Technical Specification Change Request No. 42 (Appendix A)

Replace pages 3/4 5-3, 5 and B 3/4 5-2 with the attached revised pages 3/4 5-3, 5 and B 3/4 5-2.

Proposed Change

Change the minimum HPI injection flow to $\geq$350 gpm for all combinations of 3 injection legs and $\geq$500 gpm for each HPI pump.

Reason for Proposed Change

The ECCS Small Break Analysis has shown that with a certain size break in the Reactor Coolant System on the discharge of the Reactor Coolant Pumps and a simultaneous worst case single failure, the present HPI system would not be adequate to mitigate the accident.

The modification to be able to mitigate this (and other HPI required accidents) includes the permanent opening of the HPI discharge crossover valves. Under this condition, each HPI pump will inject flow through each HPI injection leg.

The minimum HPI flow necessary for core cooling, when HPI is required, is 350 gpm. The minimum total injection flow ensures that at least 350 gpm will be injected to the reactor (through all combinations of three (3) injection legs) assuming the fourth leg is failed and has 30% of the flow from one pump.

Safety Analysis of Proposed Change

The purpose of the surveillance requirements is to provide assurance that proper ECCS flow will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Specification 4.5.2.c ensures that flow for each HPI pump will not exceed its runout conditions when the system is in its minimum resistance configuration since the stop check valves are on the pump discharge and not in the HPI injection legs. The stop check valves are set against an RCS pressure of approximately 600 psig to obtain an HPI pump flow rate of greater than 500 gpm and less than the pump runout flow rate.

The proper flow split between injection points is provided by requiring that at least 350 gpm of the minimum require 500 gpm pump flow be injected through all combinations of three injection legs. This results in the maximum flow rate through any one injection leg of no more than 30% of the minimum required pump flow of 500 gpm, i.e., any one injection leg will have a flow rate $\leq$ 150 gpm when an HPI pump is only pumping 500 gpm.

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Finally, an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analysis is assured by requiring the flow rate for all combinations of three injection legs to be equal to or greater than 350 gpm. This assumes the worst case single failure (loss of power to an HPI train), the subsequent opening of the injection valves on the train that lost power, and the assumption that all the flow through the injection leg with the maximum flow is lost through the small pipe break. It has been shown that this 350 gpm is sufficient to mitigate the worst case small break accident without exceeding the Final Acceptance Criteria of a Peak Clad Temperature of 2200°F. Therefore, this proposed change does not involve an unreviewed safety question.

ECCS SUBSYSTEMS - $T_{avg} \geq 280°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two diverse ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

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2. Verifying the correct position of each mechanical position stop for each of the stop check valves listed in Specification 4.5.2.c.

3. Verifying that the flow switches for the throttle valves listed in Specification 4.5.2.d operate properly.

4. A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

5. Verifying a total leak rate $\leq$ 6 gallons per hour for the LPI system at:

   a) Normal operating pressure or a hydrostatic test pressure of $\geq$ 150 psig for those parts of the system downstream of the pump suction isolation valve, and

   b) $\geq$ 55 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

f. At least once per 18 months, during shutdown, by

1. Verifying that each automatic valve in the flow path actuates to its correct position on a high pressure or low pressure safety injection test signal, as appropriate.

2. Verifying that each HPI and LPI pump test starts automatically upon receipt of a high pressure or low pressure safety injection test signal, as appropriate.

g. Prior to entering MODE 3 with HPI or LPI system modifications that could have altered system flow characteristics, by performance of a flow balance test during shutdown to confirm the following injection flow rates:

HPI System - Single Pump	LPI system - Single Pump
Single pump flow rate $\geq$500 gpm at 600 psig	1. Injection Leg A into the Reactor Coolant System - 2800 to 3100 gpm
Flow rate for all combinations of 3 Injection Legs into the Reactor Coolant System $\geq$350 gpm at 600 psig	2. Injection Leg B into the Reactor Coolant System - 2800 to 3100 gpm

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3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two diverse ECCS subsystems with RCS average temperature $\geq$ 280°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the core flooding tanks is capable of supplying sufficient core cooling to maintain the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures, that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

The purpose of these surveillance requirements is to provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in cold condition following mixing of the BWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

TECHNICAL SPECIFICATION CHANGE REQUEST NO. 47 (APPENDIX A)

Delete and insert, as indicated, the following pages in Appendix A of Operating License DPR-72:

Delete	Insert
V	V
IX	IX
2-6	2-6
3/4 3-2	3/4 3-2
	3/4 3-2a
3/4 3-3	3/4 3-3
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
	3/4 3-7a
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
	3/4 3-17a
3/4 3-20	3/4 3-20
	3/4 4-33
3/4 7-5	3/4 7-5
B 3/4 4-13	B 3/4 4-13

Proposed Changes

The proposed changes are:

1) Addition of flow rate indication for the emergency feedwater system;

2) Addition of the auto start circuitry for the emergency feedwater pumps;

3) Addition of the anticipatory reactor trips; and

4) Change in setpoint for electromatic relief valve actuation.

