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JRBuchanan

Docket No. 50-259 JAN 17 1979

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment No. 47 to Facility License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1. This amendment changes the Technical Specifications in response to your request of September 8, 1978 (TVA BFNP TS 115), as supplemented by your letters of October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979 and January 9, 1979. This amendment permits operation of Browns Ferry Unit No. 1 in cycle No. 3 following the current refueling outage.

The supporting analyses submitted with your application proposed to take credit for a prompt recirculation pump trip. As discussed in my letter to you of January 16, 1979, we have some reservations about the design of the RPT system. Therefore, as agreed to with your staff, we have not included credit for the RPT system in the operating limit minimum critical power ratios.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 47 to DPR-33
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

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DATE	1/ /79	1/17/79	1/ /79	1/17/79		

NRC FORM 318 (9-76) NRCM 0240 ★ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 8, 1978, as supplemented by letters dated October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979, and January 9, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7902060016

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 17, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

vii/viii	113/114
7/8	131/132
9/10	133/134
15/16	139/140
17/18	157/158
19/20	159/160
21/22	167/168
23/24	169/170
25/26	173a
27/28	181/182
29/30	218/219
73/74	220/221
	330/331

2. The underlined pages are those being changed; marginal lines on these pages indicate the revised area. The overleaf page is provided for convenience.

3. Add the following new page:

   172a

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

1.0 DEFINITIONS (Cont'd)

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

    (a) Initiating - A logic that receive signals from channels and produces decision outputs to the actuation logic.

    (b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

Z. Cumulative Downtime - The cumulative downtime for those safety components and systems whose downtime is limited to 7 consecutive days prior to requiring reactor shutdown shall be limited to any 7 days in a consecutive 30 day period.

JAN 1 0 1978

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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. 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 Scram

1. APRM Flux Scram Trip Setting (Run Mode)

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

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

where:

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

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

1.1 FUEL CLADDING INTEGRITY

2.1 FUEL CLADDING INTEGRITY

In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

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 and 8x8 R fuel, MCPR limits of Spec 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. Core Thermal Power Limit (Reactor Pressure $\leq$800 psia)

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

B. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

1.1 FUEL CLADDING INTEGRITY

or core coolant flow is less than 10% of rated, the core thermal power shall not exceed 823 MWt (about 25% of rated thermal power).

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

2.1 FUEL CLADDING INTEGRITY

$S_{RB} \leq (0.66W + 42\%)$

where:

S_{RB} = Rod block setting is percent of rated thermal power (3293 MWt)

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

In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$S_{RB} \leq (0.66W + 42\%) \frac{FRP}{CMFLPD}$ for two recirculation loop operation.

$S_{RB} \leq (0.66W + 38.7\%) \frac{FRP}{CMFLPD}$ for one recirculation loop operation.

C. Scram and isolation--reactor low water $\geq$ 538 in. above vessel zero le

D. Scram--turbine stop valve closure $\leq$ 10 percent valve closure

E. Scram--turbine control valve

1. Fast closure — Upon trip of the fast actin solenoid valve

2. Loss of control oil pressure $\geq$ 550 psig

F. Scram--low condenser vacuum $\geq$ 23 inches Hg vacuum

G. Scram--main steam line isolation $\leq$ 10 percent valve closure

H. Main steam isolation valve closure--nuclear system low pressure $\geq$ 825 psig

Amendment No. 35, 41

SEP 1 9 1978

2.1 BASES: FUEL CLADDING INTEGRITY SAFETY LIMIT

The fuel cladding represents one of the physical barriers which separate radioactive materials from environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system setpoints. While fission product migration from cladding performation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined in terms of the reactor operating conditions which can result in cladding perforation.

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset transition boiling (MCPR of 1.0). This establishes a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR >1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possiblity of clad failure. Since boiling transition is not a directly observable parameter, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 2.1.1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > limits specified in specification 3.5.K) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit 1.07 is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

.1.1 BASES

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 occurence 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 BFNP 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 of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (MCPR = 1.06) operation is constrained to a maximum LHGR of 18.5 kw/ft for 7x7 fuel and 13.4 kw/ft for 8x8 fuel. This limit is reached when the Core Maximum Fraction of Limiting Power Density equals 1.0 (CMFLPD = 1.0). For the case where Core Maximum Fraction of Limiting Power Density exceeds the Fraction of Rated Thermal Power, operation is permitted only at less than 100% of rated power and only with reduced APRM scram settings as required by specification 2.1.A.1.

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 flow will always be greater than 4.56 psi. Analyses show that with a flow of $28X10^3$ 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 $28x10^3$ 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 shut down, consideration must also be given to water level requirements due to the effect of decay heat. If water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. As long as the fuel remains covered with water, sufficient cooling is available to prevent fuel clad perforation.

1.1 BASES

The safety limit has been established at 17.7 in. above the top of the irradiated fuel to provide a point which can be monitored and also provide adequate margin. This point corresponds approximately to the top of the actual fuel assemblies and also to the lower reactor low water level trip (378" above vessel zero).

REFERENCE

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958.

2. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

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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 applicable 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 absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. 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 > limits specified in specification 3.5.K 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.

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

In summary:

1. The licensed maximum power level is 3,293 MWt.
2. Analyses of transients employ adequately conservative values of the controlling reactor parameters.
3. The abnormal operational transients were analyzed to a power level of 3440 MWT.
4. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual set points are discussed below:

A. Neutron Flux Scram

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

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during transients induced by disturbances, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses reported in Section 14 of the Final Safety Analysis Report demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin. Figure 2.1.2 shows the flow biased scram as a function of core flow.

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

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2.1 BASES

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 > limits specified in specification 3.5.K.

2. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

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 accommodate 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. IRM Flux Scram Trip Setting

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

.1 **BASES**

3. IRM Flux Scram Trip Setting (Continued)

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

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 recir-cuclation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varried with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod with-drawal. 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 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 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 state pressure.

2.1 BASES

2. Scram on loss of control oil pressure

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system, which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection.

F. Main Condenser Low Vacuum Scram

To protect the main condenser 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 vacuum 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 825 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 825 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.

2.1 BASES

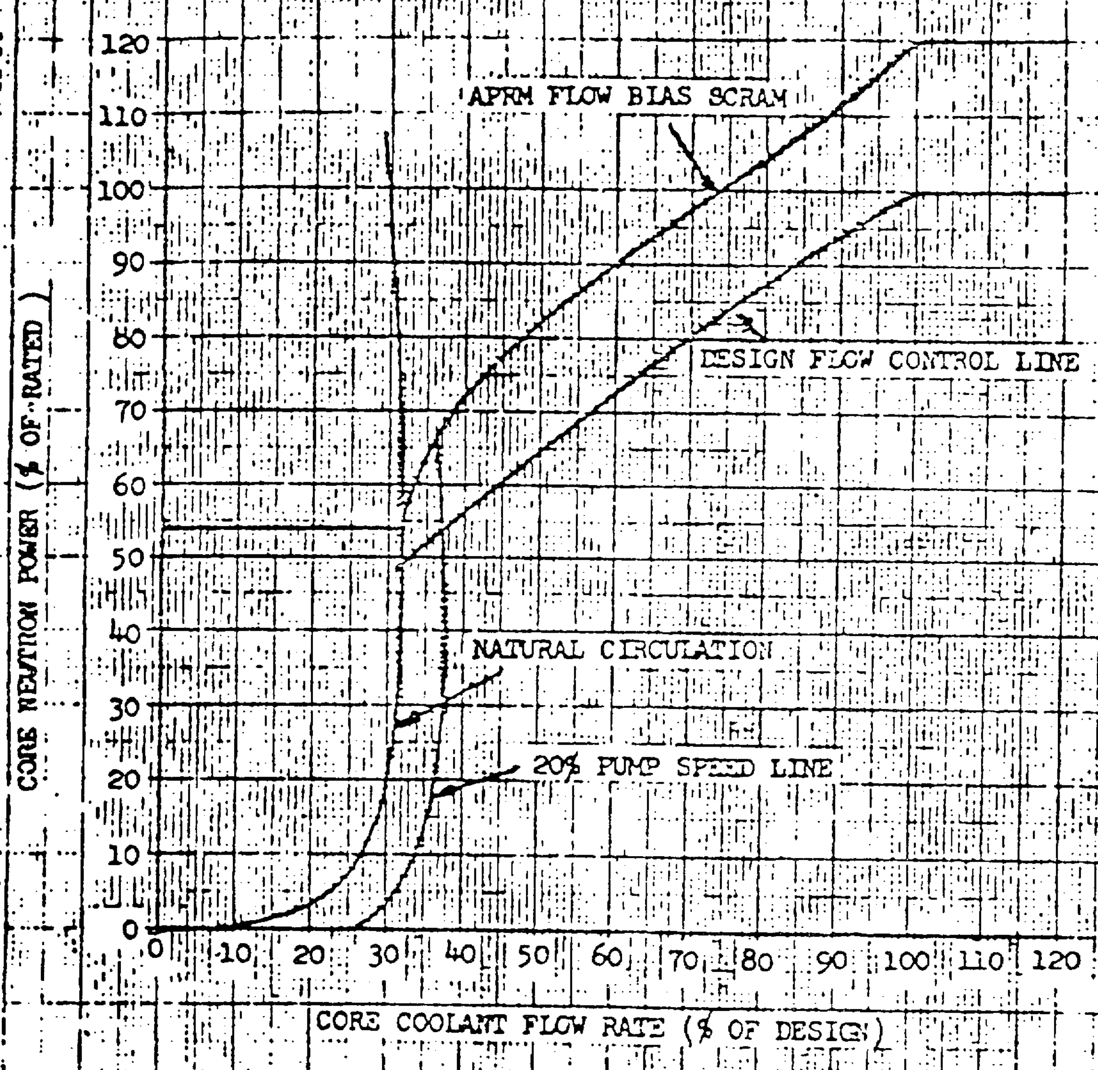
I. J. & K. Reactor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps.

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. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

2. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.



APRM FLOW BIAS SCRAM Vs. REACTOR CORE FLOW
FIG. 2.1-2

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.2 REACTOR COOLANT SYSTEM INTEGRITY	**2.2 REACTOR COOLANT SYSTEM INTEGRITY**
Applicability	**Applicability**
Applies to limits on reactor coolant system pressure	Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.
Objective	**Objective**
To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.	To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.
Specification	**Specification**
A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.	The limiting safety system settings shall be as specified below:

Protective Action	Limiting Safety System Setting
A. Nuclear system relief valves open--nuclear system pressure	1105 psig ± 11 psi (4 valves)
	1115 psig ± 11 psi (4 valves)
	1125 psig ± 11 psi (5 valves)
B. Scram--nuclear system high pressure	≤ 1,055 psig

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 ressulting directly in a reactor coolant system pressure increase is given in Reference 5.
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.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the pressure is not monitored at this point, it cannot be directly determined if this safety limit has been violated. Also, because of the potentially varying head level and flow pressure drops, an equivalent pressure cannot be a priori determined for a

1.2 BASES

pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BFNP FSAR Section 14.0)
2. ASME Boiler and Pressure Vessel Code Section III
3. USAS Piping Code, Section B31.1
4. Reactor Vessel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)
5. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

To meet the safety design basis, thirteen relief valves have been installed on the unit with a total capacity of 82.4% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1273 psig if a neutron flux scram is assumed considering 12 valves operable. This results in an 102 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the operational design, the analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in Reference 5 on page 29. This analysis shows that 12 of the 13 relief valves limit pressure in the steam line to 1201 psig. This analysisshows that peak system pressure is limited to 1229 psig which is 146 psig below the allowed vessel overpressure of 1375 psig.

