50-259

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 7, 1997

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: ISSUANCE OF AMENDMENTS - BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M95843, M95844, AND M95845) (TS 377)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment Nos.247 and 207 to Facility Operating License Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant (BFN) Units 2 and 3, respectively. These amendments are in response to your application dated June 21, 1996, with supplemental information provided on February 7, 1997, requesting changes to the minimum critical power ratio safety limit for Units 2 and 3.

The BFN Unit 1 amendment is not being issued at this time. Staff approval for Unit 1, is contingent upon that TVA documenting completion of analyses for that unit, using appropriate methodology, as was done for Units 2 and 3.

TVA's letter of June 21 also provided changes to the Technical Specification Bases to clarify requirements for supplemental spent fuel pool cooling by the residual heat removal system. These changes are being provided for all three units.

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DFOI

A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance of Amendment to Facility Operating License will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

Joseph F. Williams, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260 and 50-296

Enclosures:
1. Amendment No. 247 to License No. DPR-52
2. Amendment No. 207 to License No. DPR-68
3. Revised TS Bases, Units 1, 2 and 3
4. Safety Evaluation

cc w/enclosures: See next page

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4/29/97

cc:
Mr. O. J. Zeringue, Sr. Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Jack A. Bailey, Vice President
Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. C. M. Crane, Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

General Counsel
Tennessee Valley Authority
ET 10H
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. Raul R. Baron, General Manager
Nuclear Assurance and Licensing
Tennessee Valley Authority
4J Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Masoud Bajestani, Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Pedro Salas, Manager
Licensing and Industry Affairs
Tennessee Valley Authority
4J Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Timothy E. Abney, Manager
Licensing and Industry Affairs
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
61 Forsyth Street, SW., Suite 23T85
Atlanta, GA 30303-3415

Mr. Leonard D. Wert
Senior Resident Inspector
Browns Ferry Nuclear Plant
U.S. Nuclear Regulatory Commission
10833 Shaw Road
Athens, AL 35611

Chairman
Limestone County Commission
310 West Washington Street
Athens, AL 35611

State Health Officer
Alabama Department of Public Health
434 Monroe Street
Montgomery, AL 36130-1701

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 247
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 June 21, 1996, and supplemented on February 7, 1997, 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 Operating 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. 247 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 247

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf pages are included to maintain document completeness.

REMOVE	INSERT
1.1/2.1-1	1.1/2.1-1
1.1/2.1-2	1.1/2.1-2*
1.1/2.1-8	1.1/2.1-8
1.1/2.1-9	1.1/2.1-9
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
1.1/2.1-14	1.1/2.1-14
1.1/2.1-15	1.1/2.1-15
3.3/4.3-17	3.3/4.3-17
3.3/4.3-18	3.3/4.3-18*

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

1.1 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. Thermal Power Limits

1. 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 1.10 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.

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (RUN Mode) (Flow Biased)

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1.A Neutron Flux Trip Settings 2.1.A.1.a (Cont'd) $S \leq (0.58W + 62\%)$ where: S = Setting in percent of rated thermal power (3293 MWt) W = Loop recirculation flow rate in percent of rated b. 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.

1.1 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 perforation 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). Maintaining the MCPR greater than the Safety Limit MCPR 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 possibility 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 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 percent 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 MCPR 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 (Cont'd)

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 equal to the Safety Limit MCPR 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 1,100°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 1,400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 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 percent. Thus, a core thermal power limit of 25 percent 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.

2.1 <u>BASES</u> (Cont'd)

The bases for individual setpoints are discussed below:

A. <u>Neutron Flux Scram</u>

1. <u>APRM Flow-Biased High Flux Scram Trip Setting (RUN Mode)</u>

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 core average neutron flux.

During power increase transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased scram provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR is greater than the Safety Limit MCPR when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/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. Worth of individual rods is very low in a uniform rod pattern. Thus, of all 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 five 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 eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five 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 example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

2.1 BASES (Cont'd)

IRM Flux Scram Trip Setting (Continued)

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 percent 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. 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 discussed in paragraph 7.5.5.4 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 the Safety Limit MCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

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). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

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 prevents scram actuation. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting is selected to provide adequate margin to the flow-biased scram setpoint.

2.1 BASES (Cont'd)

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the Safety Limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than the Safety Limit MCPR.

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

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

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

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

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

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.207
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 21, 1996, and supplemented on February 7, 1997, 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 Operating License No. DPR-68 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 207 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf pages are included to maintain document completeness.

REMOVE	INSERT
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1.1/2.1-14	1.1/2.1-14
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SAFETY LIMIT

1.1 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.

Specification

A. Thermal Power Limits

1. 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 1.10 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 Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) (Flow Biased)

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1.A Neutron Flux Trip Settings 2.1.A.1.a (Cont'd) $S \leq (0.58W + 62\%)$ where: S = Setting in percent of rated thermal power (3293 MWt) W = Loop recirculation flow rate in percent of rated b. 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.

