ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION REVISIONS
BROWNS FERRY NUCLEAR PLANT UNIT 2
(TVA BFNP TS 199)

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1.0 DEFINITIONS (cont'd)

E. Operable - Operability - A system, subsystem, train, component, or device shall be Operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

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 valves 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% 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.

1.0 DEFINITIONS (Cont'd)

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% high flux trip) and the 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 sec), 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 indicate 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 satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant systems 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.

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 applicable 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 comparisons 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 > limits specified in specification 3.5.k 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 possimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

2.1 BASIS

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 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 turbine first state pressure.

2.1 BASES

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

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. References

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

2. Generic Reload Fuel Application, Licensing Topical Report NEDE-20411-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 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.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the pressure is not monitored at this point, it cannot be directly determined if this safety limit has been violated. Also, because of the potentially varying head level and flow pressure drops, an equivalent pressure cannot be a priori determined for a

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 channel 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

Min. No. of Operable Inst. Channels Per Trip System(1)(23)	Trip Function	Trip Level Setting	Shut-down	Refuel(7)	Startup/Hot Standby	Run	Action(1)
			Modes in Which Function Must Be Operable				
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	Inoperable			X	X	(5)	1.A
	APRM (16) (24)(25)						
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux (Fixed Trip)	≤120 %				X	1.A or 1.B
2	High Flux	≤15% 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-64-56 A-D)	≤2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14) (LIS-3-203-A-D)	≥538" above vessel zero		X	X	X	1.A
2	High Water Level in West Scram Discharge Tank (LS-85-45 A-D)	≤50 Gallons	X	X(2)	X	X	1.A
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤50 Gallons	X	X(2)	X	X	1.A

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

Min. No. of Operable Inst. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shut-down	Refuel (7)	Startup/Hot Standby	Run	Action (1)
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure				X(6)	1.A or 1.C
2	Turbine Cont. 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.

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 Refuelin and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekl 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 (8)	Once/Month (1)
High Drywell Pressure (PIS-64-56 A-D)	B	Trip Channel and Alarm (8)	Once/Month (1)
Reactor Low Water Level (LIS-3-203 A-D)	B	Trip Channel and Alarm (8)	Once/Month (1)
High Water Level in Scram Discharge Tank			
Float Switches (LS-85-45 C-F)	A	Trip Channel and Alarm	Once/month
Electronic Level Switches (LS-85-45A, B, G, H)	B	Trip Channel and Alarm	Once/month (8)
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/month (1)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/week

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 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-1-81 A&B, PIS-1-91 A&B)	B	Trip Channel and Alarm (8)	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. Calibration of master/slave trip units only.

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

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

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once every 7 days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/operating cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure (PIS-3-22AA, BB, C, D)	B	Standard Pressure Source	Once/Operating Cycle (9)
High Drywell Pressure (PIS-64-56 A-D)	B	Standard Pressure Source	Once/Operating Cycle (9)
Reactor Low Water Level (LIS-3-203 A-D)	B	Pressure Standard	Once/Operating Cycle (9)
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45 C-F)	A	Note (5)	Note (5)
Electronic Level Switches (LS-85 A, B, G, H)	B	Calibrated Water Column	Once/Operating Cycle
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive (PIS-1-81 A&B, PIS-1-91 A&B)	B	Standard Pressure Source	Once/Operating Cycle (9)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Turbine Cont. Valve Fast Closure on Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle

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, MSTV closure, turbine control valve fast closure, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

3.1 <u>Basis</u>

modes. In the power range the APRM system provides required protection. Ref. Section 7.5.7 FSAR. Thus, the IRM System is not required in the Run mode. The APRM's and the IRM's provide adequate coverage in the startup and intermediate range.

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

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

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

Because of the APRM downscale limit of $\geq$ 3% when in the Run mode and high level limit of $\leq$ 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-3-203 A-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-3-56 A-D)	≥ 470" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PIS-64-56 A-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

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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)	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 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)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2	Instrument Channel - Drywell High Pressure (PIS-64-58E-H)	1≤ P≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal.(15)
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PIS-3-204A-D)	≤ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PIS-64-57 A-D)	≤ 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.
2	Instrument Channel - Reactor Low Pressure (PIS-3-74A&B) (PIS-68-95, 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, 96)	230 psig ± 15	A	1. Recirculation discharge valve actuation.

