December 1, 1983

Docket No. 50-259

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. 91 to Facility License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. This amendment changes the Technical Specifications in partial response to your application of July 13, 1983 (TVA BFNP TS 190), as supplemented by your submittal of July 21, 1983.

The amendment revises the Technical Specifications to incorporate the limiting conditions for operation during Cycle 6.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

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

Enclosures:
1. Amendment No. 91 to DPR-33
2. Safety Evaluation

cc w/enclosures
See next page

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Tennessee Valley Authority
Browns Ferry Nuclear Plant, Units 1, 2 and 3

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 13, 1983, as supplemented by letter dated July 21, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical specifications contained in Appendices A and B, as revised through Amendment No. 91, are hereby incorporated in the license. The licensee shall operate the facility in accordance witht he Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 1, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 91

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

vii	37
8	168
10	168a
20	169
22	172
30	172a
33	172b
36a	219

2. The marginal lines on these pages indicate the revised area.

LIST OF TABLES (Cont'd)

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4.2.G	Surveillance Requirements for Control Room Isolation Instrumentation	106
4.2.H	Minimum Test and Calibration Frequency for Flood Protection Instrumentation	107
4.2.J	Seismic Monitoring Instrument Surveillance	108
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3.5.I	MAPLHGR Versus Average Planar Exposure	171, 172, 172a
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1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specifications

A. Thermal Power Limits

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

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

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

   a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

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

   where:

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

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

1.1 FUEL CLADDING INTEGRITY

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

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

2.1 FUEL CLADDING INTEGRITY

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

$S \leq 120$ % power .

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

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

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

2.1 BASES

In summary

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

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

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

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

The bases for individual set points are discussed below:

A. Neutron Flux Scram

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

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

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

IRM Flux Scram Trip Setting (Continued)

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

4. Fixed High Neutron Flux Scram Trip

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

B. APRM Control Rod Block

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

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

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

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

Amendment No. 91

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

Min. No. of Operable Inst. Channels Per Trip System(1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shut-down	Refuel(7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16) (24)(25)						
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux (Fixed Trip)	≤120 %				X	1.A or 1.B
2	High Flux	≤15% rated power		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative	(13)		X(21)	X(17)	X	1.A or 1.B
2	Downscale	≥3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥538" above vessel zero		X	X	X	1.A
2	High Water Level in West Scram Discharge Tank (LS-85-45A-D)	≤50 Gallons	X	X(2)	X	X	1.A
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤50 Gallons	X	X(2)	X	X	1.A

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

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

Amendment No. 91

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

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekl When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/Week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level (5)	A	Trip Channel and Alarm	Once/Month (1)
High Water Level in Scram Discharge Tank Float Switches (LS-85-45C-F)	A	Trip Channel and Alarm	Once/Month
High Water Level in Scram Discharge Tank Electronic Level Switches (LS-85-45A, B, G, H)	B	Trip Channel and Alarm (7)	Once/Month
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/Week

3.5 BASES

H. Maintenance of Filled Discharge Pipe

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

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

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

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

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

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, -2, -3, -4, -5, and -6. The analyses supporting these limiting values is presented in Reference 4.

3.5.J Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the, MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K Minimum Critical Power Ratio (MCPR)

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

3.5.L APRM Setpoints

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

3.5.M Reporting Requirements

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

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265H and P8DRB265H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265L and P8DRB265L

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

Table 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: P8DRB284L, GLTA-1, GLTA-2

