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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
License No. DPR-51

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated September 12, 1984, as supplemented November 8, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 91 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 20, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 91

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
40	40
40a	40a
41	41
--	41a
42a	42a
43a	43a
43b	43b
45	45*
45a	45a
45b	45b
45c	45c
45d	45d
--	45e
--	45f
68	68
--	68a
72	72
72a	72a
72b	72b
--	72c
--	72d
73a	73a
105	105
105a	105a

*Overleaf page provided for document completeness.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

3.4.1 The reactor shall not be heated, above 280°F unless the following conditions are met:

1. Capability to remove decay heat by use of two steam generators.

   a. A condensate pump and a main feedwater (MFW) pump, using turbine by-pass valve.

   b. A condensate pump and the auxiliary feedwater (AFW) pump using turbine by-pass valve.

*2. Fourteen of the steam system safety valves are operable.

3. A minimum of 16.3 ft. (107,000 gallons) of water is available in the condensate storage tank.

4. Both EFW pumps and their flow paths are operable.

5. Both main steam block valves and both main feedwater isolation valves are operable.

3.4.2 Initiate functions of the EFIC system which are bypassed at cold shutdown conditions shall have the following minimum operability conditions:

a. "low steam generator pressure" initiate shall be operable when the main steam pressure exceeds 750 psig.

b. "loss of 4 RC pumps" initiate shall be operable when neutron flux exceeds 10% power.

c. "main feedwater pumps tripped" initiate shall be operable when neutron flux exceeds 10% power.

* Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

3.4.3 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig.

3.4.4 Components required to be operable by Specification 3.4.1, 3.4.2, and 3.4.3 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1, 3.4.2 and 3.4.3 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1, 3.4.2, and 3.4.3 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.4.5 If the condition specified in 3.4.1.4 cannot be met:

1. With one EFW flow path inoperable, the unit shall be brought to hot shutdown within 36 hours, and if not restored to an operable status within the next 36 hours, the unit shall be brought to cold shutdown within the next 12 hours or at the maximum safe rate.

2. If both EFW trains are inoperable, restore one train to operable status within one hour or be in hot shutdown within the next 6 hours and cold shutdown within the next 12 hours or at the maximum safe rate.

3. If both EFW trains and the AFW pump are inoperable, the unit shall be immediately run back to $\leq 5\%$ full power with feedwater supplied from the MFW pumps. As soon as an EFW train or the AFW train is operable, the unit shall be placed in cold shutdown within the next 12 hours or at the maximum safe rate.

Bases

The Emergency Feedwater (EFW) system is designed to provide flow sufficient to remove heat load equal to 3½ percent full power operation. The system minimum flow requirement to the steam generator(s) is 500 gpm. This takes into account a single failure, pump recirculation flow, seal leakage and pump wear.

To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when:

1. all four RC pumps are tripped

2. both main feedwater pumps are tripped

3. the level of either steam generator is low

4. either steam generator pressure is low

5. ESAS ECCS actuation (high RB pressure or low RCS pressure)

The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic:

- If both SG's are above 600 psig, supply EFW to both SG's.

- If one SG is below 600 psig, supply EFW to the other SG.

- If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 150 psig, supply EFW only to the SG with the higher pressure.

- If both SG's are below 600 psig and the pressure difference is less than 150 psig, supply EFW to both SG's.

At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.4.2. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.

In the event of loss of main feedwater, feedwater is supplied by the emergency feedwater pumps, one which is powered from an operable emergency bus and one which is powered from an operable steam supply system. Both EFW pumps take suction from the condensate storage tank. Decay heat is removed from a steam generator by steam relief through the turbine bypass, atmospheric dump valves, or safety valves. Fourteen of the steam safety valves will relieve the necessary amount of steam for rated reactor power.

The minimum amount of water in the condensate storage tank would be adequate for about 4.5 hours of operation. This is based on the estimate of the average emergency flow to a steam generator being 390 gpm. This operation time with the volume of water specified would not be reached, since the decay heat removal system could be brought into operation within 4 hours or less.

3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig.

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows:

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be > 3115 VAC but < 3177 VAC.

b. The 460 V emergency bus undervoltage relay setpoints shall be > 423 VAC but < 431 VAC with a time delay setpoint of 8 seconds ±1 second.

3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated:

1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a, items 2 and 36 of Table 4.1-2) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)

2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a, items 2 and 42) at greater than 5% reactor power. (May be bypassed up to 20% reactor power.)

3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.

3.5.1.10 The control room ventilation chlorine detection system instrumentation shall be operable and capable of actuating control room isolation and filtration systems, with alarm/trip setpoints adjusted to actuate at a chlorine concentration of ≤ 5ppm.

3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed.

for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the N /RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. The EFIC can be tested from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activation of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the component. The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip trip may be bypassed up to 20% to allow sufficient margin for bringing the turbine on line at approximately 15%.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state, on-line repairs to the failed device, when practical, will be made and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and, in many cases, make on-line repairs.

