ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION REVISIONS
(TVA BFNP TS 179)
BROWNS FERRY NUCLEAR PLANT UNIT 2

| Section | | Page No. |
|---|---|---|

LIST OF TABLES (Cont'd)

## LIST OF ILLUSTRATIONS

## 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

## 2.1 FUEL CLADDING INTEGRITY

b. In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

c. For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR ≤13.4 kw/ft for 8x8, 8x8R, and P8x8R, and MCPR within limits of Specification 3.5.k. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

d. The APRM Rod block trip setting shall be:

$$S_{RB} \leq (0.66W + 42\%)$$

where:

$S_{RB}$ = Rod block 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

In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$$S_{RB} \leq (0.66W + 42\%)\frac{FRP}{CMFLPD}$$

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

.1.1 BASES

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.07 would not produce boiling transition. Thus, although it is not required to establish the safety limit additional margin exists between the safety limit and the actual occurence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (MCPR = 1.07) operation is constrained to a maximum LHGR of 13.4 kw/ft for 8x8, 8x8R, and P8x8R. This limit is reached when the Core Maximum Fraction of Limiting Power Density equals 1.0 (CMFLPD = 1.0). For the case where Core Maximum Fraction of Limiting Power Density exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated power and only with reduced APRM scram settings as required by specification 2.1.A.1.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flow will always be greater than 4.56 psi. Analyses show that with a flow of $28X10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

For the fuel in the core during periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

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

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

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in References 1, 2, and 3.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the normal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications as further described in Reference 4. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity has been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR $>$ limits specified in specification 3.5.k is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

## 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

2.1 BASES

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

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

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

2. Generic Reload Fuel Application, Licensing Topical Report NEDE-20411-P-A, and Addenda.

3. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-25154, NEDE-24154-P, October 1978.

4. Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980

## 1.2 REACTOR COOLANT SYSTEM INTEGRITY

**Applicability**

Applies to limits on reactor coolant system pressure

**Objective**

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

**Specification**

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.



## 2.2 REACTOR COOLANT SYSTEM 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 pressure safety limit from being exceeded.

**Specification**

The limiting safety system settings shall be as specified below:

| Protective Action | Limiting Safety System Setting |
|---|---|
| A. Nuclear system relief valves open--nuclear system pressure | 1105 psig ± 11 psi (4 valves) |
| | 1115 psig ± 11 psi (4 valves) |
| | 1125 psig ± 11 psi (5 valves) |
| B. Scram--nuclear system high pressure | ≤ 1,055 psig |

## 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. 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.

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

| Min. No. of Operable Inst. Channels Per Trip System(1)(23) | Trip Function | Trip Level Setting | Shut-down | Modes in Which Function Must Be Operable<br>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 | ≤50 Gallons | X | X(2) | X | X | 1.A |
| 2 | High Water Level in East Scram Discharge Tank | ≤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 Refuelin and Before Each Startup |
| Inoperative | C | Trip Channel and Alarm (4) | Once Per Week During Refuelin and Before Each Startup |
| APRM | | | |
| High Flux (15% scram) | C | Trip Output Relays (4) | Before Each Startup and Weekl When Required to be Operable |
| High Flux (Flow Biased) | B | Trip Output Relays (4) | Once/Week |
| High Flux (Fixed Trip) | B | Trip Output Relays (4) | Once/Week |
| Inoperative | B | Trip Output Relays (4) | Once/Week |
| Downscale | B | Trip Output Relays (4) | Once/Week |
| Flow Bias | B | (6) | (6) |
| High Reactor Pressure | A | Trip Channel and Alarm | Once/Month (1) |
| High Drywell Pressure | A | Trip Channel and Alarm | Once/Month (1) |
| Reactor Low Water Level (5) | A | Trip Channel and Alarm | Once/Month (1) |
| High Water Level in Scram Discharge Tank | | | |
| Float Switches | A | Trip Channel and Alarm | Once/month |
| Differential Pressure Switches | B | Trip Channel and Alarm | Once/month |
| Turbine Condenser Low Vacuum | A | Trip Channel and Alarm | Once/month (1) |
| Main Steam Line High Radiation | B | Trip Channel and Alarm | Once/week |

NOTES FOR TABLE 4.1.A

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

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

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

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

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

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

7. Calibration of master/slave trip units only.

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

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

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 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 | (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 |

does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of $10^{-8}$ of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high per level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (i.e., MCPR given by Specification 3.5.k or LHGR of 13.4 kw/ft. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

## Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage: i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant startup and was determined to be caused by

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. The CSS shall be operable:

(1) prior to reactor startup from a cold condition, or

(2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in specification 3.5.A.2.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and containment cooling systems when the corresponding limiting condition for operation is in effect.

Objective

To verify the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing.

| | Item | Frequency |
|---|---|---|
| a. | Simulated Automatic Actuation test | Once/ Operating Cycle |
| b. | Pump Operability | Once/ month |
| c. | Motor Operated Valve Operability | Once/ month |
| d. | System flow rate: Each loop shall deliver at least 6250 gpm against a system head corresponding to a | Once/3 months |

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 RHR may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of 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

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. | Testable check valve | Once/ operating cycle |

Each LPCI pump shall deliver 9,000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 13,000 gpm against an indicated system pressure of 235 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.

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 seven 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.

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.

