ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION REVISIONS
BROWNS FERRY NUCLEAR PLANT
UNIT 1 RELOAD 5
(TVA-BFNP-TS-190)

# ENCLOSURE 1

## PART I - CORE RELOAD CHANGES

## 2.2 BASES

### REACTOR COOLANT SYSTEM INTEGRITY

To meet the safety basis, thirteen relief valves have been installed on the unit with a total capacity of 84.1% of nuclear boiler rated steam flow at a reference pressure of (1105 + 1%) psig. The analysis of the worst overpressure transient (3-second closure of all main steam line isolation valves), neglecting the direct scram (valve position scram), results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowable vessel overpressure of 1375 psig.

## 3.5 BASES

### H. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, HPCIS, and RCICS are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for Technical Specification purposes.

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

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

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

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

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 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, -3, -4, -5, and -6. The analyses supporting these limiting values is presented in Reference 4.

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, MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K Minimum 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 MCPR value is in excess of requirments 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 APRM Setpoints

The fuel cladding integrity safety limits of section 2.1 were based on a total peaking factor within design limits (FRP/CMFLPD ≥1.0). The APRM instruments must be adjusted to ensure that the core thermal limits are not exceeded in a degraded situation when entry conditions are less conservative than design assumptions.

3.5.M 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 Specification 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 that 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.

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265H and 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 | 11.9 |
| 25,000 | 11.3 |
| 30,000 | 10.7 |
| 35,000 | 10.2 |
| 40,000 | 9.6 |

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265L and P8DRB265L

| Average Planar Exposure (MWd/t) | MAPLHGR (kW/ft) |
|---|---|
| 200 | 11.6 |
| 1,000 | 11.6 |
| 5,000 | 12.1 |
| 10,000 | 12.1 |
| 15,000 | 12.1 |
| 20,000 | 11.9 |
| 25,000 | 11.3 |
| 30,000 | 10.7 |
| 35,000 | 10.2 |
| 40,000 | 9.6 |

Table 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB284L, GLTA-1, GLTA-2

| Exposure (MWd/t) | MAPLHGR (kW/ft) |
|---|---|
| 200 | 11.2 |
| 1000 | 11.3 |
| 5000 | 11.8 |
| 10,000 | 12.0 |
| 15,000 | 12.0 |
| 20,000 | 11.8 |
| 25,000 | 11.2 |
| 30,000 | 10.8 |
| 35,000 | 10.2 |
| 40,000 | 9.5 |

Table 3.5.I-6

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB284Z

| Average Planar Exposure (MWd/t) | MAPLHGR (kW/ft) |
|---|---|
| 200 | 11.2 |
| 1,000 | 11.2 |
| 5,000 | 11.7 |
| 10,000 | 12.0 |
| 15,000 | 12.0 |
| 20,000 | 11.8 |
| 25,000 | 11.1 |
| 30,000 | 10.4 |
| 35,000 | 9.8 |
| 40,000 | 9.1 |
| 45,000 | 8.5 |