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TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum No. Operable Per Trip Sys (5)	Function	Trip Level Setting
2(1)	APRM Upscale (Flow Bias)	$\leq$ 0.66W + 42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	$\leq$ 12%
2(1)	APRM Downscale (9)	$\geq$ 3%
2(1)	APRM Inoperative	(10_b)
1(7)	RBM Upscale (Flow Bias)	$\leq$ 0.66W + 40% (2)
1(7)	RBM Downscale (9)	$\geq$ 3%
1(7)	RBM Inoperative	(10_c)
3(1)	IRM Upscale (8)	$\leq$ 108/125 of full scale
3(1)	IRM Downscale (3)(8)	$\geq$ 5/125 of full scale
3(1)	IRM Detector not in Startup Position (8)	(11)
3(1)	IRM Inoperative (8)	(10^a)
2(1)(6)	SRM Upscale (8)	$\leq 1 \times 10^5$ counts/sec.
2(1)(6)	SRM Downscale (4)(8)	$\geq$ 3 counts/sec.
2(1)(6)	SRM Detector not in Startup Position (4)(8)	(11)
2(1)(6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	$\leq$ 10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$ 110% recirculation flow
1(1)	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-85-61A & PS-85-61B)	147 psig turbine first stage pressure (approximately 30% power)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM, IRM, and APRM (Startup mode), blocks need not be operable in "Run" mode, and the APRM (Flow biased) and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition last longer than seven days, the system with the inoperable channel shall be tripped. If the first column cannot be met for both trip systems, both trip systems shall be tripped.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt). A ratio of FRP/CMFLPD <1.0 is permitted at reduced power. See Specification 2.1 for APRM control rod block setpoint.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps and IRM above range 2.

5. One instrument channel; i.e., one APRM or IRM or RBM, per trip system may be bypassed except only one of four SRM may be bypassed.

6. IRM channels A, E, C, G all in range 8 bypasses SRM channels A & C functions.

   IRM channels B, F, D, H all in range 8 bypasses SRM channels B & D functions.

7. The following operational restraints apply to the RBM only:

   a. Both RBM channels are bypassed when reactor power is $\leq$ 30%.

   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

   c. Two RBM channels are provided and only one of these may be bypassed from the console. An RBM channel may be out of service for testing and/or maintenance provided this condition does not last longer than 24 hours in any thirty day period.

   d. If minimum conditions for Table 3.2.C are not met, administrative controls shall be immediately imposed to prevent control rod withdrawal.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

High temperature in the vicinity of the HPCI equipment is sensed by 4 sets of 4 bimetallic temperature switches. The 16 temperature switches are arranged in 2 trip systems with 8 temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

3.2 BASES

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two post treatment off-gas radiation monitors are provided and, when their trip point is reached, cause an isolation of the off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or both have a downscale trip.

Both instruments are required for trip but the instruments are set so that any instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the Reactor zone ventilation exhaust ducts and in the Refueling Zone.

Trip setting of 100 mr/hr for the monitors in the Refueling Zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (ie, MCPR given by Spec. 3.5.K or LHGR of 18.5 for 7x7 or 13.4 for 8x8 and 8x8R). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage; ie, to prevent the MCPR from becoming less than 1.07. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occured during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during startup of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and startup testing is sufficient to detect anomalous drive performance.

3. The 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero an approximately 200 milliseconds later, control rod motion begins. The 200 milliseconds are included in the allowable scram insertion times specified in Specification 3.3.C.

* In order to perform scram time testing as required by specification 4.3.C.1, the relaxation of certain restraints in the rod sequence control system is required. Individual rod bypass switches may be used as described in specification 4.3.C.1.

The position of any rod bypassed must be known to be in accordance with rod withdrawal sequence. Bypassing of rods in the manner described in specification 4.3.C.1 will allow the subsequent withdrawal of any rod scrammed in the 100 percent to 50 percent rod density groups; however, it will maintain group notch control over all rods in the 50 percent density to preset power level range. In addition, RSCS will prevent movement of rods in the 50 percent density to preset power level range until the scrammed rod has been withdrawn.

3.3/4.4 BASES:

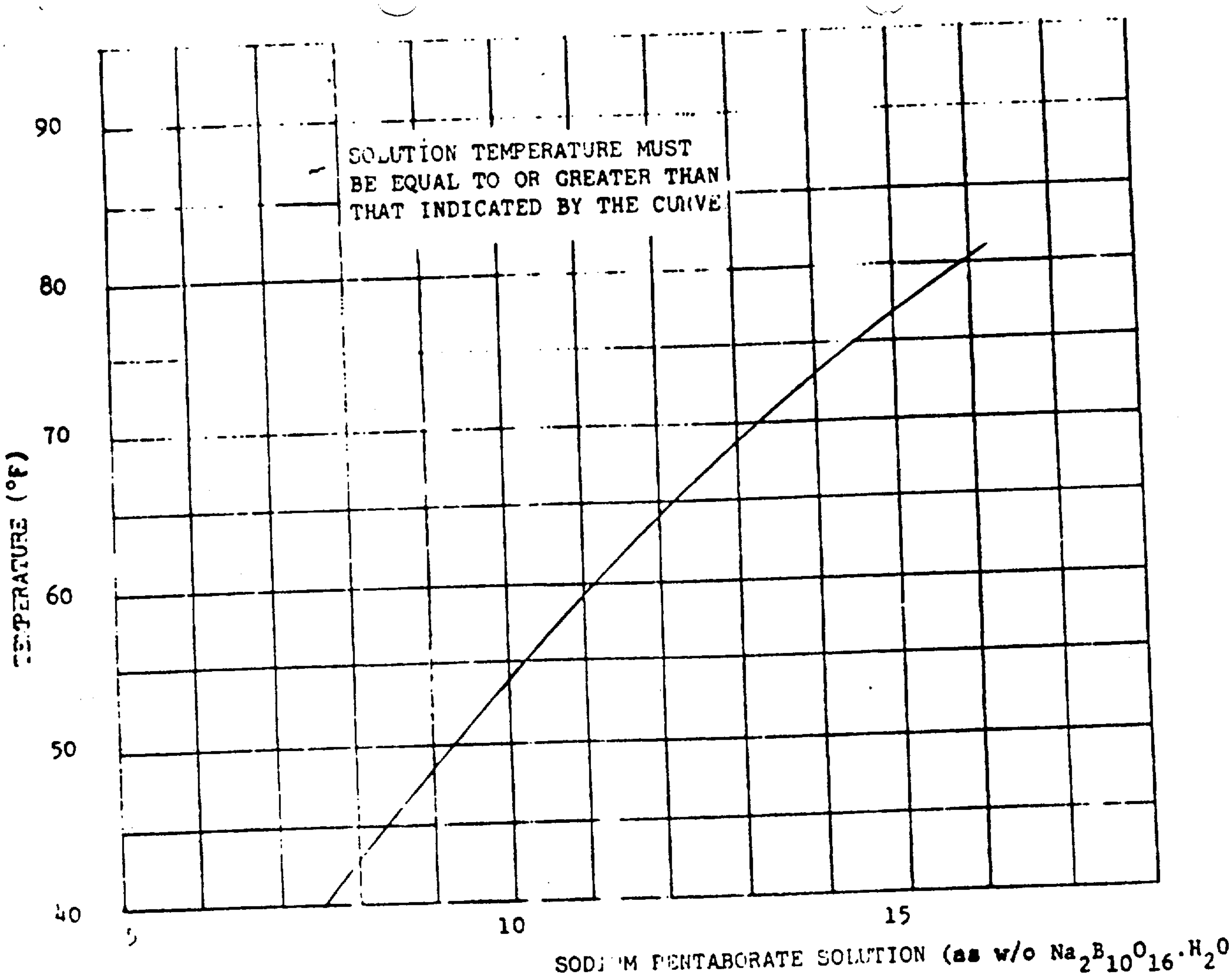
D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References

1. General Electric BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
TEMPERATURE REQUIREMENTS
FIGURE 3.4-2

.4 BASES: STANDBY LIQUID CONTROL SYSTEM

A. If no more than one operable control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the Standby Liquid Control System is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the liquid control system is required.

The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration greater than 600 ppm of boron in the reactor core in less than 125 minutes. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

T e minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long-term average availability of the system is not reduced is obtained fro a one-out-of-two system by an allowable equipment out-of-service time of one-third of the normal surveillance frequency. This method determines an equipment out-of-service time of ten days. Additional conservatism is introduced by reducing the allowable out-of-service time to seven days, and by increased testing of the operable redundant component.

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10°F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

The volume concentration requirement of the solution are such that should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature-concentration requirements are exceeded.

3.5.F Reactor Core Isolation Cooling

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is operable during such time.

3. If specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurizeed to less than 122 psig within 24 hours.

G. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be operable:

   (1) prior to a startup from a Cold Condition, or,

   (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.

2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a hot shutdown condition in 6 hours and in a cold shutdown condition in the following 18 hours.

4.5.F Reactor Core Isolation Cooling

2. When it is determined that the RCICS in inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

G. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered, in 4.6.D.2.

2. When it is determined that more than two of the ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

3.5.G Automatic Depressurization System (ADS)

3. If specifications 3.5.G.1 and 3.5.G.2 cannot be met, an orderly shutdown will be initiated and the reactor vessel pressure shall be reduced to 105 psig or less within 24 hours.

H. Maintenance of Filled Discharge Pipe

Whenever the core spray systems, LPCI, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5.G Automatic Depressurization System (ADS)

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray systems, LPCI, HPCI, and RCIC are filled:

LIMITING CONDITIONS FOR OPERATION

3.5.H Maintenance of Filled Discharge Pipe

The suction of the RCIC and HPCI pumps shall be aligned to the condensate storage tank, and the pressure suppression chamber head tank shall normally be aligned to serve the discharge piping of the RHR and CS pumps. The condensate head tank may be used to serve the RHR and CS discharge piping if the PSC head tank is unavailable. The pressure indicators on the discharge of the RHR and CS pumps shall indicate not less than listed below.

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

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, -2, -3, -4, -5, & -6. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

SURVEILLANCE REQUIREMENTS

4.5.H Maintenance of Filled Discharge Pipe

1. Every month prior to the testing of the RHRS (LPCI and Containment Spray) and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.

2. Following any period where the LPCI or core spray systems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

4. When the RHRS and the CSS are required to be operable, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.

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 $\geq$ 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

$$LHGR_{max} \leq LHGR_d [1 - (\Delta P/P)_{max} (L/LT)]$$

$LHGR_d$ = Design LHGR = 18.5 kW/ft. for 7x7 fuel
= 13.4 kW/ft for 8x8 fuel

$(\Delta P/P)_{max}$ = Maximum power spiking penalty
= 0.026 for 7X7 fuel
= 0.022 for 8X8 fuel

LT* = Total core length = 12.0 feet for 7X7 fuel
= 12.2 feet for 8X8 fuel

L = Axial position above bottom of core

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

K. Minimum Critical Power Ratio (MCPR)

The MCPR operating limit for cycle 3 is 1.34 for 7x7 fuel and 1.43 for 8x8 and 8x8R fuel. These limits apply to steady state power operation at rated power and flow. For core flows other than rated, the MCPR shall be greater than the above limits times K_f. K_f is the value shown in Figure 3.5.2. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

L. Reporting Requirements

If any of the limiting values identified in Specifications 3.5.1, J, or K are exceeded and the specified remedial action is taken, the event shall be logged and reported in a 30-day written report.

