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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated February 9, 1979, as supplemented by letters dated May 15, 1979 and May 16, 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

R. Rooney for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 25, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

vii/viii
7/8
9/10
15/16
17/18
19/20
21/22
23/24
25/26
29/30
71/72
101/102
113/114
131/132
133/134
139/140
159/160
167/168
169/170
181/182
219/220
329/330

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

3. Add the following new page:

172a

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

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.

SAFETY LIMIT

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.

LIMITING SAFETY SYSTEM SETTING

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)

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

.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 ≤ 18.5 kw/ft for 7X7 fuel and 13.4 kw/ft for 8X8 and 8X8R fuel, MCPR within limits of Specification 3.5.k. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

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

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

B. Core Thermal Power Limit (Reactor Pressure ≤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:

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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:

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

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

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 valves

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

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 = 1.07) operation is constrained to a maximum LHGR of 18.5 kw/ft for 7x7 fuel and 13.4 kw/ft for 8x8 and 8x8R 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. General Electric Reload Licensing Amendment for BFNP Unit 2 Reload No. 2, NEDO-24169. January 1979 as amended by NEDO-24169A.

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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.

2.1 BASES

In summary

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

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

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

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

The bases for individual set points are discussed below:

A. Neutron Flux Scram

1. APRM 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 possibili. of unnecessary scrams.

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 recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varried with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin

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2.1 BASES

from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. 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. 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 scram set at 10 percent of valve closure, neutron flux does not increase.

2.1 BASES

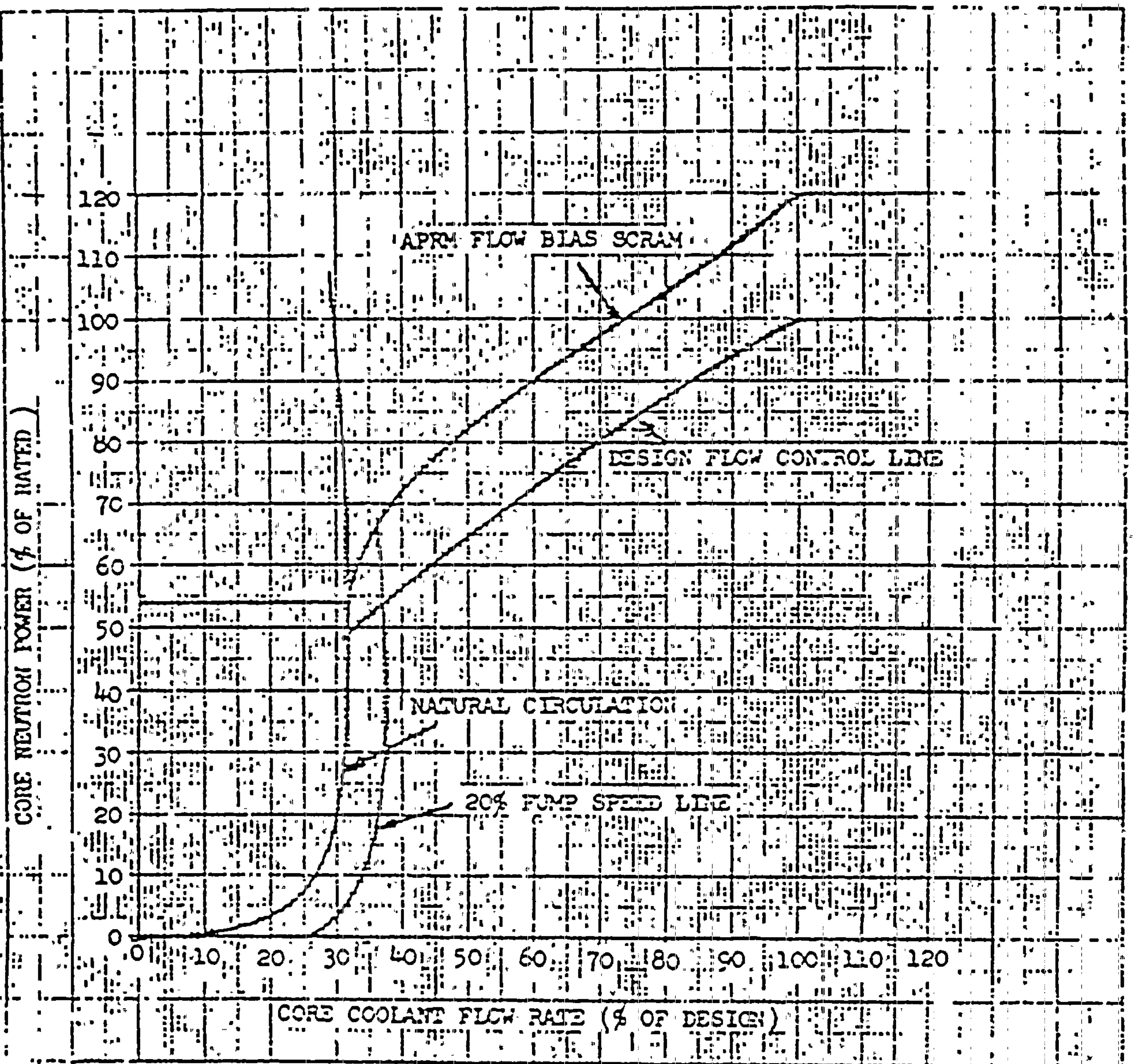
I. J. & K. Reactor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps.

