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March 29, 1982

Docket No. 50-296

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment No. 51 to Facility License No. DPR-68 for the Browns Ferry Nuclear Plant, Unit 3. This amendment changes the Technical Specifications in response to your request of December 9, 1981 (TVA BFNP TS 170) related to a reload associated with fuel Cycle 5 operation.

The changes to the Technical Specifications (1) incorporate the limiting conditions for operation during the fifth cycle 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,

Cp

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

Enclosures:
1. Amendment No. 51 to DPR-68
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

FR - amendment & concurrence only. Conditioned on change to FR Notice per Judy note to Seigel 3/25/82

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DATE	3/19/82	3/19/82	3/26/83	3/[illegible]/83	3/27/83	

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated December 9, 1981 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-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 51, 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 its 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 29, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 51

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

ii	176	274
iii	178	275
vii	181	276
viii	182	277
18	182a	278
19	182b	279
24	192	280
26	195	281
27	224	282
29	225	285
30	227	286
95	231	
123	246	
124	262	
126	263	
133	264	
146	264A	
149	265	
165	270	
166	271	
167	272	
167a	273	

2. Marginal lines on each page indicate the revised area.

Amendment No. 51

LIST OF ILLUSTRATIONS

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

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

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicalbe transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisions and results are summarized in Reference 1, 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 nominal 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 of *** is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

*** See Section 3.5.K.

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 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 (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during transients induced by disturbances, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses reported in Section N14 of the Final Safety Analysis Report demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin. Figure 2.1.2 shows the flow biased scram as a function of core flow.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM setting was selected

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

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

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram set point and initiation set points. Transient analyses reported in Section N14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. <u>References</u>

1. 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 Reactor", NEDO-24154, 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 limts 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:

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
All previous items on this page moved to page 26.	

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The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the pressure is not monitored at this point, it cannot be directly determined if this safety limit has been violated. Also, because of the potentially varying head level and flow pressure drops, an equivalent pressure cannot be (a priori) determined for a pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BFNP FSAR Section N14.0)
2. ASME Boiler and Pressure Vessel Code Section III
3. USAS Piping Code, Section B31.1
4. Reactor VEssel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

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 83.77% 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 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
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
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

3.3 REACTIVITY CONTROL

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3. a. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Sequence Control System (RSCS) shall be operable except the RSCS constraints may be suspended by means of the individual rod bypass switches for
1 - special criticality tests, or
2 - control rod scram timing per 4.3.C.1.
When RSCS is bypassed on individual rods for these exceptions RWM must be operable per 3.3.B.3.c and a second licensed operator may not be used in lieu of RWM.

4.3 REACTIVITY CONTROL

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3.a Prior to the start of control rod withdrawal at startup, the capability of the Rod Sequence Control System (RSCS) and the Rod Worth Minimizer to properly fulfill their functions shall be verified by the following checks:

3.3 REACTIVITY CONTROL

b. During the shutdown procedure, no rod movement is permitted between the testing performed above 20% power and the reinstatement of the RSCS restraints at or above 20% power. Alignment of rod groups shall be accomplished prior to performing the tests.

c. Whenever the reactor is in the startup or run modes below 20% rated power, the rod worth minimizer shall be operable. A second licensed operator may verify that the operator at the reactor console is following the control rod program in lieu of RWM except as specified in 3.3.B.3.a.

4.3 REACTIVITY CONTROL

Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences.

Group notch portion - For each of the six comparator circuits go through test initiate: comparator inhibit; verify; reset. On seventh attempt, test is allowed to continue until completion is indicated by illumination of test complete light.

b. Prior to attaining 20% rated power during rod insertion at shutdown, the tests in 4.3.B.3.a shall be performed to verify RSCS capability.

c. The capability of the rod worth minimizer (RWM) shall be verified by the following checks:

3.3 REACTIVITY CONTROL

4.3 REACTIVITY CONTROL

control rod.

5. Prior to obtaining 20% rated power during rod insertion at shutdown, verify the latching of the proper rod group and proper annunciation after insert errors.

d. When the RWM is not operable, a second licensed operator will verify that the correct rod program is followed except as specified in 3.3.B.3.a.

regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize individual control rod worth.

At power levels below 20 percent of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. Section 7.16.5.3 of the FSAR. They serve as a backup to procedure control of control rod sequences, which limit the maximum reactivity worth of control rods. Except during specified exceptions when the Rod Worth Minimizer is out of service, a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20 percent, these devices force adherence to acceptable rod patterns. Above 20 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20 percent of rated power are imposed by power distribution requirements, as defined in Section 3.5.I, 3.5.J, 4.5.I, and 4.5.J of these technical specifications. Power level for automatic bypass of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of ±10 percent of full power the nominal instrument setting is 30 percent of rated power.

Because it is allowable to bypass certain rods in the RSCS during scram time testing below 20% of rated power in the startup or run modes, a second licensed operator is not an acceptable substitute for the RWM during this testing.

4. The Source Range Monitor (SRM) system performs no automatic safety system functions; i.e., it has no scram function. It 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

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 esstential 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 specifications 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

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

1. The RHRS shall be operable:

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

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

2. With the reactor vessel pressure less than 105 psig, the RHR may be removed from service (except that two RHR pumps-containment cooling mode and associated heat exchangers must remain operable) for a period not to exceed 24 hours while being drained of

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

1.

a.	Simulated Automatic Actuation Test	Once/ Operating Cycle
b.	Pump Operability	Once/ month
c.	Motor Operated valve operability	Once/ month
d.	Pump Flow Rate	Once/3 Months
e.	Testable check valve	Once/ operating cycle

Each LPCI pump shall deliver 9,000 gpm against an indicated system pressure of 125 psig. Two LPCI pumps in the same loop shall deliver 15,000 gpm against an indicated system pressure of 200 psig.

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

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

I. Average Planar Linear Heat Generation Rate

During steady state power operation, the Maximum Average Planar 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 through 3.5.I-6. 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.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5 CORE AND CONTAINMENT COOLING SYSTEMS	4.5 CORE AND CONTAINMENT COOLING SYSTEMS
J. Linear Heat Generation Rate (LHGR)	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.	The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.
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.	

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 fully withdrawn)

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

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

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

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.