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.

3.5 BASES

3.5.G Automatic Depressurization System (ADS)

This specification ensures the operability of the ADS under all conditions for which the depressurization of the nuclear system is an essential response to station abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray subsystems can operate to protect the fuel barrier. Note that this specification applies only to the automatic feature of the pressure relief system.

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures yet be fully capable of performing their pressure relief function.

Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS.

With two ADS valves known to be incapable of automatic operation, four valves remain operable to perform their ADS function. The ECCS loss-of-coolant accident analyses for small line breaks assumed that four of the six ADS valves were operable. Reactor operation with three ADS valves inoperable is allowed to continue for seven days provided that the HPCI system is demonstrated to be operable. Operation with more than three of the six ADS valves inoperable is not acceptable.

3.5 BASES

3.5.H Maintenance of Filled Discharge Pipe

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

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

3.5.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 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1,-2,-3,-4,-5 &-6. The analyses supporting these limiting values is presented in NEDO-24056 and NEDO-24136.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with monthly tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, cause the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period was caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Whenever a CSCS system or loop is made inoperable because of a required test or calibration, the other CSCS systems or loops that are required to be operable shall be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of 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. 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 Specifications 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 logged and 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.

M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications 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. General Electric BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

TABLE 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DR265H

Average Planar Exposure (MWd/t)	MAPLHGR (kw/ft))	PCT (^{o}F)
200	11.5	1707
1000	11.6	1698
5000	11.9	1681
10,000	12.1	1666
15,000	12.1	1688
20,000	11.9	1687
25,000	11.3	1639
30,000	10.7	1580

TABLE 3.5.I-6

MAPPHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DR265L

Average Planar Exposure (MWd/t)	MAPLHGR (kw/ft)	PCT (^{o}F)
200	11.6	1711
1000	11.6	1700
5000	12.1	1692
10,000	12.1	1663
15,000	12.1	1683
20,000	11.9	1683
25,000	11.3	1637
30,000	10.7	1579

DELETE

3.6.C **Coolant Leakage**

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

D. **Relief Valves**

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

E. **Jet Pumps**

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C **Coolant Leakage**

D. **Relief Valves**

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief valve bellows shall be continuously monitored.

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

E. **Jet Pumps**

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

3.6.E Jet Pumps

3.6.F Jet Pump Flow Mismatch

1. When both recirculation pumps are in steady state operation, the speed of the faster pump shall be maintained within 122% the speed of the slower pump when core power is 80% or more of rated power or 135% the speed of the slower pump when core power is below 80% of rated power.

2. If specification 3.6.F.1 cannot be met, one recirculation pump shall be tripped.

3. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

4. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

5. Steady state operation with both recirculation pumps out of service for up to 12 hrs is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor vessel water as determined by dome pressure.

G. Structural Integrity

1. The structural integrity of the primary system shall be

4.6.E Jet Pumps

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

2. Whenever there is recirculation flow with the reactor in the Startup or Run Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

F. Jet Pump Flow Mismatch

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

G. Structural Integrity

1. Table 4.6.A together with supplementary notes, specifies the

Amendment No. ~~35~~, ~~41~~, ~~43~~, 47

3.6/4.6 **BASES:**

The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 36 rem at the exclusion distance during the 2-hour period following a steam line break. This dose is computed with the conservative assumption of a release of 140,000 lbs of coolant prior to closure of the isolation valves, and a X/Q value of 3.4 x 10^{-4} Sec/m^3.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited to 5 percent of total operation.

The sampling frequencies are established in order to detect the occurrence of an iodine transient which may exceed the equilibrium concentration limit, and to assure that the maximum coolant iodine concentrations are not exceeded. Additional sampling is required following power changes and off-gas transients, since present data indicate that the iodine peaking phenomenon is related to these events.

3.6.C/4.6.C Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mecnanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shut down to allow further investigation and corrective action.

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

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

REFERENCES

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

3.6.D/4.6.D Relief Valves

To meet the safety design basis, thirteen relief valves have been installed on the unit with a total capacity of 87.6% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1273 psig if a neutron flux scram is assumed considering 12 valves operable. This results in an 102 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the operational design, the analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in Reference 5. This analysis shows that 12 of the 13 relief valves limit pressure in the steam line to 1201 psig. This analysis shows that peak system pressure is limited to 1229 psig which is 146 psig below the allowed vessel overpressure of 1375 psig.

3.6/4.6 BASES:

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their set points are within the ± 1 percent tolerance. The relief valves are tested in place once per operating cycle to establish that they will open and pass steam.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.

3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973.

5. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

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

3.6.F/4.6.F Jet Pump Flow Mismatch

The LPCI loop selection logic has been previously described in the BFNP FSAR. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

Analyses indicate that above 80% power the loop select logic could be expected to function at a speed differential up to 14% of their average speed. Below 80% power the loop select logic would be expected to function at a speed differential up to 20% of their average speed. This specification provides margin because the limits are set at $\pm$ 10% and $\pm$ 15% of the average speed for the above and below 80% power cases, respectively. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 1 is located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The core shall consist of 442 fuel assemblies of 49 fuel rods each, 166 fuel assemblies of 63 fuel rods each, and 156 fuel assemblies of 62 fuel rods each.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of fuel in the new-fuel storage facility shall be such that k_{eff}, for dry conditions, is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

5.0 MAJOR DESIGN FEATURES (Continued)

B. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95. Fuel stored in the pool shall not contain more than 15.2 grams of uranium-235 per axial centimeter of fuel assembly.