The Degraded Voltage Monitoring relay settings are based on the short term starting voltage protection as well as long term running voltage protection. The 4.16 KV undervoltage relay setpoints are based on the allowable starting voltage plus maximum system voltage drops to the motor terminals, which allows approximately 78% of motor rated voltage at the motor terminals. The 460V undervoltage relay setpoint is based on long term motor voltage requirements plus the maximum feedwater voltage drop allowance resulting in a 92% setting of motor rated voltage.

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear power Plants to Assess Plant Conditions During and Following an Accident", December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release", February 1975.

REFERENCE

FSAR, Section 7.1.

Table 3.5.1-1 (Cont'd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM

Functional Unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
1. High pressure injection system (Note 8)					
a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
c. Manual trip pushbutton	2	1	2	1	Notes 1, 5
2. Low pressure injection system (Note 8)					
a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5
c. Manual trip pushbutton	2	1	2	1	Notes 1, 5
3. Reactor building isolation and reactor building cooling system (Note 8)					
a. Reactor building 4 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5

Table 3.5.1-1 (Cont'd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM (Cont'd)

	1	2	3	4	5
	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
4. Reactor building spray pumps (Note 8)					
a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5
5. Reactor building spray valves (Note 8)					
a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5
b. Manual trip pushbutton	2	1	2	1	Notes 1, 5

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM

	1	2	3	4	5
1. EFW Initiation					
a. Manual	2	1	2	1	Note 1

TABLE 3.5.1-1 (cont'd)

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM (Cont'd)

		1	2	3	4	5
		No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
	b. Low Level SG A or B	4/SG	2/SG	2/SG	1	Note 1
	c. Low Pressure, SG A or B	4/SG	2/SG	2/SG	1	Note 1, 19
	d. Loss of Both MFW Pumps and PWR > 10%	4	2	2	1	Note 1
	e. Loss of 4 RC Pumps	4	2	2	1	Note 1, 15
	f. ESAS Actuation Logic Tripped	2	1	2	1	Note 1
2.	SG-A Main Steam Line Isolation					
	a. Manual	2	1	2	1	Note 1
	b. Low SG A Pressure	4	2	2	1	Note 1, 19
3.	SG-B Main Steam Line Isolation					
	a. Manual	2	1	2	1	Note 1
	b. Low SG B Pressure	4	2	2	1	Note 1, 19

TABLE 3.5.1-1 (cont'd)

EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM (Cont'd)

	1	2	3	4	5
	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
OTHER SAFETY RELATED SYSTEMS					
1. Decay heat removal system isolation valve automatic closure and interlock system					
a. Reactor coolant pressure instrument channels	2	1	2	1	Notes 1, 5
b. Core flood isolation valve interlocks	2	1	2	1	Notes 1, 5

TABLE 3.5.1-1 (Cont'd)

OTHER SAFETY RELATED SYSTEMS (Cont'd)

		1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
2.	Pressurizer level channels	2	N/A	2	1	Note 10
3.	Emergency Feedwater Flow channels	2/S.G.	N/A	1	0	Note 10
4.	RCS subcooling margin monitors	2	N/A	1	0	Note 10
5.	Electromatic relief valve flow monitor	2	N/A	1	0	Note 11
6.	Electromatic relief block valve position indicator	1	N/A	1	0	Note 12
7.	Pressurizer code safety valve flow monitors	2/valve	N/A	1/valve	0	Note 10
8.	Degraded Voltage Monitoring					
	a. 4.16KV Emergency Bus Undervoltage	2/Bus	1/Bus	2/Bus	0	Note 14
	b. 460V Emergency Bus Undervoltage	*1/Bus	1/Bus	1/Bus	0	Notes 13, 14
9.	Chlorine Detection Systems	2	1	2	0	Notes 17, 18

*Two undervoltage relays per bus are used with a coincident trip logic (2-out-of-2)

TABLE 3.5.1-1 (Cont'd)

NOTES:

1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition within 12 hours if the requirements of Columns 3 and 4 are not met.

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies.

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, Specification 3.3 shall apply.

7. These channels initiate control rod withdrawal inhibits not reactor trips at ≤10% rated power. Above 10% rated power, those inhibits are bypassed.

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.

TABLE 3.5.1-1 (Cont'd)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

15. This trip function may be bypassed at up to 10% reactor power.

16. This trip function may be bypassed at up to 20% reactor power.

17. With no channel operable, within 1 hour restore the inoperable channels to operable status, or initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protective channels is as follows:

Channels A, B, C, D	Before Startup if shutdown greater than 24 hours
Channel A	One week after startup
Channel B	Two weeks after startup
Channel C	Three weeks after startup
Channel D	Four weeks after startup

The reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.