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 Specifications 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.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

L. Reporting Requirements

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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

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

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

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

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1 through 6. The analyses supporting these limiting values is presented in reference 4.

J. Linear Heat Generation Rate (LHGR)

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

3.5 BASES

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

M. References

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

Plant: BF-3

Fuel Type: Initial Core - Type 1

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.1
15,000	12.3
20,000	12.1
25,000	11.3
30,000	10.2
35,000	9.6

TABLE 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

Fuel Types: 8DRB265L and P8DRB265L

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

The values in this table are conservative for both prepressurized and non-pressurized fuel.

TABLE 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

Fuel Type: P8DRB299

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	10.9
1,000	11.0
5,000	11.5
10,000	12.2
15,000	12.3
20,000	12.2
25,000	11.9
30,000	11.3
35,000	10.9
40,000	10.4
45,000	10.0

TABLE 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

Fuel Type: P8DRB284Z

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.2
5,000	11.7
10,000	12.0
15,000	12.0
20,000	11.9
25,000	11.3
30,000	10.8
35,000	10.4
40,000	9.9
45,000	9.5

TABLE 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3 Fuel Type: P8DRB283 (LTA)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.2
5,000	11.7
10,000	12.0
15,000	12.0
20,000	11.9
25,000	11.3
30,000	10.8
35,000	10.4
40,000	10.0
45,000	9.5

TABLE 3.5.I-6

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3 Fuel Type: P8DRB314 (LTA)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	10.6
1,000	10.7
5,000	11.3
10,000	11.7
15,000	11.5
20,000	11.2
25,000	10.6
30,000	10.1
35,000	9.7
40,000	9.3
45,000	8.8

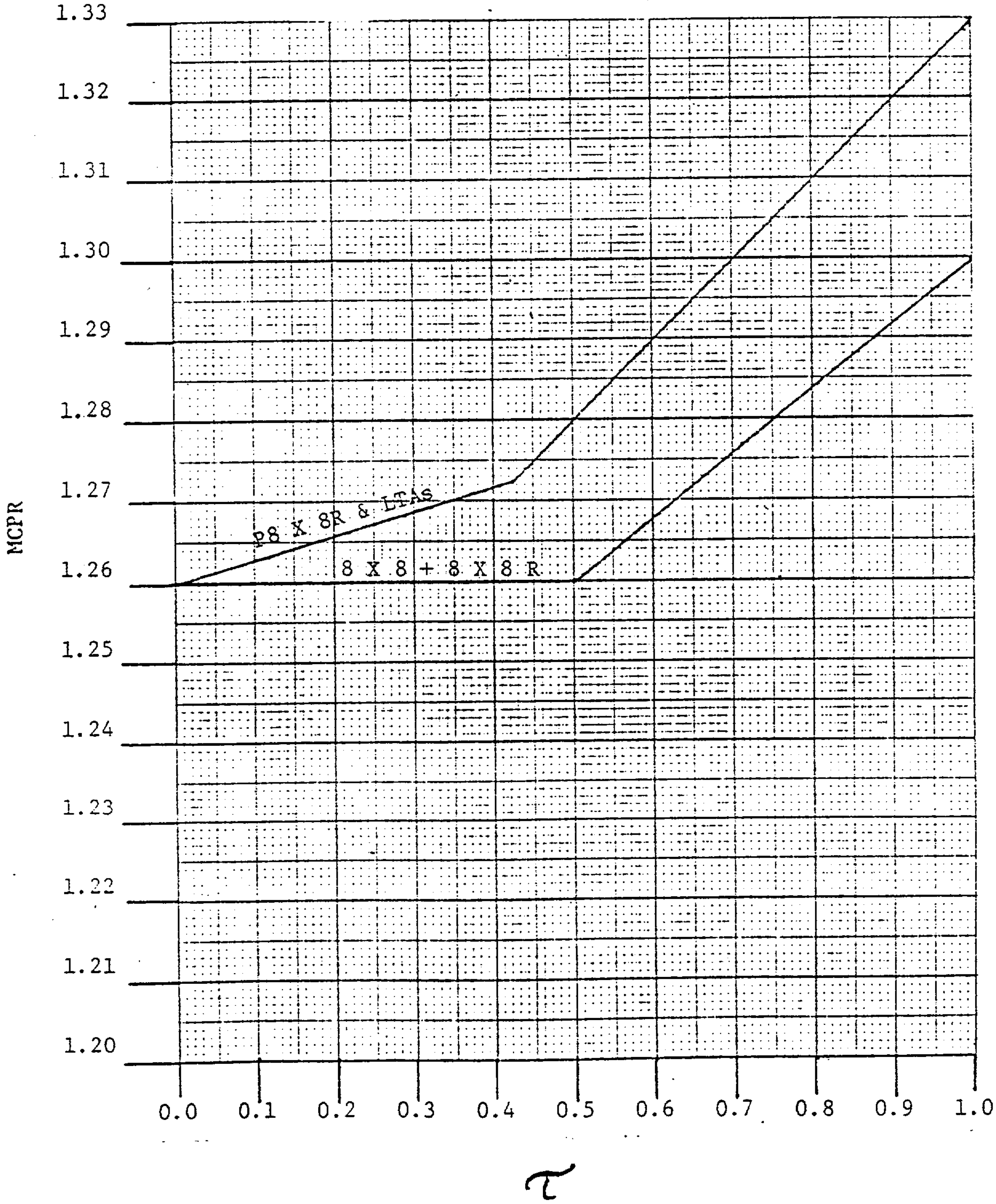

Figure 3.5.K-1
MCPR LIMITS

3.6 PRIMARY SYSTEM BOUNDARY

D. Relief Valves

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

4.6 PRIMARY SYSTEM BOUNDARY

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.

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

3.6 PRIMARY SYSTEM BOUNDARY

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.

4.6 PRIMARY SYSTEM BOUNDARY

F. Recirculation Pump Operation

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

limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shut down to allow further investigation and corrective action.

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

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

REFERENCES

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

3.6.D/4.6.D Relief Valves

To meet the safety basis, thirteen relief valves have been installed on the unit with a total capacity of 83.77% 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.

3.6/4.6 BASES

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

3.6.F/4.6.F Recirculation Pump Operation

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

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

3.6.G/4.6.G Structural Integrity

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

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

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 CONTAINMENT SYSTEMS

c. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

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.

