

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401  
400 Chestnut Street Tower II

June 3, 1982

TVA BFNP TS 167

Mr. Harold R. Denton, Director  
Office of Nuclear Reactor Regulation  
U.S. Nuclear Regulatory Commission  
Washington, DC 20555

Dear Mr. Denton:

In the Matter of the )  
Tennessee Valley Authority )

Docket No. 50-260  
50-296

By my letter to you dated September 21, 1981, we submitted license amendment request TVA BFNP TS 167 for Browns Ferry Nuclear Plant units 2 and 3. By letter from D. B. Vassallo to H. G. Parris dated February 9, 1982, we received a request for additional information concerning TS 167. Our response to that request is provided as enclosure 1.

Revisions to technical specifications are provided as enclosure 2. These revisions are made in response to the request for information, to correct typographical and omission errors, and to reflect technical specification changes approved since submittal of TS 167.

We appreciate your staff's efforts in their review of TS 167 and request their continued expeditious handling of this requested license amendment. Approval of TS 167 will increase the availability factor of Browns Ferry.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

*D S Kammer*

D. S. Kammer  
Nuclear Engineer

Subscribed and sworn to before  
me this 3<sup>rd</sup> day of June 1982.

*Paulette H. White*  
Notary Public

My Commission Expires 9-5-84

Enclosures  
cc: See page 2

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Mr. Harold R. Denton

June 3, 1982

cc (Enclosures):

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ENCLOSURE 1

RESPONSE TO D. B. VASSALLO'S LETTER  
TO H. G. PARRIS DATED FEBRUARY 9, 1982  
REGARDING TVA BFNP TS 167  
BROWNS FERRY NUCLEAR PLANT UNITS 2 AND 3

1. A discussion of the differences between the two departure-from-nuclear-boiling heat transfer correlations-Hench-Levy and GETAB-GEXL.

#### Response

A detailed technical description of the Hench-Levy and GETAB-GEXL heat flux correlations are provided in references 1 and 2, respectively.

Primarily, the Hench-Levy correlation is a variation of the "local conditions" concept for critical heat flux in the form of

$$\phi_C = (G, P, X_{CB})$$

where  $\phi_C$  is local critical heat flux and  $X_{CB}$  is the bundle average steam quality at the local elevation. This correlation was formulated as a lower limit line to existing rod bundle critical heat flux data.

The GEXL correlation is a variation of the critical quality versus boiling length concept of the form

$$X_{CB} = f(G, D, P, L_B, L, R) \text{ where}$$

$X_{CB}$  is the bundle average critical quality,  $L_B$  is the boiling length,  $L$  is the heated length, and  $R$  is a weighting factor which characterizes the local rod-to-rod peaking pattern. This type of correlation has the distinct advantage over a local conditions correlation of being able to correlate critical heat flux data for both uniform as well as nonuniform axial heat flux distributions.

As discussed in reference 2, the Hench-Levy correlation was shown to be nonconservative when predicting heat fluxes for nonuniform axial and radial heat flux data. This is because the correlation was derived from test conditions which did not duplicate actual heat flux distributions representative of BWR operating conditions. Thus, it was concluded for rod bundle heat flux distributions and heated lengths corresponding to those found in General Electric Company BWR reactor cores that the Hench-Levy correlation did not provide a lower limit line. In contrast, the GEXL correlation was based upon greater than 4,000 boiling transition data points, many of which were obtained from full size, full length rod bundles with a wide range of axial and radial heat flux profiles. Based on this extensive test data, the GEXL correlation provided a distinct improvement over the Hench-Levy correlation for predicting critical heat flux.

The figure of merit for transition boiling based on the Hench-Levy correlation is the critical heat flux ratio (CHFR), i.e., the ratio of the critical heat flux,  $\phi_C$ , for boiling transition to the existing local heat flux,  $\phi$ . Since the existing local heat flux is directly proportional to the product of reactor power and the total peaking factor at the point of CHFR, any increase in the peaking factor resulted in a corresponding decrease in CHFR. For calculations based

on GEXL, no such direct relationship exists between the peaking factor and the critical power ratio, CPR, i.e., the ratio of critical bundle power,  $P_C$ , for boiling transition to the existing bundle power,  $P$ . This infers that the calculations based on GEXL are not as strongly influenced by the assumed axial power distribution or peaking factor.

