July 11, 1984

Docket No. 50-296

Mr. Hugh G. Parris
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Tennessee Valley Authority
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Dear Mr. Parris:

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. DPR-68 for the Browns Ferry Nuclear Plant, Unit 3. This amendment changes the Technical Specifications in partial response to your request of January 23, 1984 (TVA BFNP TS 195) related to a reload associated with Fuel Cycle 6 operation.

This amendment revises the Technical Specifications to (1) incorporate the limiting safety systems settings and limiting conditions for operation during the sixth cycle and (2) reflect changes resulting from thermal power monitor modifications made during the current refueling outage.

Technical Specifications changes related to other modifications will be the subject of another amendment.

A copy of the Safety Evaluation is enclosed.

Sincerely,

Richard J. Clark

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

Enclosures:
1. Amendment No. 70 to License No. DPR-68
2. Safety Evaluation

cc w/enclosures:
See next page

DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:ORB#2	OELD	DL:AD-OR
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OK - with noted change to SER

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70
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 January 23, 1984 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 Operating 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. 70 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 11, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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.

vii	165
viii	176
9	178
12	182b
18	
19	
20	
21	
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2. The marginal lines on each page indicate the revised area.

3. Add the following new pages:

35A
182c

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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)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1 FUEL CLADDING INTEGRITY e. Fixed High Neutron Flux Scram Trip Setting - When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be: $S \leq 120\%$ power
2. Reactor Pressure $\leq$ 800 PSIA or Core Flow $\leq$ 10% of Rated When the reactor pressure is $\leq$ 800 PSIA or core flow is $\leq$ 10% of rated, the core thermal power shall not exceed 823 MWt (~25% of rated thermal 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: 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.

The void reactivity coefficient and the scram worth are described in detail in reference 1.

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 1. 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.

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.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of CMFLPD and FRP. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the CMFLPD exceeds FRP.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR > ***.

2. <u>APRM Flux Scram Trip Setting (Refuel or Start & Hot Standy Mode)</u>

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual rods is very low in a uniform rod pattern. Thus, all of possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. <u>IRM-Flux Scram Trip Setting</u>

The IRM System consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a

*** See Section 3.5.K.

5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch, and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 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. The APRM 15-percent scram will prevent 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 7.5.5.4 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 from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during the steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the CMFLPD exceeds FRP thus preserving the APRM rod block safety margin.

C. Reactor Water Low Level Scram and Isolation (Except Main Steamlines)

The set point for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 31 inches below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure- neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2).

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. Relevant transient analyses are discussed in References 1 and 2. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by the turbine first stage pressure.

F. Main Condenser Low Vacuum Scram

To protect the main concenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vaccum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 850 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP 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. Reactor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps

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

L. References

1. "BWR Transient Anlaysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

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

1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressure (1,148 psig for suction and 1,326 psig for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients are added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the reload licensing submittal for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.2 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

Amendment No. 77, 78, ~~56~~, 70

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	Inoperative			X	X	(5)	1.A
	APRM (16)(24)(25)						
2	High Flux (Fixed Trip)	≤ 120 percent				X	1.A or 1.B
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15 percent rated power (13)		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative			X(21)	X(17)	X	1.A or 1.B
2	Downscale	≥ 3 indicated on scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure	≤ 1055 psig		X (10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X (8)	X (8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A
2	High Water Level in Scram Discharge Tank	≤ 50 Gallons	X	X (2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure		X (3) (6)	X (3) (6)	X (6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure or Turbine Trip	≥ 550 psig				X (4)	1.A or 1.D

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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. 54, 70

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

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure	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	A	Trip Channel and Alarm	Once/Month
Turbine Condenser Low Vacuum	A	Trip Channel and Alarm	Once/Month (1)

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-7. 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.

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 7. The analyses supporting these limiting values is presented in reference 1.

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 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.

M. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

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

TABLE 3.5.1-7

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3

Fuel Type: BP8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.9
25,000	11.3
30,000	10.8
35,000	10.1
40,000	9.4
45,000	8.8

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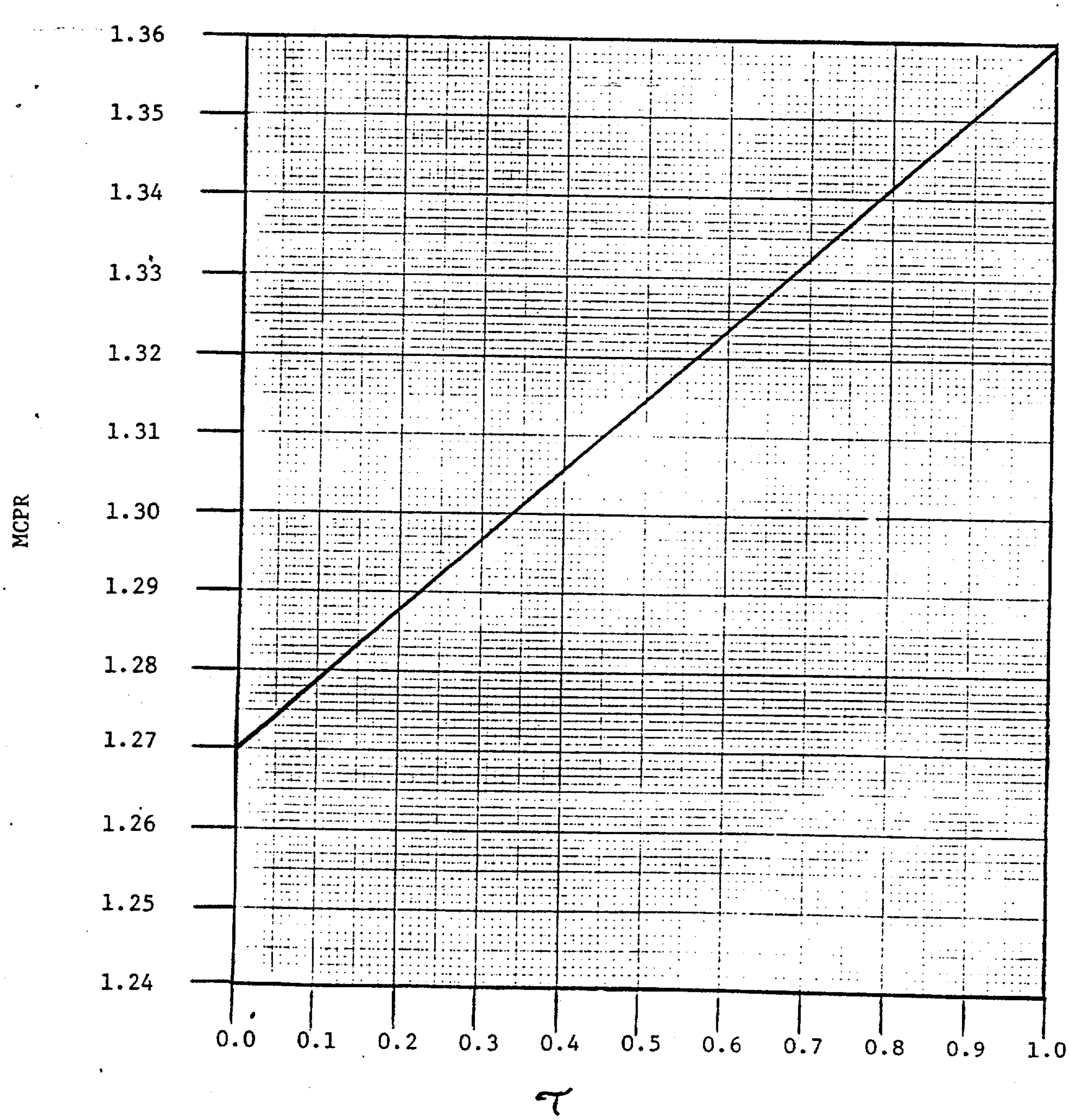

Figure 3.5.K-1

MCPR Limits for 8x8R, P8x8R and LTAs

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

DOCKET NO. 50-296

1.0 Introduction

By letter dated January 23, 1984 (TVA BFNP TS 195), 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 3. The proposed amendments and revised Technical Specifications were to: 1) incorporate the new physics and thermal-hydraulic limits associated with the sixth fuel cycle, and 2) reflect modifications performed during the refueling outage. The amendment addresses the changes to the Technical Specifications associated with the core reload and the thermal power monitor; the other changes associated with the modifications will be addressed by a separate evaluation.

