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

March 11, 1983

Docket No. 50-260

Posted
Amdt. 85
TO DPR-52

Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

The Commission has issued the enclosed Amendment No. 85 to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. This amendment is in response to your application of October 15, 1982 (TVA BFNP TS 179) as supplemented by your letters of November 17, 1982, December 10, 1982 and January 7, 1983.

The amendment revises the Technical Specifications to (1) incorporate the limiting conditions for operation during fuel Cycle 5, and (2) reflect changes resulting from design, equipment and procedural modifications made during the current refueling outage.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Richard J. Clark

Richard J. Clark, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 85 to DPR-52
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

8303300387

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 15, 1982, as supplemented by letters dated November 17, 1982, December 10, 1982 and January 7, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 85, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 85

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Review Appendix A as follows:

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

ii	27	159	234	266
iii	30	160	235a	267
iv	33	168	250	268
v	37	169	251	269
vii	39	171	252	273
viii	40	172	253	293a
8	66	172a	256	294a
9	98	181	258	297b
10	99	182	259	298
16	126	219	260	300
19	131	221	261	321
20	143	222	262	322
22	145	227	263	323
25	146	233	264	330

2. Marginal lines on the above pages indicate the area being revised.

3. Add the following new pages:

36a 160a 168a 253a

Section		Page No.

Amendment No. 85

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

SAFETY LIMIT

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.

LIMITING SAFETY SYSTEM SETTING

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 34.2×10^6 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 34.2×10^6 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/ftfor 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.

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-24011-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	Modes in Which Function Must Be Operable: Shut-down	Refuel(7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16) (24)(25)						
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux (Fixed Trip)	≤120 %				X	1.A or 1.B
2	High Flux	≤15% rated power (13)		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative			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 'larm	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 (7)
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			
40 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)	$\geq$ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
1(2)	Instrument Channel - Suppression Chamber High Level	$\leq$ 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	$\leq$ 583" above vessel zero.	A	1. Above trip setting trips RCIC turbine.
1	Instrument Channel - RCIC Turbine Steam Line High Flow	$\leq$ 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

3.3 Reactivity Control

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

F. Scram Discharge Volume (SDV)

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

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

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

4.3 Reactivity Control

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

F. Scram Discharge Volume (SDV)

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

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

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

3. No additional surveillance required.

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 12,000 gpm against an indicated system pressure of 250 psig.

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

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 $\dfrac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}$, whichever is greater

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

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

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

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

τ_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 ≥ 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. τ= 0.0 prior to initial scram time measurements for the cycle performed in accordance with Specification 4.3.C.1.

   b. τ 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	SURVEILLANCE REQUIREMENTS
3.5 CORE AND CONTAINMENT COOLING SYSTEMS	4.5 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.	

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 ≥ 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

