 and 2, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ability of the EECW System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 1 and 2. The ability to provide onsite emergency AC power is dependent on the ability of the EECW System to cool the DGs. The long term cooling capability of the RHR and core spray pumps is also dependent on the cooling provided by the EECW System.

The EECW System, together with the UHS, satisfy Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

The EECW loops are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, two EECW pumps are required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, three EECW pumps must be OPERABLE. At least two pumps will operate if the worst single active failure occurs coincident with the loss of offsite power.

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 is based on having a maximum water temperature of 95°F.

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.

(continued)

BASES (continued)

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.

The 7 day Completion Time is based on the redundant EECW System capabilities afforded by the remaining OPERABLE pumps, the low probability of an accident occurring during this time period and is consistent with the allowed Completion Time for restoring an inoperable DG.

B.1 and B.2

If the required EECW pump cannot be restored to OPERABLE status within the associated Completion Time, or two or more EECW pumps are inoperable 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 3.7.2.1

Verification of the UHS temperature ensures that the heat removal capability of the EECW System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

Verifying the correct alignment for each manual and power operated valve in the EECW System flow paths provide assurance that the proper flow paths will exist for EECW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were 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 automatically realigned to its accident position within the required time. 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.

This SR is modified by a Note indicating that 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. As such, when required EECW pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the EECW System is still OPERABLE.

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

SR 3.7.2.3

This SR verifies that the EECW System pumps will automatically start to provide cooling water to the required safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. This SR includes a functional test of the initiation logic and a functional test and calibration of the EECW pump timers (both normal power and diesel power).

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.2.3 (continued) Operating experience has shown that these components will usually pass the SR when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.
REFERENCES	1. FSAR, Chapter 5. 2. FSAR, Chapter 14. 3. FSAR, Section 10.10. 4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Ventilation (CREV) System

BASES

BACKGROUND

The CREV System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the CREV System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air. The system has a high efficiency particulate air (HEPA) filter bank in the portion of the inlet piping common to both subsystems. Each subsystem consists of a motor-driven fan, an electric duct air heater, an activated charcoal adsorber section, an electric charcoal heater, and the associated ductwork and dampers. The HEPA filter bank removes particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREV System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room. Outside air is taken in through the CREV System ventilation intake and is passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles.

The CREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. A single CREV subsystem will pressurize the control room to about 0.125 inches water gauge to prevent infiltration of air from surrounding buildings and the outdoors. CREV System operation in maintaining control room habitability is discussed in the FSAR, Section 10.12 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the CREV System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, Chapters 10 and 14 (Refs. 2 and 3, respectively). The pressurization mode of the CREV System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the FSAR, Section 14.6 (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 3. No single active failure will cause the loss of filtered outside air from the control room.

The CREV System satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

Two redundant subsystems of the CREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

The CREV System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. The electric duct heater, ductwork, and dampers are OPERABLE.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the CREV System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one CREV subsystem inoperable, the inoperable CREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREV subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREV System capability. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. 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.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

C.1, C.2.1, C.2.2, and C.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREV subsystem may be placed in the pressurization mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES

ACTIONS (continued)

D.1

If both CREV subsystems are inoperable in MODE 1, 2, or 3, the CREV System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDVRs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture that has accumulated in the charcoal as a result of humidity in the ambient air. The CREV System must be operated for ≥ 10 continuous hours with the heaters energized to dry out any moisture and to demonstrate the function of the system. Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.3.2

This SR verifies that the required CREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.3.3

This SR verifies that on an actual or simulated initiation signal, each CREV subsystem starts and operates. This SR includes verification that dampers necessary for proper CREV operation function as required. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.4 and SR 3.3.7.1.6 overlaps this SR to provide complete testing of the safety function.

SR 3.7.3.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to outdoors is periodically tested to verify proper function of the CREV System. During the emergency mode of operation, the CREV System is designed to slightly pressurize the control room ≥ 0.125 inches water gauge positive pressure

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.4 (continued)

with respect to the outdoors to prevent unfiltered inleakage. The CREV System is designed to maintain this positive pressure at a flow rate of ≥ 2700 cfm and ≤ 3300 cfm to the control room in the pressurization mode. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration systems SRs.

REFERENCES

1. FSAR, Section 10.12.
2. FSAR, Chapter 10.
3. FSAR, Chapter 14.
4. FSAR, Section 14.6.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Unit 3 Control Room AC System provides temperature control for the Unit 3 Control Room following isolation of the control room. The Unit 3 Control Room AC System consists of two redundant subsystems that provide cooling and heating of recirculated control room air. A subsystem consists of an air handling unit, a chilled water pump, a water chiller, ductwork, dampers, piping, and instrumentation and controls to provide for control room temperature control.

The Unit 3 Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. A single subsystem provides the required temperature control to maintain the control room temperature within acceptable limits for operation of equipment and for uninterrupted safe occupancy under all plant conditions. The design conditions for the control room environment are 76°F and 50% relative humidity. Each subsystem is capable of maintaining the control room temperature at or below 104°F during abnormal or accident conditions. Alternate methods of cooling the Unit 3 Control Room are available. These include, but are not limited to the Unit 1 and 2 Control Room AC System and the Relay Room AC System. The Control Room AC System operation in maintaining the control room temperature is discussed in the FSAR, Section 10.12 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for uninterrupted safe occupancy under normal and accident conditions.

The Control Room AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single failure of a component of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The Control Room AC System satisfies Criterion 3 of the NRC Policy Statement (Ref. 2).

LCO

Two redundant subsystems of the Unit 3 Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Unit 3 Control Room AC System is considered OPERABLE when the components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the air handling units, chilled water pumps, water chillers, ductwork, dampers, piping, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

(continued)

BASES

APPLICABILITY (continued)

a. During operations with a potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one Unit 3 Control Room AC subsystem inoperable, the inoperable Unit 3 Control Room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE Unit 3 Control Room AC subsystem is adequate to perform the Unit 3 Control Room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the Unit 3 control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

B.1, B.2.1, B.2.2

With both Unit 3 Control Room AC subsystems inoperable, cooling by a Unit 3 control AC subsystem must be restored without delay.

Until Unit 3 Control Room AC OPERABILITY is re-established, an alternate method of control room cooling must be placed in service within 24 hours. Alternate means should be taken as necessary to maintain the Unit 3 control room temperature during this Condition. These include, but are not limited to, the Unit 1 and 2 Control Room AC System and the Relay Room AC System. A Completion Time of 7 days (Required Action B.2.2) is provided to restore at least one Unit 3 Control Room AC subsystem to OPERABLE status. A 7 day time period is allowed to restore the function based on the low

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2 (continued)

probability of an event occurring that requires control room isolation, the alternate method of cooling, and the potential for decreased safety if the unit operator's attention is diverted from the actions necessary to restore Control Room AC to the actions associated with taking the unit to shutdown within this time limit.

C.1 and C.2

In MODE 1, 2, or 3, if the inoperable Unit 3 Control Room AC subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. 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.

D.1, D.2.1, D.2.2, and D.2.3

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE Unit 3 Control Room AC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

(continued)

BASES

ACTIONS

D.1, D.2.1, D.2.2, and D.2.3 (continued)

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 18 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

REFERENCES

1. FSAR, Section 10.12.
2. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11.3.3 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during abnormal operational transients (e.g., the feedwater controller failure-maximum demand event), as discussed in the FSAR, Section 14.5.1.1 (Ref. 2). Opening the bypass valves during the event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR and MCPR penalties.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR and MCPR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR and MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during abnormal operational transients. The 4 hour Completion Time is 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 3.7.5.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.5.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the cycle specific transient analyses performed to support the preparation of FSAR, Appendix N, Supplemental Reload Licensing Report (Ref. 4). The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 18 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 7.11.3.3.
2. FSAR, Section 14.5.1.1.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
4. FSAR, Appendix N.

