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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 78
License No. DPR-68

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

   A. The applications for amendment by Tennessee Valley Authority (the licensee) dated January 23 and June 6, 1984, comply 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-68 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78, 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 the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 27, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 78

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

   iii, iv, v, vii, 3, 4, 21, 32, 33, 34, 36, 38, 39, 40, 43, 69, 76, 81, 82, 83, 94, 99, 102, 102a, 129, 149, 242, 264, 264a, 268, 279, 285, 289, 290, 353, 354, 355

2. The marginal lines on these pages denote the area being changed.

Amendment No. ~~51~~, 78

Amendment No. 78

Amendment No. ~~51~~, ~~55~~, ~~70~~, 78

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valve closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature equal to or less than 212°F.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15 percent high flux trip and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the Startup/Hot Standby position of the mode switch.

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15 percent high flux trip) and RBM interlocks n service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 seconds), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system.

4. Refuel Mode - With the mode switch in the refuel position, interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicates at least 3 cps and the refueling crane is not over the reactor except as specified by TS 3.10.B.1.b.2. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

N. Rated Power - Rated Power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, corresponds to 3,440 MWt.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisifed:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch, and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument was on range 5, the scram setting would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. The APRM 15-percent scram will prevent higher power operation without being in the run mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough, due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5.4 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120% of rated power, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum Number of Operable Instrument Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Shutdown	Refuel (7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤ 120/125 indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)(24)(25)						
2	High Flux (Fixed Trip)	≤ 120 percent				X	1.A or 1.B
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15 percent rated power		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative	(13)		X(21)	X(17)	X	1.A or 1.B
2	Downscale	≥ 3 indicated on scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538 inch above vessel zero		X	X	X	1.A
2	High Water Level in West Scram Discharge Tank (LS-85-45A-D)	≤ 50 gallons	X	X(2)	X	X	1.A

TABLE 3.1.A (cont'd)
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum Number of Operable Instrument Channels Per Trip System (1)	(23) Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Shutdown	Refuel (7)	Startup/Hot Standby	Run	Action(1)
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤ 50 gallons	X	X(2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤ 10 percent valve closure				X(6)	1.A or 1.C
2	Turbine Control Valve Fast Closure or Turbine Trip	≥ 550 psig				X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure				X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive	not ≥ 154 psig		X(18)	X(18)	X(18)	(19)
2	Turbine Condenser Low Vacuum	≥ 23 In. Hg. Vacuum				X	1.A or 1.C
2	Main Steam Line High Radiation (14)	3X Normal Full Power Background (20)		X(9)	X(9)	X(9)	1.A or 1.C

NOTES FOR TABLE 3.1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for both trip systems, the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all operable control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% of rated.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. Deleted.

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. APRM 15% scram

8. Not required to be operable when primary containment integrity is not required.

9. Not required if all main steamlines are isolated.

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/Week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level (5)	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank			
Float Switches (LS-85-45C-F)	A	Trip Channel and Alarm	Once/Month
Electronic Level Switches (LS-85-45A, B, G, H)	B	Trip Channel and Alarm (7)	Once/Month
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be operable or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the monthly functional test program.

6. The functional test of the flow bias network is performed in accordance with Table 4.2.C.

7. Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify operability of the trip and alarm functions.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once every 7 days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/operating cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Electronic Level Switches (LS-85-45A, B, G, H)	B	Calibrated Water Column	Once/Operating Cycle (9)
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 Months
Turbine Cont. Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/operating cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be operable or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an operable status.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups , overlap between the IRM's and APRM's will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operating during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete tip system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100% power.

9 Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Ref. Section 7.5.4 FSAR. Thus, the IRM is required in the Refuel and Startup modes. In the power range the APRM system provides required protection. Ref. Section 7.5.7 FSAR. Thus, the IRM System is not required in the Run mode. The APRM's and the IRM's provide adequate coverage in the startup and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for Startup and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.1 operable in the Refuel mode is to assure that shifting to the Refuel mode during reactor power operation does not diminish the need for the reactor protection system.