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. General Electric Reload Licensing Amendment for BFNP Unit 2 Reload No. 2, NEDO-24169, January 1979 and NEDO-24169A.



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

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. General Electric Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Power Station Unit 2 Reload No. 2, NEDO-24169, January 1979 and NEDO-24169A.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The pressure relief system for each unit at the Browns Ferry Nuclear Plant has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4-1 of subsection 4.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report submitted in response to question 4.1 dated December 1, 1971.

To meet the safety design basis, thirteen safety-relief valves have been installed on unit 2 with a total capacity of 84.2% 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 1299 psig if a neutron flux scram is assumed considering one relief valve is inoperable. This results in an 76 psig margin of the code allowable overpressure limit of 1375 psig.
To meet the operational design basis, the total safety-relief capacity of 84.2% of nuclear boiler rated has been divided into 70% relief (11 valves) and 14.2% safety (2 valves). 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 that10 of 11relief valves limit pressure at the safety valves to 1226 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1250 psig which is 125 psig below the allowed vessel overpressure of 1375 psig.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by section 3.5 to be operable, there shall be two operable trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required, indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec with similar pumps starting after about 14 sec and 21 sec, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of H_2O corresponds to 300% of rated steam flow at 1140 psia and 210% at 165 psia. The HPCIS setting of 90 psi corresponds to 225% of rated flow at 1140 psia and 160% at 165 psia.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.1, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Continued)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160 V shutdown boards. See note 13.

12. In only one of the four 4160 V shutdown boards. See note 13.

13. An emergency 4160 V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level (≥ 378" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one operable trip system. Therefore one trip system may be taken out of service for functional testing and calibration for a period not to exceed 8 hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds 2 consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if 1 RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 85% within 4 hours.

TABLE 4.2.B (Continued)

Function	Functional Test	Calibration	Instrument Check
RHR Area Cooler Fan Logic	Tested during functional test of instrument channels, RHR motor start and thermostat (RHR area cooler fan). No other test required.	N/A	N/A
Core Spray Area Cooler Fan Logic	Tested during logic system functional test of instrument channels, core spray motor start and thermostat (core spray area cooler fan). No other test required.	N/A	N/A
Instrument Channel - Core Spray Motors A or D Start	Tested during functional test of core spray pump (refer to section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray Motors B or C Start	Tested during functional test of core spray pump (refer to section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray Loop 1 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
Instrument Channel - Core Spray Loop 2 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
RHRSW Initiate Logic	once/6 months	N/A	N/A
RPT initiate logic	once/month	N/A	N/A
RPT breaker	once/operating cycle	N/A	N/A

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/3 months	once/day (8)
APRM Downscale	(1) (13)	once/3 months	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (Flow Bias)	(1) (13)	once/6 months	once/day (8)
RBM Downscale	(1) (13)	once/6 months	once/day (8)
RBM Inoperative	(1) (13)	N/A	once/day (8)
IRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1)(2) (13)	N/A	N/A
SRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Detector not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1)(2) (13)	N/A	N/A
Flow Bias Comparator	(1)(15)	once/operating cycle (20)	N/A
Flow Bias Upscale	(1)(15)	once/3 months	N/A
Rod Block Logic	(16)	N/A	N/A
RSCS Restraint	(1)	once/3 months	N/A.

