ENCLOSURE 1
PROPOSED TECHNICAL SPECIFICATION REVISIONS
BROWNS FERRY NUCLEAR PLANT UNIT 3
(TVA BFNP TS 195)

LIST OF ILLUSTRATIONS

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Mode - With the mode switch in the refuel position, interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicates at least 3 cps and the refueling crane is not over the reactor except as specified by TS 3.10.B.1.b.2. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.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

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Thermal Power Limits

1. Reactor Pressure > 800 psia and Core Flow > 10% of Rated.

   When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) — (Flow Biased)

   a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq (0.66W + 54\%)$$

where:

S = Setting in percent of rated thermal power (3293 MWt)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1 FUEL CLADDING INTEGRITY e. Fixed High Neutron Flux Scram Trip Setting - When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be: S $\leq$ 120% power
2. Reactor Pressure $\leq$ 800 PSIA or Core Flow $\leq$ 10% of Rated When the reactor pressure is $\leq$ 800 PSIA or core flow is $\leq$ 10% of rated, the core thermal power shall not exceed 823 MWt (~25% of rated thermal power).	2. APRM and IRM Trip Settings (Startup and Hot Standby Modes). a. APRM - When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power. b. IRM - The IRM scram shall be set at less than or equal to 120/125 of full scale.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed throughout the spectrum of planned operating conditions up to the design thermal power condition of 3440 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 3293 MWt is the licensed maximum power level of Browns Ferry Nuclear Plant, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicalbe transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisions and results are summarized in Reference 1.

The void reactivity coefficient and the scram worth are described in detail in reference 1.

The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications as further described in Reference 1. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity has been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR of *** is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

*** See Section 3.5.K.

2.1 BASES

In summary

1. The licensed maximum power level is 3,293 MWt.

2. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

3. The abnormal operational transients were analyzed to a power level of 3440 MWt.

4. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual set points are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased High Flux Scram Trip Setting (Run Mode)

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). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

During transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120% of rated power based on recirculation drive flow according to the equations given in section 2.1.A.1 and the graph in figure 2.1.2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120% of rated power. Therefore, the flow biased provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of CMFLPD and FRP. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the CMFLPD exceeds FRP.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR > ***.

2. <u>APRM Flux Scram Trip Setting (Refuel or Start & Hot Standy Mode)</u>

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, all of possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. <u>IRM-Flux Scram Trip Setting</u>

The IRM System consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a

*** See Section 3.5.K.

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

a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during the steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the CMFLPD exceeds FRP thus preserving the APRM rod block safety margin.

C. Reactor Water Low Level Scram and Isolation (Except Main Steamlines)

The set point for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 31 inches below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure- neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2).

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control

oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. Relevant transient analyses are discussed in References 1 and 2. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by the turbine first stage pressure.

F. Main Condenser Low Vacuum Scram

To protect the main concenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vaccum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 850 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP

position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I. J. & K. <u>Reactor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps</u>

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram set point and initiation set points. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. <u>References</u>

1. "BWR Transient Anlaysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

2. Generic Reload Fuel Application, Licensing Topical Report NEDE 24011-P-A and Addenda.

1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressure (1,148 psig for suction and 1,326 psig for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients are added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the reload licensing submittal for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.2 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system.

Specification

A. When there is fuel in the vessel, The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.A.

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.A and 4.1.B respectively.

1. Daily during reactor power operation at greater than or equal to 25% thermal power, the ratio of Fraction of Rated Power (FRP) to Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked and the scram and APRM Rod Block settings given by equations in specifications 2.1.A.1 and 2.1.B shall be calculated.

2. When it is determined that a channel is failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be untripped for short periods of time to allow functional testing of the other trip system. The trip system may be in the untripped position for no more than eight hours per functional test period for this testing.

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

B. Two RPS power monitoring channels for each inservice RPS MG sets or alternate source shall be operable.

1. With one RPS electric power monitoring channel for inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

2 With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENT

4.1 REACTOR PROTECTION SYSTEM

B. The RPS power monitoring system instrumentation shall be determined operable:

At least once per 6 months by performance of channel functional tests.

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 (PIS-3-22AA, BB, C, D)	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14) (PIS-64-56A-D)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14) (LIS-3-203A-D)	≥ 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				Action(1)
			Shutdown	Refuel (7)	Startup/Hot Standby	Run	
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 (PIS-1-81A&B, PIS-1-91A&B)	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.

24. The Average Power Range Monitor scram function is varied (ref. Figure 2.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with 2.1.A.

25. The APRM flow biased neutron flux signal is fed through a time constant circuit of approximately 6 seconds. This time constant may be lowered or equivalently removed (no time delay) without affecting the operability of the flow biased neutron flux trip channels. The APRM fixed high neutron flux signal does not incorporate the time constant but responds directly to instantaneous neutron flux.

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 (PIS-3-22AA, BB, C, D)	B	Trip Channel and Alarm (7)	Once/Month (1)
High Drywell Pressure (PIS-64-56A-D)	B	Trip Channel and Alarm (7)	Once/Month (1)
Reactor Low Water Level (LIS-3-203A-D)	B	Trip Channel and Alarm (7)	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)	A	Trip Channel and Alarm	Once/Month

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)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/Week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/Month (1)
Turbine Control Valve Fast Closure or Turbine Trip	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive (PIS-1-81A&B, PIS-1-91A&B)	B	Trip Channel and Alarm	Every 3 Months
Turbine Stop Valve Closure	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. Deleted.