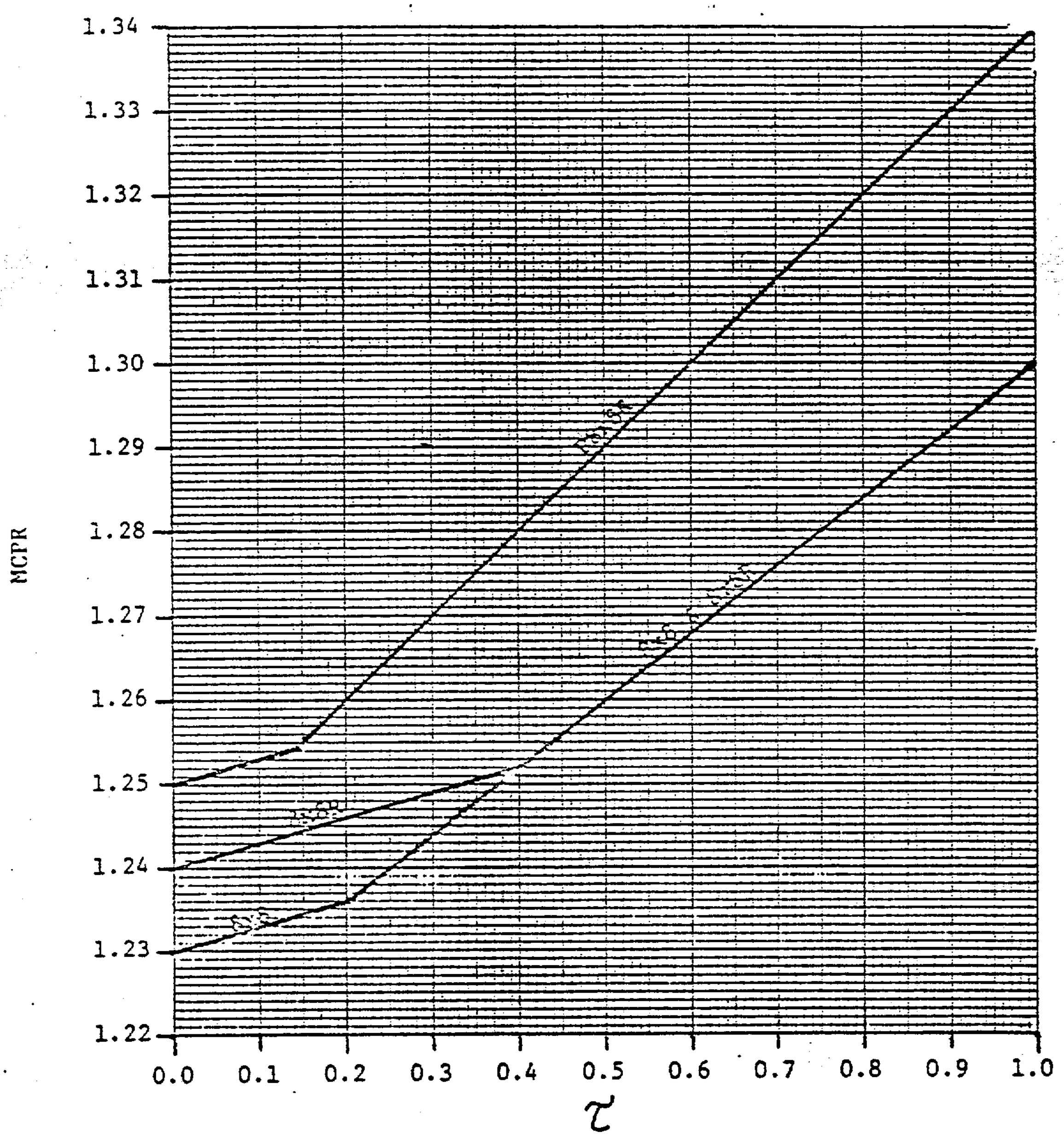

Figure 3.5.K-1
MCPR Limits

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.

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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

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 failuse 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.

3.6/4.6 BASES:

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.

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.

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	O	GC
1*	Reactor Water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves (FCV-74-48 & 47)	1	1	40	C	SC
2	RHRS - LPCI to reactor (FCV-74-53 & 67)		2	30	C	SC
2	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.

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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	O	GC
4	FCV 73-81 (Bypass around FCV 73-3)		1	10	O	GC
4	HPCIS steamline isolation valves FCV-73-2 & 3	1	1	20	O	GC
5	RCICS steamline isolation valves FCV-71-2 & 3	1	1	15	O	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

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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 ΔP air compressor suction valve (FCV-64-139)		1	10	0*	GC
6	Drywell ΔP 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

TABLE 3.7.D (Continued)

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
84-3A	Containment Atmospheric Dilution
84-3B	Containment Atmospheric Dilution
84-3C	Containment Atmospheric Dilution
84-3D	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
[illegible]	CRD Hydraulic Return
90-254A	Radiation Monitor Suction

Amendment No. [illegible], 92, 95

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)

3.7.A & 4.7.A Primary Containment

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

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

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

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

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat 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 5×10^{-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

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.

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.

demonstrate that the associated diesel generator will start.

c. The loss of voltage and degraded voltage relays which start the diesel generators from the 4-kV shutdown boards shall be calibrated annually for trip and reset and the measurements logged. These relays shall be calibrated as specified in Table 4.9.A.4.c.

d. 4-kV shutdown board voltages shall be recorded once every 12 hours.

5. 480V RMOV boards D and E

a. Once per operating cycle the automatic transfer feature for 480V RMOV boards D and E shall be functionally tested to verify auto-transfer capability.