3.2 BASES

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

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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).

3/4.3 BASES:

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 RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from 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 Specification 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 occurred during plant startup and was determined to be [illegible] 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.

3.3/4.3 BASES:

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:

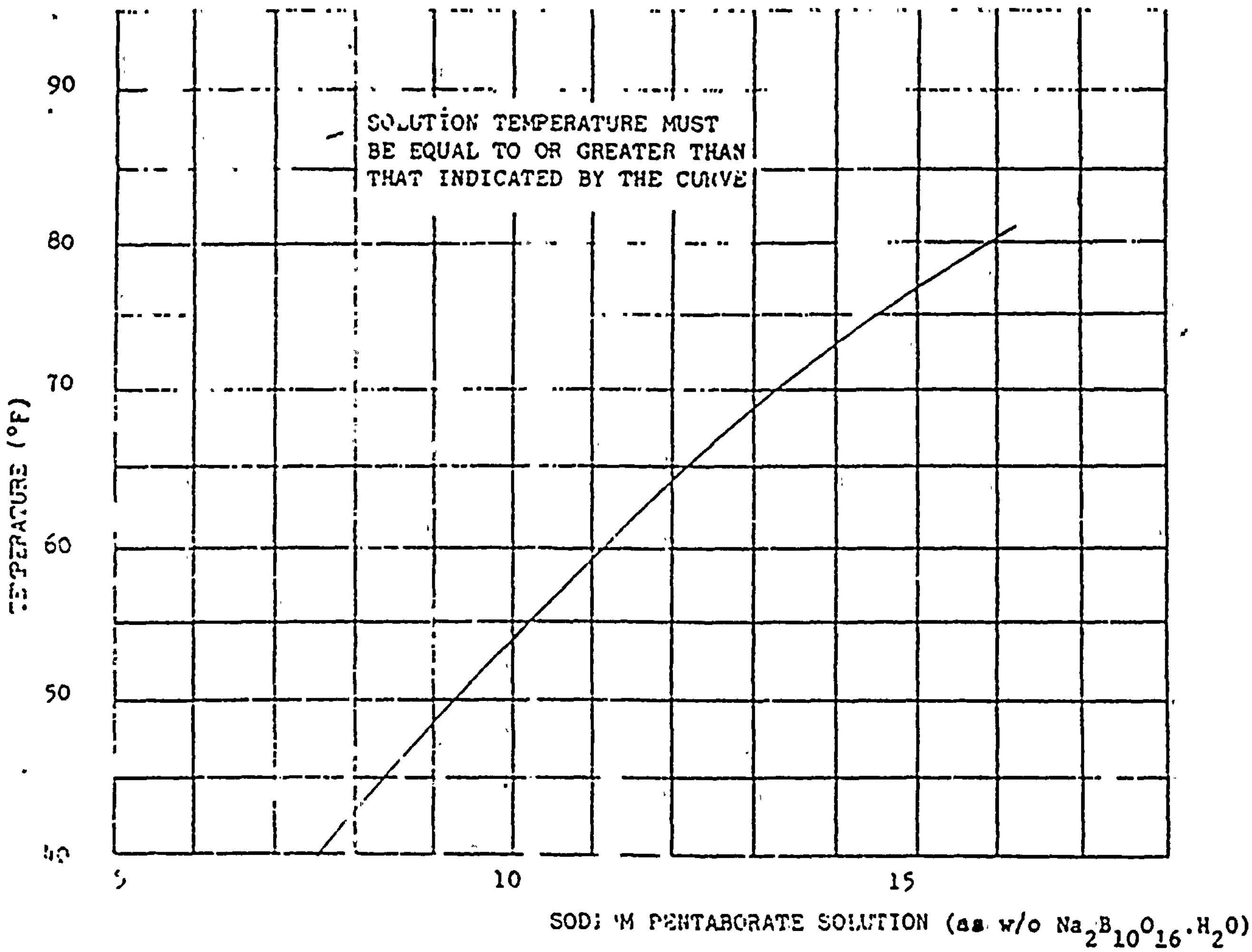
D. Reactivity Anomalies

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 Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Power Station Unit 2 Reload No. 2, NEDO-24169, January 1979 and NEDO-24169A.