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section 14.6.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the FSAR, Section 14.6.4.5 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO	The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown. When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.
SURVEILLANCE REQUIREMENTS	SR 3.7.6.1 This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 14.6.4.
3. NUREG-0800, Section 15.7.4, Revision 1, July 1981.
4. 10 CFR 100.
5. Regulatory Guide 1.25, March 1972.
6. FSAR, Section 14.6.4.5.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 1 CURRENT TECHNICAL SPECIFICATIONS MARKUP

NOTE: CTS markup for Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace Specification 3.7.2 page 2 of 8 with page 2 of 8 Rev. 1
Replace Specification 3.7.2 page 3 of 8 with page 3 of 8 Rev. 1
Replace Specification 3.7.3 page 3 of 4 with page 3 of 4 Rev. 1

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~ Applicability Modes 1, 2 + 3 A2

LCO 3.7.2 — A3 — ACTION A+B

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, the RHRSW pumps, including pump D1 or D2, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

See Justification for Changes for BFN ISTS 3.7.1

M1

LCO Requirement for UHS to be OPERABLE

ACTION for inoperable UHS

M1

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Proposed SR 3.7.2.3 M4

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. ~~Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.~~

~~M4~~ L3 LA1

b. ~~Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.~~

SR 3.7.2.2 c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

A4

M1

Proposed SR 3.7.2.1

PAGE 2 OF 8

Rev. 1

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

NOV 0 2 1995

~~LIMITING CONDITIONS FOR OPERATION~~ A1 ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)~~	~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)~~
A2 Applicability MODES 1, 2, & 3 ACTION A & B 2. During REACTOR POWER OPERATION, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.	~~2. No additional surveillance is required.~~
A3 3. During Unit 1 REACTOR POWER OPERATION, both RHRSW pumps D1 and D2 and associated valves normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below. (Note: Because standby coolant supply capability is not a short-term requirement, a component is not considered inoperable if standby coolant supply capability can be restored to service within 5 hours.)	3. Routine surveillance for these pumps is specified in 4.5.C.1. See Justification for Changes for ISTS 3.7.1

PAGE 3 OF 8
Rev. 1

~~3.7/4.7 CONTAINMENT SYSTEMS~~

APR 0 9 1993

~~LIMITING CONDITIONS FOR OPERATION~~	~~SURVEILLANCE REQUIREMENTS~~
~~3.7.E. Control Room Emergency Ventilation~~	~~4.7.E. Control Room Emergency Ventilation~~

A1

SR 3.7.3.4 M3

c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.

A2

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

SR 3.7.3.1

d. Each circuit shall be operated at least 10 hours every month.

M5 Continuous with heater operating

SR 3.7.3.3

3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.

A4 On an actual or simulated signal

ACTION A

3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.

Applicability

A3

ACTIONS B + D

M4

4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within ~~24~~ 36 hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated within 2 hours.

ACTIONS C+E

immediately M2

36 L2

M4 MODE 3 in 12hrs

~~4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the necessary dampers operate as required.~~

LA1

Proposed Required Action C.1.

L1

M1 Proposed Required Actions C.2.2, C.2.3 E.2, E.3

PAGE 3 OF 4

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 2 CURRENT TECHNICAL SPECIFICATIONS MARKUP

NOTE: CTS markup for Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace Specification 3.7.2 page 2 of 8 with page 2 of 8 Rev. 1
Replace Specification 3.7.2 page 3 of 8 with page 3 of 8 Rev. 1
Replace Specification 3.7.3 page 3 of 4 with page 3 of 4 Rev. 1

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Applicability MODES 1, 2&3

LCO 3.7.2

A3

ACTION A&B

1. PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION, A2 the RHRSW pumps, including one of pumps D1, D2, B2 or B1, shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

See Justification for Changes for BFN ISTS 3.7.1

M1

LCO Requirement for UHS to be OPERABLE

Action for inoperable UHS

M1

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Proposed SR 3.7.2.3

L3

M4

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.

LA1

b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.

SR 3.7.2.2

c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit, that is not locked, sealed, or otherwise secured in position, is in its correct position.

A4

M1

Proposed SR 3.7.2.1

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)~~	~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)~~
A2 Applicability MODES 1,2 +3 ACTION A&B A3 2. During REACTOR POWER OPERATION, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.	~~2. No additional surveillance is required.~~
3. During Unit 2 REACTOR POWER OPERATION, any two RHRSW pumps (D1, D2, B1, and B2) and associated valves normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below. (Note: Because standby coolant supply capability is not a short-term requirement, a component is not considered inoperable if standby coolant supply capability can be restored to service within 5 hours.)	3. Routine surveillance for these pumps is specified in 4.5.C.1. See Justification for Changes for BFN ISTS 3.7.1

PAGE 3 OF 8
Rev. 1

~~3.7/4.7 CONTAINMENT SYSTEMS~~

A1

~~LIMITING CONDITIONS FOR OPERATION~~ | ~~SURVEILLANCE REQUIREMENTS~~

~~3.7.E. Control Room Emergency Ventilation~~

A1

~~4.7.E. Control Room Emergency Ventilation~~

SR 3.7.3.4 M3 → c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.

A2 →

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

SR 3.7.3.1
d. Each circuit shall be operated at least 10 hours every month.

M5 continuous with heaters operating

ACTION A
3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.

Applicability

SR 3.7.3.3
3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.

A4 on an actual or simulated signal

M4 MODE 3 in 12 hrs

A3
4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within ~~24~~ hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated ~~within 2 hours~~.

ACTIONS B & D
M4
ACTIONS C & E

Immediately M2

36 L2

4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the necessary dampers operate as required.

LA1

L1 → Proposed Required Action C.1

M1 → Proposed Required Actions C.2.2, C.2.3, E.2, E.3

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

UNIT 3 CURRENT TECHNICAL SPECIFICATIONS MARKUP

NOTE: CTS markup for Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace Specification 3.7.2 page 2 of 8 with page 2 of 8 Rev. 1
Replace Specification 3.7.2 page 3 of 8 with page 3 of 8 Rev. 1
Replace Specification 3.7.3 page 3 of 4 with page 3 of 4 Rev. 1

NOV 05 1990

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

A1

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Applicability MODES 1, 2+3 A2

LCO 3.7.2

A3

ACTION A & B

1. ~~PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION,~~ the RHRSW pumps, ~~including pump B1 or B2,~~ shall be OPERABLE and assigned to service as indicated in Table 3.5-1.

See Justification for Changes for BFN ISTS 3.7.1

M1

LCO Requirements for UHS to be OPERABLE

ACTION for inoperable UHS

M1

~~SURVEILLANCE REQUIREMENTS~~

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS)~~

Proposed SR 3.7.2.3

L3

M4

1. a. Each of the RHRSW pumps normally assigned to automatic service on the EECW headers will be tested automatically each time the diesel generators are tested. ~~Each of the RHRSW pumps and all associated essential control valves for the EECW headers and RHR heat exchanger headers shall be demonstrated to be OPERABLE in accordance with Specification 1.0.MM.~~

LA1

~~b. Annually each RHRSW pump shall be flow-rate tested. To be considered OPERABLE, each pump shall pump at least 4500 gpm through its normally assigned flow path.~~

SR 3.7.2.2

c. Monthly verify that each valve (manual, power-operated, ~~or automatic~~) in the flowpath servicing safety-related equipment in the affected unit that is not locked, sealed, or otherwise secured in position, is in its correct position.

A4

M1

Proposed SR 3.7.2.1

PAGE 2 OF 8

Rev.1

~~3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS~~

~~LIMITING CONDITIONS FOR OPERATION~~ A1 ~~SURVEILLANCE REQUIREMENTS~~

~~3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)~~

A2 Applicability MODES 1,2,+3

ACTION A&B

A3

2. During REACTOR POWER OPERATION, RHRSW pumps must be OPERABLE and assigned to service as indicated in Table 3.5-1 for the specified time limits.

3. During Unit 3 REACTOR POWER OPERATION, both RHRSW pumps B1 and B2 and associated valves normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be OPERABLE except as specified in 3.5.C.4 and 3.5.C.5 below. (Note: Because standby coolant supply capability is not a short-term requirement, a component is not considered inoperable if standby coolant supply capability can be restored to service within 5 hours.)

~~4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)~~

~~2. No additional surveillance is required.~~

3. Routine surveillance for these pumps is specified in 4.5.C.1.

See Justification for Changes for BFN ISTS 3.7.1

PAGE 3 OF 8

Rev. 1

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.7.E. Control Room Emergency Ventilation~~ (A1)

SR 3.7.3.4 (M3) c. System flow rate shall be shown to within ±10% design flow when tested in accordance with ANSI N510-1975. (A2)

├ 3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.

ACTION A

Applicability

(A3)

├ 4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within ~~24~~ hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated ~~within 2 hours.~~

ACTIONS B + D

(M4) MODE 3 in 12hrs

ACTIONS C+E

Immediately (M2) 36 (L2)

(L1) → Proposed Required Action C.1

(M1) → Proposed Required Actions C.2.2, C.2.3 E.2, E.3

~~SURVEILLANCE REQUIREMENTS~~

~~4.7.E. Control Room Emergency Ventilation~~

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

SR 3.7.3.1 d. Each circuit shall be operated at least 10 hours every month.

(M5) Continuous with heaters operating

SR 3.7.3.3

3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.

(A4) on an actual or simulated signal

4. ~~During the simulated automatic actuation test of this system (see Table 4.2.C), it shall be verified that the necessary dampers operate as required.~~

(LA1)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 1)

NOTE: DOCs for Section 3.7.1 are not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace BFN ITS 3.7.2, pages 1 through 4 with pages 1 through 4, Revision 1.
Replace BFN ITS 3.7.3, pages 1 through 4 with pages 1 through 4, Revision 1.
Replace BFN ITS 3.7.5, page 2 with page 2, Revision 1.
Replace BFN ITS 3.7.6, pages 1 and 2 with pages 1 and 2, Revision 1.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.2 - EECW SYSTEM AND UHS

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The existing Applicability for Emergency Equipment Cooling Water (EECW) System Operability (3.5.C.1 & 2) requires the system to be Operable PRIOR TO STARTUP from a COLD SHUTDOWN CONDITION and during REACTOR POWER OPERATION. CTS 3.5.C.6 requires the unit to be placed in a COLD SHUTDOWN CONDITION when these two Specifications cannot be met. The proposed change (LCO 3.7.2 Applicability) requires the system to be Operable in Modes 1, 2 and 3. In Modes 1, 2, and 3, the EECW System is required to be OPERABLE to support the OPERABILITY of the support systems (e.g., Diesel Generators). This change more clearly defines the conditions when the EECW System is required to be Operable without changing the specific requirements which are currently indicated by CTS 3.5.C.1, 2, & 6. This change is administrative because the same requirements for Operability currently listed in specific specifications will be labelled APPLICABILITY and applied to ISTS Specification 3.7.2.