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	$3 < T < 5$	O	GC
1	Main steamline drain isolation valves (FCV-1-55 & 1-56)	1	1	15	O	GC
1	Reactor Water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves (FCV-74-48 & 47)	1	1	40	C	SC
2	RHRS - LPCI to reactor (FCV-74-53 & 67)		2	30	C	SC
2	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	O	GC
2	Drywell floor drain discharge isolation valves (FCV-77-2A & 2B)		2	15	O	GC

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	Outboard	Maximum Operating Time (Sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	2.5	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	5	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	5	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	5	C	SC
6	System Suction Isolation Valves to Air Compressors "A" and "B" (FCV-32-62, 63)		2	15	O	GC
6	Drywell/Suppression Chamber Nitrogen Purge Inlet (FCV-76-24)		1	5	C	SC
6	Torus Hydrogen Sample Line Valves Analyzer A (FSV-76-55, 56)		2	NA	Note 1	SC
6	Torus Oxygen Sample Line Valves Analyzer A (FSV-76-53, 54)		2	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves Analyzer A (FSV-76-49, 50)	1	1	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves Analyzer A (FSV-76-51, 52)	1	1	NA	Note 1	SC
6	Sample Return Valves - Analyzer A (FSV-76-57, 58)		2	NA	O	GC
6	Torus Hydrogen Sample Line Valves Analyzer B (FSV-76-65, 66)		2	NA	Note 1	SC

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (Sec.)	Normal Position	Action on Initiating Signal
6	Torus Oxygen Sample Line Valves-Analyzer B (FSV-76-63, 64)		2	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves-Analyzer B (FSV-76-59, 60)	1	1	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves-Analyzer B (FSV-76-61, 62)	1	1	NA	Note 1	SC
6	Sample Return Valves-Analyzer B (FSV-76-67, 68)		2	NA	0	GC
7	RCIC Steamline Drain (FSV-71-6A, 6B)		2	5	C	SC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	C	SC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-73-6A, 6B)		2	5	0	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

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

TABLE 3.7.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
	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
	Core Spray discharge to reactor check valves (FCV-75-26 and 54)	2		NA	C	Process
6	Drywell ΔP air compressor suction valve (FCV 64-139)		1	10	C	SC
6	Drywell ΔP air compressor discharge valve (FCV 64-140)		1	10	C	SC
6	Drywell CAM discharge valves (FCV 90-257A and 257B)		2	10	0	GC
6	Drywell CAM suction valves (FCV 90-254A and 254B)		2	10	0	GC
6	Drywell CAM suction valve (FCV 90-255)		1	10	0	GC

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 Suction
32-2163	Drywell Compressor Suction
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

AIR TESTED ISOLATION VALVES

Valve	Valve Identification
69-1	RWCU Supply
69-2	RWCU Supply
69-579	RWCU Return
69-624	RWCU Return
71-2	RCIC Steam Supply
71-3	RCIC Steam Supply
71-39	RCIC Pump Discharge
71-40	RCIC Pump Discharge
73-2	HPCI Steam Supply
73-3	HPCI Steam Supply
73-81	HPCI Steam Supply Bypass
73-44	HPCI Pump Discharge
73-45	HPCI Pump Discharge
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 Inlet
76-18	Drywell Nitrogen Purge Inlet
76-19	Suppression Chamber Purge Inlet
76-24	Drywell/Suppression Chamber Nitrogen Purge Inlet
76-49	Containment Inerting
76-50	Containment Inerting
76-51	Containment Inerting
76-52	Containment Inerting
76-53	Containment Inerting
76-54	Containment Inerting
76-55	Containment Inerting
76-56	Containment Inerting
76-57	Containment Inerting
76-58	Containment Inerting
76-59	Containment Inerting
76-60	Containment Inerting
76-61	Containment Inerting
76-62	Containment Inerting
76-63	Containment Inerting
76-64	Containment Inerting
76-65	Containment Inerting
76-66	Containment Inerting
76-67	Containment Inerting
76-68	Containment Inerting
77-2A	Drywell Floordrain Sump
77-2B	Drywell Floordrain Sump
77-15A	Drywell Equipment Drain Sump
77-15B	Drywell Equipment Drain Sump
84-8A	Containment Atmospheric Dilution
84-8B	Containment Atmospheric Dilution
84-8C	Containment Atmospheric Dilution
84-8D	Containment Atmospheric Dilution
84-19	Containment Atmospheric Dilution
84-20	Main Exhaust to Standby Gas Treatment

TABLE 3.7.D

AIR TESTED ISOLATION VALVES

Valve	Valve Identification
84-600	Main Exhaust to Standby Gas Treatment
84-601	Main Exhaust to Standby Gas Treatment
84-602	Main Exhaust to Standby Gas Treatment
84-603	Main Exhaust to Standby Gas Treatment
85-576	CRD Hydraulic Return
90-254A	Radiation Monitor Suction
90-254B	Radiation Monitor Suction
90-255	Radiation Monitor Suction
90-257A	Radiation Monitor Discharge
90-257B	Radiation Monitor Discharge

(DELETED)

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

(This table intentionally left blank)

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep offsite doses well below 10 CFR 100 limits.

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

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indication of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700ft^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 majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

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 of 170°F which is sufficient for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 200°F local. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

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

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

DOCKET NO. 50-296

Author: Dick Clark

1.0 Introduction

By letter dated December 9, 1981 (TVA BFNP TS 170) (1), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-68 for the Browns Ferry Nuclear Plant, Unit No. 3. The proposed amendments and revised Technical Specifications were to: 1) incorporate the limiting conditions for operation associated with the fifth fuel cycle 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 No. 3 (BF-3) shutdown for its fourth refueling on October 30, 1981 with a scheduled restart date of mid-March 1982. The initial core loading for BF-3 consisted of 764 of the single water rod 8x8 fuel assemblies, each containing 63 fuel rods. During the first refueling in September 1978, 208 of the fuel assemblies were replaced with 8x8R fuel assemblies containing 62 fuel rods in each. During the second refueling outage starting in August 1979, an additional 144 of the initial fuel bundles were replaced with P8x8R fuel assemblies, each containing 62 fuel rods. During the third refueling outage, which extended from November 23, 1980 to January 17, 1981, an additional 124 of the original 8x8 fuel assemblies were replaced with a like number of new P8x8R fuel assemblies. The prepressurized fuel assemblies (P8x8R) are essentially identical from a core physics standpoint to the two water rod fuel assemblies (8x8R) except that they are prepressurized with about three atmospheres rather than one atmosphere of helium to minimize fuel clad interaction. During the current refueling outage, an additional 272 of the P8x8R fuel assemblies will be loaded (160 of the P8DRB299 and 112 of the P8DRB284Z) along with 8 lead test assemblies (LTAs). With this reload, all but 8 of the original one-water-rod fuel assemblies will be replaced with improved fuel bundles and these 8 will be symmetrically located at the four peripheral corners of the core.

