ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATIONS REVISIONS

BROWNS FERRY NUCLEAR PLANT

UNITS 1, 2, AND 3

(TVA BFN TS 241)

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

1.1.B. Power Transient

To ensure that the Safety Limits established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

C. Reactor Vessel Water Level

Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 378 inches above vessel zero.

LIMITING SAFETY SYSTEM SETTING

2.1.B. Power Transient Trip Settings

1.	Scram and isolation (PCIS groups 2,3,6) reactor low water level	≥ 538 in. above vessel zero
2.	Scram--turbine stop valve closure	≤ 10 percent valve closure
3.	Scram--turbine control valve fast closure or turbine trip	≥ 550 psig
4.	(Deleted)	
5.	Scram--main steam line isolation	≤ 10 percent valve closure
6.	Main steam isolation valve closure --nuclear system low pressure	≥ 825 psig

C. Water Level Trip Settings

1.	Core spray and LPCI actuation--reactor low water level	≥ 378 in. above vessel zero
2.	HPCI and RCIC actuation--reactor low water level	≥ 470 in.. above vessel zero
3.	Main steam isolation valve closure--reactor low water level	≥ 378 in. above vessel zero

1.1 BASES (Cont'd)

The safety limit has been established at 378 inches above vessel zero to provide a point which can be monitored and also provide adequate margin to assure sufficient cooling. This point is the lower reactor low water level trip.

REFERENCE

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10938 and NEDE 10938.

decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108 percent 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 Steam Lines)

The setpoint 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 sufficiently below normal operating range 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 percent 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 percent 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 percent of rated, as measured by turbine first state pressure.

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at 378 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 378 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established nominal setting of three times normal background and main

SAFETY LIMIT

1.1.B. Power Transient

To ensure that the Safety Limits established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

C. Reactor Vessel Water Level

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LIMITING SAFETY SYSTEM SETTING

2.1.B. Power Transient Trip Settings

1.	Scram and isolation (PCIS groups 2,3,6) reactor low water level	≥ 538 in. above vessel zero
2.	Scram--turbine stop valve closure	≤ 10 percent valve closure
3.	Scram--turbine control valve fast closure or turbine trip	≥ 550 psig
4.	(Deleted)	
5.	Scram--main steam line isolation	≤ 10 percent valve closure
6.	Main steam isolation valve closure --nuclear system low pressure	≥ 825 psig

C. Water Level Trip Settings

1.	Core spray and LPCI actuation-- reactor low water level	≥ 378 in. above vessel zero
2.	HPCI and RCIC actuation-- reactor low water level	≥ 470 in. above vessel zero
3.	Main steam isolation valve closure-- reactor low water level	≥ 378 in. above vessel zero

1.1 BASES (Cont'd)

The safety limit has been established at 378 inches above vessel zero to provide a point which can be monitored and also provide adequate margin to assure sufficient cooling. This point is the lower reactor low water level trip.

REFERENCE

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decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108 percent 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.

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The setpoint 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 sufficiently below normal operating range to avoid spurious scrams.

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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 percent 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)

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In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

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The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

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The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

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High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established nominal setting of three times normal background and main

ENCLOSURE 2

DESCRIPTION AND JUSTIFICATION
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

Description of Change

The Browns Ferry Nuclear Plant Technical Specifications are being revised to replace water level measurements referenced to the top of active fuel with measurements referenced to the height above vessel zero. This entails a change to the section 1.1/2.1 Safety Limit for reactor vessel water level and numerous changes to the bases of sections 1.1/2.1 and 3.2/4.2. In addition, the bases of section 2.1 is being revised to correct and clarify an incorrect statement.

Reason for Change

The current technical specifications give water level setpoints referenced to both vessel zero and to the top of the active fuel. The top of the active fuel setpoints are based on fuel with 144 inches of active fuel; whereas, the fuel currently being used has 150 inches of active fuel. This change will make the technical specifications reference vessel zero throughout and will also correct the inconsistencies with the 144 inch fuel length versus the 150 inch fuel length.

In addition, this change will help eliminate possible operator confusion with having technical specification water levels referenced to different points. This was the intent of NUREG 0737, item II.K.3.27, which requires all reactor vessel water level instruments to be referenced to the same point. This change will make the technical specifications reference the same point throughout.

The bases of section 2.1 is being revised to correct a statement which is presently incorrect.

Justification for Change

Technical specification, Safety Limit 1.1.c, is affected by this change. The safety limit is changed to reference vessel zero from the present above the top of active fuel reference.

This change establishes a direct correspondence between the safety limit and the limiting safety system settings of 2.1.C as both will now be referenced to the same datum. This is not a change to the safety limit, merely a change to the reference point. Top of active fuel is 360.3 inches above vessel zero. The current safety limit is 17.7 inches above the top of active fuel. The current limit referenced to vessel zero would be 378 inches above vessel zero. This safety limit corresponds to the level that actuates low pressure coolant injection (LPCI), core spray, and main steam isolation valve closure.

<u>Justification for Change</u> (Cont'd)

The changes to the bases eliminate the top of the active fuel reference leaving the above vessel zero. This makes the technical specifications and the bases consistent in referencing all water levels to vessel zero. The change to the bases of section 2.1 will clarify and correct the margin between the lower end of the normal operating range and the scram setting.

ENCLOSURE 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

Description of Proposed Technical Specification Amendment

The proposed amendment would change the BFN Technical Specifications for units 1, 2, and 3 to replace references for the top of active fuel with the equivalent height above vessel zero.

Basis for Proposed No Significant Hazards Consideration Determination

NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from an accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

1. The proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated. There will be no change in the operation of the facility as a result of the amendment. No safety-related equipment or function will be altered. The amendment merely changes the reactor vessel water level reference.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated above, no safety-related equipment, safety functions, or plant operations will be altered as a result of this amendment. The requested change does not create any new accident mode. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident.

3. The proposed amendment does not involve a significant reduction in a margin of safety. This amendment merely changes the reference point of the level instrument to make all the instrument references identical. The safety limit is not changed and therefore there is no reduction in a margin of safety.

Determination of Basis for Proposed No Significant Hazards

Since the application for amendment involves a proposed change that is encompassed by the criteria for which no significant hazards consideration exists, TVA has made a proposed determination that the application involves no significant hazards consideration.