3.5.H Maintenance of Filled Discharge Pipe

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

| | |
|---|---|
| Pl-75-20 | 48 psig |
| Pl-75-48 | 48 psig |
| Pl-74-51 | 48 psig |
| Pl-74-65 | 48 psig |

I. Average Planar Linear Heat Generation Rate

During steady state power operation, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1, -2, -3, -4. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Linear Heat Generation Rate (LHGR)

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

4.5.H Maintenance of Filled Discharge Pipe

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

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

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

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

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

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

J. Linear Heat Generation Rate (LHGR)

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

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

3.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) as a function of scram time and core flow, shall be equal to or greater than shown in Figure 3.5.K-1 multiplied by the $K_f$ shown in Figure 3.5.2, where:

$\tau = 0$ or $\frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}$, whichever is greater

$\tau_A$=0.90 sec (Specification 3.3.C.1 scram time limit to 20% insertion from full withdrawn)

$\tau_B = 0.710+1.65 \left[\frac{N}{n}\right]^{1/2} (0.053)$ [Ref 5]

$\tau_{ave} = \frac{\sum_{i=1}^{n} \tau_i}{n}$

n = number of surveillance rod tests performed to date in cycle (including BOC test).

$\tau_i$ = scram time to 20% insertion from fully withdrawn of the $i^{th}$ rod

N = total number of active rods measured in Specification 4.3.C.1 at BOC

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

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.

2. The MCPR limit shall be determined for each fuel type 8X8, 8X8R, P8X8R, from Figure 3.5.K-1 respectively using:

   a. $\tau$ = 0.0 prior to initial scram time measurements for the cycle performed in accordance with Specification 4.3.C.1.

   b. $\tau$ as defined in Specification 3.5.K following the conclusion of each scram time surveillance test required by Specification 4.3.C.1 and 4.3.C.2.

   The determination of the limit must be completed with 72 hours of each scram time surveillance required by Specification 4.3.C.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

L. Reporting Requirements

If any of the limiting values identified in Specifications 3.5.I, J, or K are exceeded and the specified remedial action is taken, the event shall be logged and reported in a 30-day written report.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

## 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.

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. 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 $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimimum Critical Power Ratio (MCPR)

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

3.5.L. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values for MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specifications 3.5.I., .J., and .K., that if at any time during steady state power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates than an operating limit has been reached. Each event involving steady state operation beyond a specified limit shall be reported within 30 days. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative.

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

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

5. Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC request for information on ODYN computer model," September 5, 1980.

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274L

| Average Planar Exposure (Mwd/t) | MAPLHGR (kW/ft) |
|---|---|
| 200 | 11.2 |
| 1,000 | 11.3 |
| 5,000 | 11.9 |
| 10,000 | 12.1 |
| 15,000 | 12.2 |
| 20,000 | 12.1 |
| 25,000 | 11.6 |
| 30,000 | 10.9 |
| 35,000 | 9.9 |
| 40,000 | 9.3 |

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274H

| Average Planar Exposure (Mwd/t) | MAPLHGR (kW/ft) |
|---|---|
| 200 | 11.1 |
| 1,000 | 11.2 |
| 5,000 | 11.8 |
| 10,000 | 12.1 |
| 15,000 | 12.2 |
| 20,000 | 12.0 |
| 25,000 | 11.5 |
| 30,000 | 10.9 |
| 35,000 | 10.0 |
| 40,000 | 9.3 |

TABLE 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Types: 8DRB284L and P8DRB284L

| Average Planar Exposure (MWd/t) | MAPLHGR (kW/ft) |
|---|---|
| 200 | 11.2 |
| 1,000 | 11.3 |
| 5,000 | 11.8 |
| 10,000 | 12.0 |
| 15,000 | 12.0 |
| 20,000 | 11.8 |
| 25,000 | 11.2 |
| 30,000 | 10.8 |
| 35,000 | 10.0 |
| 40,000 | 9.4 |

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Types: P8DRB265H

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

4.6.E Jet Pumps

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

2. Whenever there is recirculation flow with the reactor in the Startup or Run Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3. Steady state operation with both recirculation pumps out of service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

F. Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.

2. No additional surveillance required.

3. Before starting either recirculation pump during steady state operation, check and log the loop discharge temperature and dome saturation temperature.

G. Structural Integrity

1. The structural integrity of the primary system shall be

G. Structural Integrity

1. Table 4.6.A together with supplementary notes, specifies the

[illegible]

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.