REFERENCE

FSAR Section 7.1.2.3.4

TABLE 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks
30.	Decay heat removal system isolation valve automatic closure and interlock system	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) Shall also be tested during refueling shutdown prior to repressurization at a pressure greater than 300 but less than 420 psig.
31.	Turbine overspeed trip mechanism	NA	R	NA	
32.	Diesel generator protective relaying starting interlocks and circuitry	M	Q	NA	
33.	Off-site power undervoltage and protective relaying interlocks and circuitry	W	R(1)	R(1)	(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.
34.	Borated water storage tank level indicator	W	NA	R	
35.	Reactor trip upon loss of main feedwater circuitry	M	PC	R	

TABLE 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
36. Boric acid addition tank				
a. Level channel	NA	NA	R	
b. Temperature channel	M	NA	R	
37. Degraded voltage monitoring	W	R	R	
38. Sodium hydroxide tank level indicator	NA	NA	R	
39. Incore neutron detectors	M(1)	NA	NA	(1) Check functioning
40. Emergency plant radiation instruments	M(1)	NA	R	(1) Battery check
41. Reactor trip upon turbine trip circuitry	M	PC	R	
42. Strong motion acceleographs	Q(1)	NA	Q	(1) Battery check
43. ESAS manual trip functions				
a. Switches & logic	NA	P	NA	
b. Logic	NA	M	NA	
44. Reactor manual trip	NA	P	NA	
45. Reactor building sump level	NA	NA	R	
46. EFW flow indication	M	NA	R	

TABLE 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
47. RCS subcooling margin monitor	D	NA	R	
48. Electromatic relief valve flow monitor	D	NA	R	
49. Electromatic relief block valve position indicator	D	NA	R	
50. Pressurizer safety valve flow monitor	D	NA	R	
51. Pressurizer water level indicator	D	NA	R	
52. Control room chlorine detector	D	M	R	
53. EFW initiation				
a. Manual	NA	M	NA	
b. SG low level, SGA or B	S	M	R	
c. Low pressure SGA or B	S	M	R	
d. Loss of both MFW pumps and PWR > 10%	S	M	R	

TABLE 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
e. Loss of 4 RC pumps	S	M	NA	
f. ESAS automatic logic tripped	NA	M	NA	
54. SGA main steam line isolation				
a. Manual	NA	M	NA	
b. SGA pressure low	S	M	R	
55. SGB main steam line isolation				
a. Manual	NA	M	NA	
b. SGB pressure low	S	M	R	
56. EFW valve commands (Vector)				
a. SG A pressure low	S	M	R	
b. SG B pressure low	S	M	R	
c. SG pressure difference SG A pressure> SG B pressure	S	M	R	
SG B pressure> SG A pressure	S	M	R	

TABLE 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A high range level high-high	S	M	R	
e. SG B high range level high-high	S	M	R	

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable

TABLE 4.1-2 (Continued)
Minimum Equipment Test Frequency

Item	Test	Frequency
11. Decay heat removal system isolation valve automatic closure and isolation system	Functioning	Every 18 months
12. Flow limiting annulus on main feedwater line at reactor building penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.
13. Main steam isolation valves	a. Exercise through approximately 10% travel	a. Quarterly
	b. Cycle	b. Every 18 months
14. Main feedwater isolation valves	a. Exercise through approximately 5% travel	a. Quarterly
	b. Cycle	b. Every 18 months
15. Reactor internals vent valves	Demonstrate operability by:	Each refueling shutdown.
	a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities.	
	b. Verifying that the valve is not stuck in an open position, and	
	c. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 400 lbs (applied vertically upward).	

4.8 EMERGENCY FEEDWATER PUMP

Applicability

Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.

Objective

To verify that the emergency feedwater pump and associated valves are operable.

Specification

4.8.1 Each EFW train shall be demonstrated operable:

a. By verifying on a STAGGERED TEST BASIS:

1. at least once per 31 days or upon achieving hot shutdown following a plant heatup and prior to criticality, that the turbine-driven pump starts, operates for a minimum of 5 minutes, and develops a discharge pressure of $\geq$ 1200 psig at a flow of $\geq$ 500 gpm through the test loop flow path.

2. at least once per 31 days by verifying that the motor driven EFW pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of $\geq$ 1200 psig at a flow of $\geq$ 500 gpm through the test loop flow path.

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

c. Prior to exceeding 280°F reactor coolant temperature and after any EFW flowpath manual valve alterations by verifying that each manual valve in each EFW flowpath which, if mis-positioned may degrade EFW operation, is locked in its correct position.

d. At least once per 92 days by cycling each motor-operated valve in each flowpath through at least one complete cycle.

e. At least once per 18 months by functionally testing each EFW train and:

1. Verifying that each automatic valve in each flowpath actuates automatically to its correct position on receipt of an actuation signal.

2. Verifying that the automatic steam supply valves associated with the steam turbine driven EFW pump actuate to their correct positions upon receipt of an actuation signal.

3. Verifying that the motor-driven EFW pump starts automatically upon receipt of an actuation signal.

4. Verifying that feedwater is delivered to each steam generator using the electric motor-driven EFW pump.

5. Verifying that the EFW system can be operated manually by over-riding automatic actuation signals to the EFW valves.

Bases

The monthly testing frequency will be sufficient to verify that both emergency feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps. The cycling of the emergency valves assures valve operability when called upon to function.

The functional test, performed once every 18 months, will verify that the flow path to the steam generators is open and that water reaches the steam generators from the emergency feedwater system. The test is done during shutdown to avoid thermal cycle to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater.

The automatic actuation circuitry testing and calibration will be performed per Surveillance Specification 4.1, and will be sufficient to assure that this circuitry will perform its intended function when called upon.