A3 Current Technical Specification 3.5.C.1 & 2 minimum equipment OPERABILITY requirements for the RHRSW and EECW Systems are specified in Table 3.5-1. The proposed Specification separates the CTS requirement into two separate LCOs (LCO 3.7.1 and 3.7.2). The EECW System is common to the three BFN units with two completely redundant and independent headers supplying the three BFN units. There are four pumps dedicated to EECW service and another 4 RHRSW pumps that can be aligned for EECW service. Each EECW pump is fed from a separate Shutdown board. Two EECW pumps can supply the minimum essential EECW requirements for three unit operation.

CTS Table 3.5-1, Minimum RHRSW and EECW Pump Assignment, requires 3 EECW pumps to be OPERABLE when 1, 2 or 3 units are fueled and specifies that at least one OPERABLE pump must be assigned to each header. Since only

two pumps can be assigned per header and three are required to be OPERABLE, this requirement will always be met when the LCO is met. The CTS Table 3.5-1 "row" for the 7 day time limit requires 2 pumps to be OPERABLE. The current Specification also requires the pumps be separated between headers when in the 7 day LCO time limit. This has been deleted based on Justification L1 below.

A4 The requirement to verify automatic valves are in their correct position has been deleted since there are no automatic valves in the flow path for EECW.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 LCO 3.7.2 adds the requirement that the Ultimate Heat Sink be OPERABLE. This is required since the Ultimate Heat Sink (Wheeler Reservoir) is assumed in the safety analysis. Appropriate ACTIONS and Surveillance Requirements are also added. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.

M2 An additional requirement is being added that requires the plant to be in MODE 3 within 12 hours. This change is more restrictive because it stipulates that the reactor shutdown be completed much earlier than would be required by the existing specifications (CTS 3.5.C.6). CTS require a shutdown to MODE 4 within 24 hours but does not stipulate how quickly MODE 3 must be reached. Reference Comment L2 which addresses the less restrictive change of being in MODE 4 in 36 hours rather than 24 hours.

M3 CTS Table 3.5-1 provides a 30 day allowed outage time when 3 RHRSW pumps and 2 EECW pumps are OPERABLE with one unit fueled, 5 RHRSW pumps and 2 EECW pumps OPERABLE with two units fueled, and 7 RHRSW pumps and 2 EECW pumps OPERABLE with three units fueled. The 30 days is apparently based on having RHRSW pumps in excess of what is needed. However, no credit can be taken for the RHRSW pumps if they are not aligned for EECW operation. Therefore, the 30 day allowed outage time has been deleted.

M4 An explicit requirement has been added to verify EECW pumps actuate on an actual or simulated initiation signal. The proposed SR is more restrictive since it adds an explicit Technical Specification requirement that did not exist before. This change is consistent with BWR Standard Technical Specifications, NUREG-1433.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.2 - EECW SYSTEM AND UHS

LA1 This Surveillance has been relocated to the Inservice Testing (IST) Program. The overall IST Program is still required by the BFN Improved Standard Technical Specifications (Specification 5.5.6) and requires testing of these types of pumps. Any change to this specific test will be controlled by the provisions of the licensee controlled programs. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.

LA2 Not Used.

LA3 The requirements for RHRSW (EECW) pump timers, currently located in CTS Table 3.2.B/4.2.B, are included in the LCO, Actions, and Surveillance Requirements for proposed BFN ISTS 3.7.2, "EECW System and UHS." Proposed SR 3.7.2.3 ensures the pump actuates on an actual or simulated signal and is considered to include a test of the EECW timer function. The details of the RHRSW timer functions are relocated to the Technical Requirements Manual and implementing procedures. Changes to the Technical Requirements Manual are controlled in accordance with 10 CFR 50.59. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.

"Specific"

L1 Note (A) to Table 3.5-1 requires that at least one OPERABLE pump be assigned to each header. This implies that some additional protection is provided by this requirement. However, the current BFN design criteria does not require an OPERABLE pump from each header. The proposed Specification is less restrictive in that it does not require the two subsystems (pumps) to be from separate headers. This is acceptable since either two pumps on one header or one pump on each header are capable of providing the required cooling to safety related components on the three BFN units.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.2 - EECW SYSTEM AND UHS

L2 The time to reach MODE 4, Cold Shutdown, has been extended from 24 hours to 36 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, a new (more restrictive) requirement to be in MODE 3 (Hot Shutdown) within 12 hours has been added. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

L3 CTS 4.5.C.1.a, related to EECW pumps starting automatically when a diesel generator starts is a statement of fact describing the way in which the EECW System works. Whenever a diesel generator starts, EECW pumps(s) associated with that diesel generator will automatically start, if they are not already running.

Since this same provision is not carried forward explicitly into ITS, the change is considered less restrictive. An alternative surveillance requirement, SR 3.7.2.3, is established in ITS to periodically verify that EECW pumps will automatically start on initiation signals which includes EDG starts. EECW pumps starts are also verified by plant operating instructions.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.3
CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The technical content of this requirement is being moved to Chapter 5.0 of the proposed Technical Specifications in accordance with the format of the BWR Standard Technical Specifications, NUREG 1433. Any technical changes to this requirement will be addressed with the content of proposed Specification 5.5.7. A Surveillance Requirement is added (proposed SR 3.7.3.2) to clarify that the tests of the Ventilation Filter Testing Program must also be completed and passed for determining OPERABILITY of the CREV System. Since this is a presentation preference that maintains current requirements, this change is considered administrative.

A3 This phrase has been reworded to be more specific as to what is inoperable. Specifically, Conditions D and E now specify "Two CREV subsystems inoperable...", since these are the only Conditions needed (ACTIONS A, B, and C provide the proper actions when one subsystem is inoperable). This change is a presentation preference only, therefore, it is considered administrative in nature.

A4 The phrase "on an actual or simulated signal" in reference to automatic initiation of CREVs has been added to the Surveillance Requirement. This clarifies that satisfactory automatic system initiations from either an actual or simulated signal can be used to fulfill the Surveillance Requirement. Operability is adequately demonstrated in either case since the subsystem itself cannot discriminate between "actual" or "simulated."

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.3
CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 New Required Actions have been added (proposed required Actions C.2.2, C.2.3, E.2 and E.3) to suspend CORE ALTERATIONS and operations with the potential for draining the reactor vessel, as well as the currently required suspension of irradiated fuel handling operations. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433, and is more restrictive on plant operations.

M2 The time allowed to suspend irradiated fuel handling has been changed from 2 hours to immediately, with a caveat in the Bases that suspension includes placing any fuel in a safe position. This is consistent with the BWR Standard Technical Specifications, NUREG 1433. Since the current Specification allows continued fuel handling for up to 2 hours, this change is more restrictive on plant operations.

M3 The current Technical Specifications do not require demonstration that operation of one CREV subsystem with the required flow rate of +/- 10% of design flow will result in a positive pressure in the Control Room relative to the outdoors. However, the ability of one CREV subsystem to maintain a positive pressure of 1/8 inch water gauge with one CREV fan operating at 3000 cfm +/- 10% is an assumption of the safety analyses as discussed in FSAR Section 10.12. Therefore, proposed SR 3.7.3.4 will include verification that each CREV subsystem can maintain a positive pressure of 0.125 inch water gauge relative to the outdoors during the flow verification of each CREV subsystem fan. This change is consistent with NUREG-1433.

M4 A requirement has been added (proposed Required Action B.1) requiring the unit to be in MODE 3 within 12 hours if the inoperable CREV subsystem is not restored to OPERABLE status within 7 days. In addition, Proposed ACTION D requires LCO 3.0.3 to be entered when both CREV subsystems are inoperable, whereas CTS 3.7.E.4 requires the plant be in COLD SHUTDOWN within 24 hours when both subsystems are inoperable. LCO 3.0.3 requires the unit be placed in MODE 2 within 7 hours and MODE 3 (HOT SHUTDOWN) within 13 hours. This change is consistent with BWR Standard Technical Specifications, NUREG-1433, and is more restrictive on plant operations.

M5 Proposed SR 3.7.3.1 requires each CREV subsystem to be operated with the heaters operating $\geq$ 10 continuous hours every 31 days. CTS 3.7.E.2.d requires each circuit to be operated at least 10 hours every month but does not specify the hours be continuous. The proposed SR is more restrictive since it adds an explicit Technical Specification requirement that did not exist before. However, the proposed change does not operationally represent a change since the current BFN surveillance instruction requires this test to be run over 10 continuous

hours.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 The details relating to methods of performing Surveillances have been relocated to the Bases. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications.