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 end 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 seven days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/operating cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1,000 effective full power hours
High Reactor Pressure (PIS-3-22AA, BB, C, D)	B	Standard Pressure Source	Once/operating cycle (9)
High Drywell Pressure (PIS-64-56A-D)	B	Standard Pressure Source	Once/operating cycle (9)
Reactor Low Water Level (LIS-3-203A-B)	A	Pressure Standard	Every three 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 (5)	Note (5)
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every three months
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every three months
Turbine First Stage Pressure Permissive (PIS-1-81A & B, PIS-1-91A & B)	B	Standard Pressure Source	Once/operating cycle (9)
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.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between non-class 1E power supply and the class 1E RPS bus. This will ensure that failure of a non-class 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast 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, 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.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6) (LIS-3-203A-D)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	≥ 470" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PIS-64-56A-D)	≤ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2 (3)	Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2 (12)	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.

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
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq$ 470" above vessel zero	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq$ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq$ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Mulitplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2 (16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq$ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1 (16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	$\geq$ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52, 62)	$\geq$ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum Number Operable Per Trip System (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PIS-64-58E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	$\leq$ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multipler relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	$\geq$ 470 inch above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel - Reactor High Pressure (PIS-3-204A-D)	$\leq$ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	$\leq$ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	$\leq$ 2.5 psig	A	1. Above trip setting in conjunction with low reactor water level, drywell high pressure, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.

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
2	Instrument Channel - Reactor Low Pressure (PIS-3-74A & B) (PIS-68-95) (PIS-68-96)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A&B) (PS-68-95) (PS-68-96)	230 psig ± 15	A	1. Recirculation discharge valve actuation.
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig ± 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	6≤t≤8 secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	0≤t≤1 sec.	B	1. With diesel power 2. One per motor
1	RHRSW A1, B3, C1, and D3 Timers	13≤t≤15 sec.	A	1. With diesel power 2. One per pump

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 No. Operable Per Trip Sys (5)	Function	Trip Level Setting
2(1)	APRM Upscale (Flow Bias)	≤0.66W+42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	≤12%
2(1)	APRM Downscale (9)	≥3%
2(1)	APRM Inoperative	(10b)
1(7)	RBM Upscale (Flow Bias)	≤ 0.66W+40% (2) (13)
1(7)	RBM Downscale (9)	≥ 3%
1(7)	RBM Inoperative	(10c)
3(1)	IRM Upscale (8)	≤ 108/125 of full scale
3(1)	IRM Downscale (3) (8)	≥ 5/125 of full scale
3(1)	IRM Detector not in Startup Position (8)	(11)
3(1)	IRM Inoperative (8)	(10a)
2(1) (6)	SRM Upscale (8)	≤ 1 x 10^5 counts/sec.
2(1) (6)	SRM Downscale (4) (8)	≥ 3 counts/sec.
2(1) (6)	SRM Detector not in Startup Position (4) (8)	(11)
2(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(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.

8. This function is bypassed when the mode switch is placed in Run.

9. This function is only active when the mode switch is in Run. This function is automatically bypassed when the IRM instrumentation is operable and not high.

10. The inoperative trips are produced by the following functions:

    a. SRM and IRM

        (1) Local "operate-calibrate" switch not in operate.

        (2) Power supply voltage low.

        (3) Circuit boards not in circuit.

    b. APRM

        (1) Local "operate-calibrate" switch not in operate.

        (2) Less than 14 LPRM inputs.

        (3) Circuit boards not in circuit.

    c. RBM

        (1) Local "operate-calibrate" switch not in operate.

        (2) Circuit boards not in circuit.

        (3) RBM fails to null.

        (4) Less than required number of LPRM inputs for rod selected.

11. Detector traverse is adjusted to 114 ± 2 inches, placing the detector lower position 24 inches below the lower core plate.

12. This function may be bypassed in the shutdown or refuel mode. If this function is inoperable at a time when operability is required the channel shall be tripped or administrative controls shall be immediately imposed to prevent control rod withdrawal.

13. RBM upscale flow biased setpoint clipped at 106% rated reactor power.

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-58 A LI-3-58 B	Reactor Water Level	Indicator -155" to +60"	(1) (2) (3)
2	PI-3-74A PI-3-74B	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	PR-64-50 PI-64-67B	Drywell Pressure	Recorder 0-80 psia Indicator 0-80 psia	(1) (2) (3)
2	TI-64-52AB 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 PR-64-50 and IS-64-67	Drywell temperature, pressure, and timer	Alarm if temperature > 281°F 30 minutes after drywell pressure exceeds 2.5 psig)	(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	RR-90-272CD RR-90-273CD	High Range Primary Containment Radiation Recorders	Recorder, 1 - 10^7 R/Hr	(7)
2	LI-64-159A XR-64-159	Suppression Chamber Water Level-Wide Range	Indicator, Recorder 0-240"	(1) (2) (3)
2	PI-64-39A XR-64-159 PI-64- 160A XR-64-159	Drywell Pressure Low Range Drywell Pressure Wide Range	Indicator, Recorder -5 to +5 psig 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.

