50-259

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

November 2, 1995

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: SAFETY EVALUATION OF POST-FIRE SAFE SHUTDOWN CAPABILITY AND ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M85254, M87900, M87901, AND M87902) (TS 337)

Dear Mr. Kingsley:

As discussed in the enclosed Safety Evaluation (Enclosure 4), the NRC staff has completed its review of the 10 CFR 50 Appendix R post-fire safe shutdown program for combined power operations of the Browns Ferry Nuclear Plant (BFN) Units 2 and 3. This evaluation is based predominately upon a submittal dated December 20, 1994 by the Tennessee Valley Authority (TVA). This document superseded an earlier submittal dated December 15, 1992. The staff also considered information provided by TVA on October 5, 1995. The staff has concluded that the program described by these submittals meets NRC fire protection requirements and guidance, and is, therefore, acceptable. NRC verification of proper implementation of this program will include review of inspector follow-up items discussed in Enclosure 4, Section 4.

In addition, the Commission has issued the enclosed Amendment Nos. 226, 241, and 200 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for BFN Units 1, 2, and 3, respectively. These amendments are in response to your application dated September 30, 1993, requesting revision of the operating licenses to reflect issuance of the staff safety evaluation discussed above as of the date of this letter. Enclosure 5 is the safety evaluation for these license amendments.

A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice. Please contact me at (301)415-1470 if you have questions regarding these topics.

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. 226 to License No. DPR-33
2. Amendment No. 241 to License No. DPR-52
3. Amendment No. 200 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 226
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 30, 1993, 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, Facility Operating License No. DPR-33, page 5, paragraph 2.C.(13), is hereby amended to reflect issuance of the staff safety evaluation of the Appendix R Safe Shutdown Program; to read as follows:

   (13) Browns Ferry Nuclear Plant shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for BFN as approved in the SEs dated December 8, 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989 subject to the following provision:

   The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Page 5 of License DPR-33*

Date of Issuance: November 2, 1995

*Page 5 is attached for convenience, for the composite license to reflect this change.

(11) The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Browns Ferry Physical Security Plan", with revisions submitted through May 24, 1988; "Browns Ferry Security Personnel Training and Qualification Plan", with revisions submitted through April 16, 1987; and "Browns Ferry Safeguards Contingency Plan", with revisions submitted through June 27, 1986. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

(12) The licensee is authorized to temporarily store low-level radioactive waste in an existing covered pavilion that is situated outside the security fence, as presently located, but inside the site exclusion area. The total amount of low-level waste to be stored shall not exceed 1320 curies of total activity. This authorization expires two years from the effective date of this amendment and is subject to all the conditions and restrictions in TVA's application dated January 21, 1980.

(13) Browns Ferry Nuclear Plant shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for BFN as approved in the SEs dated December 8. 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989 subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

D. This amended license is effective as of the date of issuance and shall expire midnight on December 20, 2013.

FOR THE ATOMIC ENERGY COMMISSION

S/ A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Date of Issuance: DEC 20 1973

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. 241
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 September 30, 1993, 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. 241, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Also, page 6, paragraph 2.C.(14) of the license is amended to reflect issuance of the staff safety evaluation of the Appendix R Safe Shutdown Program; to read as follows:

   (14) Browns Ferry Nuclear Plant shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for BFN as approved in the SEs dated December 8, 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989 subject to the following provision:

   The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

4. 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

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: 1. Page 6 of License DPR-52*
2. Changes to the Technical Specifications

Date of Issuance: November 2, 1995

*Page 6 is attached for convenience, for the composite license to reflect this change.

(12) The licensee is authorized to temporarily store low-level radioactive waste in an existing covered pavilion that is situated outside the security fence, as presently located, but inside the site exclusion area. The total amount of low-level waste to be stored shall not exceed 1320 curies of total activity. This authorization expires two years from the effective date of this amendment and is subject to all the conditions and restrictions in TVA's application dated January 21, 1980.

(13) Commission Order dated March 25, 1983 is modified as follows: in Attachment 1, for item II.F.1.1 and II.F.1.2 change "12/31/84" to "Prior to startup in Cycle 6."