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TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	$\leq$0.66W + 42% (2)
4(1)	APRM Upscale (Startup Mode) (8)	$\leq$12%
4(1)	APRM Downscale (9)	$\geq$3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	$\leq$0.66W + 40% (2)(13)
2(7)	RBM Downscale (9)	$\geq$3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	$\leq$108/125 of full scale
6(1)	IRM Downscale (3) (8)	$\geq$5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	$\leq 1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4) (8)	$\geq$3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	$\leq$10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$115% recirculation flow
1	Rod Block Logic	N/A
2(1)	RCSC Restraint (PS85-61A,B)	147 psig turbine first stage pressure
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	$\leq$ 25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	$\leq$ 25 gal.

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-58A LI-3-58B	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-64-67B	Drywell Pressure	Alarm at 35 psig)	
1	TS-64-52A& PIS-64-58A& IS-64-67A	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure > 2.5 psig after 30 minute delay)	(1) (2) (3) (4)
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential Pressure	Indicator 0 to 2 psid	(1) (2) (3)
1/Valve		Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe		(5)
2	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- 160A XR-64-159	Drywell Pressure Wide Range	Indicator, Recorder 0-300 psig	(1) (2) (3)
2	TI-64-161 TR-64-161 TI-64-162 TR-64-162	Suppression Pool Bulk Temperature	Indicator, Recorder 30° - 230° F	(1) (2) (3) (4) (6)
	RR-90-322A	Wide Range Gaseous Effluent Radiation Monitor	Recorder Noble Gas) 10^{-7} - 10^{+5} µCi/cc) (7) Iodine and Particulates) 10^{-12} - 10^{+2} µCi/cc)	

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 operable. If both the primary and secondary indication on any SRV tail pipe 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, 50A-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-64-58E-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
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

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 B)	Once/6 months	NA
11) Drywell Pressure (PIS-64-58A)	Once/6 months	NA
12) Drywell Temperature (TS-64-52A)	Once/6 months	NA
13) Timer (IS-64-67A)	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-Wide Range (PI-64-160A) (XR-64-159)	Once/cycle	Once/shift
22	Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162)	Once/cycle	Once/shift
23	High Range Gaseous Effluent Radiation Monitor (RR-90-322A)	Once/cycle	Once/shift

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

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.

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.

3.5.H Maintenance of Filled Discharge Pipe

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

I. Average Planar Linear Heat Generation Rate

During steady state power operation, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1, -2.

If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 kw/ft. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5.H Maintenance of Filled Discharge Pipe

1. Every month prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge, piping of these systems shall be vented from the high point and water flow determined.

2. Following any period where the LPCI or core spray systems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. When the RHRS and the CSS are required to be operable, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

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

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

The LHGR for 8X8, 8X8R, and P8X8R fuel shall be checked daily during reactor fuel operation at ≥ 25% rated thermal power.

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 ± 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, -2. The analyses supporting these limiting values is presented in Reference 1.

3.5.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 generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns, which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MPCR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values for MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specifications 3.5.I., .J., and .K., that if at any time during steady state power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates than an operating limit has been reached. Each event involving steady state operation beyond a specified limit shall be 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 circumstances, this will not be the only alternative.

3.5.M. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2, NEDO - 24088-1 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.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Types: P8DRB284L

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	11.8
25,000	11.2
30,000	10.8
35,000	10.0
40,000	9.4

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Types: P8DRB265H

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	12.0
25,000	11.6
30,000	11.2
35,000	10.9
40,000	10.5
45,000	10.0

3.6/4.6 BASES:

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their set points are within the $\pm$ 1 percent tolerance. The relief valves are tested in place once per operating cycle to establish that they will open and pass steam.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.

3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973.

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

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within $\pm$ 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

g. Local Leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves, which are not part of a water-sealed system, at not less than 49.6 psig (except for the main steam isolation valves, see 4.7.A.2.i) and not less than 54.6 psig for water-sealed valves each operating cycle. Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened and at least once per operating cycle. Acceptable methods of testing are halide gas detection, soap bubbles, pressure decay, hydrostatically pressurized fluid flow or equivalent.

The personnel air lock shall be tested at 6-month intervals at an internal pressure of not less than 49.6 psig. In addition, if the personnel air lock is opened during periods when containment integrity is not required, a test at the end of such a period will be conducted at not less than 49.6 psig. If the personnel air lock is opened during a period when containment integrity is required, a test at $\geq$ 2.5 psig shall be conducted within 3 days after being opened. If the air lock is opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings.