In support of this reload application for BF-3, TVA submitted with its application of December 9, 1981 a supplemental reload analysis[2] and a revised ECCS analysis[3] prepared by the General Electric Company (GE) for TVA.

This is the first reload for BF-3 in which the pressurization transients are calculated by GE's ODYN Code (in place of analyses previously performed by the REDY Code). Our generic letters of November 4, 1980 and January 29, 1981 (Reference 6) required that any reload submittals received after February 1, 1981 must contain appropriate ODYN analyses. (The most recent reload submittal for Browns Ferry Unit No. 1 was analyzed with the ODYN Code in accordance with our requirement. Unit 1 started up in Cycle 5 on October 1, 1981 after a 6-month outage.)

As noted above, this reload involves loading of prepressurized GE 8 x 8 retrofit (P8 x 8R) fuel. This is the same type of fuel as was loaded during the last reloads for all three Browns Ferry Units. The description of the nuclear and mechanical designs of 8 x 8 retrofit fuel is contained in Reference 4. Reference 4 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of fuel. The use and safety implications of prepressurized fuel are presented in Reference 4 and have been found acceptable per Reference 5 (enclosed in Appendix C of Reference 4).

Values for plant-specific data[a] such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other design parameters are provided in Reference 4. Additional plant and cycle dependent information is provided in the reload application [2] which closely follows the outline of Appendix A of Reference 4. Reference 5 includes a description of the NRC staff's review, approval, and conditions of approval for the plant-specific data. The above-mentioned plant-specific data have been used in the transient and accident analyses provided wth the reload application in compliance with Reference 5.

Our safety evaluation of the GE generic reload licensing topical report has also concluded that the nuclear, and mechanical design of the 8x8R and P8x8R fuels, and GE's analytical methods for nuclear and thermal-hydraulic calculations as applied to mixed cores containing 7x7, 8x8, 8x8R and P8x8R fuels, are acceptable. The staff's safety evaluation on the ODYN Code (Reference 6) concluded that this model more accurately and conservatively predicted pressure, neutron flux and ΔCPR during pressurization transients (e.g., turbine trip) than the REDY Code and was acceptable for analyzing these transients in supplemental reload licensing submittals.

2.2 Reload Evaluation

Because of our previous review of a large number of generic considerations related to use of 8X8R and P8X8R fuels in mixed core loadings, and on the basis of the evaluations which have been presented in Reference 4, only a limited number of additional areas of review needed to be included in this safety evaluation report. The areas evaluated were the proposed operating

limit minimum critical power ratios (OLMCPRs), the proposed MAPLHGR limits, the overpressurization analysis, the stability analyses, the control rod drop analyses, shutdown margins and the loss of coolant accident analyses. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 4.

For Cycle 5, 272 fresh pressurized type P8X8R fuel bundles will be loaded into the core. The remainder of the fuel bundles in the core will be a combination 8X8, 8X8R and P8X8R fuel bundles exposed during the previous four cycles.

The fresh fuel will be loaded and the previously peripheral fuel will be shuffled inward so as to constitute an octant-symmetric core pattern, which is acceptable.

2.2.1 Thermal Hydraulics

Based on the data provided in Sections 4 and 5 of Reference 2, both the control rod system and the standby liquid control system will have an acceptable shutdown capability during Cycle 5.

As stated in Reference 4, for BWR cores which reload with GE's retrofit 8x8R fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients must be equal to at least 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.

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for both the exposed 8X8, 8X8R, and P8X8R fuel and the fresh P8X8R fuel. Addition of the largest reductions in critical power ratio to the safety limit MCPR establishes the operating limits for each fuel type. The transient events analyzed were load rejection without bypass, feedwater controller failure, loss of 100°F feedwater heating, control rod withdrawal error and rotated bundle error. These events were analyzed with the ODYN Code for this reload.

The calculated system responses and reductions in CPR during each of the operational transients have been provided in Sections 9 and 10 of the GE Supplemental Reload Licensing Submittal (Reference 2). On this topic, it is acceptable if fuel specific operating limits are established for prepressurized fuel (Appendix C, Reference 4). On this basis, the transient analysis results are acceptable for use in the evaluation of the operating limit MCPR. Thus, when the reactor is operated in accordance with the proposed operating limit

MCPRs, the 1.07 SLMCPR will not be violated in the event of the most severe abnormal operational transient. This is acceptable along with the proposed changes to the Technical Specifications which incorporate the new OLMCPRs calculated by GE to be necessary to protect the fuel during cycle 5 operation.

2.2.2 ECCS Appendix K

TVA submitted errata and addenda to the BF-3 Loss of Coolant Accident Analysis[3] with this reload application. The analyses were performed for TVA by GE and evaluate the new reload fuel as well as the 8 LTAs. The Maximum Average Planar Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size were calculated by the CHASTE code.

This code is used, with suitable inputs from the other codes, to calculate the fuel cladding heatup rate, peak cladding temperature, peak local cladding oxidation, and core-wide metal-water reaction for large breaks. The detailed fuel model in CHASTE considers transient gap conductance, clad swelling and rupture, and metal-water reaction. The empirical core spray heat transfer and channel wetting correlations are built into CHASTE, which solves the transient heat transfer equations for the entire LOCA transient at a single axial plane in a single fuel assembly. Iterative applications of CHASTE determine the maximum permissible planar power where required to satisfy the requirements of 10 CFR 50.46 acceptance criteria.