(14) Browns Ferry Nuclear Plant shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for BFN as approved in the SEs dated December 8, 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989 subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

D. This amended license is effective as of the date of issuance and shall expire midnight on June 28, 2014.

FOR THE ATOMIC ENERGY COMMISSION

S/ A. Giambusso
A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendices A & B - Technical
Specifications

Date of Issuance: JUN 28, 1974

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 241

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the page identified below and inserting the enclosed page. The revised page is identified by the captioned amendment number and contains a marginal line indicating the area of change. *Overleaf page is provided to maintain document completeness.

REMOVE	INSERT
1.0-12a	1.0-12a
1.0-12.b	1.0-12.b*

DEFINITIONS (Cont'd)

NN. Appendix R Safe Shutdown Program

BFN has developed an Appendix R Safe Shutdown Program. This Program is to ensure that the equipment required by the Appendix R Safe Shutdown Analysis is maintained and demonstrated functional as follows:

1. The functional requirements of the Safe Shutdown systems and equipment, as well as appropriate compensatory measures should these systems/components be unable to perform their intended function are outlined in Section III of the Program.

2. Testing and monitoring of the Appendix R Safe Shutdown systems and equipment are defined in Section V of the Program.

OO. CORE OPERATING LIMITS REPORT (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9.1.7. Plant operation within these limits is addressed in individual specifications.

PP. Limiting Control Rod Pattern - A limiting control rod pattern shall be a pattern which results in the core being on a thermal limit, i.e. operating on a limiting value for APLHGR, LHGR, or MCPR.

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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. 200
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 September 30, 1994, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 200, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Also, page 4, paragraph 2.C.(7) of the license is amended to reflect issuance of the staff safety evaluation of the Appendix R Safe Shutdown Program; to read as follows:

   (7) Browns Ferry Nuclear Plant shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for BFN as approved in the SEs dated December 8, 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989 subject to the following provision:

   The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

4. 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

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: 1. Page 4 of License DPR-68*
2. Changes to the Technical Specifications

Date of Issuance: November 2, 1995

*Page 4 is attached for convenience, for the composite license to reflect this change.

(4) The licensee shall maintain in effect and fully implement all provisions of the Commission approved physical security plan including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plan, which contains information protected under 10 CFR 73.21, is entitled "Browns Ferry Nuclear Plant Physical Security Plan," dated May 15, 1982 (TVA letter dated June 11, 1982) and revisions submitted by TVA letters dated August 31, 1982 and October 19, 1982.

(5) The licensee shall follow all provisions of the NRC approved Guard Training & Qualification Plan, including amendments and changes made pursuant to 10 CFR 50.54(p). The approved Guard Training & Qualification Plan is identified as "Browns Ferry Nuclear Power Station Guard Training & Qualification Plan," dated August 17, 1979, as revised by pages dated January 24, 1980, May 21, 1980, October 1, 1980, and March 9, 1981 and as may subsequently be revised in accordance with 10 CFR 50.54(p). The Guard Training & Qualification Plan shall be followed, in accordance with 10 CFR 73.55(b), 60 days after the date of this amendment.

(6) The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Browns Ferry Physical Security Plan", with revisions submitted through May 24, 1988; "Browns Ferry Security Personnel Training and Qualification Plan", with revisions submitted through April 16, 1987; and "Browns Ferry Safeguards Contingency Plan", with revisions submitted through June 27, 1986. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

(7) Browns Ferry Nuclear Plant shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for BFN as approved in the SEs dated December 8, 1988, March 6, 1991, March 31, 1993, November 2, 1995 and Supplement dated November 3, 1989 subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 200

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the page identified below and inserting the enclosed page. The revised page is identified by the captioned amendment number and contains marginal lines indicating the areas of change. *Overleaf page is provided to maintain document completeness.

REMOVE	INSERT
1.0-12a	1.0-12a
1.0-12b	1.0-12b*

1.0 DEFINITIONS (Cont'd)

NN. Appendix R Safe Shutdown Program - BFN has developed an Appendix R Safe Shutdown Program. This program is to ensure that the equipment required by Appendix R Safe Shutdown Analysis is maintained and demonstrated functional as follows:

1. The functional requirements of the Safe Shutdown Systems and equipment, as well as appropriate compensatory measures should these systems/components be unable to perform their intended function are outlined in Section III of the program.

2. Testing and monitoring of the Appendix R Safe Shutdown systems and equipment are defined in Section V of the Program.

OO. CORE OPERATING LIMITS REPORT (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9.1.7. Plant operation within these limits is addressed in individual specifications.