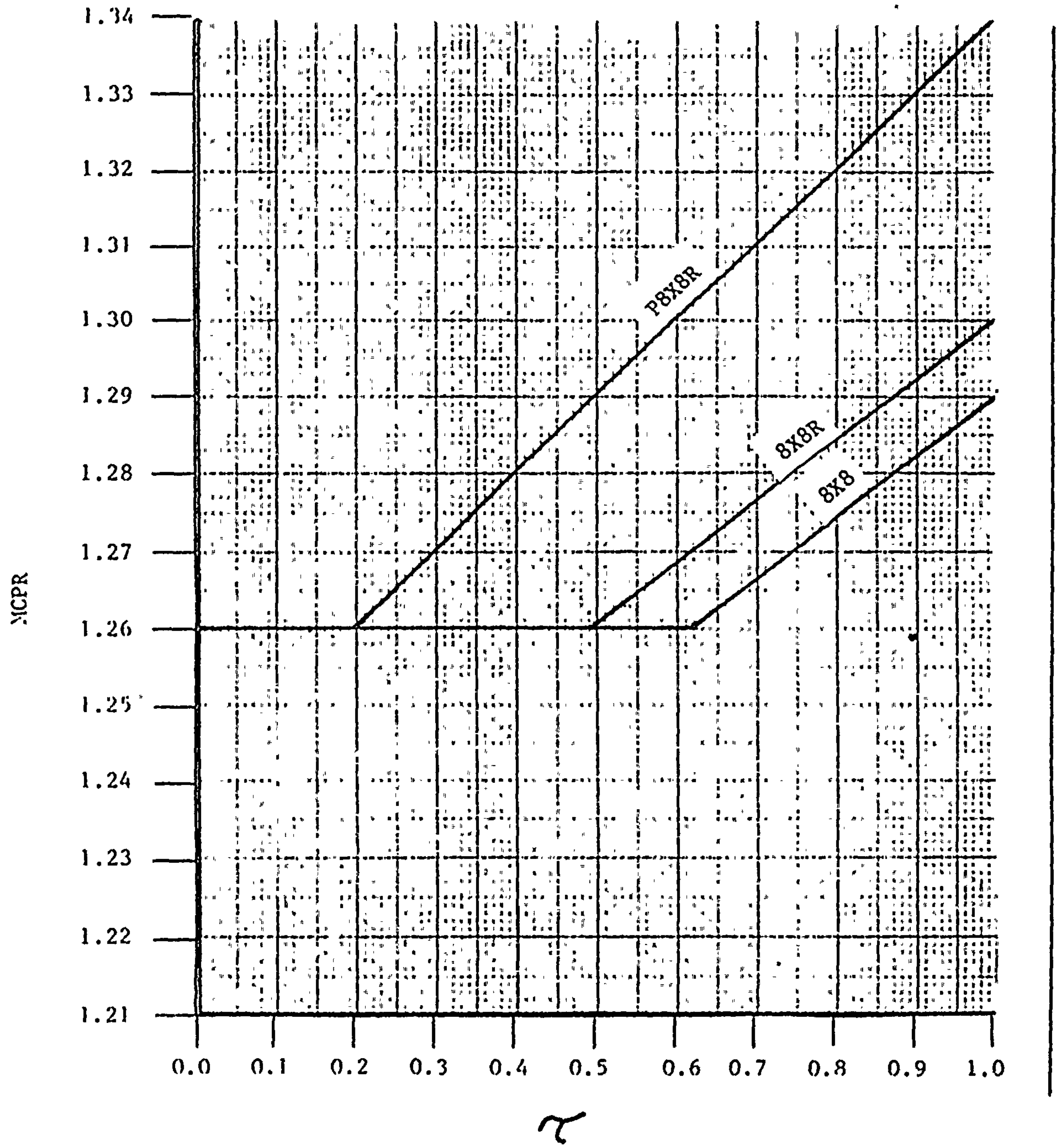


Figure 3.5.K-1

MCPR LIMITS*

*Note: Lead test assemblies are categorized as P8 x 8R bundles.

detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCES

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)

## 3.6.D/4.6.D Relief Valves

To meet the safety basis, thirteen relief valves have been installed on the unit with a total capacity of 84.1% of nuclear boiler rated steam flow at a reference pressure of (1105 + 1%) psig. The analysis of the worst overpressure transient (3-second closure of all main steam line isolation valves), neglecting the direct scram (valve position scram), results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1375 psig.

ENCLOSURE 1

PART II - PLANT MODIFICATION RELATED CHANGES

## LIST OF TABLES

## 1.1 FUEL CLADDING INTEGRITY

### Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

### Objective

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

### Specifications

A. Thermal Power Limits

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

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

## 2.1 FUEL CLADDING INTEGRITY

### Applicability

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

### Objective

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

### Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) (Flow Biased)
   a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

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

where:

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

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals $34.2 \times 10^6$ lb/hr)

## 1.1 FUEL CLADDING INTEGRITY

2. Reactor Pressure ≤800 PSIA or Core Flow ≤10% of rated.

When the reactor pressure is ≤800 PSIA or core flow is ≤10% of rated, the core thermal power shall not exceed 823 MWt (~25% of rated thermal power).

## 2.1 FUEL CLADDING INTEGRITY

d. Fixed High Neutron Flux Scram Trip Setting -- When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

$S \leq 120$ % power .

2. APRM and IRM Trip Settings (Startup and Hot Standby Modes).

a. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

b. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

## 2.1 BASES

In summary

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

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

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

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

The bases for individual set points are discussed below:

A. Neutron Flux Scram

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

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

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

## 2.1 BASES

### IRM Flux Scram Trip Setting (Continued)

example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15% scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux, and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

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

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin

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.

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.

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.

C. 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 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.

3.1 REACTOR PROTECTION SYSTEM

B.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.

4.1 REACTOR PROTECTION SYSTEM

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

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

2. At least once per operating cycle by demonstrating the operability of over-voltage, under-voltage, and under-frequency protective instrumentation including simulated automatic actuation logic, and output contactors, and verifying the following setpoints.

   a. Over-voltage $\geq$ 132 VAC
   b. Under-voltage $\leq$ 108 VAC
   c. Under-frequency $\leq$ 57 Hz

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) |
|---|---|---|---|---|---|---|---|
| 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 | ≤1055 psig | | X(10) | X | X | 1.A |
| 2 | High Drywell Pressure (14) | ≤2.5 psig | | X(8) | X(8) | X | 1.A |
| 2 | Reactor Low Water Level (14) | ≥538" above vessel zero | | X | X | X | 1.A |
| 2 | High Water Level in West Scram Discharge Tank (LS-85-45A-D) | ≤50 Gallons | X | X(2) | X | X | 1.A |
| 2 | High Water Level in East Scram Discharge Tank (LS-85-45E-H) | ≤50 Gallons | X | X(2) | X | X | 1.A |