"Specific"

L1 An alternative is proposed to suspending operations if a CREV subsystem cannot be returned to OPERABLE status within seven days and movement of irradiated fuel assemblies, CORE ALTERATIONS, or OPDRVs are being conducted. The alternative is to initiate the OPERABLE CREV subsystem and continue to conduct the operations. This action is acceptable since it ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

L2 The time allowed to place the unit in COLD SHUTDOWN (MODE 4) when an inoperable CREV subsystem can not be restored within 7 days has been changed from 24 hours to 36 hours (Proposed Required Action B.2). In addition, Proposed ACTION D requires LCO 3.0.3 to be entered when both CREV subsystems are inoperable, whereas CTS 3.7.E.4 requires the plant be in COLD SHUTDOWN within 24 hours. LCO 3.0.3 requires the unit be placed in MODE 4 within 37 hours. This provides the necessary time to shut down and cool down the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. In addition, new (more restrictive) requirements to be in MODE 2 (Startup) and MODE 3 (Hot Shutdown) (LCO 3.0.3 and Proposed Required Action B.1) within a shorter time period have been added (Refer to M4 above). These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

L3 CTS 3.7.E.1 requires CREVs to be OPERABLE at all times when any reactor vessel contains irradiated fuel. Proposed LCO 3.7.3 Applicability of MODES 1, 2, and 3, during movement of irradiated fuel in the secondary containment, during CORE ALTERATIONS, and during OPDRVs is less restrictive. The proposed LCO applicability is acceptable since the probability and consequences of a DBA are reduced in MODES 4 and 5 because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREV System OPERABLE is not required in MODES 4 and 5 except for situations under which significant radioactive releases can be postulated. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.5
THE MAIN TURBINE BYPASS SYSTEM

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 A new Specification is being added requiring the Main Turbine Bypass System to be OPERABLE. Appropriate ACTIONS and Surveillance Requirements are also added. The proposed LCO will require the Main Turbine Bypass System to be OPERABLE or APLHGR and MCPR penalties be applied. This is consistent with the BWR Standard Technical Specifications, NUREG 1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.6
SPENT FUEL STORAGE POOL WATER LEVEL

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. As a result the Technical Specifications should be more readily readable, and therefore, understandable by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

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TECHNICAL CHANGE - MORE RESTRICTIVE

M1 An ACTION is being added (ACTION A) to provide proper requirements when the water level does not meet the LCO requirements. Currently, no Actions are provided. The Action will suspend movement of irradiated fuel in the spent fuel pool, thus an accident cannot occur. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433, and is an additional restrictions on plant operations.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 The details relating to methods of performing Surveillances have been relocated to the procedures. Changes to the procedures will be controlled by the licensee controlled programs which include a review for 10 CFR 50.59 applicabillity.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.7.6
SPENT FUEL STORAGE POOL WATER LEVEL

"Specific"

L1 The proposed change deletes the requirement to maintain spent fuel pool level at all times whenever irradiated fuel is stored in the spent fuel pool. The remaining applicability will require a specified level be maintained during movement of irradiated fuel assemblies in the spent fuel storage pool. The requirement for a certain level in the spent fuel storage pool is only required when moving irradiated fuel. The current Technical Specifications imply the specification is applicable whenever irradiated fuel is stored in the pool. The fuel handling accident assumes a minimum water level above the irradiated fuel assemblies and that an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies. This proposed change, while relaxing the current Applicability, maintains the assumptions of the bounding design basis fuel handling accident. This is consistent with NUREG-1433.

L2 This change relaxes the Surveillance Requirement frequency to verify spent fuel pool water level from daily to once every 7 days. The 7 day frequency is acceptable, based on operating experience, considering that the water volume in spent fuel storage pool is normally stable, and all water level changes are controlled by plant procedures. This change is consistent with NUREG-1433.

<u>TECHNICAL CHANGES - RELOCATIONS</u>

R1 This change relocates temperature and chemistry requirements of the spent fuel pool. These requirements will be relocated to the Technical Requirements Manual (TRM). Changes to the TRM are controlled in accordance with 10 CFR 50.59. This change is consistent with NUREG-1433.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS MARKUP

NOTE: Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace page 3.7-6 with page 3.7-6 Rev1.
Replace page 3.7-12 with page 3.7-12 Rev1.
Replace page 3.7-18 with page 3.7-18 Rev1.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.2.4~~	~~Operate each [PSW] cooling tower fan for ≥ [15] minutes.~~	~~31 days~~ B3
SR 3.7.2.~~5~~ 2 P3	-------------------NOTE------------------- Isolation of flow to individual components does not render ~~[PSW]~~ EECW System inoperable. -- B1 Verify each ~~[PSW]~~ ~~sub~~system manual and power operated~~, and automatic~~ 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. P2	31 days
SR 3.7.2.~~6~~ 3 P3	Verify each EECW B1 ~~[PSW]~~ ~~subsystem~~ pump P2 actuates on an actual or required P2 simulated initiation signal.	~~[~~18~~]~~ months B2

PAGE 319 OF 478
Rev. 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.~~4~~ 3.3 P3	Verify each CREV B1 ~~[MCREC]~~ subsystem actuates on an actual or simulated initiation signal.	~~[18]~~ months
SR 3.7.~~4~~ 3.4	Verify each CREV B1 ~~[MCREC]~~ subsystem can maintain a positive pressure of ≥ ~~[0.1]~~ 0.125 B2 inches water gauge relative to the outdoors ~~[turbine building]~~ B1 during the ~~[~~pressurization~~]~~ mode of operation at a flow rate of ≤ ~~[400]~~ ≥ 2700 cfm and ≤ 3300 B2 cfm.	~~[18]~~ months B2 ~~on a STAGGERED TEST BASIS~~ "Reinstate" B1

PAGE 325 OF 478

Rev.1

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

3.7 PLANT SYSTEMS

3.7.5 ~~The~~ Main Turbine Bypass System

P11 P3

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

B1 b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the ~~{~~COLR~~}~~ ~~are made applicable~~.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~[~~Requirements of the LCO not met ~~or Main Turbine Bypass System inoperable~~~~]~~	A.1 ~~[~~Satisfy the requirements of the LCO ~~or restore Main Turbine Bypass System to OPERABLE status~~~~]~~ (B1)	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 (P3)	Verify one complete cycle of each main turbine bypass valve.	31 days

(continued)

PAGE 332 OF 478
Rev. 1

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS BASES MARKUP

NOTE: Bases Section 3.7.1 is not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace page B 3.7-7 with page B 3.7-7, Rev1.
Replace page B 3.7-13 with page B 3.7-13, Rev1.
Replace page B 3.7-18 with page B 3.7-18, Rev1.
Replace page B 3.7-19 with page B 3.7-19, Rev1.
Replace page B 3.7-23 with page B 3.7-23, Rev1.
Replace page B 3.7-24 with page B 3.7-24, Rev1.
Replace page B 3.7-25 with page B 3.7-25, Rev1.
Replace INSERT B 3.7-25A with INSERT B 3.7-25A, Rev1.
Replace page B 3.7-26 with page B 3.7-26, Rev1.
Replace page B 3.7-27 with page B 3.7-27, Rev1.
Replace INSERT B 3.7-27A with INSERT B 3.7-27A, Rev1.
Replace page B 3.7-33 with page B 3.7-33, Rev1.
Replace page B 3.7-34 with page B 3.7-34, Rev1.
Replace page B 3.7-35 with page B 3.7-35, Rev1.
Replace page B 3.7-37 with page B 3.7-37, Rev1.

B 3.7 PLANT SYSTEMS

B 3.7.2 Emergency Equipment Cooling Water (EECW) ~~[Plant Service Water (PSW)]~~ System and {Ultimate Heat Sink [UHS]} B1

BASES P2 except as marked

BACKGROUND B1 The EECW ~~[PSW]~~ System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), residual heat removal (RHR) pump coolers, and room coolers for other ~~Emergency Core Cooling System~~ equipment, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The EECW ~~[PSW]~~ System (B1) also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, nonessential loads are automatically isolated, the essential loads are provided cooling water by automatically starting RHRSW pumps aligned to EECW header ~~automatically divided between [PSW] Divisions 1 and 2, and one [PSW] pump is automatically started in each division.~~

B1 The EECW ~~{PSW}~~ System, which is common to the three BFN units, consists ~~of~~ with the {UHS} and two independent and redundant loops ~~subsystems~~. Each ~~of the two [PSW] subsystems is made up~~ loop consisting of a header, two 4500 ~~[8500]~~ gpm pumps, a suction source, valves, piping and associated instrumentation. B2 Two EECW pumps (one per loop or both on one loop) ~~Either of the two subsystems is~~ capable of providing the required cooling capacity to support the required systems ~~with one pump operating~~. The two loops ~~subsystems~~ are separated from each other so failure of one loop ~~subsystem~~ will not affect the OPERABILITY of the other loop ~~system~~. The EECW System is described in the FSAR, Section 10.10 (Ref. 3)