(7) When one of these instruments is inoperable for more than 7 days, in lieu of any other report required by specification 6.7.2, prepare and submit a Special Report to the Commission pursuant to specification 6.7.3 within the next 7 days outlining the action taken, the cause of inoperability, and the plans and schedule for restoring the system to operable status.

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D)	(1) (28)	Once/operating cycle (29)	once/day
Instrument Channel - Reactor High Pressure	(1)	once/3 months	none
Instrument Channel - Reactor Low Water-Level (LIS-3-56A-D)	(1) (28)	Once/Operating Cycle (29)	once/day
Instrument Channel - High Drywell Pressure (PIS-64-56A-D)	(1) (28)	Once/Operating Cycle (29)	N/A
Instrument Channel - High Radiation Main Steam Line Tunnel	(1)	(5)	once/day
Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	(1) (28)	Once/Operating Cycle (29)	none
Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50 A-D)	(1) (28)	Once/Operating Cycle (29)	once/day
Instrument Channel - Main Steam Line Tunnel High Temperature	(1)	once/operating cycle	none
Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	(1) (14) (22)	once/3 months	once/day (8)

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

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Water Level (LIS-3-58A-D)	(1) (28)	Once/operating cycle (29)	once/day
Instrument Channel Reactor Low Water Level (LIS-3-184 & 185)	(1) (28)	Once/Operating Cycle (29)	once/day
Instrument Channel Reactor Low Water Level (LIS-3-52 & 62)	(1) (28)	Once/Operating Cycle (29)	once/day
Instrument Channel Reactor Low Water Level (LIS-3-56A-D)	(1) (28)	once/operating cycle (29)	none
Instrument Channel Reactor High Pressure (PIS-3-204A-D)	(1) (28)	once/operating cycle (29)	none
Instrument Channel Drywell High Pressure (PIS-64-58 E-H)	(1) (28)	Once/Operating Cycle (29)	none
Instrument Channel Drywell High Pressure (PIS-64-58A-D)	(1) (28)	Once/Operating Cycle (29)	none
Instrument Channel Drywell High Pressure (PIS-64-57A-D)	(1) (28)	Once/Operating Cycle (29)	none

TABLE 4.2.B

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Pressure (PIS-3-74A & B, PS-3-74A & B) (PIS-68-95, PS-68-95) (PIS-68-96, PS-68-96)	(1) (28)	Once/Operating Cycle (29)	none
Instrument Channel Reactor Low Pressure (PS-68-93 & 94)	(1)	once/3 months	none
Core Spray Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Normal Power)	(4)	once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Diesel Power)	(4)	once/operating cycle	none

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 (LI-3-58A & B)	Once/6 months	Each Shift
2) Reactor Pressure (PI-3-74A & B)	Once/6 months	Each Shift
3) Drywell Pressure (PI-64-67B)	Once/6 months	Each Shift
4) Drywell Temperature (TI-64-52AB)	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature (TR-64-52)	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 (PR-64-50)	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. High Range Primary Containment Radiation Monitors (RR-90-272CD, RR-90-273CD)	Once/cycle(27)	Once/month
20. Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/cycle	Once/month
21. Drywell Pressure - Low Range (PI-64-39A) (XR-64-159)	Once/cycle	Once/shift
22. Drywell Pressure - Wide Range (PI-64-160A)(XR-64-159)	Once/cycle	Once/shift
23. Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162((TR-64-162)	Once/cycle	Once/shift

NOTES FOR TABLES 4.2.A THROUGH 4.2.H (Continued)

14. Upscale trip is functionally tested during functional test time as required by section 4.7.B.1.a and 4.7.C.1.a.

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of section 4.7.C.1.c.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. The Reactor Cleanup System Space Temperature monitors are RTD's that feed a temperature switch in the control room. The temperature switch may be tested monthly by using a simulated signal. The RTD itself is a highly reliable instrument and less frequent testing is necessary.

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

26. This instrument check consists of comparing the background signal levels for all valves for consistency and for nominal expected values (not required during refueling outages).

27. Calibration shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one-point source check of the detector below 10 R/hr with an installed or portable gamma source.

NOTES FOR TABLES 4.2.A THROUGH 4.2.H (Continued)

28. 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.

29. 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.

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.

3.5 BASES

testing to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a pressure suppression chamber head tank is located approximately 20 feet above the discharge line highpoint to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point serves as a backup charging system when the pressure suppression chamber head tank is not in service. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the HPCI and RCIC pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the HPCIS and RCICS piping. This assures that the HPCI and RCIC discharge piping remains filled. Further assurance is provided by observing water flow from these systems high points monthly.

I. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1 through 7. The analyses supporting these limiting values is presented in reference 1.

J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat

3.5 BASES

reported within 30 days. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumtances, this will not be the only alternative.

M. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

TABLE 3.5.1-7

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3 Fuel Type: BP8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.9
25,000	11.3
30,000	10.8
35,000	10.1
40,000	9.4
45,000	8.8

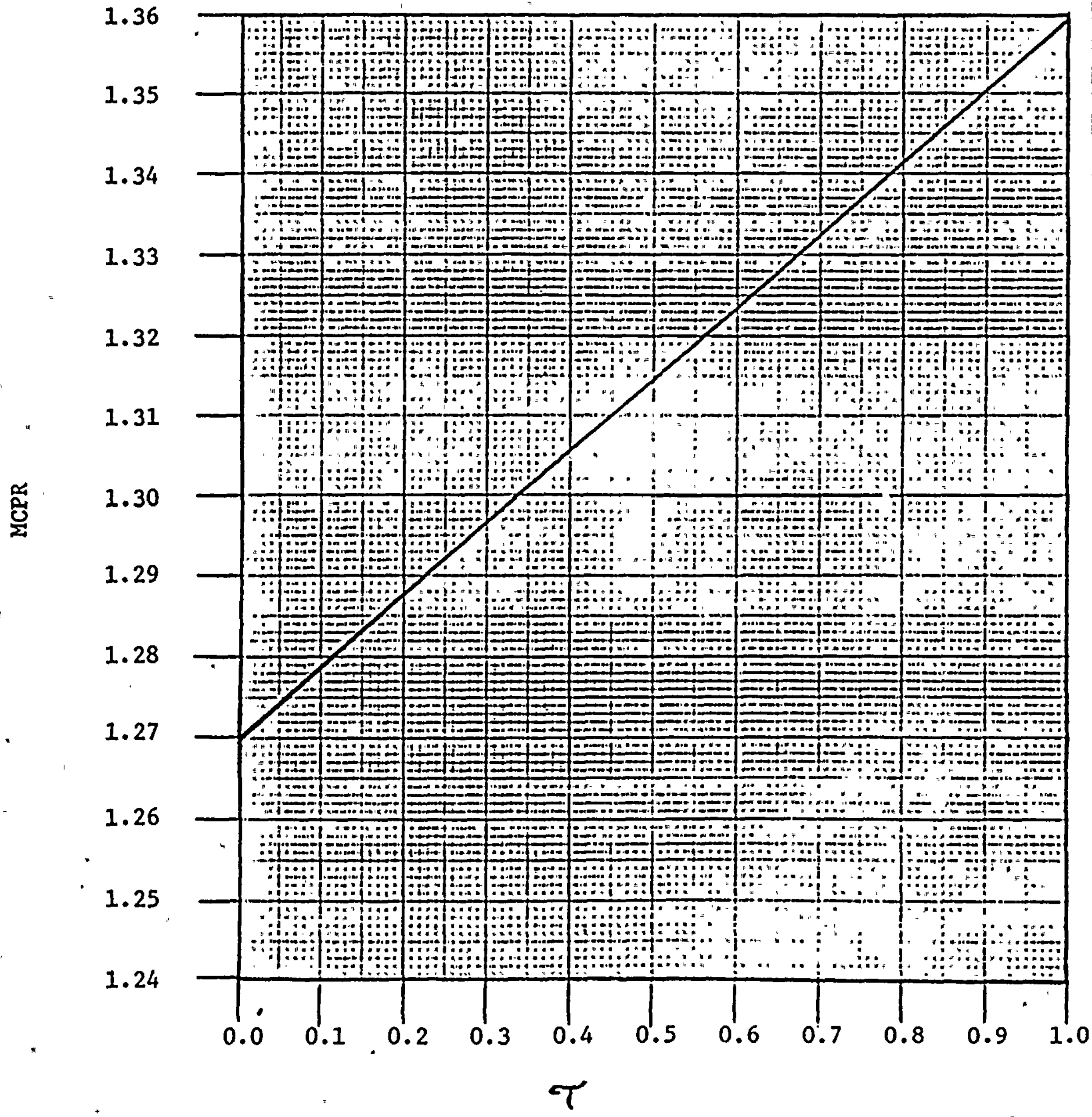


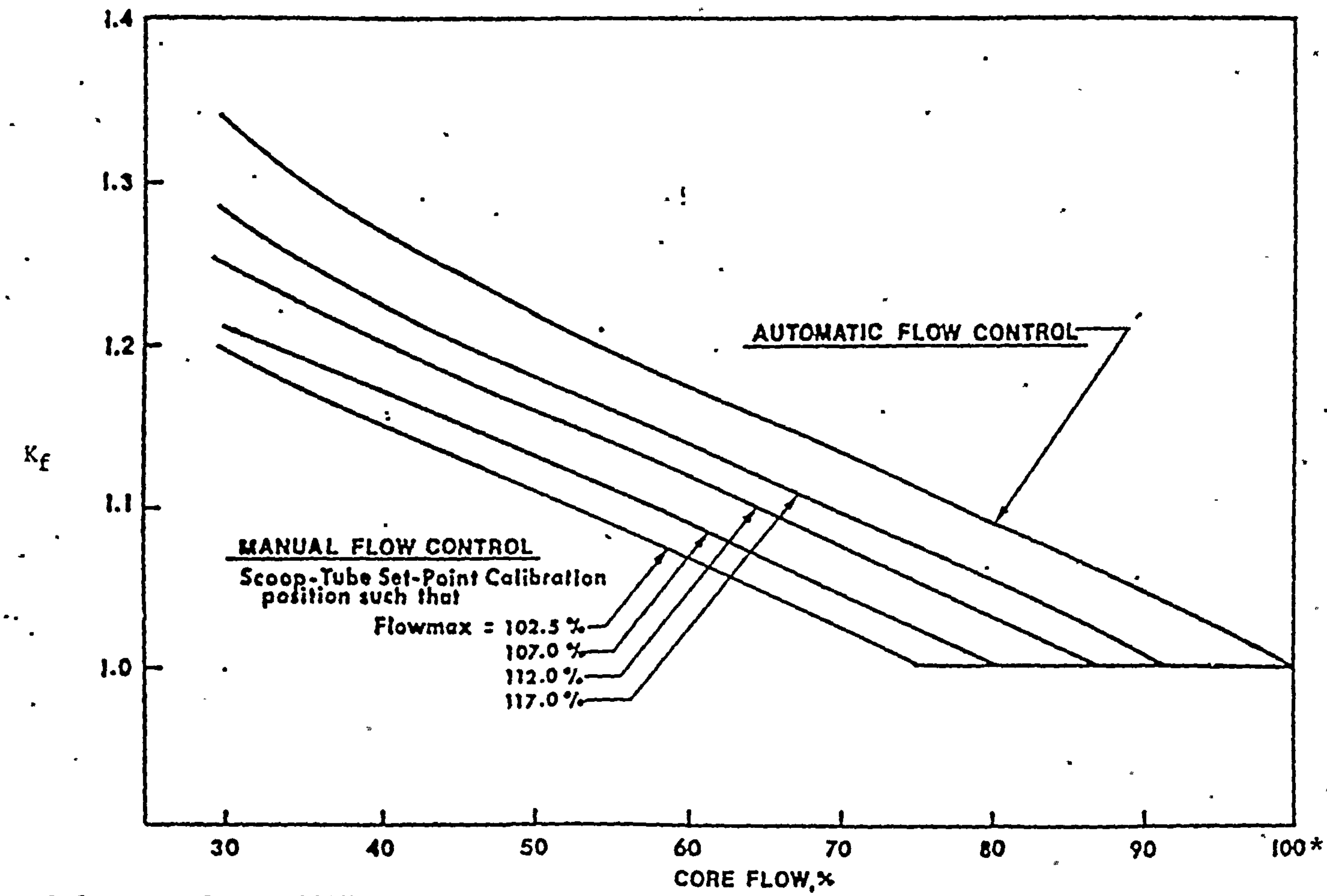
Figure 3.5.K-1

MCPR Limits for 8x8R, P8x8R and LTAs

BROWNS FERRY NUCLEAR PLANT

FIGURE 3.5.2

K_f FACTOR



*K_f = 1.0 for core flow ≥ 100%.

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	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	5	C	SC
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	O	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	O	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)

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 76-17
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.D

AIR TESTED ISOLATION VALVES

Valve	Valve Identification
1-14	Main Steam
1-15	Main Steam
1-26	Main Steam
1-27	Main Steam
1-37	Main Steam
1-38	Main Steam
1-51	Main Steam
1-52	Main Steam
1-55	Main Steam Drain
1-56	Main Steam Drain
2-1192	Service Water
2-1383	Service Water
3-554	Feedwater
3-558	Feedwater
3-568	Feedwater
3-572	Feedwater
32-62	Drywell Compressor Suction
32-63	Drywell Compressor Suction
32-336	Drywell Compressor Return
32-2163	Drywell Compressor Return
32-2516	Drywell Compressor Return
32-2521	Drywell Compressor Return
33-1070	Service Air
33-785	Service Air
43-13	Reactor Water Sample Lines
43-14	Reactor Water Sample Lines
63-525	Standby Liquid Control Discharge
63-526	Standby Liquid Control Discharge
64-17	Drywell and Suppression Chamber Air Purge Inlet
64-18	Drywell Air Purge Inlet
64-19	Suppression Chamber Air Purge Inlet
64-20	Suppression Chamber Vacuum Relief
64-c.v.	Suppression Chamber Vacuum Relief