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

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

D. Roving Fire Watch

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

3.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspection

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

F. Fire Protection Organization

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

4.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspections

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

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).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 85 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 Introduction

By letter dated October 15, 1982 (TVA BFNP TS 179), as supplemented by letters dated November 17, 1982, December 10, 1982 and January 7, 1983, the Tennessee Valley Authority (licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. The proposed amendment and revised Techncial Specifications were to: (1) incorporate the limiting conditions for operation associated with fuel Cycle 5, and (2) reflect changes resulting from design, equipment and procedural modifications made during the current refueling outage.

2.0 Discussion and Evaluation

2.1 Reload Discussion

Browns Ferry Unit 2 (BF-2) shutdown for its fourth refueling on July 30, 1982, with a projected restart date in early March 1983. BF-2 was initially fueled with 764 of the GE 7x7 fuel assemblies containing 49 fuel rods each. During the first refueling, which began March 18, 1978, 132 of the 7x7 fuel elements were replaced with one water rod 8x8 fuel assemblies. In the second refueling, which started April 27, 1979, 232 of the 7x7 fuel assemblies were replaced with a like number of two water rod, retrofit 8x8 (8x8R) bundles. During the second refueling, an additional 36 7x7 fuel assemblies were also replaced with 8x8 fuel that had originally been procured for fuel Cycle 2 but not used. During the third refueling, which began September 5, 1980, an additional 240 of the original 7x7 fuel bundles were replaced with prepressurized two water rod 8x8 retrofit (P8x8R) fuel assemblies. The prepressurized fuel assemblies are essentially identical from a core physics standpoint to the two water rod fuel assemblies (8x8R) except that they are prepressurized with about three rather than one atmospheres of helium to minimize fuel clad interaction. Our evaluation of the P8x8 fuel is discussed in the Safety Evaluation attached to our letter of April 16, 1979, to GE approving the use of this fuel in Boiling Water Reactor (BWR) reload licensing applications.

With this reload, the last of the 124 remaining initial 7x7 fuel assemblies were removed from the core. The licensee had also planned to replace 124 of the 8x8 fuel assemblies, making a total of 248 new fuel bundles to be added. During the coastdown period for BF-2 that occurred during Cycle 4 operation (from April to July 1982), higher than normal radioactivity levels were noted in the steam.

During the outage following Cycle 4 operation, the entire core was off-loaded to the spent fuel pool to permit the torus to be drained for the Mark I torus modifications. Examination of the removed fuel disclosed that many of the 8x8 fuel assemblies showed evidence of severe waterside corrosion. The cause was attributed to what has been characterized in the industry as "Crud Induced Localized Corrosion (CILC)". Detailed inspection of similarly failed fuel at Hatch 1 and Vermont Yankee disclosed that the fuel that failed was mainly gadolinia poisoned fuel rods; the method of failure was pitting corrosion perforating through the cladding. Examination of the fuel removed from BF-2 disclosed 31 fuel assemblies with significant corrosion; these were replaced with new fuel.

Thus, some of the fuel assemblies that were to be returned to the core for Cycle 5 were replaced with new fuel and some of the depleted fuel assemblies that were not going to be reinserted are now going to be reused. The net effect from reuse of fuel that was considered"spent" is that the core for Cycle 5 will have about three weeks less full power capability. The total number of fuel assemblies changed out in the current reload remains at 248.

In support of this application, the licensee submitted a Supplemental Reload Licensing Report, Y1003J01A40, (Ref. 1) and update to the LOCA analysis report, NEDO-24088-1, (Ref.2) and a number of proposed changes to the BF-2 Technical Specifications. The analyses presented in Ref.1 and 2 were based on adding 248 new fuel assemblies. Both the licensee and the General Electric Company (GE) reevaluated the analyses in light of the additional new and spent fuel assemblies being added to replace those found corroded and determined that the analyses conservatively bounded the revised coreloading, since the revised coreloading plan will have less energy than the loading described in Ref. 1.

2.2 Reload Evaluation

We reviewed the submittals and evaluated the nuclear design, the thermal hydraulic design, the transient and accident analyses, and the Technical Specification changes. The fuel mechanical design is fully described in GE report NEDE-24011-P-A-4 (Ref.3) Because of our review of a large number of generic considerations related to use of 8x8, 8x8R and P8x8R fuels in mixed loadings, and on the basis of the evaluations which have been presented in Reference 3, only a limited number of additional areas of review have been included in the Safety Evaluation. For evaluations of areas not specifically addressed in this Safety Evaluation refer to Reference 3.