The turbine condenser low vacuum scram is only required in the run mode. Below 154 psig turbine first stage pressure (30% of rated), the scram signal due to turbine stop valve closure, and turbine control valve fast closure, or turbine trip is bypassed because flux and pressure scram are adequate to protect the reactor.

Because of the APRM downscale limit of ≥ 3% when in the Run mode and high level limit of ≤ 15% when in the Startup Mode, the transition between the Startup and Run Modes must be made with the APRM instrumentation indicating between 3% and 15% of rated power or a control rod scram will occur. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the Startup Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to shutdown). When power is being reduced, if a transfer to the Startup mode is made and the IRM's have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

Table 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.
1	HPCI Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1 (2)	Instrument Channel - Condensate Header Level (LS-73-56A & B)	≥ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
2 (2)	Instrument Channel - Suppression Chamber High Level	≤7" above normal water level	A	1. Above trip setting will open HPCI suction valves to the suppression chamber.
2 (2)	Instrument Channel - Reactor High Water Level	≤ 583" above vessel zero	A	1. Above trip setting trips RCIC turbine.
1	Instrument Channel - RCIC Turbine Steam Line High Flow	≤ 450" H_2O (7)	A	1. Above trip setting isolates RCIC system and trips RCIC turbine.
4 (4)	Instrument Channel - RCIC Steam Line Space High Temperature	≤200°F.	A	1. Above trip setting isolates RCIC system and trips RCIC turbine.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	≤0.66W + 42% (2)
4(1)	APRM Upscale (Startup Mode) (8)	≤12%
4(1)	APRM Downscale (9)	≥3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	≤0.66W + 40% (2)(13)
2(7)	RBM Downscale (9)	≥3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	≤108/125 of full scale
6(1)	IRM Downscale (3) (8)	≥5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	≤ $1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4) (8)	≥3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	≤10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤115% recirculation flow
1	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-85-61A and PS-85-61B)	147 psig turbine first-stage pressure
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	≤ 25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	≤25 gal.

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-46 A LI-3-46 B	Reactor Water Level	Indicator - 155" to + 60"	(1) (2) (3)
2	PI-3-54 PI-3-61	Reactor Pressure	Indicator 0-1500 psig	(1) (2) (3)
2	PR-64-50 PI-64-67	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	TI-64-52 TR-64-52	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	TR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights	
1	N/A	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power	(1) (2) (3) (4)
1	PS-64-67	Drywell Pressure	Alarm at 35 psig	
1	TR-64-52 and PS-64-58 B and IS-64-67	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure > 2.5 psig after 30 minute delay	(1) (2) (3) (4)
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential Pressure	Indicator 0 to 2 psid	(1) (2) (3)
1/Valve		Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe		(5)
2	LI-64-159A XR-64-159	Suppression Chamber Water Level-Wide Range	Indicator, Recorder 0-240"	(1) (2) (3)
2	PI-64-160A XR-64-159	Drywell Pressure Wide Range	Indicator, Recorder 0-300 psig	(1) (2) (3)
2	TI-64-161 TR-64-161 TI-64-162 TR-64-162	Suppression Pool Bulk Temperature	Indicator, Recorder 30° - 230° F	(1) (2) (3) (4) (6)

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) From and after the date that both the acoustic monitor and the temperature indication on any one valve fails to indicate in the control room, continued operation is permissible during the succeeding thirty days, unless one of the two monitoring channels is sooner made available. If both the primary and secondary indication on any SRV tailpipe is inoperable, the torus temperature will be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV.

(6) A channel consists of 8 sensors, one from each alternating torus bay. Seven sensors must be operable for the channel to be operable.