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

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

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

| | Group (2) | Functional Test | Minimum Frequency (3) |
|---|---|---|---|
| Mode Switch in Shutdown | A | Place Mode Switch in Shutdown | Each Refueling Outage |
| Manual Scram | A | Trip Channel and Alarm | Every 3 Months |
| IRM | | | |
| High Flux | C | Trip Channel and Alarm (4) | Once Per Week During Refueling and Before Each Startup |
| Inoperative | C | Trip Channel and Alarm (4) | Once Per Week During Refueling and Before Each Startup |
| APRM | | | |
| High Flux (15% scram) | C | Trip Output Relays (4) | Before Each Startup and Weekly When Required to be Operable |
| High Flux (Flow Biased) | B | Trip Output Relays (4) | Once/Week |
| High Flux (Fixed Trip) | B | Trip Output Relays (4) | Once/Week |
| Inoperative | B | Trip Output Relays (4) | Once/Week |
| Downscale | B | Trip Output Relays (4) | Once/Week |
| Flow Bias | B | (6) | (6) |
| High Reactor Pressure (PT-3-22A-D) | B | Trip Channel and Alarm (7) | Once/Month (1) |
| High Drywell Pressure (PT-64-56A-D) | B | Trip Channel and Alarm (7) | Once/Month (1) |
| Reactor Low Water Level (LT-3-203A-D) | B | Trip Channel and Alarm (7) | Once/Month (1) |
| High Water Level in Scram Discharge Tank | | | |
| Float Switches (LS-85-45C-F) | A | Trip Channel and Alarm | Once/month |
| High Water Level in Scram Discharge Tank Electronic Level Switches (LS-85-45A, B, G, H) | B | Trip Channel and Alarm (7) | Once/month |
| Turbine Condenser Low Vacuum | A | Trip Channel and Alarm | Once/month (1) |
| Main Steam Line High Radiation | B | Trip Channel and Alarm (4) | Once/week |

TABLE 4.1.A (Continued)

| | 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 (PT-1-81A and B, PT-1-91A and B) | B | Trip Channel and Alarm (7) | 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. [illegible]

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

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

TABLE 4.1.B

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

| Instrument Channel | Group (1) | Calibration | Minimum Frequency (2) |
|---|---|---|---|
| IRM High Flux | C | Comparison to APRM on Controlled startups (6) | Note (4) |
| APRM High Flux | | | |
| Output Signal | B | Heat Balance | Once Every 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 (PT-3-22A-D) | B | Standard Pressure Source | Once/Operating Cycle (9) |
| High Drywell Pressure (PT-64-56A-D) | B | Standard Pressure Source | Once/Operating Cycle (9) |
| Reactor Low Water Level (LT-3-203A-D) | B | Pressure Standard | Once/Operating Cycle (9) |
| High Water Level in Scram Discharge Volume Float Switches (LS-85-45C-F) | A | Calibrated Water Column (5) | Note (5) |
| High Water Level in Scram Discharge Volume Electronic Level Switches (LS-85-45-A, B, G, H) | B | Calibrated Water Column | Once/Operating Cycle (9) |
| Turbine Condenser Low Vacuum | A | Standard Vacuum Source | Every 3 Months |
| Main Steam Line Isolation Valve Closure | A | Note (5) | Note (5) |
| Main Steam Line High Radiation | B | Standard Current Source (3) | Every 3 Months |
| Turbine First Stage Pressure Permissive (PT-1-81A and B, PT-1-91A and B) | B | Standard Pressure Source | Once/Operating Cycle (9) |
| Turbine Stop Valve Closure | A | Note (5) | Note (5) |

NOTES FOR TABLE 4.1.B

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

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

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

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

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

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

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

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

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

## 3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

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

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

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

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

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

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

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

TABLE 3.2.B (Continued)

| Minimum No. Operable Per Trip Sys (1) | Function | Trip Level Setting | Action | Remarks |
|---|---|---|---|---|
| 1 | Core Spray Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems. |
| 1 | ADS Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems and valves. |
| 1 | HPCI Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems. |
| 1 | RCIC Trip System bus power monitor | N/A | C | 1. Monitors availability of power to logic systems. |
| 1(2) | Instrument Channel - Condensate Header Low Level (LS-73-55A & B) | ≥ Elev. 551' | A | 1. Below trip setting will open HPCI suction valves to the suppression chamber. |
| 1(2) | Instrument Channel - Suppression Chamber High Level | ≤ 7" above normal water level | A | 1. Above trip setting will open HPCI suction valves to the suppression chamber. |
| 2(2) | Instrument Channel - Reactor High Water Level | ≤ 583" above vessel zero. | A | 1. Above trip setting trips RCIC turbine. |
| 1 | Instrument Channel - RCIC Turbine Steam Line High Flow | ≤ 450" $H_2O$ (7) | A | 1. Above trip setting isolates RCIC system and trips RCIC turbine. |