Cooling water is pumped from ~~the~~ (B1) Wheeler Reservoir ~~[Altamaha River]~~ by the B1 EECW ~~[PSW]~~ pumps to the essential components through the two main headers. After removing heat from the components, the water is discharged back to the ~~circulating water flume to replace evaporation losses from the circulating water system, or directly to the river via a bypass valve.~~ Wheeler Reservoir. P12

APPLICABLE SAFETY ANALYSES Sufficient water inventory is available for all EECW (B1) ~~[PSW]~~ System post LOCA cooling requirements for a 30 day period with no additional makeup water source available. The ability of the EECW ~~[PSW]~~ System to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the FSAR, Chapters 5 ~~[4]~~ and 14 ~~[6]~~ (Refs. 1 ~~2~~ and 2 ~~3~~ P3) respectively). These B1

(continued)

This SR includes a functional test of the initiation logic and a functional test ~~of~~ and calibration of the EECW pump timers (both normal power and diesel power).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.~~8~~ 3 (continued) P3

initiation signal. ~~This SR also verifies the automatic start capability of one of the two [PSW] pumps in each subsystem.~~ P2

Operating experience has shown that these components will usually pass the SR when performed at the ~~[~~18~~]~~ month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint. B2 P11

REFERENCES

1. FSAR, Chapter ~~[4]~~ 5. B1
2. FSAR, Chapter ~~[6]~~ 14. B1
3. FSAR, Section 10.10

P12 ~~3.~~ 4 NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7 ~~.4~~ 3 P3 ~~[Main~~ Control Room Emergency Ventilation (CREV) ~~Environmental Control (MCREC)]~~ System

BASES

P6 except as marked

BACKGROUND

B1 The CREV ~~[MCREC]~~ System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

P11 The safety related function of the CREV ~~[MCREC]~~ System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of ~~recirculated air or~~ outside supply air. Each subsystem consists of ~~a demister,~~ a motor-driven fan, an electric duct air heater ~~an electric heater, a prefilter,~~ a high efficiency particulate air (HEPA) filter bank, an activated charcoal adsorber section, ~~a second HEPA filter, a booster fan, an air handling unit (excluding the condensing unit),~~ and the associated ductwork and dampers. ~~Demisters remove water droplets from the airstream. Prefilters and~~ The HEPA filter bank removes particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

Common

charcoal

Common Sect

The ~~[MCREC]~~ System is a standby system, parts of which also operate during normal unit operations to maintain the control room environment. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the ~~[MCREC]~~ CREV System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room~~, and a part of the recirculated air is routed through either of the two filter subsystems~~. Outside air is taken in ~~at the normal~~ through the CREV system ventilation intake and is ~~mixed with the recirculated air before being~~ passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles. B1

B1 The CREV ~~[MCREC]~~ System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. A single CREV ~~[MCREC]~~ subsystem will pressurize the control room to about ~~[0.1]~~ 0.125 inches water gauge to prevent infiltration of air from surrounding buildings and the outdoors. ~~[MCREC]~~ CREV System operation in maintaining control B1 B2

B1

(continued)

BASES

BACKGROUND (continued)

room habitability is discussed in the FSAR, Section 10.12 ~~Chapters [6] and [9]~~, (Refs. 1 ~~and 2, respectively~~). B1

APPLICABLE SAFETY ANALYSES

The ability of the CREV ~~[MCREC]~~ System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, Chapters ~~[6]~~ 10 and 14 ~~[15]~~ (Refs. 2 and 3, respectively). The pressurization mode of the CREV ~~[MCREC]~~ System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the FSAR, Section 14.6 ~~[6.4.1.2.2]~~ (Ref. 4). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 3. No single active ~~or passive~~ failure will cause the loss of outside or filtered ~~recirculated~~ air from the control room. P6

The ~~[MCREC]~~ System satisfies Criterion 3 of the NRC Policy Statement (Ref. 6). P12

B1

LCO

Two redundant subsystems of the CREV ~~[MCREC]~~ System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

B1 The CREV ~~[MCREC]~~ System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. The electric duct Heater~~s~~, ~~demister~~, ductwork, ~~valves,~~ and dampers are OPERABLE~~, and air circulation can be maintained~~. P6

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

(continued)

BASES

P3

SURVEILLANCE REQUIREMENTS

SR 3.7.~~4~~3.1 (continued)

moisture that has accumulated in the charcoal as a result of humidity in the ambient air. ~~[Systems with heaters~~ The CREV must be operated for ≥ 10 continuous hours with the heaters energized ~~. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.]~~ to dry out any moisture and to demonstrate the function of the system. B1 Furthermore, the 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.~~4~~3.2 P3

B1

This SR verifies that the required ~~[MCREC]~~ CREV testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP) ~~]~~. ~~The [MCREC] filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5).~~ P17 The ~~[~~VFTP ~~]~~ includes B1 testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the B1 ~~[~~VFTP~~]~~.

SR 3.7.~~4~~3.3 P3

P3

This SR verifies that on an actual or simulated initiation signal, each ~~[MCREC]~~ CREV B1 subsystem starts and operates. P17 This SR includes verification that dampers necessary for proper CREV operation function as required. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.7.1.~~5~~ 4 and SR 3.3.7.1.b overlaps this SR to provide complete testing of the safety function. ~~The [18] month Frequency is specified in Reference 5.~~ P17

SR 3.7.~~4~~3.4 P3

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to ~~potentially contaminated adjacent areas (the turbine building),~~ outdoors P6 is periodically tested to verify proper function of the B1 ~~[MCREC]~~ CREV System. During the emergency mode of operation, the ~~[MCREC]~~ CREV System is designed to slightly pressurize the control room ≥ ~~[0.1]~~ 0.125 inches water gauge positive pressure with respect to the ~~turbine building~~ outdoors B2 to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~8~~3.4 (continued) (P3)

prevent unfiltered inleakage. The CREU ~~[MCREC]~~ (B1) System is designed to maintain this positive pressure at a flow rate of ~~≤ [400] cfm~~ ≥ 2700 cfm and ≤ 3300 cfm (B2) to the control room in the pressurization mode. The Frequency of ~~[18]~~ months ~~on a STAGGERED TEST BASIS~~ (B2) is consistent with industry practice and other filtration systems SRs. (P6)

"reinstate (B1)

REFERENCES

1. FSAR, Section 10.12 ~~Chapter [6]~~. (B1)
2. FSAR, Chapter 10 ~~[9]~~. (B1)
3. FSAR, Chapter 14 ~~[15]~~. (B1)
4. FSAR, Section 14.6 ~~[6.4.1.2.2]~~. (B1)
5. ~~Regulatory Guide 1.52, Revision 2, March 1978.~~ (P17)

(P12) 6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

Rev. 1

P7 A subsystem consists of an air handling unit, a chilled water pump, a water chiller, ductwork, dampers, piping, and instrumentation and controls to provide for control room temperature control.

B 3.7 PLANT SYSTEMS

B 3.7 ~~5~~ P3 {Control Room Air Conditioning (AC)} System B1

BASES

Unit 3 — <UNIT 3>
Unit 1 and 2 — <Unit 1 and 2>

BACKGROUND B1 The {Control Room AC} System provides temperature control for the control room following isolation of the control room. P11

B1 The {Control Room AC} System consists of two ~~independent~~, redundant subsystems that provide cooling and heating of recirculated control room air. ~~Each subsystem consists of heating coils, cooling coils, fans, chillers, compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.~~ P7 air handling units P7 100 percent capacity air handling units

<UNIT 3> UNIT 3
<UNITS 1 + 2> Unit 1 and 2
P7 motor-operated
<UNIT 3> UNIT 3 — Unit 1 and 2 P7 — <Unit 1 and 2> P7 the associated

B1 The {Control Room AC} System is designed to provide a controlled environment under both normal and accident conditions. A single subsystem provides the required temperature control to maintain ~~a suitable~~ control room ~~environment for a sustained occupancy of 12 persons~~. The design conditions for the control room environment are 76°F and 50% relative humidity. The {Control Room AC} System operation in maintaining the control room temperature is discussed in the FSAR, Section ~~[6.4]~~ 16.12 (Ref. 1). B1 B1

P7 temperature within acceptable limits for operation of equipment and for uninterrupted safe occupancy under all plant conditions

APPLICABLE SAFETY ANALYSES

The design basis of the {Control Room AC} System is to maintain the control room temperature for ~~a 30 day continuous occupancy~~. uninterrupted safe occupancy under normal and accident conditions B1 P7

P7 INSERT B3.7-25A

The {Control Room AC} System components are arranged in redundant safety related subsystems. During emergency operation, the {Control Room AC} System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single failure of a component of the {Control Room AC} System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The {Control Room AC} System is designed in accordance with Seismic Category I requirements. The {Control Room AC} System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment B1 B1 B1 B1

(continued)

P7

INSERT B3.7-25A (Unit 1 and 2)

Each subsystem is capable of maintaining the control room temperature at or below 104°F during abnormal or accident conditions. Alternate methods of cooling the Unit 1 and 2 Control Room are available. These include, but are not limited to, the use of the emergency chiller~~s~~, the Unit 3 Control Room AC System and the Relay Room AC System.