The MAPLHGR values and peak clad temperature (PCTs) for each fuel type that will be in the BF-3 core during cycle 5 were presented in reference 3 and submitted as proposed changes to the Technical Specifications in TVA's submittal[1]. The maximum PCT calculated for any fuel assembly was only 1790°F - 410°F less than the 2200°F specified in 10 CFR Part 50.46. This maximum PCT is predicted to occur at 15,000 Mwd/t average planar exposure in an older 8X8 fuel bundle loaded in the first reload. All other fuel assemblies, including the eight LTAs, are calculated to have lower PCTs throughout the entire cycle.

In NUREG-0630 ("Cladding Swelling and Rupture Models for LOCA Analysis" issued April 1980) the staff recommended that all industry ECCS models be revised to incorporate proposed new cladding correlations resulting from the NRC's confirmatory research program. On May 15, 1981, G.E. submitted a generic sensitivity study of fuel rod cladding ballooning and rupture phenomenon during a postulated LOCA. In the generic study, GE assessed the BWR ECCS sensitivity to rupture temperature by using three rupture temperature models: (1) the GE CHASTE model, (2) the NUREG-0630 model, and (3) a proposed GE model termed the adjusted model. For the 8X8 type two-water-rod fuel design, GE found that the use of the adjusted model, which may be the best of the three models and which is in fact a combination of the CHASTE and NUREG models, gave a maximum impact on PCT of $\leq 10^{\circ}$F.

Inasmuch as any combined uncertainties in the GE generic study are very much less than the 410°F minimum available margin in the highest PCT, we conclude that the issues of clad rupture and clad ballooning have been adequately accounted for in the LOCA analysis.

We have reviewed the analyses and information submitted for the reload and conclude that BF-3 conforms with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when it is operated in accordance with the Technical Specifications we are issuing with this amendment.

2.2.3 Control Rod Drop Accident

For Cycle 5, the key plant-specific and cycle-specific nuclear characteristics for the worst case control rod drop accident (CRDA) occurring during both cold and hot startup conditions are conservatively bounded by the values used in bounding CRDA analyses given in Reference 4. The results of G.E.'s analysis are presented in Section 15 of the Supplemental Reload Licensing Submittal [(2)]. The bounding analysis, which includes the adverse effects of fuel densification power spiking, shows that the peak enthalpy will not exceed the 280 cal/gm design limit. Therefore, for Cycle 5 of BF-3, the peak fuel enthalpy associated with a CRDA from the hot and cold startup condition will also be within the 280 cal/gm design limit.

Thus, we conclude that the peak enthalpy associated with a control rod drop accident occurring from any in-sequence control rod movement will be below the 280 cal/gm design limit.

2.2.4 Overpressure Analysis

For Cycle 4, the licensee has reanalyzed the limiting pressurization event (MSIV closure followed by neutron flux scram) to demonstrate that the ASME Boiler and Pressure Vessel Code requirements are met for BF-3. The methods used for this analysis, when modified to account for one failed safety valve, have also previously been approved by the staff. The acceptance criterion for this event is that the calculated peak transient pressure not exceed 100% of design pressure, i.e., 1375 psig. The reanalysis, which is presented in Section 12 of the supplemental reload submittal [(2)], shows that the peak pressure at the bottom of the reactor vessel does not exceed 1272 psig for worse case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is a decrease of 27 psig from the previous fuel cycle and is part of the reason for the changes on pages 30 and 225 of the proposed Technical Specifications. We conclude that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable on the bases outlined in Reference 4.

2.2.5 Thermal Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 4. The results, which are presented in Section 13 of the Supplemental Reload Licensing Submittal [(2)] show that the fuel dependent channel hydrodynamic stability decay ratios and reactor core stability decay ratio at the least stable operating state (corresponding to the intersection of the natural circulation power curve and the 105% rod line) are 0.29 (8X8R/P8X8R), 0.37

(8X8) and 0.79 respectively. These predicted decay ratios are all well below the 1.0 Ultimate Performance Limit decay ratio which we have found acceptable.

Prior to Cycle 3 operation, the staff as an interim measure, added a requirement to the BF-3 Technical Specifications which restricted planned plant operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins during Cycle 5. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of BF-3 during Cycle 5 to be acceptable.

2.2.6 Lead Test Assemblies

During the current refueling, eight lead test assemblies (LTAs) will be loaded in the core. Four test assemblies have an average bundle enrichment of 2.83 weight percent (vs. 2.84 and 2.99 w/o in the regular P8X8R fuel being loaded in the core). The other four test assemblies have an average bundle enrichment of 3.14 weight percent. One of each type will be loaded diagonally across from each other in the same cell in a symmetrical pattern near the center of the core. The locations of the LTAs is shown in Figure 1 of Reference 2. The LTA's incorporate the improved features of a third water rod, increased prepressurization to five atmospheres, larger pellet diameter/thinner cladding/ higher stack density, improved upper tie plate, improved spacer design, axial gadolinia distribution, and barrier fuel.

GE used NRC approved Codes, methods and procedures to evaluate the thermal-hydraulic and nuclear characteristics of the LTAs in this reload. It was also demonstrated that these approved methods are applicable to the LTAs. The results are presented in the Supplemental Reload Submittal (Reference 2). Present methods and procedures were considered to be adequate to evaluate the core response to the Control Rod Drop Analysis (CRDA), Local Rod Withdrawal Error (RWE) and Fuel Loading Errors (FLE) since sufficient nuclear inputs are available to represent the LTA bundles discretely. For RWE and CRDA, because both analyses are performed for the most limiting error rod, the rod adjacent to the LTAs was also analyzed if its worth plus any additional effects so warranted.

For the CRDA evaluation, the rod drop response for the rod adjacent to the LTAs was significantly lower than the nominal error rod and thus LTAs had no impact on the results given in the reload license submittal (Reference 2). For RWE, however, the LTA adjacent rod (although not the limiting rod) had a worth sufficient to warrant a second RWE analysis using the LTA adjacent rod as the error rod. This second RWE analysis is given in an appendix to the reload license submittal (Reference 2).