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

| Minimum No. Operable Per Trip Sys (5) | Function | Trip Level Setting |
|---|---|---|
| 2(1) | APRM Upscale (Flow Bias) | ≤0.66W+42% (2) |
| 2(1) | APRM Upscale (Startup Mode) (8) | ≤12% |
| 2(1) | APRM Downscale (9) | ≥3% |
| 2(1) | APRM Inoperative | (10b) |
| 1(7) | RBM Upscale (Flow Bias) | ≤ 0.66W+40% (2) (13) |
| 1(7) | RBM Downscale (9) | ≥3% |
| 1(7) | RBM Inoperative | (10c) |
| 3(1) | IRM Upscale (8) | ≤ 108/125 of full scale |
| 3(1) | IRM Downscale (3) (8) | ≥ 5/125 of full scale |
| 3(1) | IRM Detector not in Startup Position (8) | (11) |
| 3(1) | IRM Inoperative (8) | (10a) |
| 2(1) (6) | SRM Upscale (8) | ≤ 1 x $10^5$ counts/sec. |
| 2(1) (6) | SRM Downscale (4) (8) | ≥ 3 counts/sec. |
| 2(1) (6) | SRM Detector not in Startup Position (4) (8) | (11) |
| 2(1) (6) | SRM Inoperative (8) | (10a) |
| 2(1) | Flow Bias Comparator | ≤ 10% difference in recirculation flows |
| 2(1) | Flow Bias Upscale | ≤ 115% recirculation flow |
| 1(1) | Rod Block Logic | N/A |
| 2(1) | RSCS Restraint (PS-85-61A and PS-85-61B) | 147 psig turbine first-stage pressure |
| 1(12) | High Water Level in West Scram Discharge Tank (LS-85-45L) | ≤25 gal. |
| 1(12) | High Water Level in East Scram Discharge Tank (LS-85-45M) | ≤25 gal. |

## TABLE 3.2.F
## SURVEILLANCE INSTRUMENTATION

| Minimum # of Operable Instrument Channels | Instrument # | Instrument | Type Indication and Range | Notes |
|---|---|---|---|---|
| 2 | LI-3-46 A<br>LI-3-46 B | Reactor Water Level | Indicator - 155" to +60" | (1) (2) (3) |
| 2 | PI-3-54<br>PI-3-61 | Reactor Pressure | Indicator 0-1200 psig | (1) (2) (3) |
| 2 | TI-64-52<br>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) |
| [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
| [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
| 1 | N/A | Control Rod Position | 6V Indicating Lights ) | (1) (2) (3) (4) |
| 1 | N/A | Neutron Monitoring | SRM, IRM, LPRM 0 to 100% power ) | |
| 1 | PS-64-67 | Drywell Pressure | Alarm at 35 psig ) | (1) (2) (3) (4) |
| 1 | TR-64-52 and<br>PS-64-58 B and<br>IS-64-67 | Drywell Temperature and Pressure and Timer | Alarm if temp. > 281°F and pressure > 2.5 psig after 30 minute delay ) | |
| 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<br>$H_2M$ - 76 - 104 | Drywell and Torus Hydrogen Concentration | 0.1 - 20% | (1) |
| 2 | PdI-64-137<br>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<br>RR-90-273CD | High Range Primary Containment Radiation Recorders | Recorder, 1 - $10^7$ R/Hr | (1) (2) (3) |
| 2 | LI-64-159A<br>XR-64-159-1 | Suppression Chamber Water Level-Wide Range | Indicator, Recorder 0-240" | (1) (2) (3) |
| 2 | PI-64-39A<br>XR-64-159-2<br>PI-64-39B<br>XR-64-159-3 | Drywell Pressure Low Range<br>Drywell Pressure Wide Range | Indicator, Recorder) -5 to +5 psig )<br>Indicator, Recorder) 0-300 psig ) | (1) (2) (3) |
| 2/torus bay | TI-64-161<br>TR-64-161<br>TI-64-162<br>TR-64-162 | Suppression Pool Bulk Temperature | Indicator, Recorder $30^\circ$ - $230^\circ$ F | (1) (2) (3) |

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 ( LT-3-203A-D, SW 1) | (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, SW #1) | (1) (28) | Once/operating cycle (29) | once/day |
| Instrument Channel - High Drywell Pressure (PT-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 (PT-1-72, -76, -82, -86) | (1) (28) | Once/operating cycle (29) | none |
| Instrument Channel - High Flow Main Steam Line dPT-1-36A-D (dPT-1-13A-D/dPT-1-25A-D/dPT-1-50A-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) | once/3 months | once/day |
| Instrument Channel Reactor Low Water Level (LIS-3-184 & 185) | (1) | once/3 months | once/day |
| Instrument Channel Reactor Low Water Level (LITS-3-52 & 62) | (1) | once/3 months | once/day |
| Instrument Channel Reactor Low Water Level (LS-3-56A-D) | (1) (28) | once/operating cycle (29) | none |
| Instrument Channel Reactor High Pressure (PT-3-204A-D) | (1) (28) | once/operating cycle (29) | none |
| Instrument Channel Drywell High Pressure (PS-64-58E-H) | (1) | once/3 months | none |
| Instrument Channel Drywell High Pressure (PS-64-58A-D) | (1) | once/3 months | none |
| Instrument Channel Drywell High Pressure (PS-64-57A-D) | (1) | once/3 months | none |
| Instrument Channel Reactor Low Pressure (PS-3-74A & B) (PS-68-95) (PS-68-96) | (1) | once/3 months | none |

TABLE 4.2.B (Continued)