PP. LIMITING CONTROL ROD PATTERN - A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal limit, i.e. operating on a limiting value for APLHGR, LHGR, or MCPR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

POST-FIRE SAFE SHUTDOWN CAPABILITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) staff approved the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, fire protection program in safety evaluations (SE) dated December 8, 1988, November 3, 1989, and March 6, 1991. On January 15, 1992, the Tennessee Valley Authority (the licensee) submitted an updated Fire Protection Report (FPR) for BFN Unit 2. The revised BFN-FPR, which contained the fire protection plan, fire hazards analysis, safe shutdown analysis (SSA), and Appendix R safe shutdown program for BFN Unit 2, supported the operation of BFN Unit 2, with BFN Unit 1 and Unit 3 shutdown. In an SE issued March 31, 1993, the NRC staff approved the BFN Unit 2 FPR.

On April 1, 1993, the staff issued Amendments Nos. 192, 207, and 164 to Facility Operating Licenses DPR-33, DPR-52, and DPR-68 for BFN Units 1, 2, and 3, respectively. These amendments, in part, incorporated a fire protection license condition. The license condition requires the licensee to maintain in effect all provisions of the approved fire protection program with the provision that the licensee may make changes to the program without prior Commission approval if the changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. In the letter that transmitted the amendments, the staff stated that the BFN Appendix R safe shutdown program does not accommodate operation of BFN Units 1 and 3.

By letter dated March 19, 1993, the licensee committed to submit an application to amend the BFN licenses prior to restart of Unit 3 to revise the fire protection license condition for Units 1, 2, and 3 reflecting issuance of a new SE approving the BFN-FPR Appendix R safe shutdown program as incorporated into the BFN-FPR for combined Units 2 and 3 operation. The licensee provided this submittal on September 30, 1992, to provide the administrative mechanism to update the licenses upon issuance of this safety evaluation. The safety evaluation supporting the revision of the license conditions is dated November 2, 1995.

By letter dated December 15, 1992, the licensee submitted its combined Unit 2 and Unit 3 BFN-FPR for NRC staff review. By letter dated December 20, 1994, the licensee submitted the revised combined Unit 2 and Unit 3 BFN-FPR and a description of the changes that it made in the Unit 2 and Unit 3 BFN-FPR following the initial submittal of December 15, 1992. The revised BFN-FPR supports Unit 2 and Unit 3 operation with Unit 1 shutdown and defueled. The

Enclosure 4

revised BFN-FPR contains the fire protection plan, fire hazards analysis, SSA, and Appendix R safe shutdown program. By letter dated October 5, 1995, the licensee provided additional information. This SE documents the staff review and evaluation of the revised combined Unit 2 and Unit 3 BFN-FPR and the additional information submitted. This SE includes input that was prepared by Brookhaven National Laboratories, the staff's technical assistance contractor.

In a letter of March 22, 1995, the licensee submitted the measures it is taking to ensure that Thermo-Lag fire barriers installed at BFN meet NRC regulations.

The NRC staff conducted an inspection of the BFN safe shutdown capability during July 17-21, 1995. The results of the inspection are documented in NRC Inspection Report Nos. 50-259/95-37, 50-260/95-37, and 50-296/95-37, dated August 30, 1995.

2. FIRE PROTECTION PLAN AND FIRE HAZARD ANALYSIS

In the SE issued March 31, 1993, the staff approved the BFN Unit 2, fire protection plan and fire hazards analysis in support of the Unit 2 restart. The staff reviewed the fire protection plan and fire hazard analysis included with the revised combined Unit 2 and Unit 3 BFN-FPR and found that the licensee revised the BFN-FPR to reflect the plant configuration that will be required for combined Unit 2 and Unit 3 operation. The revised fire protection plan and fire hazards analysis for BFN Unit 2 are substantially the same as those approved by the staff in the SE of March 31, 1993. However, the licensee has changed this plan, as permitted by the license conditions issued on April 1, 1993. Except as documented below, the staff did not review these changes to the BFN-FPR as part of this SE. However, they will be subject to future NRC inspections. This review has focused on those aspects of the FPR which are necessary to support BFN Unit 3 operation.

3. POST-FIRE SAFE SHUTDOWN CAPABILITY

3.1 Separation of Safe Shutdown Functions

With the exception of areas where the staff previously approved Appendix R exemptions and fire area 16 (see Section 3.3), where redundant trains of systems necessary to achieve and maintain hot standby conditions are located within the same fire area outside the containment, the licensee has committed to use one of the following means to ensure that one train of equipment remains free of fire damage:

1. Separate equipment, components, cables and associated circuits of redundant safe shutdown systems by a fire barrier having a 3-hour fire rating.