2.2.1 Nuclear Design

With the exception of the shutdown margin and standby liquid control system analyses, the nuclear parameters applicable to the Cycle 5 core were obtained by methods and techniques described in Reference 3, which has been approved by the staff for this purpose. The results were within the range normally encountered in BWR reloads and are acceptable. The shutdown margin and standby liquid control system analyses were performed by the licensee using its core simulator and lattice physics methods which have been reviewed and approved by the staff. The shutdown margin was 1.4% reactivity change with the strongest rod out. The standby liquid control system is capable of making the unrodded core subcritical at 20°C with a margin of 2.3% reactivity change. These are acceptable margins and therefore, we conclude that the nuclear design parameters for the Cycle 5 core are acceptable.

2.2.2 Thermal-Hydraulic Evaluation

The thermal-hydraulic review includes the following areas: (1) safety limit minimum critical power ratio (MCPR), (2) operating limit MCPR, and (3) thermal hydraulic stability. The objective of this review is to confirm that the thermal hydraulic design of the reload has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients, and is not susceptible to thermal hydraulic instability.

Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8x8 fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The 1.07 SLMCPR is unchanged from the SLMCPR previously approved. The basis for this safety limit is addressed in Reference 3.

Operating Limit MCPR

Various transients could reduce the MCPR below the intended safety limit MCPR during Cycle 5 operation. The anticipated operational transients have been analyzed by the licensee to determine which could potentially induce the largest reduction in the initial (CPR). Operating cycle MCPR values for this plant specific cycle are as expected for the BWR/4 design with the fuel types that are presented in Cycle 5 of BF-2, and compare favorably with the MCPR for operating plants such as Brunswick Unit 2 Cycle 5, previously approved.

Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (Ref. 1) show that the maximum thermal-hydraulic stability decay ratio is 0.74 for this cycle. Because operation in the natural circulation mode is prohibited by Technical Specifications, there will be added margin to the core stability and therefore, we find the thermal-hydraulic stability acceptable.

Summary of Thermal-Hydraulic Evaluation

The analysis of BF-2 Cycle 5 has been performed with standard techniques described in Reference 3 and the values of MCPR and thermal-hydraulic stability decay ratio are as expected for the BWR/4 design and we, therefore, find the thermal-hydraulic design for Cycle 5 acceptable.

2.2.3 Transient and Accident Analyses

The transient and accident analyses were performed by the methods and procedures described in GE report NEDO-24011-P-A-4, which we previously approved. It was necessary to perform a cycle specific analysis of the rod drop accident since the accident reactivity shape function for Cycle 5 was not bounded by the generic shape for the cold startup case described in Reference 3. The results of this analysis, which are presented in Section 15 of Reference 1, show that the resultant peak enthalpy, cold, was 264.5 cal/gm. The resulting peak enthalpy rise was less than the acceptance criterion of 280 calories per gram and is acceptable.

The GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff and is part of the Reference 3 methodology. Potential fuel loading errors involving misoriented bundles and bundles loaded into incorrect positions have been analyzed by this methodology and the results are reported in Section 14 of the supplemental reload submittal. The analyses determined that the ΔCPR for a misoriented fuel bundle was less than the ΔCPR for the limiting transients and therefore, we conclude that the analyses performed by GE for fuel loading errors is acceptable.

2.2.4 Core Reload Technical Specification Changes for Cycle 5

During the refueling for Cycle 5 the last of the 7x7 fuel will be removed from the core. Accordingly all references to this fuel are being deleted from the Technical Specifications. This is acceptable. In addition, Technical Specifications 2.1.A, Table 3.1.A, and Table 4.1.A have been changed to reflect the alteration of the flow biased neutron flux trip to a thermal power monitor and the addition of a separate high neutron flux trip at 120% of full power. This change has been found acceptable for several boiling water reactors and therefore, is acceptable for BF-2.

The MAPLHGR values in Table 3.5.1-3 have been extended to burnup values of 40,000 MWd/t and Table 3.5.1-4 has been added to provide MAPLHGR values for the new fuel type introduced for this reload. The MAPLHGR tables were obtained by standard methods (Ref. 2) and are acceptable)

This is the first reload for BF-2 for which the overpressurization transients were analyzed with the ODYN code. Specification 3.5.k and Surveillance Requirement 4.5.k have been revised to reflect the new analyses. This procedure, which was previously reviewed and found acceptable by the staff, is being introduced on all operating BWRs and therefore, is acceptable for BF-2.