64-21	Suppression Chamber Vacuum Relief
64-c.v.	Suppression Chamber Vacuum Relief
64-29	Drywell Main Exhaust
64-30	Drywell Main Exhaust
64-32	Suppression Chamber Main Exhaust
64-33	Suppression Chamber Main Exhaust
64-31	Drywell exhaust to Standby Gas Treatment
64-34	Suppression Chamber to Standby Gas Treatment
64-139	Drywell pressurization, Compressor Suction
64-140	Drywell pressurization, Compressor Discharge
68-508	CRD to RC Pump Seals
68-523	CRD to RC Pump Seals
68-550	CRD to RC Pump Seals
68-555	CRD to RC Pump Seals

TABLE 3.7.E

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

Valve	Valve Identification
12-738	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	RCIC Turbine Exhaust
71-32	RCIC 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

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.

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 monthly.

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

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.

6.0 ADMINISTRATIVE CONTROLS

B. Source Tests

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

C. Special Reports (in writing to the Director of Regional Office of Inspection and Enforcement).

1. Reports on the following areas shall be submitted as noted:

a.	Secondary Containment Leak Rate Testing (5)	4.7.C	Within 90 days of completion of each test.
b.	Fatigue Usage Evaluation	6.6	Annual Operating Report
c.	Relief Valve Tailpipe Instrumentation	3.2.F	Within 30 days after inoperabilit of thermocouple an acoustic monitor on one valve.
d.	Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability
e.	Meteorological Monitoring Instrumentation Inoperability	3.2.I.2	Within 10 days after 7 days of inoperability
f.	High Range Primary Containment Radiation Monitors	3.2.F	Within 7 days after 7 days of inoperability.