2.0 Discussion and Evaluation

In support of the Cycle 6 reload, TVA submitted with its January 23, 1984 application a Reload Licensing Report (TVA-RLR-001) which describes results of the core design and safety analyses performed for Cycle 6. This reload is the first to be analyzed by TVA instead of the fuel vendor.

The Cycle 6 core will consist of 248 fresh fuel assemblies, and 516 burned assemblies that were originally loaded in Cycles 2 through 5. Among the burned assemblies are eight lead test assemblies that were initially installed and approved for Cycle 5. The remaining fuel is of the standard GE design described in NEDE-24011-P-A(US) GESTAR II "General Electric Standard Application for Reactor Fuel" January 1982.

Nuclear Design

The shutdown margin was determined by using the TVA BWR simulator code to calculate the core multiplication at selected exposure points for Cycle 6, with the strongest rod fully withdrawn. The shutdown margin was calculated to be 1% at the point in the cycle at which it is minimum. This exceeds the Technical Specification requirement of 0.38% and is, therefore, acceptable.

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor subcritical at any time in a cycle, from a full power, xenon free, condition to a cold, all rods out condition. The SLCS shutdown margin is calculated using the BWR simulator code to be 0.019 delta k with a 600 ppm boron concentration.

Reactivity coefficients are not used in the TVA analyses; however, their values are generated and reported. The void coefficient is calculated to be -0.757% delta k/% void at 100%-flow and -0.744% delta k/% void at 105% flow. These values are consistent with those customarily obtained for BWR reloads and are acceptable.

Thermal Hydraulics

The safety limit minimum critical power ratio (SLMCPR) of 1.07 is based on the GEXL correlation previously used for BF-3. 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 MCPR from its normal operating level. To assure that the SLMCPR will not be violated during abnormal transients, the most limiting transients have been reanalyzed for this reload. The events analyzed were load rejection without bypass, feedwater controller failure, loss of feedwater heaters, fuel loading errors, and control rod withdrawal errors. The anticipated transients are analyzed to determine that which yields the larges reduction in CPR and that value is added to the safety limit value to obtain the operating limit MCPR.

Core wide pressurization transients have been analyzed by TVA with the TVA-RETRAN code. This code has been described in a topical report (TVA-TR81-01, "BWR Transient Analysis Model Utilizing the RETRAN Program", TVA, December 1981) which also includes the verification of the code. This report was reviewed and approved by the staff with two possible restrictions by our letter to TVA of April 7, 1983. To remove these conditions, TVA submitted by letter dated November 21, 1983 a report, "Validation fo COMETHE III-J for Gap Conductance Calculations". Based on our review, our letter of May 23, 1984 advised TVA that TVA TR 81-01 was approved without conditions for referencing by TVA in core reload analyses performed by TVA for BWR facilities operated by TVA.

The non-pressurization events were analyzed with the TVA three dimensional core simulator code (TVA-TR78-03A, "Three-Dimensional Core Simulation Methods", TVA, January, 1979) which we approved by our letter to TVA of October 16, 1979. These potential transients are either steady state events or very slow transients.

The calculated MCPR's necessary to prevent SLMCPR violation during each transient are presented in the Reload Licensing Report (RLR). The limiting events for establishing the OLMCPR are the load rejection without bypass event (pressurization) and the rod withdrawal error (non-pressurization). When the reactor is operated in accordance with the proposed OLMCPR, the SLMCPR will not be violated in event of an abnormal operating transient. Changes to the Technical Specifications will incorporate the new OLMCPR.

A curve of MCPR as a function of average scram insertion time has been updated for the Technical Specifications.

Operation at 105 Percent Rated Flow

TVA proposed to operate at flow rates up to 105 percent of rated flow during Cycle 6. Analyses have been performed at both 100 and 105 percent flow and the more limiting results used to establish operating limits. The flow-biased instrumentation for the rod block monitor will be signal clipped for a setpoint of 106 percent since flow rates higher than rated would result in a delta CPR higher than reported for the rod withdrawal error event.

Such operation has been previously approved for Cycle 5 and continues to be acceptable for Cycle 6.

Loss of Coolant Accident (LOCA)

TVA submitted an addenda to the "Loss of Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3" prepared by the General Electric Company (GE) (NEDO-24194A), with the Cycle 6 reload application. The addenda covers the new BP8DRB284L fuel assemblies (FAs). The maximum average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size were calculated by General Electric using the CHASTE code.