2.2.5 Summary of Core Reload Evaluation

On the basis of our review, which has included the nuclear design, thermal-hydraulic design, transient and accident analyses, and Technical Specification changes, we conclude that operation of BF-2 for Cycle 5 will not endanger the health and safety of the public. This conclusion is based on the fact that approved methods have been used to perform the various analyses and that the results are consistent with those for other BWR/4 reactors.

3.0 Plant Modifications

3.1 Discussion

BF-2 shutdown for the present refueling and maintenance outage on July 30, 1982, and is projected to be down for over seven months. The reason for the extended outage is the time needed to complete a number of NRC-required modifications, as well as the inspections, repairs, surveillance, maintenance, and other activities normally associated with a refueling outage. During this shutdown, the licensee expects to complete numerous modifications which NRC has proposed or required for operating reactors, such as Browns Ferry, in various Bulletins, Orders, the TMI-2 Action Plan (NUREG-0737), new regulations, revisions to the Security Plan and Emergency Response Plan, resolution of generic issues, etc. Some of these modifications require changes to the Technical Specifications prior to startup and are included in this Safety Evaluation.

3.2 Evaluation

Torus Modifications

On January 13, 1981, the Commission issued an Order modifying the BF-2 license to require the licensee to promptly institute a reassessment of the containment design for suppression pool hydrodynamic loading conditions and to install any plant modifications needed to conform to the staff's Acceptance Criteria, which are contained in Appendix A to NUREG-0661 ("Safety Evaluation Report, Mark I Containment Long-Term Program" dated July 1980) by March 31, 1982. This Order was subsequently modified by an Order dated January 19, 1982, extending the time to complete some of the modifications to the Cycle 6 outage.

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 (SRV) 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. These modifications to the torus require changes to the Technical Specifications to account for water displaced by the additional structural steel and to reflect the plant unique analysis which the licensee was required to perform to assure conformance of the design to the staff's Acceptance Criteria in NUREG-0661. The specific changes to the Technical Specifications are discussed below.

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. The changes, which we have reviewed and approved, are necessary to ensure that the minimum water volume is maintained in the torus for suppression of potential LOCA loads and are acceptable. (This same change to the Technical Specifications was made by Amendment No. 51 to Facility License No. DPR-68 for BF-3 issued March 29, 1982.)

Pages 235a and 269 - In Section 3.7.A. 6.a (and the bases thereto), 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. The changes to the Technical Specifications conform to the requirements in Section 2.16, "Differential Pressure Control Requirements," in Appendix A to NUREG-0661 and are therefore, acceptable.

Pages 233, 234, 267 and 268 - The "Bases" section for Specifications 3.7.A and 4.7.A for the suppression pool temperature limits was based 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 SRV T-quenchers. During the current refueling outage, the T-quenchers are being added to the safety-relief valve discharge device. In Section 2.13.8 of Appendix A to NUREG-0661 ("Suppression Pool Temperature Limits") the staff specified that "the suppression pool local temperature shall not exceed 200°F throughout all plant transients involving SRV operations." The licensee's analyses determined that at reactor vessel pressures above approximately 555 psig, the bulk pool temperature will not exceed 180°F. At pressures below approximately

240 psig, the bulk temperature will not exceed 184^{0}F. Both temperatures are well below the acceptable limits. These temperatures 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^{0}F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure. Section 4.7.A.2.k of the present Technical Specifications requires that if extended relief valve operation causes the temperature of the suppression pool to exceed 130^{0}F, the reactor shall be shutdown and the torus and drywell visually inspected for signs of distress or displacement. 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^{0}F is not expected to be a safety concern warranting placing the reactor in cold shutdown and performing a torus inspection. Therefore, this requirement is being deleted.

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 SRV-torus integrity test program. This penetration is to be removed at the first opportunity following the test program.

Page 273 - The present Technical Specifications in the "Bases" for primary containment, discuss the specific type of protective coatings applied to the drywell and torus surfaces to protect the steel from corrosion and minimize contamination of the water. There have been significant developments in protective coating technology since the Browns Ferry units were licensed. During the torus modifications, the licensee has thoroughly sandblasted all torus surfaces in each unit and is applying coatings that offer more potential for sealing the surfaces. Therefore, the "Bases" are being generalized so that a technical specification change will not be required if a different protective coating is applied.