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage. Penetrations and isolation valves are identified as follows:

(1) Testable penetrations with double O-ring seals - Table 3.7.B,

(2) Testable penetrations with testable bellows Table 3.7.C,

(3) Isolation valves without fluid seal - Table 3.7.D,

(4) Testable electrical penetrations - Table 3.7.H, and

(5) Isolation valves sealed with fluid - Tables 3.7.E, and 3.7.F.

TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

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
1	Main steamline isolation valves (FCV-1-14, 26, 37, & 51; 1-15, 27, 38 & 52)	4	4	3 < T < 5	O	GC
1	Main steamline drain isolation valves (FCV-1-55 & 1-56)	1	1	15	O	GC
1*	Reactor Water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves (FCV-74-48 & 47)	1	1	40	C	SC
2	RHRS - LPCI to reactor (FCV-74-53 & 67)		2	30	C	SC
2	RHRS flush and drain vent to suppression chamber (FCV-74-102, 103, 119, & 120)		4	20	C	SC
2	Suppression Chamber Drain (FCV-75-57 & 58)		2	15	C	SC
2	Drywell equipment drain discharge isolation valves (FCV-77-15A & 15B)		2	15	O	GC
2	Drywell floor drain discharge isolation valves (FCV-77-2A & 2B)		2	15	O	GC

*These valves isolate only on reactor vessel low low water level (470") and main steam line high radiation of Group 1 isolations.

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-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
43-22A	RHR Suppression Chamber Sample Lines
43-22B	RHR Suppression Chamber Sample Lines
43-29A	RHR Suppression Chamber Sample Lines
43-29B	RHR Suppression Chamber Sample Lines
71-14	RCIC Turbine Exhaust
71-22	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

TABLE 3.7.F

PRIMARY CONTAINMENT ISOLATION VALVES LOCATED IN
WATER SEALED SEISMIC CLASS 1 LINES

Valve	Valve Identification
74-53	RHR LPCI Discharge
74-54	RHR
74-57	RHR Suppression Chamber Spray
74-58	RHR Suppression Chamber Spray
74-60	RHR Drywell Spray
74-61	RHR Drywell Spray
74-67	RHR LPCI Discharge
74-68	RHR LPCI Discharge
74-71	RHR Suppression Chamber Spray
74-72	RHR Suppression Chamber Spray
74-74	RHR Drywell Spray
74-75	RHR Drywell Spray
75-25	Core Spray Discharge
75-26	Core Spray Discharge
75-53	Core Spray Discharge
75-54	Core Spray Discharge

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 inoperability of thermocouple and 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.	Primary Containment Integrated Leak Rate Testing	4.7.A.2	Within 90 days of completion of each test.
	High-Range Primary Containment Radiation Monitors	3.2.F	Within 7 days after 7 days of inoperability
	High-Range Gaseous Effluent Radiation Monitor	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
BROWNS FERY NUCLEAR PLANT UNIT 2
(TVA BFNP TS 199)

A. Core-Related Changes

1. MCPR

Pages viii, 172 and 172a 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. Page 172a should be removed.

2. References in Bases

Pages 19, 23, 25, 28, 168a, and 169 - 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.

3. MAPLHGR

Pages vii, 159, 171, and 172 - Deleted unnecessary tables, revised table of contents to reflect new page numbers and revised reference to tables on page 159.

The justification and safety analysis for above revisions are described in TVA-RLR-002, 'Reload Licensing Report for Browns Ferry Unit 2 Cycle 6.'

B. Changes Related to Torus Modifications

These modifications are required by NRC to restore the originally intended margins of safety in the containment design.

Pages 79, 80, and 105a 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 78 and 105.

C. Miscellaneous Plant Modifications

1. Reactor Protection System (RPS) Modification

Pages 31, 32 and 42 - 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.

2. 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 electronic level switches to initiate a scram on high level in the SDIV.

Pages 37 and 40 of tables 4.1.A and 4.1.B - Revised to reflect the required surveillance testing on the two electronic level switches.

3. Analog Trip System

Pages 33, 34, 37, 38, 40, 55, 62, 63, 73, 85, 96, and 102 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.