D. Special Report (in writing to the Director of Regional Office of Inspection and Enforcement)

Data shall be retrieved from all seismic instruments actuated during a seismic event and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be submitted within 10 days after the event describing the magnitude, frequency spectrum, and resultant effect upon plant features important to safety.

ENCLOSURE 2

Description and Justification of Changes - TVA BFNP TS 195
(TVA Browns Ferry Nuclear Plant Unit 3 Technical Specifications)

A. Core-Related Changes

1. Increased Core Flow

Page 76 - Add note 13 and increase flow bias upscale to 115-percent recirculation

Page 78 - Add note 13 which states that the RBM upscale flow-biased setpoint is clipped at 106-percent rated reactor power.

Page 183 - Add footnote to 100-percent core flow which states that K_f is equal to 1.0 for core flows above 100 percent.

The proposed revisions are needed to operate unit 3 cycle 6 with increased core flow. The reload licensing analysis was performed for both 100- and 105-percent core flow. The most conservative results were used for determining the operating limits.

In order to operate in an increased core flow condition, the following operational conditions are necessary. The rod block monitor upscale (flow biased) must be "clipped" at 106-percent power in order to ensure adequate protection in the event of a rod withdrawal error (pages 76 and 78). The rod block on recirculation flow is being increased from 100 percent to 115 percent to ensure operational flexibility (page 76).

2. New Fuel Type

Pages vii and 182b - Add a new MAPLHGR vs. AVERAGE PLANAR EXPOSURE for the new fuel type BP8DRB284L for this cycle. The table of contents was also updated to include the new table.

3. MCPR

Pages viii and 183c - Figure 3.5.K-1 for MCPR limits was updated to reflect the limits for cycle 6 operations. The table of contents was revised to reflect the new page number.

4. References in Bases

Pages 18, 22, 23, 24, 28, 176, and 178 - Revised to reflect that the reload analyses are being done by TVA instead of GE. Changes in reference numbers and minor text changes to reflect TVA's methodology are included. Page 22 has a change to reflect that the MCPR safety limit is 1.07.

The justification and safety analysis for above revisions are described in TVA-RLR-001, "Reload Licensing Report for Browns Ferry Unit 3 Cycle 6."

B. Changes Related to Torus Modifications

Numerous modifications are being implemented in the unit 3 torus during the reload 5 refueling outage as part of the Mark I containment program. These modifications are required by NRC to restore the originally intended margins of safety in the containment design, and work will be considered complete following this outage.

Pages 82, 83, and 102a of tables 3.2.F and 4.2.F - Revised to include the surveillance instrumentation associated with the suppression pool bulk temperature. This modification provides an improved torus temperature monitoring system which consists of 16 sensors. This will provide a more accurate indication of the torus water bulk temperature as required by NUREG-0661 and will replace the suppression chamber water temperature instruments on pages 81 and 102.

Page 149 - The two-pump 15,000 gpm LPCI test surveillance 4.5.B.1 was determined to induce vibrations in the RHR return line to the torus. To eliminate the vibration, an orifice is to be installed in the return line; however, installation of this orifice plate also decreases the suppression pool cooling mode of RHR operation from 15,000 gpm to approximately 12,000 gpm. A new containment cooling analysis was performed for this configuration, and it was determined that this flow rate produces a long-term suppression pool temperature well within that necessary for stable and complete steam condensation and for adequate RHR and core spray pumps net positive suction head.

Page 242 - Since the torus is being extensively upgraded to withstand dynamic loading significantly beyond that originally expected, extended operation of relief valves above a suppression pool temperature of 130°F is not expected to be a safety concern warranting placing the reactor in cold shutdown and performing a torus inspection. This requirement originated with RO bulletin 74-14. This has previously been approved for unit 2 in amendment No. 85 and for unit 1 in Amendment No. 92.

Pages 285 and 286 - This 3.7.A and 4.7.A bases for the suppression pool temperature limits were founded on the Humboldt Bay and Bodega Bay tests. Consistent with the long-term torus integrity program of NUREG-0661 and NUREG-0783, the bases require change to account for steam mass fluxes through S/RV T-quenchers. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate RHR and core spray pump net positive suction head.

Pages 289 and 290 - The specific references to drywell and suppression chamber coatings are being deleted. There is some variation between the Browns Ferry units in the type and application of the coating, particularly due to the Mark I modification program; therefore, the technical specification bases are being generalized so that technical specification changes will not be required each cycle. Control of the torus coating will be maintained by internal TVA coating programs.

C. Miscellaneous Plant Modifications

1. Thermal Power Monitor

Page 9 - Add "Flow-Biased" to title of section 2.1.A.1.
Page 12 - Add section 2.1.A.1.e.
Page 19 & 20 - Reword bases 2.1.A.1 to reflect features of the thermal power monitors.
Page 21 - Add bases 2.1.A.4 to describe the fixed high neutron flux scram trip.
Page 32 - Change table 3.1.A to reflect addition of the fixed trip function.
Page 35a - Add notes 24 and 25.
Page 36 - Change table 4.1.A to reflect addition of fixed trip.