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel RHR Pump Discharge Pressure | (1) | once/3 months | none |
| Instrument Channel Core Spray Pump Discharge Pressure | (1) | once/3 months | none |
| Core Spray Sparger to RPV d/p | (1) | once/3 months | once/day |
| Trip System Bus Power Monitor | once/operating cycle | N/A | none |
| Instrument Channel Condensate Header Low Level (LS-73-56A,B) | (1) | once/3 months | none |
| Instrument Channel Suppression Chamber High Level | (1) | once/3 months | none |
| Instrument Channel Reactor High Water Level | (1) | once/3 months | once/day |
| Instrument Channel RCIC Turbine Steam Line High Flow | (1) | once/3 months | none |
| Instrument Channel RCIC Steam Line Space High Temperature | (1) | once/3 months | none |

TABLE 4.2.B (Continued)

| Function | Functional Test | Calibration | Instrument Check |
|---|---|---|---|
| Instrument Channel HPCI Turbine Steam Line High Flow | (1) | once/3 months | none |
| Instrument Channel HPCI Steam Line Space High Temperature | (1) | once/3 months | none |
| Core Spray System Logic | once/6 months | (6) | N/A |
| RCIC System (Initiating) Logic | once/6 months | N/A | N/A |
| RCIC System (Isolation) Logic | once/6 months | (6) | N/A |
| HPCI System (Initiating) Logic | once/6 months | (6) | N/A |
| HPCI System (Isolation) Logic | once/6 months | (6) | N/A |
| ADS Logic | once/6 months | (6) | N/A |
| LPCI (Initiating) Logic | once/6 months | (6) | N/A |
| LPCI (Containment Spray) Logic | once/6 months | (6) | N/A |
| Core Spray System Auto Initiation Inhibit (Core Spray Auto Initiation) | once/6 months (7) | N/A | N/A |
| LPCI Auto Initiation Inhibit (LPCI Auto Initiation) | once/6 months (7) | N/A | N/A |

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

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

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

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

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

| Instrument Channel | Calibration Frequency | Instrument Check |
|---|---|---|
| 17 Relief valve Tailpipe Thermocouple Temperature | NA | Once/month (24) |
| 18 Acoustic Monitor on Relief Valve Tailpipe | Once/cycle (25) | Once/month (26) |
| 19. High Range Primary Containment Radiation Monitors | Once/cycle(27) | Once/month |
| 20. Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159-1) | Once/cycle | Once/month |
| 21. Drywell Pressure - Low Range (PI-64-39A) (XR-64-159-2) | Once/cycle | Once/shift |
| 22. Drywell Pressure - Wide Range (PI-64-39B) (XR-64-159-3) | Once/cycle | Once/shift |
| 23. Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162( (TR-64-162) | Once/cycle | Once/shift |

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.3 Reactivity Control

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

F. Scram Discharge Volume (SDV)

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

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

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

4.3 Reactivity Control

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

F. Scram Discharge Volume (SDV)

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

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

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

3. No additional surveillance required.

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

1. The RHRS shall be operable:

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

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

2. With the reactor vessel pressure less than 105 psig, the RHRS may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of suppression chamber quality water and filled with primary coolant quality water provided that during cooldown two loops with one pump per loop or one loop with two pumps, and associated diesel generators, in the core spray system are operable.

3. If one RHR pump (LPCI mode) is inoperable, the reactor may remain in operation for a period not to exceed 7 days provided the remaining RHR pumps (LPCI mode) and both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators remain operable.

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

1.

| | | |
|---|---|---|
| a. | Simulated Automatic Actuation Test | Once/ Operating Cycle |
| b. | Pump Operability | Once/ month |
| c. | Motor Operated valve operability | Once/ month |
| d. | Pump Flow Rate | Once/3 months |
| e. | Test Check Valve | Once/ Operating Cycle |

Each LPCI pump shall deliver 9000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 12000 gpm against an indicated system pressure of 250 psig.

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

3. When it is determined that one RHR pump (LPCI mode) is inoperable at a time when operability is required, the remaining RHR pumps (LPCI mode) and active components in both access paths of the RHRS (LPCI mode) and the CSS and the diesel generators shall be demonstrated to be operable immediately and daily thereafter.

3.6.C Coolant Leakage

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

D. Relief Valves

1. When more than one relief valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief/safety valve bellows shall be continuously monitored.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

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.

3.7.A Primary Containment

4.7.A Primary Containment

within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

j. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. Drywell and Torus Surfaces

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

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.7.A PRIMARY CONTAINMENT | 4.7.A PRIMARY CONTAINMENT |
| 3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers | 3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers |
| a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 psid. | a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised and the associated instrumentation including setpoint shall be functionally tested for proper operation each three months. |
| b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate primary containment integrity. | b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage. |
| 4. Drywell-Pressure Suppression Chamber Vacuum Breakers | 4. Drywell-Pressure Suppression Chamber Vacuum Breakers |
| a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and c, below. | a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every month. |
| b. One drywell-suppression chamber vacuum breaker may be non-fully closed so long as it is determined to be not more than 3" open as indicated by the position lights. | b. When it is determined that two vacuum breakers are inoperable for opening at a time when operability is all other vacuum breaker |

