

ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGES  
BROWNS FERRY NUCLEAR PLANT UNIT 2

8007170 566

1. FUEL CLADDING INTEGRITY2.1 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, 8x8R, and P8x8R, and MCPR within limits of Specification 3.5.k. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

2. APRM--When the reactor mode switch is in the STARTUP POSITION, the APRM scram shall be set at less than or equal to 15% of rated power.
3. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

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

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

## 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 occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNW 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, 8x8R, and P8x8R. 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  $28 \times 10^3$  lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than  $28 \times 10^3$  lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

For the fuel in the core during periods when the reactor is 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.

## 2.1 BASES

1. 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," NEDC-10802, Feb., 1973.
2. Generic Reload Fuel Application, Licensing Topical Report NEDE-20411-P-A, and Addenda.

## 1.2 BASES:

### REACTOR COOLANT SYSTEM INTEGRITY

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

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

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

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

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the 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)

## 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 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 which, if a neutron flux scram is assumed considering one relief valve is inoperable, has adequate margin to the code allowable overpressure limit of 1375 psig. To meet the operation 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 limiting plant isolation transient is presented in the supplemental reload licensing submittal for the current cycle. This analysis shows that 10 of 11 relief valves limit pressure at the safety valves to a value which is below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to a value which is well below the allowed vessel overpressure of 1375 psig.



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 per 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, (i.e., MCPR given by Specification 3.5.k or LHCR of 18.5 kw/ft for 7 x 7 or 13.4 for 8 x 8, 8 x 8R & P8 x 8R fuel). 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: i.e., 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 caused by

### 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  $1\% \Delta K$ . Deviations in core reactivity greater than  $1\% \Delta K$  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. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A, and Addenda.

OPERATING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

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.

PI-75-20	48 psig
PI-75-48	48 psig
PI-74-51	48 psig
PI-74-69	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 of 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.

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:

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.

$$\text{LHGR}_{\text{max}} < \text{LHGR}_G [1 - (\Delta P/P)_{\text{max}} (L/LT)]$$

$$\text{LHGR}_d^{\text{max}} = \text{Design LHGR} = 18.5 \text{ kW/ft for } 7 \times 7 \text{ fuel}$$

$$= 13.4 \text{ kW/ft for } 8 \times 8, 8 \times 8\text{R, and } P8 \times 8\text{R fuel}$$

$(\Delta P/P)_{\text{max}}$  = Maximum power spiking penalty  
 = 0.026 for 7x7 fuel  
 = 0.022 for 8x8, 8x8R, and P8x8R fuel

LT = Total core length = 12.0 ft for 7x7 & 8x8  
 = 12.5 ft for 8x8R & P8x8R

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 BFN 2 cycle 4 is 1.32 for 7X7, 1.27 for 8X8, 8x8R, and P8x8R 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  $\geq 25\%$  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 RA2XS

#### 3.5.H Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, NPCIS, 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 will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the NPCI and RCIC pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the NPCIS and RCICS piping. This assures that the NPCI 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 LOCFRSO, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than  $\pm 20^\circ\text{F}$  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 LOCFRSO 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 reference 4.

### 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 R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

### 3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at maximum 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.

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

TABLE 3.5.I-5

## MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB284 and P8DRB284

AVERAGE PLANAR EXPOSURE (MWD/t)	MAPLHGR (kW/ft)	PCT (°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

### 3.6/4.6 PACES

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 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 which, if a neutron flux scram is assumed considering one relief valve is inoperable, has adequate margin to 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 limiting plant isolation transient is presented in the supplemental reload licensing submittal for the current cycle. This analysis shows that 10 of 11 relief valves limit pressure at the safety valves to a value which is below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to a value which is well below the allowed vessel overpressure of 1375 psig.



### 3.6/4.6 BASES:

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  $\pm 1$  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. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A, and Addenda.

### 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  $\pm 5$  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.

## 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 reactor core may contain 764 fuel assemblies consisting of 7x7 assemblies having 49 fuel rods each, 8x8 assemblies having 63 fuel rods each, and 8x8R (and P8x8R) assemblies having 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).

ENCLOSURE 2