INSERT B3.7-25A (Unit 3)

Each subsystem is capable of maintaining the control room temperature at or below 104°F during abnormal or accident conditions. Alternate methods of cooling the Unit 3 Control Room are available. These include, but are not limited to, ~~the use of the emergency chillers,~~ the Unit 1 and 2 Control Room AC System and the Relay Room AC System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The {Control Room AC} System satisfies Criterion 3 of the NRC Policy Statement. (Ref. 2) (P12) (B1)

LCO

Two ~~independent and~~ (P7) redundant subsystems of the {Control Room AC} System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits. (B1) Unit 1 and 2 / Unit 3 <Units 1 and 2> <Unit 3>

(B1) The {Control Room AC} System is considered OPERABLE when the ~~individual~~ (P7) components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the ~~cooling coils, fans, chillers, compressors,~~ air handling units, chilled water pumps, water chillers, (P7) ductwork, dampers, piping, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the {Control Room AC} System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation. (B1)

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the {Control Room AC} System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated: (B1)

a. During operations with a potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the {secondary} containment. (B1)

Rev. 1
PAGE 697 OF 939

(continued)

BASES (continued)

<Unit 3> <Units 1 and 2> P7 Unit 3 Unit 1 and 2

ACTIONS

A.1

B1 With one [control room AC] subsystem inoperable, the inoperable [control room AC] subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE [control room AC] subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate safety and nonsafety cooling methods. B1

P7 INSERT B 3.7-27A

C.1 and C.2 (~~B.1~~ and ~~B.2~~) P3

<Unit 3> <Units 1 and 2> P7 Unit 3 Unit 1 and 2

In MODE 1, 2, or 3, if the inoperable [control room AC] subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. 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. B1

D.1, D.2.1, D.2.2, and D.2.3 (~~C.1, C.2.1, C.2.2, and C.2.3~~) P3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

B1 During movement of irradiated fuel assemblies in the [secondary] containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE [control room AC] subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, B1

<Units 1 and 2> Unit 1 and 2 <Unit 3> Unit 3 P7

PAGE 698 OF 939

(continued)

INSERT B3.7-27A Rev. 1 (P7)

B.1, B.2.1, B.2.2 <Unit 3>

Unit 3 <Unit 1 and 2>

With both Unit 1 and 2 Control Room AC subsystems inoperable, cooling by a Unit 1 and 2 Control Room AC subsystem must be restored without delay.

<Unit 1 and 2> Unit 3 <Unit 3>

Until Unit 1 and 2 Control Room AC OPERABILITY is re-established, an alternate method of control room cooling must be placed in service within 24 hours. Alternate means should be taken as necessary to maintain the Unit 1 and 2 control room temperature during this Condition. These include, but are not limited to, the use of the emergency chiller, the Unit 3 Control Room AC System and the Relay Room AC System. A Completion Time of 7 days (Required Action B.2.2) is provided to restore at least one Unit 1 and 2 Control Room AC subsystem to OPERABLE status. A 7 day time period is allowed to restore the function based on the low probability of an event occurring that requires control room isolation, the alternate method of cooling, and the potential for decreased safety if the unit operator's attention is diverted from the actions necessary to restore Control Room AC to the actions associated with taking the unit to shutdown within this time limit.

B 3.7 PLANT SYSTEMS

B 3.7.~~7~~ 5 P3 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is [25]% of the Nuclear Steam Supply System rated steam flow. B2 Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine ~~three~~ P8 valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these ~~three~~ nine P8 valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine ~~Electro~~ Generator ~~Hydraulic~~ Control System, as discussed in Pressure Regulator and P8 the FSAR, Section 7.11.3.3 B1 ~~[7.7.4]~~ (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during abnormal operational transients (e.g., the feedwater controller failure - maximum demand event) P8 ~~the turbine generator load rejection transient,~~ as discussed in the FSAR, Section 14.5.1.1 B1 ~~[15.1.1]~~ (Ref. 2). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in ~~an~~ APLHGR and P18 MCPR penalt~~y~~ies.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement (Ref. 3) P12.

(continued)

BASES (continued)

LCO

P18

APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the

B1

APLHGR and

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. ~~[~~With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met ~~]~~ The MCPR limits for the inoperable Main Turbine Bypass System ~~is~~ are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

P8

abnormal operational

LCO 3.2.1 and

P18

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit ~~and the cladding 1% plastic strain limit are~~ is not violated during ~~the turbine generator load rejection~~ abnormal operational transients. As discussed in the Bases for ~~LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and~~ LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

"Rise... P16

ACTIONS

A.1

P18

APLHGR and

B1

P18

APLHGR and

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR and MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

P18 APLHGR and

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during ~~the turbine generator load rejection~~ abnormal operational P8 transients. The 4 hour Completion Time is 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.~~7~~ 5.1 P3

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.~~7~~ 5.2 P3

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. B2 The ~~[~~18~~]~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the B2 ~~[~~18~~]~~ month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

Rev. L

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.~~8~~ 6 Spent Fuel Storage Pool Water Level

P3

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

10.3 B1

A general description of the spent fuel storage pool design is found in the FSAR, Section ~~[ ]~~ (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section B1 14.6.4 ~~[15.1.4]~~ (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% of 10 CFR 100 (Ref. 3) exposure guidelines NUREG-0800 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the FSAR, Section B1 14.6.4.5 ~~[9.1.2.2.2]~~ (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion~~n~~ a 2 and 3 of the NRC Policy Statement (Ref. 7). P12

P20

PAGE 709 OF 739 (continued)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

JUSTIFICATION FOR CHANGES TO NUREG-1433

NOTE: All pages are provided.

Replace pages 1 through 3 with pages 1 through 4 Revision 1.

JUSTIFICATION FOR CHANGES TO NUREG-1433
SECTION 3.7 - PLANT SYSTEMS

BRACKETED PLANT SPECIFIC INFORMATION

B1 Brackets removed and optional wording preferences revised as necessary to reflect appropriate plant specific requirements.

B2 Brackets removed and values revised as necessary to reflect plant specific design.

B3 Bracketed requirements removed and optional wording deleted.

NON-BRACKETED PLANT SPECIFIC CHANGES

P1 The Conditions of LCO 3.7.1 have been revised to reflect the BFN plant specific design and analyses for the RHRSW System. The BFN RHRSW System is common to the three units and includes four loops with two pumps per loop with each loop providing water to one RHR heat exchanger on each unit. Analyses (which includes consideration for a single failure) requires at least one pump per loop be OPERABLE and a total number of pumps be OPERABLE dependent upon the number of units fueled. Since the BFN design includes excess redundancy, a Condition has been added (similar to the 30 day allowed outage times for RHR Suppression Pool Cooling and Spray) to allow a 30 day allowed outage time with one subsystem or required pump inoperable. The worst additional single failure could not result in a complete loss of RHRSW function, however it would result in reduced containment cooling capability. Also, SR 3.7.1.1 has been revised since the RHRSW System does not include automatic valves.

P2 The Conditions of LCO 3.7.2 have been revised to reflect BFN plant specific design and analyses for the EECW System. The EECW system is common to the three BFN units and includes two loops with two pumps per loop with each loop providing cooling water to safety related components on all three units. Each pump is fed by a separate 4 kV shutdown board and the worst case single failure could take out only one pump. Analyses (which includes consideration for a single failure) require three pumps to be OPERABLE. Also, SR 3.7.2.2 has been revised since there are no automatic valves in the flow paths servicing safety related systems or components.

P3 Renumbering and/or relettering due to additions or deletions.

P4 NUREG-1433 Specification 3.7.3, Diesel Generator (DG) [1/B] Standby Service Water (SSW) System was deleted because no comparable system exists at BFN.

P5 This note has been deleted since BFN does not have a toxic gas protection mode. Per FSAR Section 10.12.5.3, this mode is not necessary.

P6 Changes to reflect BFN plant specific design for the Control Room Emergency Ventilation (CREV) System.

P7 Changes to the Bases were made to reflect BFN plant specific design for the Control Room (CR) Air Conditioning (AC) System. Added proposed BFN ISTS Action B to allow a 7 day allowable out of service time to restore at least one subsystem when both control room subsystems are inoperable provided action is taken immediately to restore the subsystem and that an alternate method of cooling is placed in operation within 24 hours. BFN has the capability of cooling the Units 1 and 2 CR using other AC systems within the CR isolation boundary. These include, but are not limited to, the Unit 3 Control Room AC and Relay Room AC Systems. In addition to the proposed Required Actions, BFN also has procedural controls in place that ensure actions are taken based on Control Room temperature. NUREG LCO 3.7.5, Action B (proposed BFN ISTS Action C), has been modified to require a shutdown to MODE 3 in 12 hours and MODE 4 in 36 hours when the Required Actions and associated Completion Time of proposed BFN ISTS Conditions A or B are not met in MODE 1, 2, and 3. NUREG LCO 3.7.5, Action D, has been deleted. NUREG LCO 3.7.5, Actions C and E, have been consolidated into one Action (proposed Action D) which addresses the condition where the Required Actions of Conditions A or B are not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs. The proposed changes are justified based on the current BFN licensing basis which requires the control room to be maintained habitable and at a temperature that does not affect equipment operability but has no Technical Specification requirements for the CR air conditioning system. BFN uses administrative controls to ensure control room temperature is acceptable and that alternate means for maintaining CR temperatures are taken as necessary.