An evaluation of the potential impact of a rotated bundle error was performed for both the standard reload bundle as well as the LTAs. The results are given in Section 14 and Appendix A of the Supplemental Reload Submittal (Reference 2). TVA has committed to perform additional surveillance during loading activities to preclude a mislocated or misoriented LTA so that the calculated ΔCPR for a FLE on these fuel assemblies will not affect the operating limit MCPR. We find the analyses and special surveillance procedures on the LTAs to be acceptable.

3.0 Changes to Technical Specifications - Reload

Our evaluation of the specific changes to the Technical Specifications resulting from the current reload is presented below:

Pages 18, 24, and 178 - This is the first reload for BF-3 in which the transients were analyzed by GE's ODYN code as required by us. An additional citation is being added to the technical specifications to reference our approval of this code for core reloads.

Page 29 - A reference to the NRC approved GE Generic Reload Topical Report, NEDE-24011-P-A and Addenda, was added to provide further support for the Section 2.1 bases.

Pages vii, 165, 176, 181, 182 and 182a - During the reload 4 refueling, the last of the initial core type-2 fuel assemblies will be removed. Therefore, reference to this fuel type is being deleted. In addition, the exposure limits for the presently installed fuel types have been extended as supported by NEDO-24194A. Finally, MAPLHGR tables have been added for the new fuel types and LTAs loaded in this cycle (i.e., Tables 3.5.I-4, 5 and 6) as discussed in Section 2.2.2 above.

Pages ii, viii, 166, 167, 167a, and 182b - As supported by the reload submittal, the operating limit MCPRs are being changed. Since the MCPRs were determined by the ODYN code (rather than the REDY code), the OLMCPRs are now calculated from two curves rather than being a single value (or a ramp change with fuel exposure). Our evaluation was covered in Section 2.2.1 above.

Pages 123, 124, 126, and 133 - As discussed in Section 2.2.6 above, eight lead test assemblies (LTAs) will be loaded. In order to obtain additional physics data, special cold criticality tests have been planned for this cycle. These criticality tests require suspension of the rod sequence control system (RSCS) constraints by means of the individual rod bypass switches. This testing is planned as part of the Lead Test Assembly Program in which TVA and GE are participating.

The RSCS is a backup to the Rod Worth Minimizer (RWM). It independently imposes restrictions on control rod movement to mitigate the effects of a postulated rod drop accident. The RWM, in turn, serves as a backup to procedural controls to limit control rod worth during startup and low power operation. During low power and startup operation, unrestrained rod patterns can create rods of sufficient worth to exceed design limits on a Rod Drop Accident (RDA).

Neither the RSCS or RWM are required at high power, so both systems are bypassed at greater than 20% power. The RWM is a computer monitoring system which minimizes individual control rod worths by blocking rod movement if the existing control rod pattern deviates from a specific sequence. The sequences are developed by the Plant Nuclear Engineers and loaded into the RWM memory. Actual rod positions are obtained for comparison to the sequence from the Rod Position Information System. Rod movement sequences are developed to limit rod worth to a level below which, if an RDA were to occur at a free-fall rate limited by the velocity limiter, the fuel enthalpy from the transient would be less than 280 cal/gm.

The RSCS also imposes restrictions on control rod movements to reduce rod worths, thus reducing the consequences of a postulated RDA. As such, it is a backup to and complements the RWM. The components of the RSCS are grouped as belonging to the Sequence Control Mode or the group Notch Control Mode of operation. The Sequence Control Mode controls rod movement from rods full in to the 50% rod density by imposing rod select blocks. The Group Notch Control Mode controls rod movement from the 50% rod density level to the 30% power bypass by imposing rod withdrawal and insert blocks.

The RSCS circuitry does allow limited manual bypass cabability. In the Sequence Control Logic, the full in <u>or</u> fully out position for each rod can be bypassed (rod simulated as being full in <u>or</u> full out). This is necessary for scram time surveillance and system surveillance. The present Technical Specifications (Section 3.3.B.2) require that the RSCA shall be operable whenever the reactor is in the startup or run modes below 20% rated power. Since the RSCS is a backup to RWM, the present Technical Specifications permit (Section 3.3.B.3.C) a second licensed operator to verify that the operator at the reactor console is following the control rod program below 20% rated power if the RWM is inoperable (except during scram time testing). As noted above, the proposed change to the Technical Specifications to be able to test the LTAs is to permit the RSCS restraints to be suspended by means of the individual rod bypass switches for special criticality tests or control rod scram timing. If the RSCS is bypassed, the RWM must be operable (i.e., a second licensed operator could not substitute for the RWM).

The purpose of both the RSCS and RWM is to mitigate the effects of a postulated rod drop accident. Since the RSCS is a backup to the RWM, it is reasonable to permit one or the other to be bypassed under certain controlled conditions as long as the other system is operational. Furthermore, the constraints imposed by the RSCS and RWM are the results of analyses programmed into the process computer. For the LTAs, TVA performed an analysis to show that a postulated rod drop accident involving control rods withdrawn during the cold critical test would not exceed the peak fuel enthalpy design limit of 280 cal/gm. The rod worth minimizer (RWM) will be programmed to ensure adherence to the withdrawal sequence specified in the cold critical test procedure. The RWM must be operable for this test; a second licensed operator may not be used in lieu of the RWM for this testing.

Based on the analyses and the compensatory actions to be taken when the RSCS is bypassed (i.e., RWM system operable), the proposed changes to the Technical Specifications are acceptable. During the September 1981 reload for Browns Ferry Unit 1, four LTAs were placed in the core. The proposed changes to the Technical Specifications for BF-3 are the same as those we approved for BF-1 by Amendment No. 76 to the Unit 1 license on September 15, 1981. BF-1 started up on October 1, 1981. Our experience with the BF-1 startup program reinforces the acceptability of the proposed controls and changes to the BF-3 Technical Specifications.

4.0 <u>Plant Modifications</u>

4.1 <u>Discussion</u>

BF-3 shutdown for the present refueling and maintenance outage on October 30, 1981 is projected to be down for over five 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, TVA expects to complete 63 of the 300 plus 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 for convenience.

4.2 Evaluation

(a) Torus Modifications

On January 13, 1981 the Commission issued an Order modifying the BF-3 license to require TVA 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 TVA was required to perform to conform the design to the staff's Acceptance Criteria in NUREG-0661. The specific changes to the Technical Specifications are discussed below.

Pages 231 and 285 - 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.