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

Table 3.5.I-6

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: P8DRB284Z

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

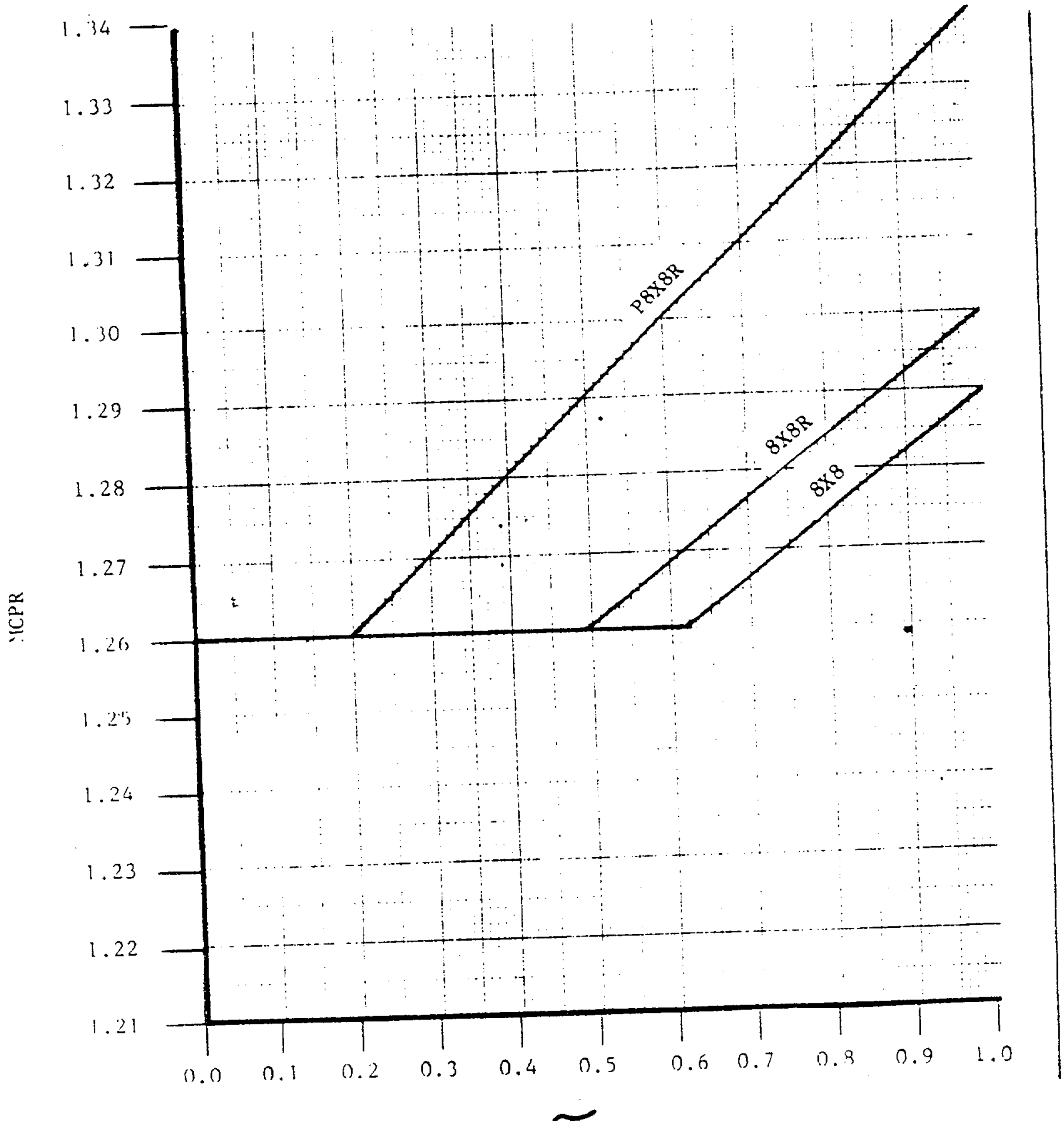

Figure 3.5.K-1

MCPR LIMITS*

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

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

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

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

REFERENCES

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

3.6.D/4.6.D Relief Valves

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGUALTION

SUPPORTING AMENDMENT NO. 91 TO FACILITY LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 Introduction

By letter dated July 13, 1983 (TVA BFNP TS 190) (References 1 and 2) and supplemented by letter dated July 21, 1983, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. The proposed amendment and revised Technical Specifications would (1) incorporate the limiting conditions for operation of the facility in the sixth fuel cycle following the fifth refueling of the reactor and (2) reflect modifications performed during the outage. This amendment addresses the changes to the Technical Specifications associated with the core reload; the changes associated with the modifications will be addressed separately.