Reasons for Proposed Change

In Mr. H. R. Denton's (USNRC) letter to Mr. W. P. Stewart (FPC) of July 6, 1979, Mr. Denton stated that "Appropriate Technical Specifications for Limiting Conditions for Operation and for surveillance requirements should be ... provided to the staff within seven days ... (and) should cover:

(1) Addition of flow rate indication for the emergency feedwater system;

(2) Addition of the auto start circuitry for the emergency feedwater pumps;

(3) Addition of the anticipatory reactor trips; and

(4) Changes in setpoint for high pressure reactor trip and PORV (or electromatic relief valve) actuation."

All of the above additions and changes have been performed at Crystal River Unit 3 and this Technical Specification Change Request incorporates them into the Technical Specifications save the change in setpoint for high pressure reactor trip. This change was already incorporated into the Technical Specifications by Amendment 19 to the Crystal River Facility Operating License.

Safety Analysis of Proposed Change

The addition of flow rate indication for the emergency feedwater system will enhance the operation of this system. The operator will be able to determine if adequate feedwater is being supplied to steam generators for heat removal. This change does not diminish any of the requirements applicable to the Safety Analysis and no unreviewed safety question is involved.

The addition of the auto start circuitry for the motor-driven emergency feedwater pump enhances the response of the emergency feedwater system in that there is greater certainty that at least one emergency feedwater pump will start when an emergency feedwater actuation signal is initiated. None of the requirements applicable to the Safety Analysis are diminished by the proposed change and no unreviewed safety question is involved.

The addition of the anticipatory reactor trip insures that the reactor will trip sooner than it would have before the anticipatory trips were installed. The events that will initiate the anticipatory trips would have resulted in a high pressure reactor trip and these anticipatory trips will therefore limit the reactor pressure to a value below the high pressure trip setpoint. None of the requirements applicable to the Safety Analysis are dimimished by the proposed change and no unreviewed safety question is involved.

The change in the setpoint of the electromatic relief valve insures that the reactor will reach its high pressure trip setpoint first. This will reduce the number of challenges to the electromatic relief valve and therefore reduces probability of failure. None of the requirements applicable to the Safety Analysis are diminished by the proposed change and no unreviewed safety question is involved.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

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TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Reactor Containment Vessel Pressure High	≤4 psig	≤4 psig
9. Main Turbine Auto Stop Oil Low(2)	≥45 psig	≥45 psig
10. Main Feedwater Pump Turbine A and B Control Oil Low(2)	≥55 psig	≥55 psig
11. OTSG A and B Level Low-Low(2)	≥18 inches	≥18 inches

(1) Trip may be manually bypassed when RCS pressure ≤1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is ≤5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of ≤1720 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure >1800 psig.

(2) Trip not part of Reactor Protection System

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TABLE 3.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	1	1	1	1, 2, and *	8
2. Nuclear Overpower	4	2	3	1, 2	2#
3. RCS Outlet Temperature--High	4	2	3	1, 2	3#
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	4	2(a)	3	1, 2	2#
5. RCS Pressure--Low	4	2(a)	3	1, 2	3#
6. RCS Pressure--High		2	3	1, 2	3#
7. Variable Low RCS Pressure	4	2(a)	3	1, 2	3#
8. Reactor Containment Pressure--High	4	2	3	1, 2	3#
9. Intermediate Range, Neutron Flux and Rate	2	0	2	1, 2 and *	4
10. Source Range, Neutron Flux and Rate					
A. Startup	2	0	2	2## and *	5
B. Shutdown	2	0	1	3, 4 and 5	6

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TABLE 3.3-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Control Rod Drive Trip Breakers	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
12. Reactor Trip Module	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
13. Shutdown Bypass RCS Pressure--High	4	2	3	2**, 3**, 4**, 5**	6#
14. Main Turbine Auto Stop Oil--Low(1)	1	1	1	1***	8
15. Main Feedwater Pump Turbines A and B Control Oil--Low(1)	2	2	2	1###	8
16. OTSG A and B Level--Low-Low(1)	2	2	2	1, 2, 3, 4	8

(1) Trip not part of Reactor Protection System

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

** When Shutdown Bypass is actuated.

*** Trip function may be bypassed in this MODE with reactor power below 20% RATED THERMAL POWER. Bypass shall be automatically removed when reactor power exceeds 20% RATED THERMAL POWER.