Pages 39 and 110a - Revised to add note 8 and note 28 respectively 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 8 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 8 now applies to the reactor water level instruments.

Pages 41 and 110a - Revised to add note 9 and note 29 respectively 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.

4. Scram Permissive Pressure Switches at 1055 PSIG

Pages 3, 4, 34, 35 and 44 - 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.

5. Drywell Temperature and Pressure

Pages 78 and 105 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.

6. NUREG-0737 Items

Pages 79, 80, 105a, and 110a of tables 3.2.F and 4.2.F - 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
d. High-range gaseous effluent radiation monitors

These four items are in response to NUREG-0737, items II.F.1.3, II.F.1.1, II.F.1.2, II.F.1.4, and II.F.1.5.

Pages 78 and 105 - Delete the drywell pressure and suppression chamber water level instruments. They are being replaced by items b and c above.

Pages 356 - The proposed change is addition of new reporting requirements for a condition of inoperability for the high range primary containment radiation monitors and for the high range gaseous effluent radiation monitors. These instruments were added to address the requirements of TMI Plan NUREG-0737 Item Nos. II.F.1.3 (High Range Radiation Monitors) and II.F.1.1 (High Range Gaseous Effluent Monitors).

7. Testable Penetrations

Page 256 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 is is a 'Spare.' The drywell head is being added to this table. It was inadvertently not listed, but was included in the surveillance program. Penetration X-213A was removed. It was previously removed from unit 3.

8. Redundant Air Supply to Drywell

Page 258 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 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.

9. Demineralized Water

Page 262 - 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.

10. Residual Heat Removal Head Spray

Pages 250 and 263 - Revised to delete primary containment isolation valves 74-77 and 74-78 of the RHR system head spray from tables 3.7.A and 3.7.F. The removal of the head spray line is part of the Intergranular Stress Corrosion Cracking Study being done on Browns Ferry. No safety related functions will be adversely affected by disconnecting this line; therefore, the margin of safety will not be reduced.

D. <u>Administrative Technical Specification Changes</u>

Pages iv, vi and vii - Technical specification titles for section 3.7.D-G were modified to correctly reflect the respective technical specifications. Page numbers were corrected.

Pages 231 and 232 - Revise 4.7.G surveillance requirement for personnel air lock to be consistent with units 1 and 3.

Page 220 - The proposed change is deletion of the reference to safety valves in conjunction with relief valves. The safety valves with unpiped discharge have been removed and replaced with relief valves. The relief valves have their discharge piped to below the torus water for steam condensation. There were no safety concerns associated with replacing the safety valves with relief valves. This modification was done on a previous refueling outage. This is an administrative change primarily because it updates the specifications to reflect actual plant configuration.

ENCLOSURE 3
DETERMINATION OF NO SIGNIFICANT HAZARDS
BROWNS FERRY NUCLEAR PLANT UNIT 2
(TVA BFNP TS 199)

Description of amendment request:

The amendment would revise the Technical Specifications (T.S.) of the operating license to: (1) modify the core physics, thermal and hydraulic limits to be consistent with the reanalyses associated with replacing about 1/3 of the core during the cycle 5 refueling outage for unit 2 and (2) reflect plant modifications performed during the cycle 5 refueling and modification outage. Specifically, the amendment would result in changes to the T.S. in the following twelve areas:

1. Changes to the license related to the Cycle 6 core reload involving removal of depleted fuel assemblies in about one-third of the nuclear reactor core and replacement with new fuel of the same type previously loaded in the core with attendant license changes in the core protection safety limits and reactor protection system setpoints. The actual changes are slight adjustment (by 0.01 in initial core life) in the Operating Limit Minimum Critical Power Ratio (OLMCPR), deleted two of four tables on maximum average planar linear heat generation rate (MAPLHGR) versus average planar exposure that will not be needed due to the fuel change and a change to the references in the bases to reflect that TVA performed the reload transient analysis.

   The loading pattern also includes four Westinghouse QUAD+ demonstration assemblies loaded in peripheral locations. Evaluations performed by Westinghouse indicate that the results of licensing analyses for the previous type assembies bound those for the QUAD+ assemblies. Cycle specific analyses performed by TVA confirm this conclusion.