AMENDMENT NO. 190

1.1 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 perforation 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). Maintaining the MCPR greater than the Safety Limit MCPR 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 possibility 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 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 percent 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 MCPR 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 <u>BASES</u> (Cont'd)

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 equal to the Safety Limit MCPR 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 1,100°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 1,400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a flow of 28×10^3 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 28×10^3 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 percent. Thus, a core thermal power limit of 25 percent 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.

2.1 BASES (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased 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 core average neutron flux.

During power increase transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased scram provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR is greater than the Safety Limit MCPR when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/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. Worth of individual rods is very low in a uniform rod pattern. Thus, of all 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 eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five 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 example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions for that range.

2.1 BASES (Cont'd)

IRM Flux Scram Trip Setting (Continued)

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 percent 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. 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 discussed in paragraph 7.5.5.4 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 the Safety Limit MCPR. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

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). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

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 prevents scram actuation. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting is selected to provide adequate margin to the flow-biased scram setpoint.

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the Safety Limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit MCPR. The limiting power transients are given in Reference 1. Analysis of these transients shows that the negative reactivity rates resulting from the scram with the average response of all drives as given in the above specifications provide the required protection and MCPR remains greater than the Safety Limit MCPR.

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

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

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

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

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

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

Browns Ferry Nuclear Plant

Revised Technical Specification Bases

Clarification of Supplemental Spent Fuel Cooling Requirements

	REMOVE	INSERT
Unit 1	3.10/4.10-13 3.10/4.10-14	3.10/4.10-13* 3.10/4.10-14
Unit 2	3.10/4.10-13 3.10/4.10-14	3.10/4.10-13* 3.10/4.10-14
Unit 3	3.10/4.10-13 3.10/4.10-14	3.10/4.10-13 3.10/4.10-14*

* Denotes overleaf page

3.10 BASES (Cont'd)

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

The SRMs are provided to monitor the core during periods of unit shutdown and to guide the operator during refueling operations and unit startup. Requiring two OPERABLE SRMs (FLCs) during CORE ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

Each SRM (FLC) is not required to read ≥ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

3.10 BASES (Cont'd)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

C. Spent Fuel Pool Water

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

The fuel pool cooling system is designed to maintain the pool water temperature less than 125°F during normal heat loads. If the reactor core is completely unloaded when the pool contains two previous discharge batches, the temperature may increase to greater than 125°F. The RHR system supplemental fuel pool cooling mode can be used under these conditions to maintain the pool temperature to less than 125°F.

3.10.D/4.10.D BASES

Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The five-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

3.10.E/4.10.E

Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to

3.10 BASES (Cont'd)

REFERENCES

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The SRMs are provided to monitor the core during periods of unit shutdown and to guide the operator during refueling operations and unit startup. Requiring two OPERABLE SRMs (FLCs) during CORE ALTERATIONS assures adequate monitoring of the fueled region(s) and the core quadrant where CORE ALTERATIONS are being performed. The fueled region is any set of contiguous (adjacent) control cells which contain one or more fuel assemblies. An SRM is considered to be in the fueled region when one or more of the four fuel assembly locations surrounding the SRM dry tube contain a fuel assembly. An FLC is considered to be in the fueled region if the FLC is positioned such that it is monitoring the fuel assemblies in its associated core quadrant, even if the actual position of the FLC is outside the fueled region.

Each SRM (FLC) is not required to read $\geq$ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be OPERABLE. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

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The fuel pool cooling system is designed to maintain the pool water temperature less than 125°F during normal heat loads. If the reactor core is completely unloaded when the pool contains two previous discharge batches, the temperature may increase to greater than 125°F. The RHR system supplemental fuel pool cooling mode can be used under these conditions to maintain the pool temperature to less than 125°F.

D. Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The five-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

E. Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

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3.10 BASES (Cont'd)

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3.10.F Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety links provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety links are removed during decontamination.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all required refueling equipment interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its function.

B. Core Monitoring

Requiring the SRMs to be functionally tested prior to any CORE ALTERATION assures that the SRMs will be OPERABLE at the start of that alteration. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY.

REFERENCES

1. Fuel Pool Cooling and Cleanup System (BFNP FSAR Subsection 10.5)

2. Spent Fuel Storage (BFNP FSAR Subsection 10.3)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 247 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated June 21, 1996, the Tennessee Valley Authority (the licensee) requested amendments of the Technical Specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. The proposed amendments revise the safety limit minimum critical power ratio (SLMCPR) to correct a non-conservative value. On May 24, 1996, in accordance with 10 CFR Part 21, the General Electric Company informed the Nuclear Regulatory Commission that the generic calculated SLMCPR may be nonconservative for some reactor core and fuel designs. The licensee has determined that for BFN Unit 2 Cycle 9, the SLMCPR given in TS 1.1.A.1 (SLMCPR = 1.07) is nonconservative. The licensee has requested that the SLMCPR calculated to bound BFN Unit 2 Cycle 9 operation (SLMCPR = 1.10) be used for all three BFN reactors pending long-term resolution of the issue. The licensee provided supplemental information on February 7, 1997, which did not affect the staff's proposed finding of no significant hazards consideration.