C. Loads greater than 1000 pounds shall not be carried over spent fuel assemblies stored in the spent fuel pool.

5.6 SEISMIC DESIGN

The station class I structures and systems have been designed to withstand a design basis earthquake with ground acceleration of 0.2g. The operational basis earthquake used in the plant design assumed a ground acceleration of 0.1g (see Section 2.5 of the FSAR).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 47 TO FACILITY LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-259

1.0 Introduction

By letter dated September 8, 1978 (TVA BFNP TS115), as supplemented by letters dated October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979 and January 9, 1979, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1. The proposed amendment and revised Technical Specifications would incorporate the limiting conditions for operation of the facility in the third fuel cycle following the second refueling of the reactor. In support of this reload application for Browns Ferry Unit No. 1 (BF-1), the licensee has submitted a reload licensing document[1] prepared by the General Electric Company (GE), a supplemental reload licensing document[5] also prepared by GE and proposed changes to the Technical Specifications (1,2,5,22).

2.0 Discussion

Browns Ferry Unit No. 1 (BF-1) shutdown for its second refueling on November 26, 1978. During the refueling, 156 irradiated 7x7 fuel assemblies were replaced with a like number of new, two water rod, retrofit 8x8 (8x8R) fuel assemblies designed and fabricated by the General Electric Company (GE). During initial operation in fuel cycle 2 (January to November 1978), an increase in fission product activity was noted in the off-gas. During the outage, all of the irradiated fuel was "sipped" to check for possible leakage of fission products through the cladding. As a result of this operation, it was found that two of the 168 8x8 fuel assemblies that had been installed during the previous refueling evidenced a slight amount of leakage and were replaced with two 7x7 fuel assemblies irradiated during the initial fuel cycle. The development of minor leakage in two 8x8 fuel assemblies is not considered significant. The fact that all of the fuel was inspected (sipped) provided confidence that the 8x8 fuel is acceptable for use in the forthcoming fuel cycle.

This reload (Reload 2) is the first for BF-1 to incorporate GE's 8x8R fuel design on a batch basis. The description of the nuclear and mechanical design of the Reload 2 8x8R fuel and the exposed fuel designs used for initial core and Reload 1 is contained in GE's generic licensing topical report for BWR reloads.[6] Reference 6 also contains a complete set of references to GE's topical reports which describe GE's BWR reload analysis methods for the nuclear, mechanical, thermal-hydraulic, transient and accident calculations, together with information on the applicability of these methods to cores containing a mixture of different fuel designs. Portions of the plant-specific data, such as operating conditions and design parameters which are used in transient and accident calculations, have also been included in the topical report.

Our safety evaluation[7] of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, as applied to cores containing mixtures of 7x7, 8x8, and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was expressed in the staff's evaluation[8] of the information in Reference 9.

As part of our evaluation[7] of Reference 6 we found the cycle-independent input data for the reload transient and accident analyses for BF-1 to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 5, which follows the format and content of Appendix A of Reference 6.

As a result of the staff's generic evaluation[7] of a substantial number of safety considerations related to use of 8x8R fuel in mixed core loadings with 8x8 and 7x7 fuel, only a limited number of additional review items are included in this evaluation. These include the plant and cycle-specific input data and results presented in Reference 5, the LOCA-ECCS analysis results for the reload fuel design, and those items identified in Reference 7 as requiring special attention during reload reviews.

3.0 Evaluation

3.1 Nuclear Characteristics

For Cycle 3, 156 fresh 8x8R fuel bundles, with a bundle average enrichment of 2.65 wt/% U-235 will be loaded into the core, replacing a like number of exposed 7x7 assemblies. The remainder of the 764 fuel assembly reload core will consist of the irradiated 7x7 and 8x8 fuel assemblies exposed during the first two fuel cycles. The reference core loading for Cycle 3 will result in eighth core symmetry, which is consistent with previous cycles.

The information provided in Section 6 of Reference 5 indicates that the fuel temperature and void dependent behavior of the reconstituted core is not significantly different from that of previous cycles. Additionally, scram effectiveness, Figure 2 of Reference 5, is also similar to earlier cycles. The 1.7%Δk/k calculated shutdown margin for the reconstituted core meets the Technical Specification requirement that the core be subcritical by at least 0.38%Δk/k in the most reactive operating state with the single most reactive control rod fully withdrawn and all other rods fully inserted. Finally, Reference 5 indicates that a boron concentration of 600 ppm in the moderator has been calculated to make the reactor subcritical by at least 3.0%Δk at 20°C, and xenon free conditions. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System. We have reviewed these analyses and compared them to the Technical Specification requirements and find them acceptable.

3.2 Thermal-Hydraulics

3.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 7, for BWR cores which reload with GE's retrofit 8x8R fuel, the allowable minimum critical power ratio (MCPR), resulting from either core-wide or localized abnormal operational transients, is equal to 1.07. With this MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee for Cycle 3 represents a .01 increase from the 1.06 SLMCPR applicable during Cycle 2. The basis for the revised safety limit is addressed in Reference 6, while our generic approval of the new limit is given in Reference 7. This change continues to meet the recommendation of Standard Review Plan 4.4 and on that basis has been found acceptable in Reference 7.

3.2.2 Operating Limit MCPR

Various transient events will reduce the MCPR from its normal operating value. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed by the licensee to determine which event results in the largest reduction in critical power ratio. Each of the events has been conservatively analyzed for each of the several fuel types (i.e., 7x7, 8x8, 8x8R) and for the full range of exposure through the cycle.