TABLE 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
ADS Timer	(4)	once/operating cycle	none
Instrument Channel RHR Pump Discharge Pressure	(1)	once/3 months	none
Instrument Channel Core Spray Pump Discharge Pressure	(1)	once/3 months	none
Core Spray Sparger to RPV d/p	(1)	once/3 months	once/day
Trip System Bus Power Monitor	once/operating cycle	N/A	none
Instrument Channel Condensate Header Level (LS-73-56A,B)	(1)	once/3 months	none

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/3 months	once/day (8)
APRM Downscale	(1) (13)	once/3 months	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (Flow Bias)	(1) (13)	once/6 months	once/day (8)
RBM Downscale	(1) (13)	once/6 months	once/day (8)
RBM Inoperative	(1) (13)	N/A	once/day (8)
IRM Upscale	(1) (2) (13)	once/3 months	once/day (8)
IRM Downscale	(1) (2) (13)	once/3 months	once/day (8)
IRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1) (2) (13)	N/A	N/A
SRM Upscale	(1) (2) (13)	once/3 months	once/day (8)
SRM Downscale	(1) (2) (13)	once/3 months	once/day (8)
SRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1) (2) (13)	N/A	N/A
Flow Bias Comparator	(1) (15)	once/operating cycle (20)	N/A
Flow Bias Upscale	(1) (15)	once/3 months	N/A
Rod Block Logic	(16)	N/A	N/A
RSCS Restraint	(1)	once/3 months	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter	once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter	once/operating cycle	N/A

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level	Once/6 months	Each Shift
2) Reactor Pressure	Once/6 months	Each Shift
3) Drywell Pressure	Once/6 months	Each Shift
4) Drywell Temperature	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature	Once/6 months	Each Shift
8) Control Rod Position	NA	Each Shift
9) Neutron Monitoring	(2)	Each Shift
10) Drywell Pressure (PS-64-67)	Once/6 months	NA
11) Drywell Pressure (PS-64-58B)	Once/6 months	NA
12) Drywell Temperature (TR-64-52)	Once/6 months	NA
13) Timer (IS-64-67)	Once/6 months	NA
14) CAD Tank Level	Once/6 months	Once/day
15) Containment Atmosphere Monitors	Once/6 months	Once/day
16) Drywell to Suppression Chamber Differential Pressure	Once /6 months	Each Shift

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
17 Relief valve Tailpipe Thermocouple Temperature	NA	Once/month (24)
18 Acoustic Monitor on Relief Valve Tailpipe	Once/cycle (25)	Once/month (26)
19. Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/cycle	Once/month
20. Drywell Pressure - Wide Range (PI-64-160A)(XR-64-159)	Once/cycle	Once/shift
21. Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162)	Once/cycle	Once/shift

3.3 REACTIVITY CONTROL

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δ k. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. If specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the shutdown condition within 24 hours.

F. Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be operable any time that the reactor protection system is required to be operable except as specified in 3.3.F.2.

2. In the event any SDV drain or vent valve becomes inoperable, reactor operation may continue provided the redundant drain or vent valve is operable.

3. If redundant drain or vent valves become inoperable, the reactor shall be in hot stand-by within 24 hours.

4.3 REACTIVITY CONTROL

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

E. Surveillance requirements are as specified in 4.3.C and .D above.

F. Scram Discharge Volume (SDV)

1.a. The scram discharge volume drain and vent valves shall be verified open prior to each starup and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

1.b. The scram discharge volume drain and vent valves shall be demonstrated operable monthly.

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated operable immediately and weekly thereafter.

3. No additional surveillance required.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

B. Residual Heal Removal System (RHRS) (LPCI and Containment Cooling)

1. The RHRS shall be operable:

(1) prior to a reactor startup from a Cold Condition; or

(2) when there is irradiated fuel in the reactor vessel and when the reactor vessel pressure is greater than atmospheric, except as specified in specifications 3.5.B.2, through 3.5.B.7

2. With the reactor vessel pressure less than 105 psig, the RHR may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)

1. a. Simulated Automatic Actuation Test — Once/ Operating Cycle

b. Pump Operability — Once/ month

c. Motor Operated valve operability — Once/ month

d. Pump Flow Rate — Once/3 Months

e. Testable check valve — Once/ operating cycle

Each LPCI pump shall deliver 9,000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12,000 gpm against an indicated system pressure of 250 psig.