The licensee also provided revised TS Bases to resolve a discrepancy between the Bases and the Final Safety Analysis Report description of the supplemental spent fuel pool cooling mode of the residual heat removal system.

2.0 DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATIONS CHANGES

The changes consist of a revision to Safety Limit 1.1.A.1, as follows:

> When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.10 shall constitute violation of the fuel cladding integrity safety limit.

Changes to the TS Bases, which refer to this safety limit, delete references to a specific MCPR value.

Enclosure 4

An unrelated change to the TS Bases regarding the capability of the residual heat removal (RHR) system to provide supplemental spent fuel pool cooling changes the word "will" to "can."

3.0 EVALUATION

The change to the SLMCPR restores margin lost when it was determined a generic SLMCPR value was nonconservative for some fuel and core designs. The BFN reactors are designed such that for transients caused by a single operator error or equipment malfunction are limited so that, considering uncertainties in monitoring core operations, more than 99.9% of the fuel rods are expected to avoid boiling transition.

A cycle-specific calculation has been performed for the current BFN Unit 2 Cycle 9 which resulted in an SLMCPR of 1.09. A similar calculation for the current BFN Unit 3 Cycle 8 yields an SLMCPR of 1.10. BFN Unit 1 is defueled, and is not expected to operate for at least several years, so analytical results have not been documented for that unit. The licensee proposes an SLMCPR of 1.10 for all three units.

The SLMCPR in TS 1.1.A.1 is proposed to change from 1.07 to 1.10 when the reactor pressure is greater than 800 psia and its associated Bases 1.1, 2.1.A.1, 2.1.A.3, 2.1.C, and 3.3/4.3.C are proposed to change from numerical number of 1.07 to the wording of the SLMCPR based on the cycle-specific analysis performed by General Electric (GE) for BFN Unit 2 Cycle 9 mixed core of GE11/GE9 fuel, which is also applicable to BFN Unit 1 and Unit 3 Cycle 8. The cycle-specific parameters were used including the actual core loading, the most limiting permissible control blade patterns, actual exposure-dependent rod power for R-factor distributions, and calculation made for several points in the cycle.

The staff has reviewed the proposed TS and its associated Bases changes which are based on the analyses performed using BFN Unit 2 Cycle 9 cycle-specific inputs and approved methodologies including GESTAR II (NEDE-24011-P-A-11, Sections 1.1.5 and 1.2.5) and NEDO-10985-A, January 1977, and found them acceptable. Because the R-factor methodology referenced in NEDE-24011-P-A-11 is not applicable to the part-length GE11 fuel, a revised R-factor methodology described in NEDC-32505P, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," November 1995 was used. The revised R-factor calculation method uses the same NRC-approved equation stated in GESTAR (NEDE-24011-P-A) with the correction factors to account for the peaking factor effects due to the part-length-rod design. The staff has reviewed the R-factor calculation method for the GE11, the relevant information provided in the proposed Amendment 25 to GESTAR II, NEDE-24011 (which is under staff review) and the supplemental information dated February 7, 1997, on the Browns Ferry Unit 2 Cycle 9 (BFN2C9) and Unit 3 Cycle 8 SLMCPR calculation. The staff has found that the methodologies discussed above apply to the BFN design, and the justification for analyzing and determining the SLMCPR of 1.10 for all three Browns Ferry units based on the result of the analysis for the BFN Unit 3 Cycle 8 (BFN3C8) is acceptable, since (1) all three units are not an equilibrium core; (2) the fresh GE11 bundles for BFN3C8 have the flattest R-factor distribution compared with that in the BFN2C9; and (3) BFN3C8 is loaded with a higher batch fraction.

Based on its review, the staff concludes that the changes to the TS and its associated Bases for the SLMCPR are acceptable for BFN Units 2 and 3, since the changes are analyzed based on the NRC-approved method and a conservative cycle-specific SLMCPR is used for these units. The new values will ensure that greater than 99 percent of the fuel rods will avoid transition boiling.

Staff approval of similar changes for BFN Unit 1 is dependent on the licensee providing appropriate documentation of similar calculations for that unit. Therefore, an amendment to implement the revised SLMCPR for Unit 1 is not approved at this time.

In addition, a correction of the discrepancy in the description of the RHR supplemental fuel pool cooling mode in Bases 3.10.C by changing the wording from "will" to "can" is proposed to denote that the RHR system is a means of providing additional fuel pool decay heat removal. This revision is an appropriate clarification of the Bases for all three units.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official (Kirk Whatley) was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 42285). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based upon the considerations discussed above, that: (1) the amendments do not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) create the possibility of a new or different kind of accident from any previously evaluated, or (c) significantly reduce a margin of safety, and therefore, the amendments do not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (3) such activities will be conducted in compliance with the Commission's regulations; and (4) issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. Huang

Dated: May 7, 1997