2. Separate equipment, components, cables and associated circuits of redundant safe shutdown systems by a horizontal distance of more than 20 feet free of intervening combustibles or fire hazards and install automatic fire detection and suppression systems in the areas.

3. Separate equipment, components, cables and associated circuits of redundant safe shutdown systems by a fire barrier having a 1-hour fire rating and install automatic fire detection and suppression systems in the area.

The licensee's proposed criteria for providing fire protection for redundant trains of systems necessary to achieve and maintain hot standby conditions meets the technical requirements of Section III.G of Appendix R to 10 CFR Part 50 and are, therefore, acceptable.

In Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers," dated December 17, 1992, the staff informed the licensee of areas of concern regarding the use of Thermo-Lag 330-1 (Thermo-Lag) fire barriers to meet NRC regulations. A Thermo-Lag fire barrier is installed in the BFN intake pumping structure to provide a 1-hour barrier for one division of residual heat removal service water (RHRSW) cables. In a letter of March 22, 1995, the licensee submitted the measures it is taking to ensure that this barrier meets NRC regulations. The licensee will replace this barrier with a Thermo-Lag fire barrier (barrier material and installation configuration) that it has qualified by its Watts Bar Nuclear Power Plant qualification test program prior to performing reactor pressure vessel hydrostatic testing for Unit 3 restart (approximately one month prior to initial criticality). The licensee's schedule for replacing the Thermo-Lag fire barrier currently installed in the intake pumping structure is acceptable. The staff will review the licensee's corrective measures during a future inspection.

The licensee has abandoned other Thermo-Lag fire barriers that are not needed to meet NRC regulations. The licensee plans to remove the abandoned barriers that are accessible and for which it will be cost effective. For those barriers that are not accessible, or for which it will not be cost effective to remove, the licensee will address ampacity derating, seismic effects, and combustibility. In its letter of March 22, 1995, the licensee stated that it will complete its analyses by December 22, 1995, and will remove the barriers by June 20, 1996. The licensee's schedule is acceptable. The staff will review the licensee's analyses during a future inspection.

Closure of staff review of Thermo-Lag issues for the Browns Ferry site is dependent on review of results of ampacity derating testing performed for the Watts Bar facility.

3.2 Post-Fire Safe Shutdown Methodology - General Plant Areas

3.2.1 Safe Shutdown Analysis Methodology

The licensee's methodology for assessing compliance with the separation and fire protection requirements of Section III.G of Appendix R consisted of:

1. Determining the functions required to achieve and maintain safe shutdown.

2. Grouping specific plant locations into fire areas.

3. Identifying for each fire area one or more paths that will satisfy each required shutdown function.

4. Developing functional criteria that defined the required equipment for the shutdown paths.

5. Identifying power and control cables for shutdown-related equipment and associated circuits that are not isolated from shutdown cabling.

6. Relocating cables and equipment, providing fire barriers, and providing fire detection and fire suppression systems to meet the separation and fire protection requirements of Section III.G of Appendix R or providing justification where deviations from these requirements occur.

On the basis of this methodology, and subject to the previously approved exemptions from the requirements of Section III.G, the licensee's general methodology conforms to the requirements of Appendix R to 10 CFR Part 50 for protection of safe shutdown capability and is, therefore, acceptable.

3.2.2 Post-Fire Safe Shutdown Capability

The post-fire safe shutdown capability for the combined operation of Unit 2 and Unit 3 has not changed significantly since the staff issued its SEs of December 8, 1988, November 3, 1989 and March 6, 1991. The licensee's SSA demonstrated that sufficient redundancy exists for systems needed for safe shutdown. The SSA included components, cabling, and support equipment needed to achieve and maintain safe shutdown in the event of a fire anywhere in the plant. At least one train of systems would be available to achieve and maintain safe shutdown conditions including the components, cabling, and support equipment necessary to achieve cold shutdown. Support equipment includes associated electrical distribution and ventilation systems.