TABLE 3.7.A (Continued)

| Group | Valve Identification | Number of Power Operated Valves Inboard | Outboard | Maximum Operating Time (Sec.) | Normal Position | Action On Initiating Signal |
|---|---|---|---|---|---|---|
| 6 | Torus Hydrogen Sample Line Valves Analyzer A (FSV-76-55, 56) | | 2 | NA | Note 1 | SC |
| 6 | Torus Oxygen Sample Line Valves Analyzer A (FSV-76-53, 54) | | 2 | NA | Note 1 | SC |
| 6 | Drywell Hydrogen Sample Line Valves Analyzer A (FSV-76-49, 50) | | 2 | NA | Note 1 | SC |
| 6 | Drywell Oxygen Sample Line Valves Analyzer A (FSV-76-51, 52) | | 2 | NA | Note 1 | SC |
| 6 | Sample Return Valves - Analyzer A (FSV-76-57, 58) | | 2 | NA | 0 | SC |
| 6 | Torus Hydrogen Sample Line Valves Analyzer B (FSV-76-65, 66) | | 2 | NA | Note 1 | SC |
| 6 | Torus Oxygen Sample Line Valves-Analyzer B (FSV-76-63, 64) | | 2 | NA | Note 1 | SC |
| 6 | Drywell Hydrogen Sample Line Valves-Analyzer B (FSV-76-59, 60) | | 2 | NA | Note 1 | SC |
| 6 | Drywell Oxygen Sample Line Valves-Analyzer B (FSV-76-61, 62) | | 2 | NA | Note 1 | SC |
| 6 | Sample Return Valves-Analyzer B (FSV-76-67, 68) | | 2 | NA | 0 | SC |

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

TABLE 3.7.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 |

BASES

3.7.A & 4.7.A Primary Containment

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

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

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

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

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

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately 3 feet and water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached.

The maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

BASES

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

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

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

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

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

BASES

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

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

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

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

Establishing the test limit of 2.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly

## 3.11 FIRE PROTECTION SYSTEMS

### D. ROVING FIRE WATCH

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

## 4.11 FIRE PROTECTION SYSTEMS

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

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

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

### D. ROVING FIRE WATCH

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

3.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspection

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

F. Fire Protection Organization

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

4.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspections

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

F. Fire Protection Organization

No additional surveillance required.

## 3.11 FIRE PROTECTION SYSTEMS

G. Air Masks and Cylinders

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

H. Continuous Fire Watch

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

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

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

## 4.11 FIRE PROTECTION SYSTEMS

G. Air Masks and Cylinders

No additional surveillance required.

H. Continuous Fire Watch

No additional surveillance required.

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

No additional surveillance required.

ENCLOSURE 2
DESCRIPTION AND JUSTIFICATION
TVA BFNP TS 190
BROWNS FERRY NUCLEAR PLANT
UNIT 1 RELOAD 5
PART I - CORE RELOAD CHANGES

Pages 30 and 219, sections 2.2 (Bases for Reactor Coolant System Integrity) and 3.6.D/4.6.D (Bases for Relief Valves) - The value for the total capacity of the 13 relief valves is being increased to 84.1 percent of nuclear boiler rated (NBR) steam flow at a reference pressure of (1105 + 1 percent) psig. The value of 84.1-percent total relief capacity is derived from the value of 78.1-percent NBR steam flow for 12 SRVs operable out of 13 SRVs as listed in the Browns Ferry unit 1 reload 5 supplemental licensing submittal. The Browns Ferry unit 1 reload 5 supplemental licensing submittal value of 78.1-percent NBR steam flow was calculated based on certified valve capacity for a 5.125-inch throat diameter valve (870,000 lbs/hr at 1090 + 3-percent psig) issued by the ASME National Board of Boiler and Pressure Vessel Inspectors. The certified values are obtained by testing and are listed as 90 percent of the measured capacity values for conservatism.

Page 172b, figure 3.5.K-1 - The operating limit MCPRs are being revised to reflect the adjusted MCPR values for all events provided in the unit 1 reload 5 supplemental licensing submittal.

Pages vii, 172, and 172a - Add new MAPLHGR table for fuel type P8DRB284Z and fuel type P8DRB265H added to table 3.5.I-3. This reflects the latest revision of the "Loss-of-Coolant Accident Analysis Report for Browns Ferry Unit 1," General Electric Company, May 1983, NEDO-24056, Revision 1 as referenced in the Browns Ferry unit 1 reload 5 supplemental licensing submittal.

Page 168a - Reference to the MAPLHGR tables is being corrected to include tables 3.5.I-5 and -6.

ENCLOSURE 2

PART II - PLANT MODIFICATION RELOAD CHANGES

A. Changes Related to Torus Modifications

Numerous modifications are being implemented in the unit 1 torus during the reload 5 refueling outage as part of the Mark I Containment Program. These modifications are required by NRC to restore the originally intended margins of safety in the containment design, and work will be considered complete following this outage.

Pages 79 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.

Page 145 - The two-pump 15,000-gpm LPCI test surveillance 4.5.B.1 was determined to induce vibrations in the RHR return line to the torus. To eliminate the vibration, an orifice is to be installed in the return line; however, installation of this orifice plate also decreases the suppression pool cooling mode of RHR operation from 15,000 gpm to approximately 12,000 gpm. A new containment cooling analysis was performed for this configuration, and it was determined that this flow rate produces a long-term suppression pool temperature well within that necessary for stable and complete steam condensation and for adequate RHR and core spray pumps net positive suction head.