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
TEMPERATURE REQUIREMENTS

FIGURE 3.4-2

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.

.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,and -5. 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.

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:

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 and 8x8R fuel

LT = Total core length = 12.0 feet for 7x7 and 8x8 fuel
= 12.5 feet for 8x8R 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 BFNP 2 cycle 3 is 1.33 for 7X7, 1.30 for 8X8, and 1.28 for 8X8R fuels. 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.I, 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 ≥ 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. The analyses supporting these limiting values is presented in NEDO-24088 and NEDO-24169.

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

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 LicensingAmendment for BFNP Unit 2, NEDO-24169, January 1979 and NEDO-24169A.

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.

TABLE 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB284

AVERAGE PLANAR EXPOSURE (MWd/t)	MAPLHGR (kW/ft)	PCT (^{o}F)
200	11.2	1685
1,000	11.3	1667
5,000	11.8	1671
10,000	12.0	1647
15,000	12.0	1669
20,000	11.8	1672
25,000	11.2	1633
30,000	10.8	1596

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.C Coolant Leakage	4.6.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. Safety and Relief Valves	D. Safety and Relief Valves
1. When more than one relief valve or one or more safety valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.	1. At least one safety valve and approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves (2 safety and 11 relief) 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/safety valve bellows shall be continuously monitored. 4. At least one relief valve shall be disassembled and inspected each operating cycle.
E. Jet Pumps	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.	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.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.E Jet Pumps	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%.
3.6.F Jet Pump Flow Mismatch	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.	1. Recirculation pump speeds shall be checked and logged at least once per day.
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. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hrs.	
G. Structural Integrity	G. Structural Integrity
1. The structural integrity of the primary system shall be	1. Table 4.6.A together with supplementary notes, specifies the

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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 Safety and Relief Valves

The safety and relief valves are required to be operable above the pressure (105 psig) at which the core spray systems is not designed to deliver full flow. The pressure relief system for each unit at the Browns Ferry Nuclear Plant has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4-1 of subsection 4.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance, as modified by Reference 4, with the ASME Code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report submitted in Amendment 22 in response to question 4.1 dated December 6, 1971

To meet the safety design basis, thirteen safety-relief valves have been installed on unit 2 with a total capacity of 84.2% 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 1299 psig if a neutron flux scram is assumed considering one relief valve is inoperable. This results in an 76 psig margin of the code allowable overpressure limit of 1375 psig.
To meet the operational design basis, the total safety-relief capacity of 84.2% of nuclear boiler rated has been divided into 70% relief (11 valves) and 14.2% safety (2 valves). The analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming

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a turbine trip scram is presented in Reference 5 on page 29. This analysis shows that10 of 11relief valves limit pressure at the safety valves to 1226 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1250 psig which is 125 psig below the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief and safety 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. General Electric BWR Reload 2 Licensing Amendment for BFNP Unit 2, NEDO-24169,January 1979.and NEDO-24169A.

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.

4. Daily tests of annunciation lights and audible devices are performed as a routine operation function.

5. The CO_2 system manufacturer recommends semiannual testing of CO_2 system fire detection circuits.

Figure 6.3-1 describes the in-plant fire protection organization including the roving fire watch. In addition, other operating personnel periodically inspect the plant during their normal operating activities for fire hazards and other abnormal conditions.

Smoke detectors will be tested "in-place" using inert freon gas applied by a pyrotronics type applicator which is accepted throughout the industrial fire protection industry for testing products of combustion detectors or by use of the MSA chemical smoke generators. At the present time the manufacturers have only approved the use of "punk" for creating smoke. TVA will not use "punk" for testing smoke detectors.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 2 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 364 fuel assemblies of 49 fuel rods each, 168 fuel assemblies of 63 fuel rods each, and 232 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).