Pages 246 and 286 - 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.

Page 286 - In the bases for the limits established for primary containment, there is a discussion of steam condensing loads associated with relief valve operation. The peak temperature of the torus water used in the evaluation is being changed from 160°F to 200°F local temperature. 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 Technical Specifications are being changed to conform to the staff's acceptance criteria in NUREG-0661 to avoid excessive steam condensing loads and are therefore acceptable.

(b) Replacement of Safety Valves

During the present refueling outage, the two presently installed main steam line safety valves are being replaced with two-stage Target Rock safety/relief valves (SRVs) identical to the other 11 SRVs. Thus, BF-3 in the future will have 13 SRVs. The capacities of these valves were factored into GE's "Overpressurization Analysis" in Section 12 of the Supplemental Reload Submittal (Reference 2) as discussed in Section 2.2.4 of this Safety Evaluation.

The value of 83.77 percent total relief capacity is derived from the values of 77.33 percent for 12 SRVs operable out of a total of 13 SRVs. The capacity of 77.33 percent of nuclear boiler rated steam flow, as listed in the BF-3 Reload 4 Supplemental Licensing Submittal, was calculated based on certified valve capacity for a 5.125-inch throat diameter valve (870,000 lbs/hour at 1,090 +3 psig) issued by the ASME National Board of Boiler and Pressure Vessel Inspectors. The certified values are obtained by testing and are listed as 90 percent of the measured capacity values for conservatism. As noted in Section 2.2.4, the licensee's analysis of the limiting overpressure event is acceptable. Since the number, type, and capacities of the SRVs are specified in the Technical Specifications (and bases thereto), changes need to be made on pages 26, 27, 30, 192, 224 and 225 to reflect 13 rather than 11 SRVs and the 83.77 percent capacity.

(c) Containment Vent and Purge Modifications

Our letters of November 29, 1978, September 27, 1979, and October 22, 1979 to all licensees identified concerns regarding containment venting and purging during normal operation. All licensees were requested to implement certain corrective actions and to evaluate their systems with respect to our positions in Standard Review Plan Section 6.2.4 Revision 1 and Branch Technical Position CSB 6-4 Revision 1. The TMI Action Plan Requirements, NUREG-0737, Item II.E.4.2, "Containment Isolation Dependability," imposed additional requirements on the design of containment systems. In our letter to TVA of December 17, 1981 we advised TVA that except for certain areas in which our review had not been completed (e.g., environmental qualification) TVA's proposed actions and modifications satisfactorily resolved Multiplant Action B-24 (Venting and Purging

Containments While at Full Power) and NUREG-0737 Items II.E.4.2.1 through II.E.4.2.5, provided certain testing requirements were included in the Technical Specifications. Some of these requirements were included in TVA's submittal of December 9, 1981.

In response to our generic letters of September 27, 1979 and October 22, 1979 to licensees of all light water reactors, TVA is modifying the containment purge system for BF-3 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 263 and 264 are being revised to reflect the significant reduction in the maximum allowable operating time for the purge valves. On the nitrogen purge valves, the operating time is being reduced from 10 seconds to 5 seconds and on the purge inlet and exhaust isolation valves, the operating time is being reduced from 100 and 90 seconds, respectively, 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. The changes to the Technical Specifications are those specified in our letter of December 17, 1981 and are acceptable.

(d) Primary Containment Isolation Valves

Tables 3.7.A through 3.7.H list the various valves 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, TVA has proposed revisions to these tables to reflect plant modifications and the requirements in NUREG-0737 Item II.E.4.2.

Table 3.7.A

Page 262 - FCV-1-55 and 1-56 drain valves are required to be open for extended periods during power operation. Therefore, these valves will be considered as normally open and technical specification surveillance requirement 4.7.D.1.b will apply.

Page 263 - TVA has proposed to delete valve FCV-69-12 on the Reactor Water Cleanup System from Table 3.7.A. This valve is not a containment isolation valve. Isolation is provided by check valves 69-579 and 3-572. Based on our review, we find this acceptable.

Page 263 - FCV-73-81, the bypass valve around the HPCI steam supply outboard isolation valve (FCV-73-3), was added to BF-3 during the 1980 refuel outage. During quarterly surveillance testing on HPCI isolation valve FCV-73-3 in which the valve is closed and reopened, the steamline downstream from FCV-73-3 is subject to thermal stresses from the closure and subsequent reopening. FCV-73-81 was added to relieve those thermal stresses. This is a one-inch

valve. It is an isolation group 4 valve with a maximum closure time of 10 seconds. Since the valve was added to the HPCI system, it must be periodically verified as being operable. The addition of this valve to Table 3.7.A is both necessary and acceptable.

Pages 263, 264 and 264A - During the 1980 refueling outage on BF-3, the hydrogen-oxygen analyzer system was replaced with the new Hays-Republic Hydrogen-Oxygen analyzer system. Our safety evaluation for this new system was covered in Amendment No. 37 to Facility License No. DPR-68 for BF-3 issued January 12, 1981. All of the system 76 (containment inerting system) valves were installed in the plant during the last outage as part of the new hydrogen-oxygen monitoring system. These valves are being added to Table 3.7.A to require periodic verification that these valves are operable. We have reviewed the changes and find them acceptable.

Page 264 - System suction isolation valves to the drywell air compressors "A" and "B" trains, FCV-32-62 and FCV-32-63, have been installed in the plant as part of a system modification. The drywell air compressors supply air to the air-operated valves in containment. These valves are being added to Table 3.7.A to reflect this modification and to insure that their operability is periodically verified. The proposed change is acceptable.

Page 264A - The normal position of FCV-71-7A, 7B is closed rather than open. The normal position and action on receiving an initiating signal are being changed to show the correct positions.

Page 265 - The core spray discharge to reactor check valves FCV-75-26 and FCV-75-54 should be included in this table. They are primary containment isolation valves and were inadvertently omitted from this table.

Page 265 - It is proposed to add to Table 3.7.A valves FCV-64-139 and FCV-64-140. These are the drywell ΔP air compressor suction and discharge valves. They are containment isolation valves and should be verified for operating time. These valves are a part of the addition of the drywell pressurization system. Their addition to Table 3.7.A is acceptable.