2. Identification of pertinent thermal limits and a discussion of why the change in thermal limits justifies a longer time to take corrective action (or possibly justifications why the limits can be removed from the Technical Specifications).

#### Response

The pertinent thermal limits which must be considered are the fuel cladding integrity safety limits for departure from nucleate boiling (DNB) and 1-percent plastic strain of the cladding.

Under the technical specification limits based upon the Hench-Levy correlation, the DNB thermal limit was characterized as the minimum critical heat flux ratio (MCHFR). Safety analyses were performed assuming a particular design power shape which initialized some point in the core at the MCHFR operating limit at rated power and flow. For rated conditions and the assumed design power peaking factor, the operating limit MCHFR was determined to be 1.9 to ensure a safety limit MCHFR of 1.0 could be maintained for all transient and steady-state conditions. Since the operating limit was dependent on the assumed power peaking factor, it was necessary to consider the change in peaking factor as a function of the core operating range in setting trip setpoints to maintain thermal margins. Thus, a setdown of the APRM flow-biased rod block/scram setpoints as a function of the total peaking factor was utilized to maintain the thermal margin to a MCHFR of 1.9. It should be noted that the 1.9 MCHFR corresponds to a limiting condition for operation and not a fuel cladding integrity safety limit setpoint. Thus, for total peaking factors greater than the design value peaking factor (higher peaking factors sometime occur during startup), allowance was made for the setpoints to be lowered.

With introduction of the GEXL correlation, the functional form of the safety limit and operating limit for preventing DNB became the minimum critical power ratio (MCPR). Determination of an operating limit MCPR is based on analyses initiated from rated (105-percent power) core conditions assuming a fixed 120-percent flux scram. The operating limit is determined from results of transient analyses both core-wide and localized events. The resultant CPR due to the transients is added to the safety limit MCPR to determine the necessary operating limit MCPR to ensure adequate thermal margin. Additionally, the required operating limit is increased at reduced core flow to ensure the safety limit is not violated in the event of a flow increase transient. Since the MCPR limit is not directly functionally dependent on the power

shape; only on bundle characteristics of power, flow, and pressure; it is not necessary to reduce the scram setpoints as a function of peaking factor. The MCPR safety analyses take no credit for an APRM flow-biased scram and, consequently, this flow-biased scram does not ensure additional margin to the safety limit MCPR.

The fuel cladding safety limit to prevent 1-percent plastic strain is represented by the maximum linear heat generation rate (MLHGR). The current operating limit MLHGR is 13.4 kW/ft for 8x8 fuel lattices with the safety limit MLHGR (for 1-percent strain) being approximately 160 percent of the design operating limit. The use of the flow-biased rod block/scram lines and the adjustments based on the ratio of core maximum fraction of limiting power density to core fraction of rated power ensures that operation along the allowable flow control line will maintain the MLHGR below 13.4 kW/ft. Safety analyses have demonstrated that transients initiated from the operating limit MLHGR will not violate the 1-percent plastic strain LHGR limit. Typically, transients result in approximately a 20-percent increase in LHGR compared to the 60-percent margin to 1-percent plastic strain.

The proposed change in the Browns Ferry technical specifications does not eliminate the flow-biased rod block/scram setpoints or the setdown adjustments. However, as discussed above, the additional conservatism resulting from these constraints is not required to maintain the margin of safety, and it is appropriate that a relaxed time allowance be incorporated. Operationally, the setdown adjustment enforces an optimal upper bound on core total peaking factor which in turn ensures conformance to LHGR limits at full power rod patterns. This is operationally important since it is difficult to make rod adjustments at high powers due to fuel preconditioning constraints. However, because of changes in Xenon distribution during startup and following power shape changes and the tendency of rod movement to exaggerate peaking at very low powers, it is difficult to strictly maintain the optimum design peaking factor during the earlier stages of a startup or rod pattern change and still reach the desired steady-state power distribution. As the startup continues, the normal progression of rod movements and flow increases will flatten peaking and the limit will naturally be reestablished. Thus, we request that 6 hours be allowed for this limit to be achieved before corrective action is required. This change will allow optimum plant operation while preserving additional safety margin above and beyond that required for anticipated transients initiated from analyzed operating conditions.