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

The MAPLHGR values and peak cladding temperatures for each type FA that will be in the BF-3 Cycle 6 reload are presented in NEDO-21494A (as addended). The limit MAPLHGR values for the new BP8DRB284L fuel are included as proposed Technical Specifications changes in TVA's submittal, the values for other type FAs having been previously included. These MAPLGHR values will, in event of a LOCA, limit PCT to less than that allowed by 10 CFR 50 Appendix K and are, therefore, acceptable.

Control Rod Drop Accident

The rod drop accident (RDA) was reanalyzed for Cycle 6 by TVA using the TVA RDA transient simulation program with input from the TVA 3D simulator code. The RDA simulation model is described in Appendix A of the RLR. The TVA code has been checked against a test problem using a method similar to that of the fuel vendor and shown to be conservative. The staff therefore concludes that the RDA analysis method used by TVA is acceptable. The results of the analysis for BF-3, Cycle 6 is 240 cal/gram maximum fuel enthalpy. This value meets the staff acceptance criterion of 280 cal/gram and is therefore acceptable.

Overpressure Analysis

The licensee has reanalyzed the limiting pressurization event - main steamline isolation valve (MSIV) closure followed by direct neutron flux scram, using the TVA-RETRAN code. The results indicate a peak vessel pressure of 1287.6 psia. This is substantially identical to that of Cycle 5, reported as 1272 psig in our March 29, 1982 Cycle 5 evaluation and found acceptable therein.

Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for Cycle 6 using a model based on the LAPUR code which is applicable to both core and channel hydrodynamic stability. This model is currently being reviewed by the staff. The review has not progressed to the point where the staff can give generic approval to the TVA methodology. However, the review has progressed sufficiently to approve the Cycle 6 reload for the following reasons:

1. There are no significant changes in fuel loading between Cycle 6 and Cycle 5.

2. The decay ratio (core) as calculated by TVA for Cycle 6 is 0.87 which is very similar to the Cycle 5 calculated decay ratio and is acceptable.

3. The TVA model adequately predicts the results of the Peach Bottom Thermal-Hydraulic Stability Tests.

Thermal Power Monitor

The APRM flow-biased flux trip will be altered by the insertion of a damping circuit having a six-second time constant. This circuit simulates the time constant for heat transfer from fuel to coolant such that the flow-biased trip is based on heat flux as opposed to neutron flux. The fixed trip will still respond directly to neutron flux.

The Technical Specifications will be revised to reflect the modification.

The thermal power monitor has been previously approved for use on other BWRs including Browns Ferry Units 1 and 2 (i.e. BF-2 Amendment 85, BF-1, Amendment 91). The staff therefore concludes that it is acceptable for BF-3.

3.0 Changes to Technical Specifications-Reload

Specification 3.5.I and the Table of Contents will be changed to include the "MAPLHGR vs AVERAGE PLANAR EXPOSURE" table for the new BP8DRB284L type fuel.

Specification 3.5.K will be changed to update MCPR limits for Cycle 6. The Table of Contents will be revised to reflect the new page number of Figure 3.5.K-1.

Bases for Limiting Safety System Settings Related to Fuel Cladding Integrity, and Reactor Coolant System Integrity will be revised to reflect that reload analyses are being done by TVA instead of GE. Changes in text and references reflect TVA methodology.

The staff has reviewed these changes and concludes they are acceptable. This conclusion is based on the following:

1. Approved methods were used to perform the design and analysis of the Cycle 6 reload or the approval could be granted on other grounds.

2. Appropriate criteria for operational limits and accident consequences were met.

4.0 Changes to Technical Specifications - Thermal Power Monitor

Technical Specifications Sections 2.1.A Fuel Cladding Integrity Limiting Safety System Settings and Bases), 3.1 (Reactor Protection System, Limiting Conditions for Operation) and 4.1 (Reactor Protection System Surveillance Requirements) will be changed to reflect the addition of the thermal power monitors. The staff has reviewed the changes and found them to be acceptable. The changes are consistent with those issued for Unit 2 in Amendment 85 of the Unit 2 Technical Specifications.

5.0 Environmental Considerations

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for catergorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 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.

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Dated: July 11, 1984