The provisions of Specification 3.0.4 are not applicable.

High voltage to detector may be deenergized above 10^{-10} amps on both Intermediate Range channels.

Trip function may be bypassed in this MODE with reactor power below 10% RATED THERMAL POWER. Bypass shall be automatically removed when reactor power exceeds 10% RATED THERMAL POWER.

(a) Trip may be manually bypassed when RCS pressure $\leq$1720 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is $\leq$5% of RATED THERMAL POWER,

(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of $\leq$1720 psig is imposed, and

(3) The Shutdown Bypass is removed when RCS pressure >1800 psig.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1,

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIMES
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower*	$\leq$0.3 seconds
3. RCS Outlet Temperature--High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE*	$\leq$1.4 seconds
5. RCS Pressure--Low	$\leq$0.5 seconds
6. RCS Pressure--High	$\leq$0.5 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Reactor Containment Pressure--High	Not Applicable
9. Main Turbine Auto Stop Oil Low**	Not Applicable
10. Main Feedwater Pump Turbines A and B Control Oil Low**	Not Applicable
11. OTSG A and B Level Low-Low**	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.
**Trip not part of Reactor Protection System

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TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Nuclear Overpower	S	D(2) and Q(7)	M	1, 2
3. RCS Outlet Temperature--High	S	R	M	1, 2
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	S(4)	M(3) and Q(7,8)	M	1, 2
5. RCS Pressure--Low	S	R	M	1, 2
6. RCS Pressure--High	S	R	M	1, 2
7. Variable Low RCS Pressure	S	R	M	1, 2
8. Reactor Containment Pressure-High	S	R	M	1, 2
9. Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	1, 2 and *
10. Source Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	2, 3, 4, and 5
11. Control Rod Drive Trip Breaker	N.A.	N.A.	M and S/U(1)	1, 2 and *

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TABLE 4.3-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
12. Reactor Trip Module	N.A.	N.A.	M	1, 2 and *
13. Shutdown Bypass RCS Pressure-High	S	R	M	2**, 3**, 4**, 5**
14. Main Turbine Auto Stop Oil--Low#	S	R	M	1
15. Main Feedwater Pump Turbine A and B Control Oil--Low#	S	R	M	1
16. OTSG A and B Level--Low-Low#	S	R	M	1, 2, 3, 4

#Trip not part of Reactor Protection System

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Steam Line Rupture Matrix					
1. Low SG Pressure	2 per steam generator	1 per steam generator	2 per steam generator	1, 2, 3***	10
2. Automatic Actuation Logic	1 per steam generator	1 per steam generator	1 per steam generator	1, 2, 3	10
c. Emergency Feedwater					
1. Main Feedwater Pump Turbines A and B Control Oil Low	2	2	2	1, 2, 3##	10
2. OTSG A and B Level Low-Low	2	2	2	1, 2, 3, 4	10

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TABLE 3.3-3 (Continued)

TABLE NOTATION

* Trip function may be bypassed in this MODE with RCS pressure below 900 psig. Bybass shall be automatically removed when RCS pressure exceeds 1700 psig.

** Trip function may be bypassed in this MODE with RCS pressure below 900 psig. Bypass shall be automatically removed when RCS pressure exceeds 900 psig.

*** Trip function may be bypassed in this MODE when steam generator pressure below 725 psig. Bypass shall be automatically removed when steam generator pressure exceeds 765 psig.

The provisions of Specification 3.0.4 are not applicable.

Trip function may be bypassed in this MODE prior to stopping the operating main feedwater pump. Bypass shall be manually removed after starting the first main feedwater pump.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1.