3. A discussion of why the system is adequately protected by the 120-percent fixed scram in the absence of any peaking factor adjustment, and that the calibration techniques used in setting the APRM scram trip settings (under high APRM Gain Adjustment Factor) ensure that the 120-percent setpoint is not exceeded.

Response

As discussed previously, the current safety analyses assume a 120-percent fixed scram and have been shown to conservatively bound reactor operation with or without a peaking factor adjustment. No credit is taken for setdown adjustment or flow bias scrams.

The APRM 120-percent trip setting is calibrated quarterly and functionally checked weekly. The APRM output signal is calibrated weekly by a heat balance such that the percent power reading on the APRM is greater than the percent thermal power calculated by heat balance.

4. The bases for these limits in the Technical Specifications, if you propose to retain the equations as a limiting condition for operation (as in your submittal).

Bases

3.5.L APRM Setpoints

Operation is constrained to a maximum LHGR of 18.5 kW/ft for 7x7 fuel and 13.4 kW/ft for 8x8, 8x8R, and P8x8R. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

### References

1. J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," September 1966, (APED-5286).
2. S. Levy, et.al, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," January 1977, (NEDO-10958-4).
3. NEDE-24011-P-A, "Generic Reload Fuel Application," General Electric Company Licensing Topical Report, July 1979



UNIT 2

PROPOSED CHANGES

<u>Section</u>	<u>Page No.</u>
D. Reactivity Anomalies . . . . .	125
E. Reactivity Control . . . . .	126
3.4/4.4 Standby Liquid Control System . . . . .	135
A. Normal System Availability . . . . .	135
B. Operation with Inoperable Components . . . . .	136
C. Sodium Pentaborate Solution . . . . .	137
3.5/4.5 Core and Containment Cooling Systems . . . . .	143
A. Core Spray System . . . . .	143
B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling) . . . . .	145
C. RHR Service Water System and Emergency Equipment Cooling Water System (EECWS) . . . . .	151
D. Equipment Area Coolers . . . . .	154
E. High Pressure Coolant Injection System (HPCIS) . . . . .	154
F. Reactor Core Isolation Cooling System (RCICS) . . . . .	156
G. Automatic Depressurization System (ADS) . . . . .	157
H. Maintenance of Filled Discharge Pipe . . . . .	158
I. Average Planar Linear Heat Generation Rate . . . . .	159
J. Linear Heat Generation Rate . . . . .	159
K. Minimum Critical Power Ratio (MCPR) . . . . .	160
L. APRM Setpoints . . . . .	160A
M. Reporting Requirements . . . . .	160 A
3.6/4.6 Primary System Boundary . . . . .	174
A. Thermal and Pressurization Limitations . . . . .	174
B. Coolant Chemistry . . . . .	176

## SAFETY LIMIT

### 1.1 FUEL CLADDING INTEGRITY

#### B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia)

When the reactor pressure is less than or equal to 800 psia,

## LIMITING SAFETY SYSTEM SETTING

### 2.1 FUEL CLADDING INTEGRITY

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

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are  $LHGR \leq 18.5$  kw/ft for 7x7 fuel and  $\leq 13.4$  kw/ft for 8x8, 8x8R, and P8x8R, and MCPR within limits of Specification 3.5.k.

If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

2. 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.
3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

#### B. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

## 2.1 BASES

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 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.