Page 145 - Section 4.5.B.1 of the Technical Specifications requires that every three months, the LPCI capability of the RHR pumps shall be demonstrated. In the tests, the pumps take suction from the torus and return the water to the torus. The pumps are required to demonstrate that two pumps in the same loop can deliver at least 15,000 gpm against an indicated system pressure (head) of 200 psig.

The two-pump 15,000 gpm LPCI test surveillance was determined to induce vibrations in the RHR return line to the torus. To eliminate the vibration, an orifice has been installed in the return line. However, installation of this orifice plate also decreases the suppression pool cooling mode of RHR operation from 15,000 gpm to approximately 12,000 gpm. A new containment cooling analysis was performed for this configuration, and it was determined that this flow rate 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. The revised test requirement is that the two pumps demonstrate that they can deliver 12,000 gpm against a higher head - 250 psig. The orifice is in the return line to the torus and does not change the volume of water that would be injected into the reactor during the LPCI mode. The 12,000 gpm at higher pump head pressure is equivalent to 15,000 gpm at lower discharge pressure. We conclude that the change has no adverse impact on the LPCI or containment cooling modes of RHR operation and is acceptable.

480V MOV Boards Tie-In and LPCI M-G Sets Installation

Pages 293a, 297b, 298, 300 and 330 - Amendment No. 45 to Facility License No. DPR-52 for BF-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 modifications are designed to eliminate the recirculation loop selection logic and to rewire the accident initiation signals to direct both LPCI injection valves to open upon detection of accident conditions. The modifications include installation of qualified Class 1E motor-generator (MG) sets to serve as isolation devices between the redundant divisional 480V shutdown boards (power sources) and the swing bus (auto-transfer) of the 480V reactor MOV boards that supply motive power to the LPCI valve operators. In 1976, the NRC staff requested the licensee to propose modifications to eliminate the LPCI systems recirculation loop selection logic to eliminate a potential single failure concern. As noted above, the design was approved by Amendment Nos. 51, 45 and 23 for Units 1, 2 and 3, respectively, on May 11, 1979. The modifications require changes to the Technical Specifications which are incorporated herein. The associated Technical Specifications are consistent with those approved in Amendment No. 75 to Facility License No. DPR-33 for BF-1 dated September 3, 1981.

Thermal Power Monitor

Pages 8, 10, 20, 22, 33, 36a and 37 - During this outage, the licensee has installed a flow-biased simulated thermal power monitor. These monitors are installed on most all BWRs; the justification for these monitors is discussed in the "Bases" for the APRM settings in the BWR Standard Technical Specifications (BWR/4 STS, Section 2.2.1, page B2-7). The monitors are installed to have the APRM flow biased neutron flux signal respond to the thermal flux rather than the neutron flux by accounting for the approximately six-second thermal time constant of the fuel. The proposed changes to the Technical Specifications are acceptable, since they are based on previously reviewed and accepted changes for similar BWRs.

Scram Discharge Instrument Volume

Pages 37, 39, 40 and 126 - The long-term modifications to the scram discharge instrument volume (SDIV) necessary to resolve problems related to the partial rod insertion event are being implemented during this outage for BF-2. To upgrade the reliability of the SDIV instrumentation, two of the float-type pressure switches are being replaced by diverse differential pressure switches. Tables 4.1.A and 4.1.B are therefore being revised to add these switches to the list of instruments that require surveillance testing.

Containment Vent and Purge System

In response to NRC generic letters of September 27, 1979 and October 22, 1979 to "All Light Water Reactors," the licensee is modifying the containment purge system for BF-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.

These same changes to the Units 1 and 3 Technical Specifications were made respectively by Amendment No. 76 to License No. DPR-33, issued September 15, 1981, and by Amendment No. 51 to License No. DPR-58 issued March 29, 1982. Since the changes to the Technical Specifications for BF-2 are those requested by our letter of December 17, 1981 and have been previously approved for BF-1 and BF-3, they are acceptable for BF-2.