Where available, high pressure core injection (HPCI) will be used to maintain vessel inventory for the first 2 hours of the event, followed by depressurization of the vessel by use of the main steam relief valves (MSRVs), and use of the residual heat removal (RHR) system for establishing cold shutdown via the alternative shutdown cooling mode. Changes to the safe shutdown and alternative shutdown capabilities that support the combined operation of Unit 2 and Unit 3 are:

1. Using the reactor core isolation cooling (RCIC) system as an additional high pressure system for the first 2 hours of the event for Unit 3.

2. Assuring the availability of four MSRVs for Unit 3 (three required).

3. Deleting drywell temperature instrumentation for Unit 3.

4. Using reactor water cleanup (RWCU) automatic isolation on high temperature for Unit 3, (high/low pressure interface).

The major components and systems which are required to shutdown Units 2 and 3 include:

1. Three to eight diesel generators (DGs), 250V DC batteries, and the electrical boards for the necessary equipment. The number of DGs required for safe shutdown varies from fire area to fire area from a minimum of three (shutdown of Unit 2 only) to a maximum of eight (shutdown of both Unit 2 and Unit 3).

2. One RHR pump per unit.

3. Two emergency equipment cooling water (EECW) pumps.

4. One RHRSW pump per unit.

5. Three MSRVs per unit.

6. Control bay Units 2 and 3 shutdown board room heating ventilation and air conditioning (HVAC) and RHR pump room coolers

7. Instrumentation.

8. HPCI and RCIC for certain plant areas.

The above listed components and systems agree with the description provided in the SE of December 8, 1988, and are, therefore, acceptable. However, the post-fire safe shutdown capability requires the main steam isolation valves (MSIVs) be closed and remain closed. As such, the MSIVs form an integral part of the shutdown capability and should, therefore, be included in the list of minimum required safe shutdown equipment. By letter dated October 5, 1995, the licensee stated that it will modify the safe shutdown analysis portion of the Units 2 and 3 BFN-FPR by revising Paragraph 5.4.3, "High-Low Pressure Interface," and adding the MSIVs to Table 5-2, "High-Low Pressure Interface Components." This is acceptable.

The licensee has performed an electrical separation study to ensure that at least one train of equipment is available in the event of a fire in areas that might adversely affect the aforementioned components. The licensee identified and traced safe shutdown equipment and cabling through each fire area from the component to the power source. In addition, associated circuits whose fire-induced spurious operation could adversely affect post-fire safe shutdown were identified by a system review to determine those components whose maloperation could affect the safe shutdown capability.

The changes for Unit 3 operation are consistent with the SSA the NRC staff approved in its SEs of December 8, 1988, November 3, 1989 and March 6, 1991, and are, therefore, acceptable.

During the July 17-21, 1995, inspection, the staff reviewed the BFN-FPR, in which, the licensee stated that the maximum drywell temperature of 281°F may be temporarily exceeded. However, the licensee informed the inspectors that on the basis of revised calculations, it determined that the maximum drywell temperature will not exceed 244°F. This provides sufficient margin below the design temperature limit of 281°F. As a result of this conclusion, the licensee deleted drywell temperature monitoring as a Unit 3 Appendix R required component. During the inspection the NRC staff reviewed the revised analysis for maximum drywall temperature and found it acceptable. However, the required revision of the SSA should be formalized. This revision will be verified by inspection prior to Unit 3 restart.

3.3 Areas Where Alternative Safe Shutdown Capability is Used

The only area outside containment where redundant safe shutdown divisions are not separated and protected in accordance with Section III.G.2 of Appendix R is fire area 16 (control building elevations 593-feet, 606-feet, and 617-feet). Fire area 16 consists of the following rooms:

1. Main control rooms
2. Cable spreading rooms
3. Mechanical equipment rooms
4. Auxiliary instrument rooms
5. Process computer room
6. Communications room
7. Motor generator (MG) set rooms

The licensee has provided alternate shutdown capability for fires in fire area 16 in accordance with Sections III.G.3 and III.L of Appendix R.

3.3.1 Alternative Shutdown Capability

The alternative shutdown system was described in the SEs of December 8, 1988 and November 3, 1989. In the latter document, the staff noted that in the event of a fire in fire area 16, if the safe shutdown instructions (SSIs) are executed, fire area 16 will be abandoned and remote shutdown panel 25-32 and other locations will be manned to achieve post-fire safe shutdown.

A fire in fire area 16 could adversely affect Unit 2 and Unit 3 HPCI operation. A rapid depressurization of the Unit 2 reactor pressure vessel would be required within 20 minutes of the entry conditions being met. Unit 3 RCIC operation would not be affected and Unit 3 would not require rapid depressurization of the reactor pressure vessel for two hours.