P8 Changes to reflect BFN plant specific design and analyses for the Main Turbine Bypass System.

P9 Since the same Note is applicable to all Required Actions, it has been consolidated into one and relocated to apply to the ACTIONS.

P10 The 72 hour Completion Time in NUREG-1433 is based on the redundant System capabilities afforded by the OPERABLE subsystem, the low probability of an accident occurring during this time period, and is consistent with the allowed Completion Time for restoring an inoperable DG, which in NUREG-1433 is 72 hours. However, the allowed Completion Time for restoring an inoperable DG in BFN proposed ISTS Specification 3.8.1 is 7 days. Therefore, for consistency the allowed Completion Time for EECW has been changed to 7 days.

P11 Editorial/grammatical correction.

P12 Appropriate reference provided.

P13 Not Used.

P14 Due to the BFN EECW System design, a loss of one subsystem (pump or header) does not result in a loss of EECW to the DGs or affect the RHR shutdown cooling subsystems operability. Therefore, these Notes are inappropriate for BFN and have been deleted.

P15 The gaseous radwaste system at BFN is designed to prevent inadvertent release of significant quantities of gaseous and particulate radioactive material from the restricted area of the plant, so that the resulting radiation exposures are within guideline values of 10 CFR 20. Current BFN Technical Specifications do not include a limit for the air ejector offgas release rate. BFN data for the offgas radiation monitor demonstrates that typical release rates are far below the NUREG limit. Since BFN current licensing basis does not include an air ejector offgas release rate limit and BFN operating experience demonstrates release rates are far below the NUREG limit, the NUREG specification for main condenser offgas has not been included in the proposed BFN ISTS.

P16 Deleted (APLHGR limits have been reinstated in LCO).

P17 References to Regulatory Guide 1.52 have been deleted to be compatible with the previously submitted ITS 5.5.7, Ventilation Filter Testing Program.

P18 Incorporated APLHGR correction factor for an inoperable Main Turbine Bypass System. Discussions with fuel vendor indicates an APLHGR correction factor will likely be needed as a result of future core reload analyses. The limits will be listed in the COLR.

P19 A statement has been added to the BASES for ITS SR 3.7.3.3 to require the SR to include a verification that dampers necessary for proper CREV operation function as required. This addition incorporates a CTS requirement.

P20 In ITS B 3.7.6, Applicable Safety Analyses, a correction has been made to the NRC Policy Statement criteria numbers referenced to incorporate a generic change (TSTF-139, Rev.1).

P21 A statement has been added to the bases for ITS SR 3.7.2.3 to require the SR to include a functional test of the initiation logic and EECW pump timers, and a calibration of the EECW pump timers. This addition incorporates CTS requirements.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.7
LIST OF REVISED PAGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS (Revised pages marked Revision 1)

NOTE: NSHC for Section 3.7.1 are not included. A beyond scope change (TS-395) is being submitted separately, which revises all of that section.

Replace pages 6 through 15 with pages 6 through 16 Revision 1.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.7.2 - EECW SYSTEM AND UHS

<u>TECHNICAL CHANGES - LESS RESTRICTIVE</u>
<u>(L1)</u>

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The EECW System is used to mitigate the consequences of an accident, but is not considered as an initiator of any previously analyzed accident. As such the proposed change, which no longer requires the OPERABLE EECW pumps be from separate headers, does not increase the probability of any accident previously evaluated. Since any two OPERABLE EECW pumps will provide the necessary cooling for the three BFN units during a DBA on one unit, the proposed change does not involve any increase to the consequences of any accident previously evaluated.

2. <u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

   The proposed change does not involve a significant reduction in a margin of safety since the EECW System can provide necessary cooling with any two OPERABLE EECW PUMPS.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L2)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware or operating procedure changes. The EECW System is not assumed to be an initiator of any analyzed event. The change will not allow continuous operation such that a single failure will preclude the affected component's function from being performed. This change allows an additional 12 hours to reach MODE 4, which provides a reasonable amount of time to perform an orderly shutdown, thus further minimizing a potential upset from a too rapid decrease in plant power. Additionally, the consequences of an event occurring while the unit is being shutdown during the extra 12 hours is the same as the consequences of an event occurring for the current 24 hours. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The increased time allowed for reach MODE 4 with inoperable EECW components is acceptable based on the small probability of an event requiring the inoperable EECW components to function and the desire to minimize plant transients. The requested 12 hour extension will provide sufficient time for the unit to reach MODE 4 in an orderly manner. As a result, the potential for human error will be reduced. In addition, the unit is now required to be in MODE 3 within 12 hours (a shutdown

TECHNICAL CHANGES - LESS RESTRICTIVE
(L2) (continued)

condition). As such, any reduction in a margin of safety will be insignificant and offset by the benefit gained from providing sufficient time to reach MODE 4, thus avoiding potential plant transients from attempting to reach MODE 4 in the current time and the benefit of being subcritical (MODE 3) in a shorter required time.

TECHNICAL CHANGES - LESS RESTRICTIVE
(L3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The EECW System is used to mitigate the consequences of an accident, but is not considered as an initiator of any previously analyzed accident. As such, the proposed change, which deletes the verification that associated EECW pump(s) automatically start whenever a diesel generator is started, does not increase the probability of any accident previously evaluated. Since the ITS contains a new surveillance requirement which verifies that EECW pumps start automatically on all initiation signals, the proposed change does not involve any increase to the consequences of any accident previously analyzed.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the ITS contains a requirement which will verify EECW pump automatic start capability.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.7.3
CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM

<u>TECHNICAL CHANGES - LESS RESTRICTIVE</u>
<u>(L1)</u>

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   An alternative is proposed to suspending operations if a CREV subsystem cannot be returned to operable status that would allow continued movement of irradiated fuel assemblies, CORE ALTERATIONS, or operations with the potential for draining the reactor vessel. The alternative is to initiate the operable CREV subsystem and continue to conduct the operations. Operation of the CREV System is not considered as an initiator of a previously evaluated accident. Therefore, the operation does not significantly increase the probability of an accident previously identified. Since one subsystem is sufficient for any accident, the consequences of any previously evaluated accident are not significantly increased.

2. <u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

   This change provides for continued performance of previously evaluated operations. Since these operations have been previously considered, their continued performance does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

   The margin of safety considered in performance of these operations is maintained by starting and running the system that would be required to initiate should an accident occur. Operation of the remaining subsystem ensures that no failures that would prevent actuation will occur, and that any active failure will be readily detected. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.7.3
CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE
(L2)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The change does not result in any hardware or operating procedure changes. CREVs is not assumed to be the initiator of any analyzed event. The change will not allow continuous operation such that a single failure will preclude the affected function from being performed. This change allows an additional 12 or 13 hours to reach MODE 4 when the 7 day allowed outage time for one CREV subsystem is not met or both CREV subsystems are inoperable, which provides a reasonable amount of time to perform an orderly shutdown, thus further minimizing a potential upset from a too rapid decrease in plant power. Additionally, the consequences of an event occurring while the unit is being shutdown during the extra 12 or 13 hours is the same as the consequences of an event occurring in the current time period. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

   The possibility of a new or different kind of accident from any previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve a physical modification to the plant.

3. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

   The increased time allowed for reaching MODE 4 is acceptable based on the small probability of an event requiring it to function and the desire to minimize plant transients. The requested 12 or 13 hour extension will provide sufficient time for the unit to reach MODE 4 in an orderly manner. As a result, the potential for human error will be reduced. In addition, the unit is now required to be in MODE 2 and/or 3 within interim time periods. As such, any reduction in a margin of safety will be insignificant and offset by the benefit gained from providing sufficient time to reach MODE 4, thus avoiding potential plant

TECHNICAL CHANGES - LESS RESTRICTIVE
(L2) (CONTINUED)

transients from attempting to reach MODE 4 in the current time and the benefit of being subcritical (MODE 3) in a shorter required time. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.7.3
CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE
(L3)

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed amendment changes the LCO applicability from at all times irradiated fuel is in any reactor vessel to MODES 1, 2, and 3, during movement of irradiated fuel in the secondary containment, during CORE ALTERATIONS, and during OPDRVs. The proposed LCO applicability is acceptable since the probability and consequences of a DBA are reduced in MODES 4 and 5 because of the pressure and temperature limitations in these MODES. The proposed LCO applicability continues to require the CREV System to be OPERABLE in those situations under which significant radioactive releases can be postulated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This change provides for continued performance of previously evaluated operations. Since these operations have been previously considered, their continued performance does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The ability of the CREV System to maintain the habitability of the control room is not affected since the CREV system is only assumed to operate following a LOCA, fuel handling accident, main steam line break, and control rod drop accident. Since the CREV System continues to be required OPERABLE in those situations in which these types of event could occur, the proposed change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
SECTION 3.7.6
SPENT FUEL STORAGE POOL WATER LEVEL

<u>TECHNICAL CHANGES - LESS RESTRICTIVE</u>
<u>(L1)</u>

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   The proposed change deletes the requirement to maintain a specified level in the spent fuel pool at all times. The remaining applicability will require a specified level be maintained at only during movement of irradiated fuel assemblies in the spent fuel storage pool. The proposed change does not affect the probability of an accident. The spent fuel pool water level is not assumed to be an initiator of any analyzed event. The consequences of an accident are not affected by changing the Applicability to only when moving irradiated fuel assemblies in the spent fuel storage pool. The fuel handling accident assumes a minimum water level above the irradiated fuel assemblies and that an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies. This proposed change, while relaxing the current Applicability, maintains the assumptions of the bounding design basis fuel handling accident. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

   This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>TECHNICAL CHANGES - LESS RESTRICTIVE</u>
<u>(L1) (continued)</u>

3. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

The proposed change only requires the LCO be applicable during movement of irradiated fuel assemblies in the spent fuel storage pool. The margin of safety is not significantly reduced because the fuel handling accident assumes a minimum water level above the irradiated fuel assemblies and that an irradiated fuel assembly is dropped onto an array of irradiated fuel assemblies. This proposed change, while relaxing the current Applicability, maintains the assumptions of the bounding design basis fuel handling accident. The safety analysis assumptions will still be maintained, thus no question of safety exist. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
SECTION 3.7.6
SPENT FUEL STORAGE POOL WATER LEVEL

<u>TECHNICAL CHANGES - LESS RESTRICTIVE</u>
<u>(L2)</u>

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

   This change relaxes the Surveillance Requirement frequency to verify spent fuel storage pool water level from daily to once every 7 days. The proposed change does not affect the probability of an accident. The spent fuel pool water level is not assumed to be an initiator of any analyzed event. The proposed change still provides assurance spent fuel pool water level is maintained consistent with analysis assumptions. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

   This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The proposed amendment does not involve a significant reduction In a margin of safety.</u>

   This change relaxes the Surveillance Requirement frequency to verify spent fuel storage pool water level from daily to once every 7 days. The increased interval for the verification of water level is acceptable since the 7 day frequency has been shown, based on industry operating experience, to be adequate for maintaining the spent fuel storage pool water level within limits. Therefore, the margin of safety is not significantly reduced because the proposed changes to the surveillance Frequency will continue to provide the necessary assurance that spent fuel storage pool water level is being maintained within limits. The safety analysis assumptions will still be maintained, thus no question of safety exist. Therefore, this change does not involve a significant reduction in a margin of safety.

BFN UNIT 1, 2, AND 3 CROSS-REFERENCE MATRIX

CTS NUMBER [*]	BFN ITS NUMBER	NUREG NUMBER	DELETED	RELOCATED TO BASES	RELOCATED TO TRM	RELOCATED TO PROC	RELOCATED CONTROL
Table 3.2.B RHRSW Timers	None	None			YES		10 CFR 50.59
Table 3.5-1 Note A	None	None	YES				
3.5.C.1, 3.5.C.2	3.7.1 Applicability	3.7.1 Applicability					
3.5.C.1, 3.5.C.2	3.7.2 Applicability	3.7.2 Applicability					
3.5.C.1, 3.5.C.2, 3.5.C.6	3.7.2 Action B	3.7.2 Action E					
3.5.C.1, 3.5.C.2, Table 3.5-1	3.7.1 Action A	3.7.1 Action C					
3.5.C.1, 3.5.C.2, Table 3.5-1	3.7.1 Action B	None					
3.5.C.1, 3.5.C.2, Table 3.5-1	3.7.2 Action A	3.7.2 Action D	YES				
3.5.C.1, Table 3.5-1	3.7.1 LCO	3.7.1 LCO					
3.5.C.1, Table 3.5-1	3.7.2 LCO	3.7.2 LCO					
3.5.C.3	None	None				YES	LCP
3.5.C.4	None	None				YES	LCP
3.5.C.5	None	None				YES	LCP
3.5.C.6	3.7.1 Action D	3.7.1 Action E					
3.5.C.7	3.7.1 Applicability	3.7.1 Applicability					
3.7.E.1	3.7.3 LCO	3.7.4 LCO					
3.7.E.1, 3.7.E.3	3.7.3 Applicability	3.7.4 Applicability					
3.7.E.2	5.5.7	5.5.8					
3.7.E.2.c	SR 3.7.3.4	SR 3.7.4.4					
3.7.E.3	3.7.3 Action A	3.7.4 Action A					
3.7.E.4	3.7.3 Action B	3.7.4 Action B					
3.7.E.4	3.7.3 Action C	3.7.4 Action C					
3.7.E.4	3.7.3 Action D	3.7.4 Action D					
3.7.E.4	3.7.3 Action E	3.7.4 Action E					
3.10.C.1	3.7.6 Applicability	3.7.8 Applicability					
3.10.C.1	3.7.6 LCO	3.7.8 LCO	YES				
3.10.C.2	None	None			YES		10 CFR 50.59
3.10.C.3	None	None			YES		10 CFR 50.59

BFN UNIT 1, 2, AND 3 CROSS-REFERENCE MATRIX

CTS NUMBER [*]	BFN ITS NUMBER	NUREG NUMBER	DELETED	RELOCATED TO BASES	RELOCATED TO TRM	RELOCATED TO PROC	RELOCATED CONTROL
4.5.C.1.a, Table 4.2.B - RHRSW	SR 3.7.2.3	SR 3.7.2.6	YES			YES	LCP
4.5.C.1.b	None	None				YES	LCP
4.5.C.1.c	SR 3.7.1.1	SR 3.7.1.1					
4.5.C.1.c	SR 3.7.2.2	SR 3.7.2.5	YES				
4.5.C.3	None	None				YES	LCP
4.7.E.1.a, b, c	5.5.7	5.5.8					
4.7.E.2d	SR 3.7.3.1	SR 3.7.4.1					
4.7.E.3, Table 4.2.G	SR 3.7.3.3	SR 3.7.4.3					
4.7.E.4	None	None		YES			ITS 5.5.10
4.10.C.1	SR 3.7.6.1	SR 3.7.8.1	YES			YES	LCP
4.10.C.2	None	None			YES		10 CFR 50.59
4.10.C.3	None	None			YES		10 CFR 50.59
None	3.7.1 Action C	3.7.1 Action D					
None	3.7.1 Action E	None					
None	3.7.1 Action F	None					
None	3.7.1 Action G	None					
None	3.7.1 Actions Note	3.7.1 Action C					
None	3.7.2 LCO	3.7.2 LCO					
None	3.7.2 Action B	3.7.2 Action E					
None	3.7.3 B.1	3.7.3 B.1					
None	3.7.3 C1	3.7.3 C1					
None	3.7.3 C.2.2	3.7.3 C.2.2					
None	3.7.3 C.2.3	3.7.3 C.2.3					
None	3.7.3 E.2	3.7.3 E.2					
None	3.7.3 E.3	3.7.3 E.3					
None	3.7.4 LCO	3.7.5 LCO					
None	3.7.4 Applicability	3.7.5 Applicability					
None	3.7.4 Action A	3.7.5 Action A					
None	3.7.4 Action B	None					
None	3.7.4 Action C	3.7.5 Action B					
None	3.7.4 Action D	3.7.5 Action C					
None	3.7.5 LCO	3.7.7 LCO					

BFN UNIT 1, 2, AND 3 CROSS-REFERENCE MATRIX

CTS NUMBER [*]	BFN ITS NUMBER	NUREG NUMBER	DELETED	RELOCATED TO BASES	RELOCATED TO TRM	RELOCATED TO PROC	RELOCATED CONTROL
None	3.7.5 Applicability	3.7.7 Applicability					
None	3.7.5 Action A	3.7.7 Action A					
None	3.7.5 Action B	3.7.7 Action B					
None	3.7.6 Action A	3.7.8 Action A					
None	SR 3.7.2.1	SR 3.7.2.3					
None	SR 3.7.3.2	SR 3.7.4.2					
None	SR 3.7.4.1	SR 3.7.5.1					
None	SR 3.7.5.1	SR 3.7.7.1					
None	SR 3.7.5.2	SR 3.7.7.2					
None	SR 3.7.5.3	SR 3.7.7.3					
None	None	3.7.1 Action A					
None	None	3.7.1 Action B					
None	None	3.7.2 Action A					
None	None	3.7.2 Action B					
None	None	3.7.2 Action C					
None	None	3.7.5 Action D					
None	None	3.7.5 Action E					
None	None	Section 3.7.3					
None	None	Section 3.7.6					
None	None	SR 3.7.2.1					
None	None	SR 3.7.2.2					
None	None	SR 3.7.2.4					