Pages 233 and 234 - Since the torus is being extensively upgraded to withstand dynamic loading significantly beyond that originally expected, extended operation of relief valves above a suppression pool tempature of 130°F is not expected to be a safety concern warranting placing the reactor in cold shutdown and performing a torus inspection. This requirement is therefore unnecessary and deletion is proposed. This requirement originated with RO bulletin 74-14. This has previously been approved for unit 2 in amendment No. 85.

Pages 267 and 268 - The 3.7.A and 4.7.A bases for the suppression pool temperature limits were founded on the Humboldt Bay and Bodega Bay tests. Consistent with the long-term torus integrity program of NUREG-0661 and NUREG-0783, the bases require change to account for steam mass fluxes through the S/RV T-quenchers. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate RHR and core spray pump net positive suction head.

Page 273 - The specific references to drywell and suppression chamber coatings are being deleted. There is some variation between the Browns Ferry units in the type and application of the coating, particularly due to the Mark I Modification Program; therefore, the technical specification bases are being generalized so that technical specification changes will not be required each cycle. Control of the torus coating will be maintained by internal TVA coating programs.

B. Miscellaneous Plant Modifications and Administrative Technical Specification Changes

Thermal Power Monitor

Page 8 - Add "Flow Biased" to title of section 2.1.A.1.

Page 10 - Add section 2.1.A.1.d.

age 20 - Reword basis 2.1.A.1 to reflect features of the thermal power monitor.

Page 22 - Add basis 2.1.A.4 to describe high neutron flux scram fixed trip.

Page 33 - Change table 3.1.A to reflect addition of the fixed trip function.

Page 36a - Add notes 24 and 25.

Page 37 - Change table 4.1.A to reflect addition of fixed trip.

Justification

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

Safety Analysis

The APRM flow-biased scram will occur when the fuel surface heat flux resulting from a neutron flux transient reaches a point equivalent to the thermal power trip setpoint. This is done by passing the neutron flux signal through a filter network with a time constant shorter than that representative of the fuel thermal time constant. There is a separate trip unit which initiates a scram at less than 120-percent

instantaneous neutron flux. This scram function is the basis for transient and accident analysis, and no credit is taken for the flow-biased scram function. Any flow-biased scram function therefore provides additional margin from fuel damage beyond that of the transient analysis. This has previously been approved for unit 2 in amendment No. 85.

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 nonclass 1E power supplies and the RPS. This will ensure that failure of a nonclass 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. B. Hughes dated August 7, 1978.

Scram Discharge Instrument Volume

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

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

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

Implementation of the SDV and SDIV modifications adding the redundant vent and drain valves provides increased assurance that the second purpose will be fulfilled while decreasing the probability of

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

Pages 33, 73, and 102 of tables 3.1.A, 3.2.C, and 4.2.C - Revised to reflect an east and west scram discharge instrument volume.

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.

Page 126 - Sections 3.3.F and 4.3.F are being revised to reflect the addition of the redundant drain and vent valve.

Analog Trip System

Pages 37, 38, 40, 85, and 96 of tables 4.1.A, 4.1.B, and 4.2.B - Revised to reflect the modifications associated with the addition of an analog trip system. The associated functional test methods and calibration methods are being revised accordingly and notes have been added to the appropriate instrumentation to relect this. The calibration frequency is being extended to once/operating cycle due to the high reliability of the analog trip system. Note 5 was also removed from page 37.

Pages 39, 41, and add page 110a - The notes describing the functional test and calibration methods for the analog trip system were added. Note 5 on page 39 was deleted since it is no longer being referenced.*

NUREG-0737 Items

Pages 79, 105a, and 110 of tables 3.2.F and 4.2.F - Revised to include the surveillance instrumentation associated with the following accident instrumentation: (1) containment high-range radiation monitors, (2) drywell pressure-wide range, and (3) suppression chamber wide-range water level. These three items are in response to NUREG-0737, items II.F.1.3, II.F.1.4, and II.F.1.5, respectively.

Pages 78 and 105 - Delete the drywell pressure and suppression chamber water level instruments. They are being replaced by items 2 and 3 above.

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*The functional test discussed in Note 7 replaces the water level perturbation previously discussed in Note 5.

10 CFR 50, Appendix J Testing

Pages 231 and 232 - Revised to reflect the installation of strongbacks on the personnel airlock doors which will allow testing in accordance with 10 CFR 50, appendix J.

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. This line was added to provide the capability for isolation of approximately one-half of the drywell supplied equipment in the case of a drywell line leak. This air supply will be used to supply two inboard MSIV's, approximately one-half of the main steam relief valves, and approximately one-half of all other air-operated equipment in the drywell. This will significantly reduce the possibility of any one control air pipe break inside containment from requiring immediate shutdown and isolation due to MSIVs, MSRVs, and drywell coolers being inoperable. This will, in turn, significantly increase the margin of safety.

Administrative Changes

Pages ii, iii, and v - Technical specification titles for sections 3.5/4.5.A, 3.5/4.5.J, 3.5/4.5.L, 3.6/4.6.D, 3.6/4.6.H, 3.7/4.7.H, 6.9, 6.10, and 6.11 were modified to correctly reflect the respective technical specifications.