2. An air test on the drywell and torus headers and nozzles shall be conducted once/5 years. A water test may be performed on the torus header in lieu of the air test.

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (Sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	2.5	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	5	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	5	0	GC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	5	0	GC
6	System Suction Isolation Valves to Air Compressors "A" and "B" (FCV-32-62, 63)		2	15	0	GC
6	Drywell/Suppression Chamber Nitrogen Purge Inlet (FCV-76-24)		1	5	C	SC
6	Torus Hydrogen Sample Line Valves Analyzer A (FSV-76-55, 56)		2	NA	Note 1	SC
6	Torus Oxygen Sample Line Valves Analyzer A (FSV-76-53, 54)		2	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves Analyzer A (FSV-76-49, 50)		2	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves Analyzer A (FSV-76-51, 52)		2	NA	Note 1	SC
6	Sample Return Valves - Analyzer A (FSV-76-57, 58)		2	NA	0	GC
6	Torus Hydrogen Sample Line Valves Analyzer B (FSV-76-65, 66)		2	NA	Note 1	SC

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (Sec.)	Normal Position	Action on Initiating Signal
6	Torus Oxygen Sample Line Valves-Analyzer B (FSV-76-63, 64)		2	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves-Analyzer B (FSV-76-59, 60)		2	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves-Analyzer B (FSV-76-61, 62)		2	NA	Note 1	SC
6	Sample Return Valves-Analyzer B (FSV-76-67, 68)		2	NA	0	GC
7	RCIC Steamline Drain (FSV-71-6A, 6B)		2	5	C	SC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	C	SC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-73-6A, 6B)		2	5	0	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

NOTE: 1: Analyzers are such that one is sampling drywell hydrogen and oxygen (valves from drywell open - valves from torus closed) while the other is sampling torus hydrogen and oxygen (valves from torus open - valves from drywell closed)

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TABLE 3.7.B

TESTABLE PENETRATIONS WITH DOUBLE O-RING SEALS

Penetration No.	Identification
X-1A	Equipment Hatch
X-1B	Equipment Hatch
X-4	Head Access, Drywell
X-6	CRD Removal Hatch
X-25	Flange on 64-18
X-25	Flange on 64-19
X-25	Flange on 84-8A
X-25	Flange on 84-8D
X-26	Flange on 64-31
X-26	Flange on 64-34
X-35a	TIP Drive
X-35B	TIP Drive
X-35c	TIP Drive
X-35d	TIP Drive
X-35e	TIP Drive
X-35f	TIP Indexer Purge
X-35g	Spare
X-47	Power Operation Test
X-200A	Suppression Chamber Access Hatch
X-200B	Suppression Chamber Access Hatch
-	Drywell Head
-	Shear Lug No. 1
-	Shear Lug No. 2
-	Shear Lug No. 3
-	Shear Lug No. 4
-	Shear Lug No. 5
-	Shear Lug No. 6
-	Shear Lug No. 7
-	Shear Lug No. 8
X-205	Flange on 64-20
X-205	Flange on 64-21
X-205	Flange on 84-8B
X-205	Flange on 84-8C
X-205	Flange on 76-19
X-205	Flange on 76-18
X-219A	Spare (Unit 3 Only)
X-223	Suppression Chamber Access Hatch
X-231	Flange on 64-29
X-231	Flange on 64-32