2. Changes in the T.S. to reflect modifications to the torus as part of the Mark I containment program.

   This includes revising the tables listing surveillance instrumentation for suppression pool bulk temperature reflecting the installation of 16 sensors for an improved torus temperature monitoring system and a revision to the basis for the existing limits on torus water temperature.

3. Change to the T.S. to reflect modifications to the scram discharge instrument volume (SDIV); each of the SDIVs now have new, diverse level instrumentation. The changes to the T.S. are to add operability, surveillance and calibration requirements on the new level instrumentation.

4. Change to T.S. surveillance instrumentation tables to add new instrumentation for containment high-range radiation monitors and to add new instrumentation; and delete current instrumentation for drywell pressure-wide range and suppression chamber wide-range water level in response to requirements in NUREG-0737; items II.F.1.3, II.F.1.4 and II.F.1.5.

5. Changes to T.S. RPS instrumentation requirement tables to delete the bypass function if reactor pressure is less than 1055 psig and the mode switch not in the RUN mode.

6. Change to T.S. surveillance instrumentation tables to reflect new instrument numbers for the new upgraded drywell temperature and pressure instrumentation.

7. Revisions to the table of testable penetrations to reflect the new testable penetrations as a result of modifications to make the flange side of several isolation valves testable.

8. Revision to the T.S. table for containment isolation valve surveillance to add two new isolation valves that are part of a newly installed redundant discharge line from the drywell compressor into containment and to delete one isolation valve which was removed from the demineralized water system.

9. Revision to the T.S. table for containment isolation valve surveillance to delete two isolation valves for the residual heat removal head spray line which is being removed.

10. Revision of T.S. to provide limiting conditions for operation and surveillance requirements for electric power monitoring for the reactor protection system power supply.

11. Modify the T.S. to apply to the new analog (continuous measuring) instrumentation. The analog instrumentation replaces certain mechanical-type pressure and level switches with a more accurate and more stable electronic transmitter/electronic switch system and will provide improved performance of trip functions for reactor protection system actuation, and containment isolation. The changes to the T.S. include:

    a. in the tables on functional test frequencies, calibration frequencies and surveillance requirements, for each switch replaced, add the instrument number and type of sensor beneath the parameter being monitored and/or controlled.

    b. add notes to the above tables to specify how the functional and calibration tests are to be conducted.

    c. in addition to the above administrative changes, the calibration requirements have been changed to incorporate extended calibration intervals. However, the required setpoints, functional test frequencies and channel check frequencies for the instrumentation will not be changed. The new calibration requirements, together with the new instrumentation, are expected to provide a more reliable instrumentation system.

12. Administrative changes to the T.S. involving changes to the Table of Contents to reflect the above license changes and miscellaneous editorial changes.

Bases for proposed no significant hazards consideration determination:

The Commission has provided guidance concerning the application of the standards by providing examples of actions that are likely, and are not likely, to involve significant hazard considerations (48 FR 14870). Four examples of actions not likely to involve significant hazards considerations are:

"(i) A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more significant surveillance requirement.

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable...

(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

Each of the twelve changes to the T.S. described previously is encompassed by one of the above example of actions not likely to involve a significant hazards consideration. The basis for this determination on each of the twelve changes is discussed below.

1. Core Reload

1:a Fuel Changes

The changes to the T.S. associated with removing depleted spent fuel from the reactor and replacing these with new fuel assemblies is encompassed by example (iii) above of those actions not likely to involve a significant hazards consideration.

The proposed reload involves fuel assemblies which have been shown to be analytically similar or which are of the same type as previously found acceptable by the staff and loaded in the core in previous cycles. The analytical methods used by the licensee to demonstrate conformance to the technical specifications have been previously approved by the staff. In addition, no changes have been made to the acceptance criteria for the technical specification changes involved.

Since the replacement fuel assemblies are analytically similar or of the same type previously added to all three Browns Ferry units and other BWRs and since the codes, models, and analytical techniques used to analyze the reload have been generically approved by the NRC, the changes to the technical specifications associated with the reload are clearly encompassed by example (iii) of the guidance provided by the Commission for an action not likely to involve a significant hazards consideration.

1.b References in the Bases

The changes in the T.S. associated with changing the references in the Bases to reflect that the reload transient analysis is now being performed by TVA is encompassed by examples (i) and (iii) above of those actions not likely to involve a significant hazards consideration.