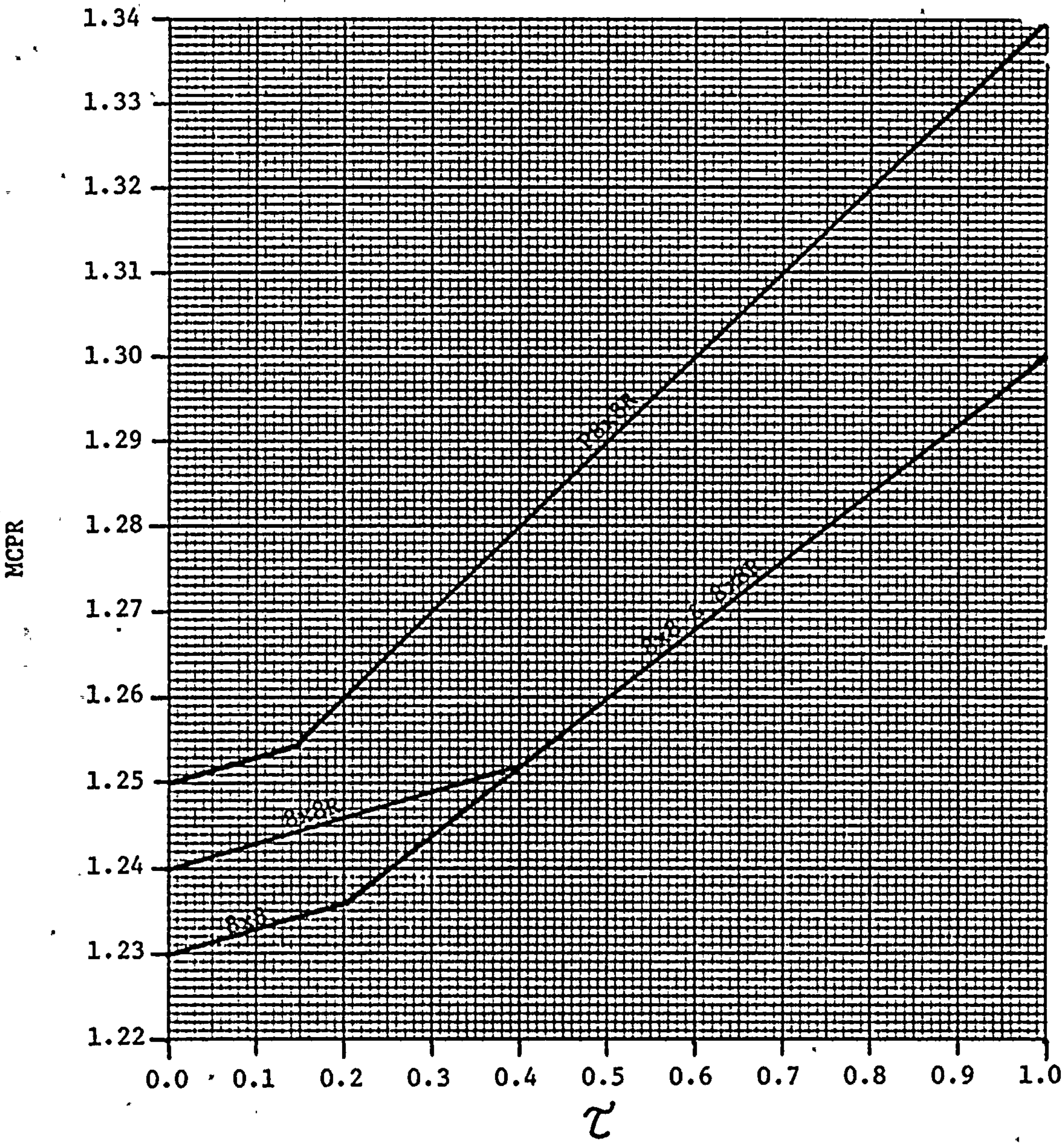


Figure 3.5.K-1
MCPR Limits

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.

REFERENCE

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. 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.

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

## 3.6.F/4.6.F Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgement from actual plant obsrevation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in there additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

## 3.7 CONTAINMENT SYSTEMS

### Applicability

Applies to the operating status of the primary and secondary containment systems.

### Objective

To assure the integrity of the primary and secondary containment systems.

### Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits except as specified in 3.7.A.2.

   a. Minimum water level = -6.25" (Differential pressure control >0 psid)

      -7.25" (0 PSID Differential pressure control)

   b. Maximum water level = -1"

## 4.7 CONTAINMENT SYSTEMS

### Applicability

Applies to the primary and secondary containment integrity.

### Objective

To verify the integrity of the primary and secondary containment.

### Specification

A. Primary Containment

1. Pressure Suppression Chamber

   a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

[illegible]

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.

3.7.A Primary Containment

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.

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.

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.

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.

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

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. 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

a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every month.

b. When it is determined that two vacuum breakers are inoperable for opening at a time when operability is requir all other vacuum breaker

## 3.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

      (1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

      (2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

   b. If the differential pressure of specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

## 4.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

   a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

[illegible]

TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

| Group | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec.) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 1 | Main steamline isolation valves (FCV-1-14, 26, 37, & 51; 1-15, 27, 38 & 52) | 4 | 4 | 3 < T < 5 | 0 | GC |
| 1 | Main steamline drain isolation valves (FCV-1-55 & 1-56) | 1 | 1 | 15 | 0 | GC |
| 1* | Reactor Water sample line isolation valves | 1 | 1 | 5 | C | SC |
| 2 | RHRS shutdown cooling supply isolation valves (FCV-74-48 & 47) | 1 | 1 | 40 | C | SC |
| 2 | RHRS - LPCI to reactor (FCV-74-53 & 67) | | 2 | 30 | C | SC |
| 2 | Reactor vessel head spray isolation valves (FCV-74-77 & 78) | 1 | 1 | 30 | C | SC |
| 2 | RHRS flush and drain vent to suppression chamber (FCV-74-102, 103, 119, & 120) | | 4 | 20 | C | SC |
| 2 | Suppression Chamber Drain (FCV-75-57 & 58) | | 2 | 15 | C | SC |
| 2 | Drywell equipment drain discharge isolation valves (FCV-77-15A & 15B) | | 2 | 15 | 0 | GC |
| 2 | Drywell floor drain discharge isolation valves (FCV-77-2A & 2B) | | 2 | 15 | 0 | GC |

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

TABLE 3.7.A (Continued)

| Group | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec.) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 3 | Reactor water cleanup system supply isolation valves FCV-69-1, & 2 | 1 | 1 | 30 | 0 | GC |
| 4 | FCV 73-81 (Bypass around FCV 73-3) | | 1 | 10 | 0 | GC |
| 4 | HPCIS steamline isolation valves FCV-73-2 & 3 | 1 | 1 | 20 | 0 | GC |
| 5 | RCICS steamline isolation valves FCV-71-2 & 3 | 1 | 1 | 15 | 0 | GC |
| 6 | Drywell nitrogen purge inlet isolation valves (FCV-76-18) | | 1 | 5 | C | SC |
| 6 | Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19) | | 1 | 5 | C | SC |
| 6 | Drywell Main Exhaust isolation valves (FCV-64-29 and 30) | | 2 | 2.5 | C | SC |
| 6 | Suppression chamber main exhaust isolation valves (FCV-64-32 and 33) | | 2 | 2.5 | C | SC |
| 6 | Drywell/Suppression Chamber purge inlet (FCV-64-17) | | 1 | 2.5 | C | SC |
| 6 | Drywell Atmosphere purge inlet (FCV-64-18) | | 1 | 2.5 | C | SC |