### 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 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 Scram

#### 1. Fast Closure Scram

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failures of the turbine bypass valves. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in 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. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

#### 4.1 BASES

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, and MAPLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

### 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 power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 as modified in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds design linear heat generation rate due to power spiking. The LHGR (as a function of core height for 7x7 fuel and as a constant for 8x8, 8x8R, and P8x8R fuel) shall be checked daily during reactor operation at  $\geq 25\%$  power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

### 3.5.K. 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 requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

### 3.5.L. APRM Setpoints

Operation is constrained to a maximum LHGR of 18.5 kW/ft for 7x7 fuel and 13.4 kW/ft for 8x8, 8x8R, and P8x8R. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

## BASES

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

### 3.5.N References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.
2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).
3. Communication: V. A. Moore to I.S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.
4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.



UNIT 3

PROPOSED CHANGES

<u>Section</u>		<u>Page No.</u>
	C. Scram Insertion Times	128
	D. Reactivity Anomalies	129
	E. Reactivity Control	129
3.4/4.4	Standby Liquid Control System	137
	A. Normal System Availability	137
	B. Operation with Inoperable Components	139
	C. Sodium Pentaborate Solution	139
3.5/4.5	Core and Containment Cooling Systems	146
	A. Core Spray System	146
	B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	149
	C. RHR Service Water System and Emergency Equipment Cooling Water System (EECWS)	155
	D. Equipment Area Coolers	158
	E. High Pressure Coolant Injection System (HPCIS)	159
	F. Reactor Core Isolation Cooling System (RCICS)	160
	G. Automatic Depressurization System (ADS)	161
	H. Maintenance of Filled Discharge Pipe	163
	I. Average Planar Linear Heat Generation Rate	165
	J. Linear Heat Generation Rate	166
	K. Minimum Critical Power Ratio (MCPR)	167
	L. APRM Setpoints	167A
	M. Reporting Requirements	167A
3.6/4.6	Primary System Boundary	184
	A. Thermal and Pressurization Limitations	184

uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

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

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

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

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of  $28 \times 10^3$  lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than  $28 \times 10^3$  lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

For the fuel in the core during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If water level

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.

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 N14.5 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 Scram

1. Fast Closure Scram

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failures of the turbine bypass valves. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control

#### 4.1 BASES

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, and MAPLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

(NO CHANGE PROPOSED TO PAGE 167)

generation if fuel pellet densification is postulated.

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

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 requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above  $25\%$  rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at thermal limit.

3.5.L APRM Setpoints

Operation is constrained to a maximum LHGR of  $18.5 \text{ kW/ft}$  for  $7 \times 7$  fuel and  $13.4 \text{ kW/ft}$  for  $8 \times 8$ ,  $8 \times 8R$ , and  $P8 \times 8R$ . This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.



### 3.5 BASES

#### 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.

#### N. 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.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION REVISIONS  
REFERENCE: TVA BFP TS 167  
BROWNS FERRY NUCLEAR PLANT UNITS 2 AND 3

## DESCRIPTION OF CHANGES

### Unit 2 Pages

- 11 - Update Table of Contents
- 9 - Reference to "LHGR  $\leq$  18.5kW/ft for 7 x 7 fuel" was removed in TS 167 submittal. Reference to 7 x 7 fuel should be retained.
- 23 - This page was not submitted in TS 167. It should be included since the change proposed is consistent and in agreement with all other changes being requested in TS 167.
- 48 - This page was correctly included in TS 167 but bars in page margin did not properly indicate location of changes requested.
- 169 - BASES "3.5.L APRM Setpoints" being added per recommendation of NRC.
- 169A - This page is only being reformatted and renumbered. No change to specifications is proposed.

### Unit 3 Pages

- 11 - Update Table of Contents
- 16 - This page was not included in TS 167 but should have been because the change proposed is consistent with all others proposed in TS 167.
- 22 - (Same reason as page 16 above)
- 47 - This page was correctly included in TS 167 but bar in page margin was left off.
- 167 - Page 167 was included in TS 167 with proposed revisions. However, because of changes made to page 167 since submittal of TS 167, no change is needed and proposed page 167 is withdrawn.
- 177 - BASES "3.5.L APRM Setpoints" being added per recommendation of NRC.
- 178 - This page is only being reformatted and renumbered. No change to specifications is proposed.