The reload analysis, in the past, has been performed by General Electric Company. This reload analysis has been performed by TVA using analytical methods described in TVA-TR81-01-A. The analytical methods have been approved by the staff. Since NRC has previously found these methods acceptable and the T.S. changes are being made to achieve consistency between the methods used and the references in the Bases, these changes to the T.S. are clearly encompassed by examples (i) and (iii) of the guidance provided by the Commission for an action not likely to involve a significant hazards consideration.

2. Changes Related to Torus Modifications

One of the changes to the T.S. is to revise the tables that list 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 presently listed in the T.S.

The change to the T.S. are necessary administrative follow up actions essential to the implementation of this improvement. The changes to the T.S. place operability and calibration requirements on the new temperature monitoring system. Since these are new instruments, the surveillance requirements are not presently in the T.S. Thus, adding those restrictions and controls is encompassed by example (ii) provided by the Commission.

3. Scram Discharge Instrument Volume

The SDVs and 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 electronic level switches to initiate a scram on a high level in the SDIV. Thus, the changes to the T.S. are necessary administrative follow up actions essential to the implementation of this improvement. Adding these new restrictions and controls, which otherwise would not be in the T.S., is encompassed by example (ii) of the guidance provided by the Commission.

4. Accident Monitoring Instrumentation

Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," requires all licensees to install five new monitoring systems and to provide onsite sampling/analysis capability for a specified range of radionuclides. For all six categories, NUREG-0737 states: "Changes to technical specifications will be required." During this refueling outage, the licensee has installed: (a) a containment high-range monitoring system, (b) a drywell wide-range pressure monitoring system and (c) a suppression chamber wide-range water level monitoring system. These three items were required by NUREG-0737, items II.F.1.3, II.F.1.4 and II.F.1.5, respectively. The changes to the T.S., which track the model T.S. provided to the licensee by the staff, are to add operability and surveillance requirements on the new monitoring systems to the T.S.

The revisions also delete the present drywell pressure and suppression chamber water level instruments since they are being replaced by items b and c above. The changes to the technical specifications are necessary administrative follow up actions required by the Commission. Adding the new surveillance requirements and controls is encompassed by example (ii) of the guidance provided by the Commission.

5. Scram Permissive Pressure Switches at 1055 psig

Present configuration on unit 2 has a bypass function which allows a scram in the refuel and startup/hot standby modes of operation by the scram functions main steamline isolation valve closure and turbine condenser low vacuum 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 steamline 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 become available, the reactor is already scrammed. Since the reactor is protected by the high-pressure scram function, the proposed change does not result in any reduction in the margin of safety. The T.S. changes therefore are encompassed by example (vi) of the guidance provided by the Commission.

6. Drywell Temperature and Pressure

The drywell temperature and pressure surveillance instrumentation is being upgraded this outage to provide qualified, more reliable instrumentation. The T.S. are being revised to reflect new instrument numbers. The surveillance requirements remain unchanged. The changes to the technical specifications are necessary administrative follow up actions required by the Commission and are clearly encompassed by example (i) of the guidance provided by the Commission.

7. Testable Penetrations

Modifications are being made to the flange side of fourteen containment isolation valves which cannot be isolated from primary containment to be tested. This modification will provide two gaskets with a pressure tap between the gaskets to allow the flange to be leak tested. Operability of the valve will not be affected by this modification. Fourteen new testable penetrations resulted and they were added to the table of testable penetrations with double o-ring seals. New surveillance requirements are being added. The change is encompassed by example (ii) of the guidance provided by the Commission.

Several editorial changes were also made to this table. They include revising the identification name on several penetrations, adding a penetration that was tested but was inadvertently left out of the table and removing penetration X-213A which no longer exists. These changes are purely administrative and are encompassed by example (i) of the guidance provided by the Commission.

8. Redundant Air Supply to Drywell

During the current outage, TVA has installed a second discharge line from the drywell compressor into containment. This line was added to provide the capability for isolation of approximately one-half of the drywell suppression equipment in the case of a drywell line leak. This air supply will be used to supply two inboard main steam isolation valves (MSIVs), approximately one-half of the main steam relief valves (MSRVs), 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 as a result of MSIVs, MSRVs, and drywell coolers being inoperable, Since any line penetrating containment requires two isolation valves, the table in the Technical Specifications listing the isolation valves that must be periodically tested is being revised to add these two new isolation valves. TVA has concluded that this modification will increase the margin of safety. The changes to the technical specifications are necessary administrative follow up actions essential to the implementation of this improvement. The two isolation valves being added to the T.S. are new valves not presently listed in the T.S. If they are not added to the table of valves to be periodically tested, there would be no T.S. requirement to test these valves. Adding these additional controls is encompassed by example (ii) of the guidance provided by the Commission.