TABLE 3.7.A (Continued)

| Group | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec.) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 6 | Suppression Chamber purge inlet (FCV-64-19) | | 1 | 2.5 | C | SC |
| 6 | Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17) | | 1 | 5 | C | SC |
| 6 | Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31) | | 1 | 5 | O | GC |
| 6 | Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34) | | 1 | 5 | O | GC |
| 6 | System Suction Isolation Valves to Air Compressors "A" and "B" (FCV-32-62, 63) | | 2 | 15 | O | GC |
| 7 | RCIC Steamline Drain (FCV-71-6A, 6B) | | 2 | 5 | O | GC |
| 7 | RCIC Condensate Pump Drain (FCV-71-7A, 7B) | | 2 | 5 | C | SC |
| 7 | HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B) | | 2 | 5 | C | SC |
| 7 | HPCI steamline drain (FCV-73-6A, 6B) | | 2 | 5 | O | GC |
| 8 | TIP Guide Tubes (5) | | 1 per guide tube | NA | C | GC |

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 |
|---|---|---|---|---|---|---|
| | Standby liquid control system check valves (CV 63-526 & 525) | 1 | 1 | NA | C | Process |
| | Feedwater check valves (CV-3-558, 572, 554 & 568) | 2 | 2 | NA | 0 | Process |
| | Control rod hydraulic return check valves (CV-85-576 & 573) | 1 | 1 | NA | 0 | Process |
| | RHRS - LPCI to reactor check valves (CV-74-54 & 68) | 2 | | NA | C | Process |
| 6 | CAD System Torus/Drywell Exhaust to Standby Gas Treatment (FCV-84-19,20) | | 2 | 10 | C | SC |
| 6 | Drywell/Suppression Chamber Nitrogen Purge Inlet (FCV-76-24) | | 1 | 5 | C | SC |
| | Core Spray Discharge to Reactor Check Valves FCV-75-26,54 | 2 | | NA | C | Process |

TABLE 3.7.A (Continued)

| Group | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec.) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 6 | Drywell $\Delta$P air compressor suction valve (FCV-64-139) | | 1 | 10 | 0* | GC |
| 6 | Drywell $\Delta$P air compressor discharge valve (FCV-64-140) | | 1 | 10 | 0 | GC |
| 6 | Drywell CAM suction valves (FCV-90-254A and 254B) | | 2 | 10 | 0 | GC |
| 6 | Drywell CAM discharge valves (FCV-90-257A and 257B) | | 2 | 10 | 0 | GC |
| 6 | Drywell CAM suction valve (FCV-90-255) | | 1 | 10 | 0 | GC |

*This valve cycles open and closed during normal operation.

TABLE 3.7.B
TESTABLE PENETRATIONS WITH DOUBLE O-RING SEALS

| | |
|---|---|
| X-1A | Equipment Hatch |
| X-1B | " " |
| X-4 | DW Head Access Hatch |
| X-6 | CRD Removal Hatch |
| X-35A | T.I.P. Drives |
| X-35B | " " |
| X-35C | " " |
| X-35D | " " |
| X-35E | " " |
| X-35F | " " |
| X-35G | " " |
| X-47 | Power Operations Test |
| X-200A | Supp. Chamber Access Hatch |
| X-200B | " " " " |
| X-213A | Suppression Chamber Drain |
| X-223 | Supp. Chamber Access Hatch |
| | DW Flange-Top Head |
| | Shear Lug Inspection Cover #1 |
| | " " " Hatch #2 |
| | " " " " #3 |
| | " " " " #4 |
| | " " " " #5 |
| | " " " " #6 |
| | " " " " #7 |
| | " " " " #8 |

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 |
| 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 |