Primary Containment Isolation Valves

Tables 3.7.A through 3.7.H list the various valves and penetrations associated with primary containment isolation. Specifically, Table 3.7.A lists the primary containment isolation valves that must be operable during reactor power operation (in accordance with Section 3.7.D of the Technical Specifications) along with the maximum operating times and normal position. Table 3.7.D lists the primary containment isolation valves on which local leak rate tests must be performed each cycle in accordance with Section 4.7.2.g. Tables 3.7.E, 3.7.F and 3.7.G list the stop-check and check valves on the torus and drywell influent lines that must be similarly tested. As discussed below, the licensee has proposed revisions to these tables to reflect plant modifications and the requirements in NUREG-0737 Item II.E.4.2.

Tables 3.7.D through 3.7.G have been completely revised to be more consistent with the BWR/4 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. Similar changes were made to the same tables for BF-3 in Amendment No. 51 issued March 29, 1982. As part of the revisions to these tables, the licensee has proposed to air test certain isolation valves that were previously water tested. Appendix J to 10 CFR Part 50 specifies air testing as the recommended leak testing method, except for those valves that are fluid sealed. In addition, the staff considers air testing of valves to be a more conservative method than water testing. On the basis of the information provided by the licensee in the submittal of October 15, 1982, and the requirements of 10 CFR Part 50, Appendix J, we conclude that the proposed changes to the Technical Specifications with respect to the test medium are acceptable.

Most of the changes to the tables on isolation valves are to add or delete valves which the inservice pump and valve testing program indicated should be verified for operating time, to correct valve numbers or to correct valve positions. Each proposed change is described in detail in the licensee's submittal of October 15, 1982. We have reviewed each change and concluded they are acceptable, since they are consistent with modifications either deleting or adding valves.

NUREG-0737, Item II.K.3.15

TMI Action Plan Item II.K.3.15 requires licensees of BWRs to modify pipe-break-detection circuitry so that pressure spikes resulting from HPCI and RCIC initiation will not cause inadvertent system isolation. The licensee elected to employ the BWR Owners Group modification which incorporates a three second time delay relay (TDR) to prevent spurious isolation. In our letter to the licensee of October 13, 1981, we requested the licensee to provide certain analyses and to "propose the appropriate Surveillance Requirements and Limiting Conditions of Operation for the HPCI and RCIC systems which address this item." The safety evaluation was provided by the licensee's letter of December 16, 1981. All of the Browns Ferry units have had a three-second TDR on the HPCI systems. During the current outage for BF-2, a TDR was added to the RCIC system. The proposed changes to the Technical Specifications requiring calibration and surveillance of the time delay relays was submitted with the licensee's application of October 15, 1982. Table 4.2.B (p99) is being modified to require a logic system functional test, including calibration of the RCIC and HPCI system isolation logic. The changes to the Technical Specification reflect the surveillance requirements requested in our letter of October 13, 1981 on Item II.K.3.15 and are acceptable. The same changes were made to the BF-3 Technical Specifications in Amendment No. 51 issued March 29, 1982.

4.0 Administrative Changes

The licensee has proposed eight administrative changes to the BF-2 Technical Specifications; the licensee has described and justified each change in its submittal of October 15, 1982. The changes are to revise the Table of Contents, to reformat one section, to correct or add references or to delete reference to a table that was removed by a previous amendment. These changes do not affect any actual limiting conditions for operation. We conclude that these proposed changes are editorial in nature and do not alter the techncial bases of the specifications and therefore, are acceptable.

5.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

6.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a safety margin, the amendment does not involve a significant hazards consideration (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 11, 1983

Principal Contributors: Walt Brooks, Dick Clark, Jim Hall

7.0 References

1. "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant, Unit 2, Reload No. 4 (Cycle 5)," Y1003J01A40.

2. NEDO-24088-1, "LOCA Analysis for Browns Ferry Nuclear Plant Unit 2."

3. NEDE-24011-P-A-4, "General Electric Standard Application for Reactor Fuel," January 1982.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 85 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority (the licensee), which revised the Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit 2 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications to (1) incorporate the limiting conditions for operation during fuel Cycle 5, and (2) reflect changes resulting from design, equipment and procedural modifications made during the current refueling outage.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and the pursuant to 10 CFR 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated October 15, 1982, as supplemented by letters dated November 17, 1982, December 10, 1982 and January 7, 1983, (2) Amendment No. to License No. DPR-52, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 11th day of March 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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