In the transient analyses of Reference 7, credit was taken for an end-of-cycle (EOC) recirculation pump trip (RPT). (The EOC RPT is different from and should not be confused with the ATWS RPT). We have reviewed the design of the EOC RPT and conclude that it is unacceptable for reasons as given in our January 16 letter (Reference 23). Since there is no available analysis which is specific to this core, we require a conservative bound on operating limit MCPR. The previous cycle transient analyses (Reference 24) were evaluated from this standpoint. The input parameters for that cycle analyses are conservative when compared to this cycle input paramters at the EOC. This includes comparisons of void reactivity coefficient, scram reactivity insertion, and Doppler reactivity coefficient which are the key parameters for core-wide transient behavior. The key parameters for CPR evaluations, which are also conservative for last cycle's analysis, are power peaking factor, bundle flow rate and initial CPR. With these conservative input parameters, the transient results for last cycle are bounding for this cycle at EOC. Therefore, we have proposed and the licensee has agreed to operating limit MCPRs of 1.34 for 7x7 fuel and 1.43 for 8x8 and 8x8R fuel. These were derived from a safety limit MCPR of 1.07 and ΔCPR of 0.27 for 7x7 fuel and 0.36 for 8x8 fuel. This assures that an abnormal operational transient will result in a CPR no lower than the 1.07 safety limit which we find acceptable as discussed in the previous section.

3.3 Accident Analysis

3.3.1 ECCS Appendix K Analysis

The licensee has reevaluated the adequacy of ECCS performance in connection with the new reload fuel design, using methods previously approved by the staff. The results of these plant-specific analyses are given in Reference 5.

We have reviewed the information submitted by the licensee and conclude that all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 will be met when the reactor is operated in accordance with the MAPLHGR versus Average Planar Exposure values given in Section 6 of Reference 5 and which have been incorporated in the revised Technical Specifications.

Continuous operation with four of six automatic depressurization system (ADS) valves operable (instead of the previous five out of six requirement) has been found acceptable in References 16 and 17. We have reviewed this acceptance and its applicability to BF-1. On the bases of our review and the conclusions reached in References 16 and 17, we find the proposed change from five to four to be acceptable.

3.3.2 Control Rod Drop Accident

Because the characteristic accident analysis input parameters for the worst case CRDA did not satisfy all of the assumptions of the bounding analysis, the licensee reanalyzed this event on a plant-specific basis. The results showed the peak fuel enthalpy to be less than the 280 cal/gm limit which is acceptable.

3.3.3 Failure of Trip Inputs from Turbine Building to Reactor Protection System

During our review of the reactor protection system, we noted that the trip inputs for the recirculation pump trip and reactor scram following load rejection or turbine trip originate in the turbine building. The turbine building, as is the case of most boiling water reactor plants, is not seismically qualified, hence, its integrity and functions cannot be assured in the event of an earthquake.

For these reasons, the licensee was requested to analyze the consequences of a safe shutdown earthquake concurrent with the limiting transient event without taking credit for reactor scram or recirculation pump trip from the turbine building inputs. Browns Ferry Unit 1 has referenced a Hatch Unit 2 analysis. We have compared the significant parameters for these two plants (bundle power level and critical power ratio change) and have concluded that the Hatch 2 analysis conservatively bounds the Browns Ferry Unit 1 conditions. We agree with the licensee that this analysis is applicable to Browns Ferry and on the basis of previous staff findings on this analysis,[19] we find the results acceptable.

3.3.4 Fuel Loading Error

The licensee has also considered the effect of a possible fuel loading error on bundle CPR. An analysis of the most severe misoriented fuel loading error using GE's new methodology,[13, 14] which as modified, has been approved[15] by the staff, shows that the worst possible rotation of a fuel bundle will not cause a violation of the 1.07 safety limit MCPR. Additionally, an analysis of the most severe mislocated fuel bundle with GE's new, approved methodology shows that the worst potential mislocation will not violate the MCPR safety limit. We find the results of these analysis acceptable.

3.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 7. As specified in Reference 20, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable.

3.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed with the methods described in Reference 6. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state (corresponding to the intersection of the natural circulation curve and 105% rod line on the power-flow map) are below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. Although a final test report has not as yet been received by the staff for review, it is expected that the test results will aid considerably in resolving the staff concerns.

For the previous operating cycle, the staff, as an interim measure, added a requirement to the Technical Specifications which restricted planned operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins for the current cycle so that the decay ratio is <1.0 in all operating modes. On the basis of the foregoing, the staff considers the plant thermal-hydraulic stability characteristics to be acceptable.

4.0 Physics Startup Testing

The licensee will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during Cycle 2. The tests will check that the core is loaded as intended, that the incore monitoring system is functioning as expected, and that the process computer has been reprogrammed to properly reflect changes associated with the reload. The test program is consistent with that previously found acceptable for Browns Ferry Unit 3.[11] We find this test program to be acceptable.

5.0 Technical Specification Changes

The proposed Technical Specification changes include a revised fuel cladding integrity safety limit MCPR, a revised operating limit minimum critical power ratios (MCPR) for each fuel type, addition of a MAPLHGR vs average planar exposure table and addition of a design maximum total peaking factor for the reload 8x8R fuel assemblies. The revised 1.07 safety limit MCPR results in a .01 increase from the 1.06 safety limit MCPR (SLMCPR) of the previous cycle. Based on our generic review,[7] we find the use of a 1.07 SLMCPR to be acceptable (Section 3.2.1, herein). Also, based on the discussions appearing in Section 3.2.2 herein, the staff finds the proposed operating limit MCPRs, as modified to reflect analysis uncertainties, to be acceptable. We find that the proposed MAPLHGR vs average planar exposure table is adequate to assure conformance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 for the reload 8x8R fuel assemblies per Section 3.3.1, herein.