| Valve | Valve Identification |
|---|---|
| 69-1 | RWCU Supply |
| 69-2 | RWCU Supply |
| 69-579 | RWCU Return |
| 71-2 | RCIC Steam Supply |
| 71-3 | RCIC Steam Supply |
| 71-39 | RCIC Pump Discharge |
| 71-40 | RCIC Pump Discharge |
| 73-2 | RCIC Steam Supply |
| 73-3 | RCIC Steam Supply |
| 73-44 | HPCI Pump Discharge |
| 73-45 | HPCI Pump Discharge |
| 73-81 | HPCI Steam Supply Bypass |
| 74-47 | RHR Shutdown Suction |
| 74-48 | RHR Shutdown Suction |
| 74-661 | RHR Shutdown Suction |
| 74-662 | RHR Shutdown Suction |
| 76-17 | Drywell/Suppression Chamber Nitrogen Purge |
| 76-18 | Drywell Nitrogen Purge Inlet |
| 76-19 | Suppression Chamber Purge Inlet |
| 76-24 | Drywell/Suppression Chamber Nitrogen Purge |
| 76-49 | Containment Atmospheric Monitor |
| 76-50 | Containment Atmospheric Monitor |
| 76-51 | Containment Atmospheric Monitor |
| 76-52 | Containment Atmospheric Monitor |
| 76-53 | Containment Atmospheric Monitor |
| 76-54 | Containment Atmospheric Monitor |
| 76-55 | Containment Atmospheric Monitor |
| 76-56 | Containment Atmospheric Monitor |
| 76-57 | Containment Atmospheric Monitor |
| 76-58 | Containment Atmospheric Monitor |
| 76-59 | Containment Atmospheric Monitor |
| 76-60 | Containment Atmospheric Monitor |
| 76-61 | Containment Atmospheric Monitor |
| 76-62 | Containment Atmospheric Monitor |
| 76-63 | Containment Atmospheric Monitor |
| 76-64 | Containment Atmospheric Monitor |
| 76-65 | Containment Atmospheric Monitor |
| 76-66 | Containment Atmospheric Monitor |
| 76-67 | Containment Atmospheric Monitor |
| 76-68 | Containment Atmospheric Monitor |
| 77-2A | Drywell Floordrain Sump |
| 77-2B | Drywell Floordrain Sump |
| 77-15A | Drywell Equipment Drain Sump |
| 77-15B | Drywell Equipment Drain Sump |
| 84-8A | Containment Atmospheric Dilution |
| 84-8B | Containment Atmospheric Dilution |
| 84-8C | Containment Atmospheric Dilution |
| 84-8D | Containment Atmospheric Dilution |
| 84-19 | Containment Atmospheric Dilution |
| 84-20 | Main Exhaust to Standby Gas Treatment |
| 84-600 | Main Exhaust to Standby Gas Treatment |
| 84-601 | Main Exhaust to Standby Gas Treatment |
| 84-602 | Main Exhaust to Standby Gas Treatment |
| 84-603 | Main Exhaust to Standby Gas Treatment |
| 85-576 | CRD Hydraulic Return |
| 90-254A | Radiation Monitor Suction |

TABLE 3.7.D (Continued)

| Valve | Valve Identification |
|---|---|
| 90-254B | Radiation Monitor Discharge |
| 90-255 | Radiation Monitor Discharge |
| 90-257A | Radiation Monitor Discharge |
| 90-257B | Radiation Monitor Discharge |

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TABLE 3.7.E

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

| Valve | Valve Identification |
|---|---|
| 12-738 | Auxiliary Boiler to RCIC |
| 12-741 | Auxiliary Boiler to RCIC |
| 43-28A | RHR Suppression Chamber Sample Lines |
| 43-28B | RHR Suppression Chamber Sample Lines |
| 43-29A | RHR Suppression Chamber Sample Lines |
| 43-29B | RHR Suppression Chamber Sample Lines |
| 2-1143 | Demineralized Water |
| 71-14 | RCIC Turbine Exhaust |
| 71-32 | RCIC Vacuum Pump Discharge |
| 71-580 | RCIC Turbine Exhaust |
| 71-592 | RCIC Vacuum Pump Discharge |
| 73-23 | HPCI Turbine Exhaust |
| 73-24 | HPCI Turbine Exhaust Drain |
| 73-603 | HPCI Turbine Exhaust |
| 73-609 | HPCI Exhaust Drain |
| 74-722 | RHR |
| 75-57 | Core Spray to Auxiliary Boiler |
| 75-58 | Core Spray to Auxiliary Boiler |
| | Core Spray to Auxiliary Boiler |

TABLE 3.7.F

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

| Valve | Valve Identification |
|---|---|
| 74-53 | RHR LPCI Discharge |
| 74-54 | RHR |
| 74-57 | RHR Suppression Chamber Spray |
| 74-58 | RHR Suppression Chamber Spray |
| 74-60 | RHR Drywell Spray |
| 74-61 | RHR Drywell Spray |
| 74-67 | RHR LPCI Discharge |
| 74-68 | RHR LPCI Discharge |
| 74-71 | RHR Suppression Chamber Spray |
| 74-72 | RHR Suppression Chamber Spray |
| 74-74 | RHR Drywell Spray |
| 74-75 | RHR Drywell Spray |
| 74-77 | RHR Head Spray |
| 74-78 | RHR Head Spray |
| 75-25 | Core Spray Discharge |
| 75-26 | Core Spray Discharge |
| 75-53 | Core Spray Discharge |
| 75-54 | Core Spray Discharge |

TABLE 3.7.G

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TABLE 3.7.H (Continued)

| | |
|---|---|
| X-107B | Spare (testable) |
| X-108A | Power |
| X-108B | CRD Rod Position Indic. |
| X-109 | " " " " |
| X-110A | Power |
| X-110B | CRD Rod Position Indic. |
| X-230 | Containment Air Monitoring System |
| X-200A-SC | S/RV Test Instrumention (Temporary) |

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, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement 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 at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep offsite doses well below 10 CFR 100 limits.

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 released 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 from 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 indication of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 $FT^3$ with or 128,700 $FT^3$ 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 a water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will assure 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.

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

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The <4% hydrogen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant accident.

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

3.9 AUXILIARY ELECTRICAL SYSTEM

b. The units 1 and 2 4-kV shutdown boards are energized.

c. The 480-V shutdown boards associated with the unit are energized.

d. The units 1 and 2 diesel auxiliary boards are energized.

e. Loss of voltage and degraded voltage relays operable on 4-kV shutdown boards A, B, C, and D.

f. Shutdown busses 1 and 2 energized.

g. The 480V Rx. MOV Boards D & E are energized with M-G sets 2DN, 2DA, 2EN, and 2EA in service.

5. The 250-volt unit and shutdown board batteries and a battery charger for each battery boards are operable.

6. Logic Systems

a. Common accident signal logic system is operable.

b. 480-V load shedding logic system is operable.

7. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.

4.9 AUXILIARY ELECTRICAL SYSTEM

with instructions based on the manufacturer's recommendations.

e. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged.

2. D. C. Power System - Unit Batteries (250-Volt) Diesel Generator Batteries (125-Volt) and Shutdown Board Batteries (250-Volt)

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

c. A battery rated discharge (capacity) test shall be performed and the voltage, time, and output current measurements shall be logged at intervals not to exceed 24 months.

3.9 AUXILIARY ELECTRICAL SYSTEM

shutdown boards and undervoltage relays are operable. (Within the surveillance schedule of 4.9.A.4.b).

12. When one 480-volt shutdown board is found to be inoperable, the reactor will be placed in hot standby within 12 hours and cold shutdown within 24 hours.

13. If one 480-V RMOV board M-G set is inoperable, the reactor may remain in operation for a period not to exceed seven days, provided the remaining 480-V RMOV board m-g sets and their associated loads remain operable.

14. If any two 480-V RMOV board M-G sets become inoperable, the reactor shall be placed in the cold shutdown condition within 24 hours.

15. If the requirements for operating in the conditions specified by 3.9.B.1 through 3.9.B.14 cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown and in the cold condition within 24 hours.

4.9 AUXILIARY ELECTRICAL SYSTEM

3.9 AUXILIARY ELECTRICAL SYSTEM

C. Operation in Cold Shutdown

Whenever the reactor is in cold shutdown condition with irradiated fuel in the reactor, the availability of electric power shall be as specified in Section 3.9.A except as specified herein.

1. At least two units 1 and 2 diesel generators and their associated 4-kV shutdown boards shall be operable.

2. An additional source of power consisting of at least one of the following:

   a. The unit 1 or 2 unit station service transformers energized.

   b. One 161-kV transmission line and its associated common station service transformer energized.

   c. Either 161-kV line, one cooling tower transformer and the bus tie board energized and capable of supplying power to the units 1 and 2 shutdown boards energized.

   d. A third operable diesel generator.

3. At least one 480-V shutdown board for each unit must be operable.

4. One 480-V RMOV board motor-generator (M-G) set is required for each RMOV board (D or E) required to support operation of the RHR system in accordance with 3.5.B.9.

4.9 AUXILIARY ELECTRICAL SYSTEM

control functions, operative power for unit motor loads, and alternative drive power for a 115-volt a-c unit preferred motor-generator set. One 250-volt d-c system provides power for common plant and transmission system control functions, drive power for a 115-volt a-c plant preferred motor-generator set, and emergency drive power for certain unit large motor loads. The four remaining systems deliver control power to the 4160-volt shutdown boards.

Each 250-Volt d-c shutdown board control power supply can receive power from its own battery, battery charger, or from a spare charger. The chargers are powered from normal plant auxiliary power or from the standby diesel-driven generator system. Zero resistance short circuits between the control power supply and the shutdown board are cleared by fuses located in the respective control power supply. Each power supply is located in the reactor building near the shutdown board it supplies. Each battery is located in its own independently ventilated battery room.

The 250-volt d-c system is so arranged, and the batteries sized such, that the loss of any one unit battery will not prevent the safe shutdown and cooldown of all three units in the event of the loss of offsite power and a design basis accident in any one unit. Loss of control power to any engineered safeguards control circuit is annunciated in the main control room of the unit affected. The loss of one 250-Volt shutdown board battery affects normal control power only for the 4160-Volt shutdown board which is supplies. The station battery supplies loads that are not essential for safe shutdown and cooldown of the nuclear system. This battery was not considered in the accident load calculations.

There are two 480-V ac Reactor Motor-Operated Valve (RMOV) Boards that contain motor-generator (M-G) sets in their feeder lines. These 480-V ac RMOV boards have an automatic transfer from their normal to alternate power source (480-V ac shutdown boards). The M-G sets act as electrical isolators to prevent a fault from propagating between electrical divisions due to an automatic transfer. The 480-V ac RMOV boards involved provide motive power to valves associated with the LPCI mode of the RHR system. Having an M-G set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required. Since sufficient equipment is available to maintain the minimum complement required for RHR (LPCI) operation, a 7-day servicing period is justified. Having two M-G sets out of service can considerably reduce equipment availability. Therefore, the affected unit shall be placed in cold shutdown within 24 hours.

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.

## 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

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 2 is located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The reactor core may contain 764 fuel assemblies consisting of 8x8 assemblies having 63 fuel rods each, and 8x8R and P8x8R assemblies having 62 fuel rods each.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder ($B_4C$) compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of fuel in the new-fuel storage facility shall be such that $k_{eff}$, for dry conditions, is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

ENCLOSURE 2
DESCRIPTION AND JUSTIFICATION
(TVA BFNP TS 179)

Enclosure 2 is comprised of the following documentation:

- detailed written description and justification.
- Unit 2 Reload 4 Supplemental Reload Licensing Submittal (Y1003J01A40, July 1982).
- Errata and Addenda No. 4, July 1982, and No. 5 August 1982, to the Browns Ferry LOCA Analyses, NEDO-24088-1.

ENCLOSURE 2

Description and Justification of Changes
(TVA Browns Ferry Nuclear Plant Technical Specifications)

A. Changes Related to Reload 4

(Reference attachments C and D)

Pages vii, 9, 16, 131, 159, 168, 171, 172, 172a, and 330 - During the reload 4 refueling, the last of the initial core types-1, -2, and -3 7X7 fuel assemblies will be removed. Therefore, references to these fuel types are being deleted. In addition, a MAPLHGR table for a new fuel type has been added (P8DRB265H).