2.0 Discussion

Browns Ferry Unit 1 (BF-1) shutdown for its fifth refueling on April 16, 1983. BF-1 was initially fueled with 764 of the General Electric Co. (GE) 7 x 7 fuel assemblies containing 49 fuel rods each. During the first refueling, 166 of the 7 x 7 fuel assemblies were replaced with a like number of one water rod 8 x 8 fuel assemblies containing 63 fuel rods each. During the second refueling, an additional 156 of the original fuel assemblies were replaced with two water rod retrofit 8 x 8R fuel bundles containing 62 fuel rods each. During the third refueling outage, another 232 of the 7 x 7 fuel bundles were replaced with P8 x 8 fuel assemblies, each containing 62 fuel rods. A total of 260 new fuel assemblies were loaded in the core during the fourth refueling outage in April 1981. During this fourth refueling, all of the remaining 214 original 7 x 7 fuel bundles were replaced along with 46 of the 8 x 8 fuel assemblies from the second reload. During this same refueling, four lead test assemblies were also loaded in the core. The four lead test assemblies (LTAs) (two GLTA-1 and two GLTA-2) are exactly the same as the standard P8DRB284L (P8 x 8R) reload bundle fuel except for a small axial section of increased Gadolinia content in some rods. Test measurements were performed on these bundles during Cycle 5 to benchmark the effect of this increased Gadolinia content.

During this current reload, 252 new P8 x 8R fuel assemblies are being loaded in the core.

The Cycle 6 core consists of 17 8x8D (one water rod) fuel assemblies from the second reload, 743 8x8DR (two water rod) assemblies and 4 lead test assemblies. The 8x8D and 8x8DR assemblies are of the GE design described in Reference 3 and have been approved by the staff for use in boiling water reactors. The lead test assemblies were inserted for Cycle 5 and their continued presence is acceptable. The staff concludes that no further review of the fuel mechanical design is required.

The LOCA analyses report has been revised for Cycle 6 and includes the MAPLHGR values calculated for the 8x8 fuel types (including the LTAs). On the basis that approved calculation methods and procedures have been used to perform the analysis we concluded that these are acceptable.

3.0 Evaluation

3.1 Nuclear Design

Except as noted below, nuclear design and analysis were performed with the methods and procedures described in Reference 3 which has been approved by the staff for use in reload applications. The nuclear parameters for Cycle 6 are within the range of those normally obtained and are acceptable.

The core cold, clean, effective multiplication factor, shutdown margin, and standby liquid control system capability were calculated by TVA with methods which have been approved by the staff (Memo, Denise to Gammill, dated August 27, 1979) and we find these analyses to be acceptable.

3.2 Thermal-Hydraulic Design

The objective of the review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients, and is not susceptible to thermal-hydraulic instability.

The review includes the following areas: (1) safety limit minimum critical power ratio (MCPR), (2) operating limit MCPR, and (3) thermal-hydraulic stability.

The licensee has submitted the analysis report for Cycle 6 operation at 105% of rated core flow conditions (Reference 2). Operation at this higher core flow condition was approved by the staff in Reference 5. Discussion of the review concerning the thermal-hydraulic design for Cycle 6 operation follows:

Safety Limit MCPR

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in Reference 3, the approved safety limit MCPR is 1.07. The safety limit MCPR of 1.07 is used for BF-1 Cycle 6 operation.