Pages ii, iv, and v - Technical specification titles for sections 3.5/4.5.M, 3.10/4.10.C, 3.11/4.11.A, and 6.2 were corrected to reflect the actual page numbers.

Pages iv, 321, 322, and 323 - Technical specifications 3.11/4.11.F through 3.11/4.11.I were given titles to be consistent with the present format. The table of contents were also corrected to reflect this change.

Page vi - This page number for table 4.2.B was corrected to reflect the actual page number.

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

Pages 66 and 98 - An editorial change was made to more accurately indicate that HPCI suction switchover is made on condensate header level rather than condensate tank level.

Page 99 - Surveillance requirements due to addition of RCIC steam flow isolation time delay have been added. Surveillance on HPCI time delay relay required by NUREG-0737, item II.K.3.15 is also added.

Page 169 - Remove reference to fuel types to make this more consistent with unit 2. It in no way changes the content of the bases.

Page 181 - References to safety valves were removed from 3.6.D.1 and 4.6.D.1. The safety valves were previously removed from unit 1.

Page 251a - The Hays-Republic $H^2$ - $O^2$ monitoring system was installed on unit 1 during the reload 4 outage. The primary containment isolation valves associated with this system were installed with both valves on each line being outside of containment. Table 3.7.A is being revised to indicate the correct position of the valves. This revision is clerical in nature only.

ENCLOSURE 3

BROWNS FERRY NUCLEAR PLANT

SIGNIFICANT HAZARDS CONSIDERATION

FOR

PROPOSED TECHNICAL SPECIFICATION CHANGES - TS 190

PART A - CORE RELOAD CHANGES

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The "Supplemental Reload Licensing Submittal for Browns Ferry Unit 1, Reload 5" shows that there is no increase in the probability or consequence of an accident.

2. Does the proposed amendment create the proability of a new or different kind of accident from any accident previously evaluated?

   No. The proposed amendments to unit 1 technical specifications are in support of reload 5. It does not create the probability of any new or different kind of accident.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   As stated in 2 above, the proposed revisions are in support of reload 5 on unit 1. These revisions are needed to maintain the margin of safety during cycle 6, and no reduction to this margin will result.

ENCLOSURE 3

PART B - SIGNIFICANT HAZARDS CONSIDERATION

SIGNIFICANT HAZARDS CONSIDERATIONS FOR BROWNS FERRY NUCLEAR PLANT UNIT 1
RELOAD 5 MODIFICATION-RELATED TECHNICAL SPECIFICATIONS

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   A. Changes Related to Torus Modifications

      No. The changes described in section A of attachment 1 were made in response to an NRC requirement to restore the originally intended margins of safety in the containment design. These modifications therefore will decrease the probability of an accident and increase the margin of safety.

      An improved torus temperature monitoring system was added. This will also increase the margin of safety.

      Most of the associated revisions in this section have previously been approved for unit 2 in amendment No. 85.

   B. Miscellaneous Plant Modifications and Administrative Technical Specification Changes

      <u>Thermal Power Monitor</u> - No. The addition of the thermal power monitor will prevent a flow-biased neutron flux scram when a transient-induced neutron flux spike occurs that is a short-time duration and does not result in an instantaneous heat flux in excess of transient limits. This feature will reduce the number of scrams due to small fast flux transients such as those which result from control valve and MSIV testing and small perturbations in water level and pressure. No credit is taken in the transient and accident analysis for the flow-biased scram function; therefore, any flow-biased scram function provides additional margin from fuel damage. As a result, this modification will decrease the probability of an accident and increase the margin of safety.

      <u>Reactor Protection System (RPS) Modification</u> - No. The modifications on the RPS will provide a fully redundant class 1E protection at the interface of the nonclass 1E power supplies and the RPS. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E reactor protection system thus increasing the margin of safety and decreasing the probability of an accident.

Scram Discharge Instrument Volume - No. As stated in attachment 1, 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. These modifications will decrease the probability of this event occurring thus increasing the margin of safety.

Analog Trip System - No. The analog trip system adds to the margin of safety of the plant due to its high reliability. It therefore does not involve a significant increase in the probability or consequences of an accident.

NUREG-0737 Items - No. This modification adds containment high-range radiation monitors, drywell pressure wide-range monitors, and suppression chamber wide-range water level instruments. These have been added to increase the safety of the plant in response to NUREG-0737, items II.F.1.3, II.F.1.4, and II.F.1.5, respectively. The modifications do not increase the probability or consequences of an accident or reduce the margin of safety.

10 CFR 50, Appendix J Testing - No. This modification is being made to facilitate testing the drywell airlock doors in accordance with 10 CFR 50, appendix J. The testing will not involve a reduction in the margin of safety or increase the probability of an accident.

Redundant Air Supply to Drywell - No. This modification increases the margin of safety and reduces the probability of an accident by providing a redundant air supply to the drywell. The modification 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.

Administrative Changes - No. These revisions are administrative in nature only and do not affect the probability of an accident or the margin of safety.

2. Does the proposed amendment create the probability of a new or different kind of accident from any accident previously evaluated?

   No. See (1).

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   No. See (1).