Pages 19, 25, and 169 - This is the first reload for Browns Ferry unit 2 in which the transients were analyzed by General Electric (GE) Company's ODYN code as required by the staff. Additional citations are being added to the technical specifications to reference NRC's approval of this code for core reloads.

Pages iii, 27, 30, 181, and 219 - During the reload 4 outage, the two presently-installed main steam line safety valves will be replaced with 2-stage target rock safety/relief valves (S/RVs) identical to the other 11 S/RVs. This change was recognized in the reload analyses. In addition, in 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 84.10 percent. The value of 84.10 percent total relief capacity is derived from the values of 77.63 percent for 12 SRVs operable out of a total of 13 SRVs. The capacity of 77.63 percent of nuclear boiler rated steam flow, as listed in the Browns Ferry unit 2, reload 4 Supplemental Licensing Submittal, was calculated based on certified valve capacity for a 5.125-inch throat diameter valve (870,000 lbs/hour at 1,090 + 3 percent) 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.

Pages ii, viii, 160, 160a, and 172a - As supported by the reload submittal, the operating limit MCPRs are being changed since the MCPRs were determined by the ODYN code rather than the REDY code. OLMCPRs are now calculated from two curves rather than being a single value (or a ramp change with fuel exposure).

Pages 160 and 169 - The 7X7 fuel power spiking penalty and the supporting 3.5.J bases are being deleted since all 7X7 fuel assemblies are being removed. The deletion of the power spiking penalties for 8X8, 8X8R, and P8X8R fuel assemblies were previously approved for Browns Ferry unit 3 by Amendment No. 67 to Facility License DPR-52 on June 12, 1981.

B. Changes Related to Torus Modifications

Numerous modifications are being implemented in the unit 2 torus during the reload 4 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. The structural modifications to the torus containment include addition of torus tiedowns, addition of ring girder reinforcement and reinforcing attached piping nozzles. Vent system modifications include shortening the downcomers, adding local reinforcement to the vent header, and adding new tie bars to the downcomers. Attached piping is being strengthened including modification of the ECCS header support. Many changes are being made to the safety/relief valve piping system including adding quencher arms to the ramshead, adding quencher arm and ramshead supports, adding 10-inch vacuum valves, reinforcing the ring girder at the SRV hanger attachment, rerouting of piping, and adding new snubbers and supports for the piping.

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 13,000 gpm. A new containment cooling analysis was performed for this configuration, and it was determined that this flow rate induces 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 227, 267, and 269 - The minimum torus water level limits in section 3.7.A.1.a and in the bases for this section are being changed from -7 inches (differential pressure control greater than 0 psid) to -6.25 inches and from -8 inches (0 psid differential pressure control) to -7.25 inches, a change in each case of 0.75 inch. There are 15-inch by 15-inch sealed box beams being added as support for the safety/relief valve lines and HPCI-RCIC internal supports. Addition of these supports will result in appreciable water displacement. Calculations indicate that the box beams and HPCI-RCIC supports will increase the torus water level approximately 3/4-inch due to their presence. This rise in the torus water level is reflected in these revised technical specification values.

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 temperature of 130°F is not expected to be a safety concern warranting placing the reactor in cold shutdown and performing a torus inspection. This requirement is therefore unnecessary and deletion is proposed.

Pages 235a and 269 - In section 3.7.A.6.a and the bases thereof the setpoint for the drywell suppression chamber (wetwell) differential pressure control (ΔP) is being changed from 1.3 psid to 1.1 psid. Downcomer water clearing loads are greatly reduced by physically shortening the downcomers (by almost

one foot) and imposing a drywell-wetwell ΔP. The Browns Ferry unique loads were determined by considering a differential pressure of 1.10 psid at the maximum allowable torus water level. In order to be consistent with this analysis, the technical specification associated with the ΔP control has been established at 1.10 psid.

Page 256 - Table 3.7.B has been revised to include penetration X-223. This penetration has been installed to provide another suppression chamber access hatch to facilitate the torus modifications.

Page 266 - Table 3.7.H has been revised to include temporary electrical penetration X-200A-SC which is integral to torus access hatch X-200A. This electrical penetration is designed to accommodate instrumentation for the S/RV-torus integrity test program. This penetration is to be removed at the first opportunity following the test program.

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.

C. <u>Miscellaneous Plant Modifications and Administrative Technical Specification Changes</u>

1. <u>480V MOV Boards Tie-In and LPCI M-G Sets Installation</u>

Pages 293a, 297b, 298, 300, and 330 - Amendment No. 45 to Facility License No. DPR-52 for Browns Ferry unit 2 dated May 11, 1979 adds a license condition authorizing modifications to the power supply for certain LPCI valves. The modification ensures that the 480V ac reactor MOV boards, with the associated autotransfer feature, will be isolated from the redundant divisional power supplies. The associated technical specifications are consistent with those approved in Amendment No. 75 to Facility License No. DPR-33 for Browns Ferry unit 1 dated September 3, 1981.

2. <u>Thermal Power Monitor</u>

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

Page 10 - Add section 2.1.A.2.c.

Page 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.

3. Scram Discharge Instrument Volume

Pages 37, 39, and 40. The long-term modifications to the scram discharge volume and instrument volume (SDIV) necessary to resolve problems related to the partial rod insertion event are being implemented during this outage. To upgrade the reliability of the SDIV instrumentation, two of the float-type level switches are being replaced by diverse differential pressure switches. Tables 4.1.A and 4.1.B are therefore being revised to reflect that these analog-type devices will also require surveillance testing.