Page 265 - The following valves are being added to Table 3.7.A because these valves were inadvertently omitted in the original Technical Specifications. These valves are all containment isolation valves and need to be included in Table 3.7.A to ensure that their required operating times are periodically tested.

FCV-90-254A and B, drywell CAM suction valves
FCV-90-257A and B, drywell CAM discharge valves
FCV-90-255, drywell CAM suction valve

Tables 3.7.D through 3.7.G

TVA is proposing to revise Tables 3.7.D through 3.7.G to be more consistent with the BWR Standard Technical Specifications (NUREG-0123, Rev. 3 issued Fall 1980). These tables presently list a "Test Medium" (i.e., air, water

or nitrogen) and a "Test Method" (i.e., the specific valves between which the test medium is to be applied). The Standard Technical Specifications (Table 3.6.3-1) do not specify a test method, since this is more appropriately left to the pump and valve testing procedures. TVA has proposed to include the test medium in the title of the tables (i.e., separate tables for those valves to be tested by air vs. water or nitrogen) and to delete the test method. We find the proposed changes are consistent with the BWR Standard Technical Specifications and are therefore acceptable.

Table 3.7.D (presently p277, proposed pgs 270-272)

The following valves have been added to table 3.7.D:

Valve 76-66 on the new Hays-Republic H_2-O_2 analyzer
Valve 73-81, bypass valve around HPCI outboard isolation valve 73-3

Test connections were added to BF unit 3 so that the following valves could be tested.

FCV-2-1192	Service Water
FCV-2-1383	Service Water
FCV-33-1070	Service Air
FCV-33-785	Service Air
FCV-68-508	CRD to RC Pump Seals
FCV-68-523	CRD to RC Pump Seals
FCV-68-550	CRD to RC Pump Seals
FCV-68-555	CRD to RC Pump Seals

Valves 74-54 and 74-68 have been added to table 3.7.D due to inadvertently omitting them from this table. They are tested and should be included in the Technical Specifications.

Valves 76-215 to 76-254 have been deleted from table 3.7.D due to the replacement of the H_2-O_2 analyzer system. Valve 64-141 has also been deleted because it is not an isolation valve and is not tested. Valve 85-573 has been omitted from this table due to a plant modification that eliminated the containment penetration for this CRD return line.

Table 3.7.E (p 279)

Table 3.7.E has been revised to include valve 2-1143 on the demineralized water line into the torus, which is now required to be tested, since it is an isolation valve.

(e) NUREG-0737, Item II.K.3.15

TMI Action Plan item II.K.3.15 requires licensees of BWR's to modify pipe-break-detection circuitry so that pressure spikes resulting from HPCI and RCIC initiation will not cause inadvertent system isolation. TVA elected to employ the BWR Owner's Group modification which incorporates a three-second time delay relay (TDR) to prevent spurious isolation. In our letter to TVA 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 TVA'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-3, 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 TVA's application of December 9, 1981. Table 4.2.B (p95) 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 Specifications reflect the surveillance requirements requested in our letter of October 13, 1981 on item II.K.3.15 and are acceptable.

5.0 Administrative Changes

Browns Ferry and other BWRs are not presently permitted to operate in the natural circulation mode (i.e., without one of the recirculation pumps in operation). This restriction is presently a paragraph in Section 2.1 (top of pg 19) which contains the bases for the "Limiting Safety System Settings Related to Fuel Cladding Integrity." This paragraph is being moved, verbatim, to the bases for recirculation pump operation on page 227, which is a more appropriate location. There is no safety significance to this reformatting of the Technical Specifications. Also, the title of Section 3.6.F (p 195), which is presently entitled "Jet Pump Flow Mismatch," is being changed to "Recirculation Pump Operation" which is what the section encompasses.

Section 3.5.A.1.(2) of the Technical Specifications on the Core Spray System (page 146) and Section 3.5.B.1.(2) on the Residual Heat Removal System (page 149) presently contain references to non-applicable sections of the Technical Specifications. Since these changes do not affect any actual limiting conditions for operation, plant safety is not affected and the non-applicable references are being removed.

Section 3.5.L of the Bases specifies the reporting requirements if any of the thermal-hydraulic limits associated with fuel rod integrity (e.g., MAPLHGR, LHGR or MCPR limits) are exceeded. The present Technical Specifications (pages 177 and 178) require that "Each event involving steady state operation beyond a specified limit shall be logged and reported quarterly." Actually, TVA has been notifying NRC promptly of any such incident and has been filing a 30 day LER as for other incidents. This section is being changed to require a report within 30 days. Also, on page 178, an additional reference is being added to reflect that this reload was analyzed by the ODYN Code.

6.0 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.

7.0 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 accidents previously considered and does not involve a significant decrease 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 29, 1982

References

1. Letter, L. M. Mills, TVA to Harold R. Denton, NRC, dated December 9, 1981.

2. "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 3, Reload 4 (Cycle 5)," Y1003J01A30, dated November 1981.

3. Errata and Addenda sheets dated September 1981 to NEDO-24194A, "Loss of Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, dated July 1979.

4. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A, August 1979.

5. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE), April 16, 1979, and enclosed SER.

6. Generic letter 81-08 to all holders of Construction Permits and Operating Licenses for Boiling Water Reactors, Subject: ODYN Code, dated January 29, 1981 forwarding the staff's "Safety Evaluation for the General Electric Topical Report, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDE-24154-P, Volumes I, II and III," dated June 1980.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 51 to Facility Operating License No. DPR-68 issued to the Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit 3, located in Limestone County, Alabama. The amendment is effective upon startup of Unit 3 in the fifth fuel cycle.

This amendment changes the Technical Specifications to: (1) incorporate the limiting conditions for operation during the fifth fuel cycle and (2) (a) impose minimum torus water level requirements and drywell differential setpoints (b) reflect changes in safety relief valves (c) reflect reduced valve closure times for the containment purge system (d) reflect changes to the primary containment isolation valves, (e) impose surveillance requirements on certain valves, (f) impose surveillance and calibration requirements for pipe-break detection circuitry and (g) make administrative changes to the bases for certain technical specifications and to reporting requirements, all reflective of 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 this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated December 9, 1981 (2) Amendment No. 51 to License No. DPR-68, 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, N. W., 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 29th day of March 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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