ACTION 11 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed provided the inoperable channel is placed in the bypassed condition and the minimum channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. REACTOR BLDG. SPRAY		
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	≤ 30 psig See 1.a.2, 3, 4	≤ 30 psig See 1.a.2, 3, 4
b. Automatic Actuation Logic	Not Applicable	Not Applicable
4. OTHER SAFETY SYSTEMS		
a. Reactor Bldg. Purge Exhaust Duct Isolation on High Radioactivity		
Gaseous	*	Not Applicable
b. Steam Line Rupture Matrix		
1. Low SG Pressure	≥ 600 psig	≥ 600 psig
2. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Emergency Feedwater		
1. Main Feedwater Pump Turbines A and B Control Oil Low	≥ 55 psig	≥ 55 psig
2. OTSG A and B Level Low-Low	≥ 18 inches	≥ 18 inches

*Determined by requirements of Appendix "B" Tech. Specs. Section 2.4.2 - Crystal River 3 Operating License No. DPR-72.

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
1. Manual	
a. High Pressure Injection	Not Applicable
b. Low Pressure Injection	Not Applicable
c. Reactor Building Cooling	Not Applicable
d. Reactor Building Isolation	Not Applicable
e. Reactor Building Spray	Not Applicable
f. Reactor Building Purge Isolation	Not Applicable
g. Steam Line Rupture Matrix	
1) Emergency Feedwater Actuation	Not Applicable
2) Feedwater Isolation	Not Applicable
3) Steam Line Isolation	Not Applicable
2. Reactor Building Pressure-High	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*
c. Reactor Building Cooling	25*
d. Reactor Building Isolation	60*
3. Reactor Building Pressure High-High (with HPI Signal)	
a. Reactor Building Spray	56*
4. RCS Pressure Low	
a. High Pressure Injection	25*
5. RCS Pressure Low-Low	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*

TABLE 3.3-5 (continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
6. Low Steam Generator Pressure	
a. Feedwater Isolation	34
b. Steam Line Isolation	5
7. Main Feedwater Pump Turbines A and B Control Oil Low	
a. Emergency Feedwater Actuation	Not Applicable
8. OTSG A and B Level Low-Low	
a. Emergency Feedwater Actuation	Not Applicable

*Diesel Generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BLDG. SPRAY				
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	S	R	M(4)	1, 2, 3
b. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3
4. OTHER SAFETY SYSTEMS				
a. Reactor Bldg. Purge Isolation on High Radioactivity				
1. Gaseous	S	R	M	All Modes
b. Steam Line Rupture Matrix				
1. Low SG Pressure	N/A	R	N/A	1, 2, 3
2. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3(5)
c. Emergency Feedwater				
1. Main Feedwater Pump Turbines A and B Control Oil Low	S	R	N/A	1, 2, 3
2. OTSG A and B Level Low-Low	S	R	N/A	1, 2, 3, 4

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RELIEF VALVE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.11 The pressurizer electromatic relief valve shall be OPERABLE with a lift setting of ≤550 psig.

APPLICABILITY: MODES 4, 5 and 6*

ACTION:

With the pressurizer electromatic relief valve not OPERABLE, restore the valve to OPERABLE status within 1 hour or vent the Reactor Coolant System within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 The pressurizer electromatic relief valve shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the pressurizer electromatic relief isolation valve is open, and

b. At least once per 18 months, during shutdown, by verifying the lift setting.

*With the reactor vessel head not removed.

SURVEILLANCE REQUIREMENTS (CONTINUED)

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

3. Verifying that the emergency feedwater ultrasonic flow rate detector is zero-checked.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on each emergency feedwater actuation test signal.

2. Verifying that the steam turbine driven pump and the motor driven pump start automatically:

a. Upon receipt of an emergency feedwater actuation OTSG A and B level low-low test signal, and

b. Upon receipt of an emergency feedwater actuation main feedwater pump turbines A and B control oil low test signal.

3. Verifying that the operating air accumulators for FWV-39 and FWV-40 maintain $\geq$27 psig for at least one hour when isolated from their air supply.

BASES

All pressure-temperature limit curves are applicable up to the fifth effective full power year. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, and 3.4-3, and 3.4-4.

The pressure and temperature limits shown on Figures 3.4-2 and 3.4-4 for reactor criticallity and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components, except steam generator tubes, ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrate that the valves are fully open at the forces assumed in the safety analysis.

3/4.4.11 RELIEF VALVE

The OPERABILITY of the electomatic relief valve ensures that sufficient overpressure protection is provided and that the RC System pressure stays within Appendix G limits with $T_{avg} \leq 280°F$.

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