4. <u>Revisions to Tables 3.7.A through 3.7.G, Primary Containment Isolation</u>

Tables 3.7.A to 3.7.G are being revised to reflect several plant modifications. All revisions to these tables that were submitted in previous technical specification submittals are also being incorporated in these proposed revisions. The revisions to these tables will supersede revisions of the following submittals: T.S. 92, 123, 133, and 146.

<u>Table 3.7.A</u>

FCV-1-55 and -56 drain valves are required to be open for extended periods during power operation. Therefore, these valves will be considered as normally open and technical specification surveillance requirement 4.7.D.1.b will apply.

A footnote is being added to reactor water sample line isolation valves to clarify the actual isolation trip signals. These valves isolate only on reactor vessel low-low water level (470 inches) and main steam line high radiation.

An error which inadvertently listed the suppression chamber drain valves as FCV-74-57, 58 is being corrected to read FCV-75-57, -58.

Valve FCV-69-12 is being deleted from table 3.7.A. This valve is not a containment isolation valve. Isolation is provided by check valves 69-579 and 3-572.

In response to NRC generic letters of September 27, 1979 and October 22, 1979 to "All Light Water Reactors," TVA is modifying the containment purge system for unit 2 during this outage to satisfy applicable requirements of NRC Branch Technical Position CSB 6-4 regarding valve closure times and addition of debris screens. Pages 251 and 252 are being revised to reflect the significant reduction in the maximum allowable operating time. On the nitrogen purge valves, the operating time is being reduced from 10 seconds to 5 seconds and on the purge inlet and isolation valves, the operating time is being reduced from 90 seconds to only 2.5 seconds. The faster valve closure times significantly reduce potential offsite doses. The addition of the debris screens provides protection against foreign material entering the purge ducting and interfering with closure of the purge valves. In our letter of June 2, 1981, we provided the data and analysis to demonstrate that the purge valves are adequate for closure against the design basis loss-of-coolant accident forces.

The drywell exhaust valve bypass to the SBGT system (FCV-64-31) and the suppression chamber exhaust valve bypass to the SBGT system (FCV-64-34) are being changed from "C" to "O" and from "SC" to "GC." This will permit operation of the drywell to torus $\Delta$P compressor in automatic as it was orginally designed to be operated.

The normal operation of FCV-71-7A, 7B are being changed from "O" to "C" and the action on initiating signal from "GC" to "SC." The normal position of these valves is closed.

Change "FCV-75-57, -58" to "FCV-73-6A, -6B." These valves were incorrectly numbered in the table.

Valves FCV-64-139 and -140 are being added to table 3.7.A. These drywell $\Delta$P air compressor suction and discharge valves are containment isolation valves and should be verified for operating time. These valves are a part of the addition of the drywell pressurization system.

The following additions are proposed for table 3.7.A because these valves were inadvertently omitted in this table. These valves are all containment isolation valves and should be verified for operating time.

FCV-84-19, -20, CAD system torus/drywell exhaust to SBGT
FCV-76-24, drywell to suppression chamber nitrogen purge inlet
FCV-75-26, -54, core spray discharge to reactor check valves
FCV-90-254A and B, drywell CAM suction valve
FCV-90-257A and B, drywell CAM discharge valves
FCV-90-255, drywell CAM suction valve

Tables 3.7.D through 3.7.G

Tables 3.7.D through 3.7.G have been completely revised to be more consistent with standard technical specifications. These tables contain a "Test Medium" and "Test Method." The proposed tables have been revised to contain the "Test Medium" within the title of the table and eliminate the "Test Method" altogether. The standard technical specifications do not contain a test method for testing isolation valves. In addition, the test methods for these valves are contained in their specific testing instructions and therefore should not be contained in the technical specifications. The deletion of the test methods from the table does not have any adverse impact on safety.

Table 3.7.D

Test connections were added to unit 2 so that the following valves could be tested.

FCV-2-1192, service water
FCV-2-1383, service water
FCV-33-1070, service air
FCV-33-785, service air

FCV-64-141 has been deleted from this table. It is not an isolation valve and is not tested.

Table 3.7.E

Table 3.7.E has been revised to include valve 2-1143 which is now being tested.

FCV-74-722 has also been added to table 3.7.E. This valve was inadvertently omitted.

Summary

The plant modifications and changes described above are significant improvements in plant safety. Tables 3.7.A through 3.7.G have been revised to reflect all plant modifications affecting the primary containment isolation system. These proposed revisions reflect changes which do not adversely affect plant safety.

4. Administrative Changes

Pages ii, iii, and v - Technical specification titles for sections 3.5/4.5.A, 3.5/4.5.J, 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 specification.

Pages iii, 19, 182, 221, and 222 - The paragraph title pertaining to recirculation pump operation has been changed to be more consistent with the specification intent. Additionally, section 2.1 of the technical specifications contains the bases for the "limiting safety system settings related to fuel cladding integrity." At the bottom of page 19 there is presently a paragraph relating to operation in the natural circulation mode. This paragraph is being moved verbatim to the bases for recirculation pump operation on page 221 which is a more appropriate location. There is no safety significance to this reformatting of the technical specifications.

Page iv - The page numbers for technical specification titles for 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 vii - 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 has previously been removed from the technical specifications by amendment No. 48.

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 relay have been added. Surveillance on HPCI time delay relay required by NUREG-0737, item II.K.3.15 is also added.

Pages 143 and 145 - These changes are administrative changes that remove references to nonapplicable technical specification requirements. These changes do not affect any actual limiting conditions for operations; therefore, plant safety is not affected.

Page 169 - Reporting requirements changed to be consistent with other units.