Operating Limit MCPR

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (CPR). The CPR values given in Section 10 of Reference 2 are plant-specific values calculated by using approved methods including ODYN methods. The calculated CPRs are adjusted to reflect either Option A or Option B CPR by employing the conversion methods described in Reference 4. The MCPR values are determined by adding the adjusted CPRs to the safety limit MCPR. Section 12 of Reference 2 presents the cycle MCPR values of both the pressurization and non-pressurization transients. The maximum cycle MCPR values (Options A and B) in Section 12 are specified as the operating limit MCPRs and incorporated into the Technical Specifications. We find that the approved method was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated transients. We therefore conclude that these limits are acceptable.

Thermal-Hydraulic Stability

The results of thermal-hydraulic analyses (Reference 2) show that the maximum core stability decay ratio is 0.87, which is the same as that of the previously approved Cycle 5 core. Since the calculated maximum core stability decay ratio is less than that of some of the operating plants (for example, Peach Bottom Units 2 and 3 have decay ratios of 0.98) and since additional stability margin is assured by Technical Specification restrictions which prevent operation in the natural circulation mode, we conclude that the thermal-hydraulic stability results remain acceptable for Cycle 6 operation.

3.3 Transient and Accident Analayses

Rod Withdrawal Error

The licensee has elected to use the generic bounding analysis described in Reference 3 for this event. That analysis has been accepted by the staff (Memo, Rubenstein to Lainas, dated February 15, 1983) and we find its use acceptable for Browns Ferry. The RBM output is signal clipped at 106% power in order to permit operation at 105% core flow.

Fuel Loading Error

This event has been analyzed by the methods described in Reference 3 which has been approved by the staff (Letter to Gridley, GE, from Eisenhut, NRC, dated May 12, 1978) and we find its use acceptable for Browns Ferry. This event is not limiting for Cycle 6.

Control Rod Drop Accident

A cycle specific analysis has been performed for this event using approved methods described in Reference 3. The resultant peak fuel enthalpies meet our acceptance criterion for this event and are therefore acceptable.

3.4 Technical Specification Changes

The Bases for Relief Valves (Bases 3.6.D/4.6.D) are being revised to reflect the full flow equivalent of the flow (12 out of 13 valves) assumed in the safety analysis. This is acceptable.

The operating limit MCPR values are being altered to conform to the results of the safety analysis for Cycle 6 and are acceptable. Additional MAPLHGR curves for the new fuel being inserted for this cycle are included. These are acceptable changes.

Changes have been made in the Technical Speficiation to reflect the addition of a flow biased thermal power monitor and a fixed high level neutron flux trip to the Reactor Protection System. These changes have been made on several BWRs and are acceptable for BF-1.

3.5 Summary

We conclude that the analysis of the Cycle 6 reload (Reload 5) for BF-1 is acceptable and that the reactor may be reloaded and operated for Cycle 6 without undue risk to the public health and safety. This conclusion is based on the following:

1. The fuel mechanical design is the current standard design for GE reactors and has been previously reviewed and accepted.

2. The nuclear and thermal-hydraulic design analyses have been performed by previously approved methods and the design parameters are within the range expected for GE reactors.

3. The results of the cycle specific transients and accident analyses meet applicable criteria.

4. The proposed Technical Specifications are consistent with the reloaded core and with the results of the analyses.

4.0 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.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: December 1, 1983

Principal Reviewers: W. Brooks, A. Gill

References

1. TVA BFNP TS 190, Letter to H. R. Denton, NRC from L. M. Mills, Tennessee Valley Authority, dated July 13, 1983, with supplemental information submitted in a letter from D. S. Kammer, TVA, to H. R. Denton, NRC, dated July 21, 1983.

2. General Electric report 22A8559, "Supplemental Reload Licensing Submittal for Browns Ferry Unit 1, Reload 5", dated 1983, submitted with TVA's letter of July 13, 1983.

3. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (US), GESTAR-II, January 1982.

4. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," GE Report NEDE-24154-P, October 1978.

5. Memorandum from L. S. Rubenstein (NRC) to G. C. Lainas (NRC), "Browns Ferry-1 Technical Specification Changes Regarding Core Flow Conditions Above Rated Flow During Cycle 5 (TACS 49718)", March 10, 1983.