The proposed flow biased APRN upscale rod block has been revised. The revision reduces the setpoint for this rod block by 1% of rated power. This reduction will result in a less severe rod withdrawal error, because the transient will be terminated earlier. The rod withdrawal error analysis utilized this revision. Our evaluation of these results showed that the safety limit MCPR criteria was met and therefore, the revision is acceptable.

The Technical Specifications have been modified to adjust the number of operable ADS valves based on the findings, as discussed in Section 3.3.1, herein.

A calculational constant (Total Core Length) for the 8x8R LHGR evaluation has been added to the Technical Specifications. This has been previously found acceptable in Reference 5. Since the fuel for this reload is identical to that of the Reference 5 evaluation, we find this addition acceptable.

Finally. the Technical Specifications, which are associated with safety/relief valve number and operability, are being revised. The revisions allow replacement of two safety valves with two safety/relief valves which will be aligned identically to the present safety/relief valves. Section 3.4, herein, has found that acceptable overpressurization protection is provided by these specifications. Therefore, the modification is acdeptable.

Environmental Considerations

The revised operating limit minimum critical power ratios (OLMCPR) discussed in Section 3.2.2 may result in a restriction on the attainable power generation. The reduction in rated power level is estimated to be minimal (a few percent) for the first part of the fuel cycle. It is expected that the present OLMCPRs will be revised to be less restrictive when satisfactory documentation is received relating to the testing of the EOC RPT. Thus, the reduction in rated power level, if any, will be for a limited period of time. The small reduction in power from one unit of the BFNP for a limited period does not affect the environmental evaluation contained in the Final Environmental Statement (FES) related to operation of the Browns Ferry Nuclear Plant, Units 1, 2 and 3, issued September 1, 1972, There will be no significant change in the other environmental impacts identified in the FES. This amendment does not authorize a change in effluent types or total amounts. We conclude that this amendment will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 17, 1979

<u>References:</u>

1. Tennessee Valley Authority (TVA) letter (Gilleland) to USNRC (Denton) dated September 8, 1978.

2. TVA letter (Gilleland) to USNRC (Denton) dated October 5, 1978.

3. TVA letter (Gilleland) to USNRC (Denton) dated November 30, 1978.

4. TVA letter (Gilleland) to USNRC (Denton) dated December 5, 1978.

5. "Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 1 Reload 2," NEDO-24136, Rev. 1, November 1978.

6. "Generic Reload Fuel Application," General Electric Report, NEDE-24011-P-3, dated March 1978.

7. USNRC letter (Eisenhut) to General Electric (Gridley) dated May 12, 1978, transmitting "Safety Evaluation for the General Electric Topical Report, 'Generic Reload Fuel Application,' (NEDE-24011-P)."

8. "Status Report on the Licensing Topical Report, General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by the Division of Technical Review, Office of Nuclear Reactor Regulation, USNRC, April 1975.

9. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360 Revision 1, Supplement 4, April 1, 1976.

10. Memo from P. S. Check (RSB-NRC) to T. A. Ippolito (ORB#3-NRC), "Browns Ferry 3 - Cycle 2 Reload (TACS #8026)," November 8, 1978.

11. NRC letter (Ippolito) to TVA (Hughes), Amendment Nos. 45, 41, and 18 to Facility License No. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant Units Nos. 1, 2, and 3, dated November 18, 1978.

12. GE letter (Fuller) to NRC (Ross) dated January 13, 1978.

13. GE letter (Engle) to NRC (Eisenhut), "Fuel Assembly Loading Error" dated June 1, 1977.

14. GE letter (Engle) to NRC (Eisenhut) dated November 30, 1977.

15. NRC letter (Eisenhut) to GE (Engle) dated May 8, 1978.

16. NRC Safety Evaluation Report on Operation of Browns Ferry Units 1 and 2 with Four of the Six ADS Valves Operable, May 7, 1978.

17. NRC letter (Ippolito) to TVA (Hughes), Amendment No. 35 to Facility License No. DPR-52 for Browns Ferry Nuclear Plant Unit No. 2 Cycle 2, dated June 21, 1978.

18. Edwin I. Hatch Nuclear Plant, Unit 2, FSAR Question 212.64 (15.1.1), (15.2.2).

19. Safety Evaluation Report related to operation of Edwin I. Hatch Nuclear Plant, Unit No. 2, Georgia Power Company, et. al., USNRC, Office of Nuclear Reactor Regulation, Docket No. 50-366, NUREG-0411, June 1978.

20. Letter, E. D. Fuller (GE) to USNRC (Ross), "Impact of One-Dimensional Transient Model on Plant Operations Limits," June 26, 1978.

21. Carmichael, L. A., and Niemi, R. O., "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2," EPRI-NP-564, June 1978.

22. TVA letter (Gilleland) to USNRC (Denton) dated January 8, 1979.

23. Letter, T. A. Ippolito (NRC) to H. G. Parris (TVA), January 16, 1979.

24. General Electric Boiling Water Reactor Reload-1 Licensing Amendment for Browns Ferry Nuclear Plant Unit 1, NEDO-24020, May 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-259

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 47 to Facility Operating License No. DPR-33 issued to Tennessee Valley Authority (the licensee), which revised the Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 1 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment permits operation of Browns Ferry Unit No. 1 in Cycle No. 3 following the second refueling outage.

The application for this amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated September 8, 1978, as supplemented by letters dated October 5, 1978, November 30, 1978, December 5, 1978, December 14, 1978, January 8, 1979 and January 9, 1979, (2) Amendment No. 47 to License No. DPR-33, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 17th day of January 1979

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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