Justification

The addition of the thermal power monitor will prevent a flow-biased neutron flux scram when a transient-induced neutron flux spike occurs that is a short-time duration and does not result in an instantaneous heat flux in excess of transient limits. Neutron flux is damped by approximately a 6-second fuel time constant. This feature will reduce the number of scrams due to small, fast flux transients such as those which result from control valve and MSIV testing and small perturbations in water level and pressure.

Safety Analysis

The APRM flow-biased scram will occur when the fuel surface heat flux resulting from a neutron flux transient reaches a point equivalent to the thermal power trip setpoint. This is done by passing the neutron flux signal through a filter network with a time constant shorter than the representative of the fuel thermal time constant. There is a separate trip unit which initiates a scram less than 120-percent instantaneous neutron flux. This scram function is the basis for transient and accident analysis, and no credit is taken for the flow-biased scram function. Any flow-biased scram function therefore provides additional margin from fuel damage beyond that of the transient analysis. This has previously been approved for unit 2 in amendment No. 85.

2. Reactor Protection System (RPS) Modification

Pages 31, 31A, and 41 - Sections 3.1.B and 4.1.B are being added to reflect the limiting conditions for operation and surveillance requirements associated with the RPS modifications. Page 42 is being modified to add a description of these sections in the bases. The RPS is being modified to provide a fully redundant class 1E protection at the interface of the non-class 1E power supplies and the RPS. This will ensure that failure of a non-class 1E reactor protection power supply will not cause adverse interaction to the class 1E reactor protection system. This is in response to a finding at Hatch 2 identified in T. A. Ippolito's (NRC) letter to N. P Hughes dated August 7, 1978.

3. Scram Discharge Instrument Volume

The scram discharge volumes (SDVs) and scram discharge instrument volumes (SDIVs) are being modified to address inadequacies identified by the partial rod insertion event on Browns Ferry unit 3 in June 1980. One of the modifications includes adding another valve in series to the existing drain and vent valve on the SDV and SDIV. Another modification includes adding electronic level switches to initiate a scram on high level in the SDIV.

As designed, the drain and vent valves serve two purposes neither of which is a direct safety function. The first purpose, when the valves are in the open position, is to provide a positive means of maintaining the SDV drained to ensure adequate volume to accept water discharged during a scram. The direct safety function of adequate volume is ensured, however, by the redundant and diverse SDIV level instruments described above.

The second purpose, when the valves are in the closed position, is to limit the amount of water discharged to the radioactive waste system following a scram. There is no direct safety function associated with this purpose, but two other means are designed to alleviate this operational inconvenience. The first is in the control rod drive (CRD) seal design which serves as the reactor pressure boundary to limit leakage. The second is the proven ability to reset the scram under most conditions in less than five minutes thereby closing the scram outlet valves and stopping flow to the SDV.

Implementation of the SDV and SDIV modifications adding the redundant vent and drain valves provides increased assurance that the second purpose will be fulfilled while decreasing the probability of fulfillment of the first purpose. Given the high level of confidence of the level switches meeting the first purpose versus the somewhat lower confidence of being able to reset the scram and fulfilling the second purpose, it is prudent to specify and maintain a certain level of operability to meet the second purpose. In the case of the drain and vent valves not serving a direct safety function, one of a redundant pair of valves is fully adequate for continued power operation with increased surveillance of the operable valve. The revisions described have been previously approved for unit 2 in amendment No. 85.

Pages 32, 76, and 99 of tables 3.1.A, 3.2.C, and 4.2.C - Revised to reflect an east and west scram discharge instrument volume.

Pages 36 and 39 of tables 4.1.A and 4.1.B - Revised to reflect the required surveillance testing on the two electronic level switches.

Page 129 - Sections 3.3.F and 4.3.F are being revised to reflect the addition of the redundant drain and vent valve.

4. <u>Analog Trip System</u>

Pages 32, 33, 36, 37, 39, 57, 64, 66, 88, 92, and 93 of tables 3.1.A, 4.1.A, 4.1.B, 3.2.A, 3.2.B, 4.2.A, and 4.2.B - Revised to add instrument numbers and references to descriptions of the functional tests and calibrations. The calibration frequencies are being extended to "once/operating cycle" due to the high reliability of the analog trip system.

Page 38 - Revised to add note 7 to describe the functional test for analog instruments. Setpoints of all instrumentation are checked with each functional test and verified to be within the range dictated by the plant setpoint methodology for functional tests. The surveillance criteria are not satisfied unless the setpoints are within that range. Note 7 of table 4.1.A is included to clarify that analog trip functional tests involve a simulated electronic signal as opposed to a simulated process variable as is used to test the mechanical trip switches. The general requirements for a functional test are defined in section 1.V.3. Note 5 is being removed since the new note 7 now applies to the reactor water level instruments.

Page 40 - Revised to add note 9 which gives a description of a calibration for analog instrumentation. The purpose of this note is to augment the definition of instrument calibration (TS 1.V.1) to clarify its applicability to analog trip instruments and associated components. Note 9 states that calibration involves adjustment of components such that the instrument reading corresponds to known values of the process variable, and the trip circuitry be adjusted such that the trip output relay changes state at the proper analog value. In accordance with note 9, the channel calibration performed at 18-month intervals encompasses all of the components including sensors, alarm interlocks, and/or trip functions out to and including the trip output relay. The remainder of the trip components are logic devices only and are tested during instrument functional tests on a more frequent interval as required by table 4.1.A.