TABLE 3.7.E

PRIMARY CONTAINMENT ISOLATION VALVES WHICH TERMINATE
BELOW THE SUPPRESSION POOL WATER LEVEL

Valve	Valve Identification
12-733	Auxiliary Boiler to RCIC
12-741	Auxiliary Boiler to RCIC
43-28A	RHR Suppression Chamber Sample Lines
43-28B	RHR Suppression Chamber Sample Lines
43-29A	RHR Suppression Chamber Sample Lines
43-29B	RHR Suppression Chamber Sample Lines
71-14	RC Turbine Exhaust
71-32	RC Vacuum Pump Discharge
71-580	RCIC Turbine Exhaust
71-592	RCIC Vacuum Pump Discharge
73-23	HPCI Turbine Exhaust
73-24	HPCI Turbine Exhaust Drain
73-603	HPCI Turbine Exhaust
73-609	HPCI Exhaust Drain
74-722	RHR
75-57	Core Spray to Auxiliary Boiler
75-58	Core Spray to Auxiliary Boiler
	Core Spray to Auxiliary Boiler

BASES

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the changes of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occuring.

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 49.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting form isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately 3 feet and water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. The maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much at 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

BASES

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and LOCA cases. The actions required by specification 3.7.c-f assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for controlled blowdown anytime during RCIC operation and assures margin for complete condensation of steam from the design basis loss-of-coolant accident.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed ("Torus Support System and Attached Piping Analysis for the Browns Ferry Nuclear Plant Units 1, 2, and 3," dated September 9, 1976 and supplemented October 12, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.1 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.06 feet to 3.58 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit along on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure and held constant. The 2 psig set point will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

With a differential pressure of greater than 1 psig, the rate of change of the suppression chamber pressure must not exceed .25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.14 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5 percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635 percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of 3, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

Establishing the test limit of 2.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate (49 psig Method) or the allowable test leak rate (25 psig Method) by 0.75 thereby providing a 25%

3.11 FIRE PROTECTION SYSTEMS

4.11 FIRE PROTECTION SYSTEMS

3. The class A supervised detector alarm circuits will be tested once each two months at the local panels.

4. The circuits between the local panels in 4.11.C.3 and the main control room will be tested montnly.

5. Smoke detector sensitivity will be checked in accordance with manufacturer's instruction annually.

D. ROVING FIRE WATCH

A roving fire watch will tour each area in which automatic fire suppression systems are to be installed (as described in the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2," Section X) at intervals no greater than 2 hours. A keyclock recording type system shall be used to monitor the routes of the roving fire watch. The patrol will be discontinued as the automatic suppression systems are installed and made operable for each specified area.

D. ROVING FIRE WATCH

A monthly walk-through by the Safety Engineer will be made to visually inspect the plant fire protection system for signs of damage, deterioration, or abnormal conditions which could jeopardize proper operation of the system.

3.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspection

All fire barrier penetrations, including cable penetration barriers, fire doors and fire dampers, in fire zone boundaries protecting safety related areas shall be functional at all times. With one or more of the required fire barrier penetrations non-functional within one hour establish a continuous fire watch on at least one side of the affected penetration or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol until the work is completed and the barrier is restored to functional status.

F. Fire Protection Organization

The minimum in-plant fire protection organization and duties shall be as depicted in Figure 6.3-1.

4.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspections

Each required fire barrier penetration shall be verified to be functional at least once per 18 months by a visual inspection, and prior to restoring a fire barrier to functional status following repairs or maintenance by performance of a visual inspection of the affected fire barrier penetration.

F. Fire Protection Organization

No additional surveillance required.

3.11 FIRE PROTECTION SYSTEMS

G. Air Masks and Cylinders

A minimum of fifteen air masks and thirty 500 cubic inch air cylinders shall be available at all times except that a time period of 48 hours following emergency use is allowed to permit recharging or replacing.

H. Continuous Fire Watch

A continuous fire watch shall be stationed in the immediate vicinity where work involving open flame welding, or burning is in progress.

I. Open Flames, Welding, and Burning in the Cable Spreading Room

There shall be no use of open flame, welding, or burning in the cable spreading room unless the reactor is in the cold shutdown condition.

4.11 FIRE PROTECTION SYSTEMS

G. Air Masks and Cylinders

No additional surveillance required.

H. Continuous Fire Watch

No additional surveillance required.

I. Open Flames, Welding, and Burning in the Cable Spreading Room

No additional surveillance required.