One isolation valve on the demineralized water system was removed from unit 2. The demineralized water system is no longer used. The isolation valve was removed and the line capped. The T.S. are being revised to remove this valve from the table of valves to be tested. The changes to the technical specifications are necessary administrative follow up actions essential to the implementation of the improvement. The changes are clearly encompassed by example (i) provided by the Commission.

9. Residual Heat Removal Head Spray Line

Two isolation valves on the residual heat removal head spray line were removed from unit 2. The head spray line was removed and the penetration capped. The T.S. are being revised to remove these valves from the table of valves to be tested. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of the improvement. The changes are clearly encompassed by example (i) provided by the commission.

10. Monitoring of RPS Power Supply

By letter dated August 7, 1978, the Commission advised TVA that during review of Hatch unit 2, the staff had identified certain deficiencies in the design of the voltage regulator system of the motor generator sets which supply power to the reactor protection system (RPS). Pursuant to 10 CFR 50.54(f), TVA was required to evaluate the RPS power supply for Browns Ferry 1, 2, and 3 in light of the information set forth in our letter. By letter dated September 24, 1980, the staff informed TVA (and most other BWRs) that "we have determined that modifications should be performed to provide fully redundant Class IE protection at the interface of non-Class IE power supplies and the RPS." The staff also advised TVA that "we have found that the conceptual design proposed by the General Electric Company and the installed modification on Hatch are acceptable solutions to our concern." By letter dated December 4, 1980, TVA committed to install the required modifications. By letters dated October 30, 1981 and July 28, 1982, NRC sent TVA model Technical Specifications for electric power monitoring of the RPS design and modification. During the current outage of unit 2, the RPS is being modified to provide a fully redundant Class IE protection at the interface of the non-Class IE power supplies and the RPS. This will ensure that failure of a non-Class IE reactor protection power supply will not cause adverse interaction to the class IE reactor protection system.

The Technical Specifications are being revised similar to the model T.S. provided to TVA 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 changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The additional limitations and controls, which are presently not in the T.S., are encompassed by example (ii) of the guidance provided by the Commission.

11. Analog Trip System

The RPS, the primary containment isolation system (PCIS), and the core standby cooling systems (CSCS) use mechanical-type switches in the sensors that monitor plant process parameters. These mechanical-type switches are very subject to drift in the setpoint as is evident from the many licensee event reports (LERs) that have been submitted reporting calibration drifts in these switches.

Advances in technology make it possible to replace the mechanical-type switches with a more accurate and more stable electronic transmitter/ electronic switch system. For several years, TVA has been planning to replace existing pressure switches that sense drywell and reactor pressures with analog loops and modify the reactor water level indication loops to improve the reliability, accuracy and response time of this instrumentation. The modification involves removing one device and substituting other devices to perform the same function. Changes in design bases, protective function, redundancy, trip point and logic are not involved. Similar modifications have been approved for other BWRs.

As described previously, most of the changes to the T.S. are administrative in nature (i.e., adding the specific number and types of sensor and adding notes to describe how testing is conducted). As such, they are encompassed by example (i) of the guidance provided by the Commission. The changes in surveillance requirements relates to example (ii) of the guidance provided by the Commission. Some of the surveillance intervals have been decreased as appropriate for each new instrument. However, the overall effect of the changes in technical specifications will be to increase the total surveillance requiremens in support of a more reliable instrumentation system.

12. Administrative Changes

Several administrative changes are being made to the Technical Specific-tions. These include revising the Table of Contents to reflect the change discussed above, and miscellaneous editorial changes. The surveillance requirement for the personnel air lock is being changed to be consistent with the surveillance for units 1 and 3. These changes are editorial in nature and have no safety significance. These changes are encompassed by example (i) cited by the Commission as an action not likely to pose a significant hazards consideration.

Since all of the changes to the T.S. given in the twelve areas above are encompassed by an example in the guidance provided by the Commission of actions not likely to involve a significant hazards consideration, the staff has made a proposed determination that the application for amendment involves no significant hazards consideration.