5. Scram Permissive Pressure Switches at 1055 PSIG

Pages 3, 4, 33, 34, and 43 - Deleted the bypass function if reactor pressure is less than 1055 psig and the mode switch not in the RUN mode. This affects the main steamline isolation valve closure and the turbine condenser low vacuum scram functions. These two functions will only be operable in the RUN mode.

The bypass function only allows a scram in the refuel and startup/hot standby modes of operation by the two scram functions listed above when the reactor pressure is greater than 1055 psig. The reactor high-pressure scram is set at 1055 psig and is operable in these two modes of operation. If reactor pressure exceeds 1055 psig, the reactor scrams due to the reactor high-pressure scram function, and the main steam line isolation valve closure and the turbine condenser low vacuum functions become operable. The bypass circuit therefore serves no real purpose. When the two scram functions are available, the reactor is already scrammed. Since the reactor is protected by the reactor high-pressure scram function, the proposed change does not result in any reduction in the margin of safety. See the attached General Electric NEDO 20697.

6. Drywell Temperature and Pressure

Pages 81 and 102 of tables 3.2.F and 4.2.F - Revised to reflect new instrument numbers for the new upgraded drywell temperature and pressure instrumentation. The surveillance requirements remain the same.

7. NUREG-0737 Items

Pages 82, 83, 102a, and 107 of tables 3.2.F and 4.2.F and page 386 - Revised to include the surveillance instrumentation associated with the following accident instrumentation:

a. Containment high-range radiation monitor
b. Drywell pressure-wide range
c. Suppression chamber-wide range water level

These three items are in response to NUREG-0737, items II.F.1.3, II.F.1.4, and II.F.1.5, respectively.

Pages 81 and 102 - Delete the drywell pressure and suppression chamber water level instruments. They are being replaced by items b and c above.

8. H_2/O_2 Analyzer System

Pages 264 and 264A of table 3.7.A - Revised to reflect modification on the Hays-Republic hydrogen/oxygen monitoring systems. Inboard isolation valves were moved outside the containment.

The isolation valves for the H_2/O_2 system are class B valves, those that isolate lines that connect directly with the containment free air space. This type of lines generally has two isolation valves in series both on the outside of containment. Four valves on this system were previously installed inside containment. They are being moved outboard. The isolation logic for these valves is not being altered; therefore, there is no reduction in the margin of safety.

9. Testable Penetrations

Page 268 of table 3.7.B - Revised to include new testable penetrations as a result of modifications made to make the flange side of the following valves testable 64-18, 64-19, 64-20, 64-21, 64-29, 64-31, 64-32, 64-34, 76-17, 76-18, and 84-8A-D.

Minor corrections to this table were also made. Penetrations X-35G was listed in this table for "T.I.P Drives" and is being revised to reflect that it is a "Spare." Penetrations X-219A, X-223, and the drywell head are being added to this table. They were inadvertently not listed, but they were included in the surveillance program. Penetration X-213A was removed. It was previously removed from unit 3.

10. Redundant Air Supply to Drywell

Page 270 of table 3.7.D - Revised to include primary containment isolation valves 32-2516 and 32-2521 for the drywell compressor discharge line. The line was added to provide the capability for isolation of approximately one-half of the drywell supplied equipment in case of a drywell line leak. This air supply will be used to supply two inboard MISVs, approximately one-half of the main steam relief valves, and approximately one-half of all other air-operated equipment in the drywell. This will significantly reduce the possibility of any one control air pipe break inside containment from requiring immediate shutdown and isolation due to MSIVs, MSRVs, and drywell coolers being inoperable. This will, in turn, significantly increase the margin of safety.

11. Demineralized Water

Page 279 - Revised to delete primary containment isolation valve 2-1143 of the demineralized water system. This valve isolated the demineralized water line to the torus ring header. The line is no longer used, so the valve will be removed and the line capped. No safety-related functions will be adversely affected by disconnecting this line; therefore, the margin of safety will not be reduced.

D. Administrative Technical Specification Changes

Pages iii, iv, and v - Technical specification titles for sections 3.6/4.6.D, 3.6/4.6.H, 3.7/4.7.H, 3.11/4.11.D, 3.11/4.11.E, and 6.9 were modified to correctly reflect the respective technical specifications. Titles were added for sections 6.10 and 6.11.

Pages iii and iv - Technical specification titles for sections 3.9/4.9.B and 3.11/4.11.B were corrected to reflect the actual page number.

Pages iv, 353, 354, and 355 - Technical specifications 3.11/4.11.F through 3.11/4.11.I were given titles to be consistent with the present format. The table of contents was also corrected to reflect this change.

Page vii - Tables 3.5-1, 4.9.A.4.c, and 3.11.A were added to the "List of Tables." These tables were inadvertently omitted from this list. In addition, table 6.3.A was removed from the technical specifications by amendment No. 48. The titles for tables 3.7.D, 3.7.E, 3.7.F, and 3.7.G were corrected to reflect the actual title.

Page 21 - The bases for the IRMs were revised to read like units 1 and 2. Also, the reference to the FSAR was corrected.

Pages 69 and 94 - An editorial change was made to more accurately indicate that HPCI suction switchover is made on condensate header level rather than condensate tank level.