The alternative safe shutdown systems available for fire area 16 are:

1. Diesel generators A, B, C, D, and 3A; 250V DC batteries; and the electrical boards for the necessary equipment.
2. One RHR pump per unit.
3. Two emergency equipment cooling water (EECW) pumps, i.e., EECW Loop B South Header Pumps B3 and D3.
4. One RHRSW pump per unit, i.e., RHRSW Pumps C2 and A1.

5. Four MSRVs per unit, i.e., 2(3)-PCV-1-5, 2(3)-PCV-1-22, 2(3)-PCV-1-30, and 2(3)-PCV-1-34.

6. Unit 2 shutdown board room air conditioning unit 2A, Unit 3 shutdown board room air conditioning unit 3A, and RHR pump room coolers.

7. Unit 3 RCIC system.

Instrumentation available for a fire in fire area 16 consists of the reactor pressure vessel level and pressure and the torus suppression pool level and temperature. For Unit 2, neither drywell pressure nor temperature is available. For Unit 3, for which the RCIC system will be relied upon, drywell pressure indication will be available.

The systems identified by the licensee for achieving and maintaining alternative safe shutdown in the event of a fire meet the requirements of Section III.L of Appendix R and are, therefore, acceptable.

3.4 Alternative Shutdown Performance Goals

The alternative shutdown system was designed to allow the alternative shutdown performance goals specified in Section III.L of Appendix R to be achieved.

3.4.1 Reactivity Control

Reactivity control was previously addressed in the SEs of December 8, 1988 and November 3. 1989. The licensee did not change its method of providing reactivity control function for combined Unit 2 and Unit 3 operation, or for Unit 3 operation alone. Therefore, the licensee's reactivity control is acceptable.

3.4.2 Reactor Coolant Inventory Control

Reactor coolant inventory control (makeup) was previously addressed in the SE of December 8, 1988. The licensee did not change its method of reactor coolant inventory control for combined Units 2 and 3 operation as opposed to Unit 2 operation only. Therefore, the licensee's reactor coolant inventory control is acceptable.

3.4.3 Reactor Coolant Pressure Control

As documented in the SE of December 8, 1988, reactor coolant pressure control is provided by the MSRVs. Overpressurization protection prior to depressurization of the reactor pressure vessel is provided by the self-actuating pressure mode of the MSRVs. The licensee's method of providing the reactor coolant pressure control function has not changed from that approved by the staff in its SE of December 8, 1988, and is, therefore, acceptable.

3.4.4 Decay Heat Removal

In its SE of December 8, 1988, the staff noted that during hot shutdown, decay heat will be removed by the self-actuating mode of MSRV operation.

Specifically, during high pressure isolation operation, decay heat is removed from the reactor through the MSRVs using the torus as a heat sink.

The HPCI and RCIC systems are only credited for maintenance of coolant inventory. The decay heat removed from the vessel via the operation of the HPCI and RCIC systems is transferred to the torus as steam through the HPCI and RCIC pump turbine exhaust lines, but is less than the total decay heat generated in the reactor core. The remainder of the decay heat is transferred to the torus through the MSRVs.

The HPCI and RCIC systems are not able to transfer the decay heat from the torus. The RHR and RHRSW systems transfer the decay heat from the torus to the ultimate heat sink. Since HPCI and RCIC cannot complete the entire function of transferring the decay heat load to the ultimate heat sink, they are not considered as decay heat removal systems.

During cold shutdown, decay heat removal is achieved by utilizing the "alternate shutdown cooling mode" (i.e., RHR system used in conjunction with MSRVs and RHRSW system).

The licensee's methods for decay heat removal meet the requirements of Section III.L of Appendix R, and are, therefore, acceptable.

3.4.5 Process Monitoring

As documented in the SE of December 8, 1988, in addition to the normal control room instrumentation, local instrumentation is provided at backup control panels for direct indications of process variables. Instrumentation available to support the minimum safe shutdown systems include reactor vessel pressure, reactor water level, suppression pool level, suppression pool temperature, drywell temperature and pressure indication. In addition, breaker and valve position indications are provided at the shutdown boards.

The drywell pressure and drywell temperature indication is included in the safe shutdown equipment so that the operators can monitor the drywell conditions when either the HPCI system or the RCIC system is used to maintain reactor inventory. Monitoring of these parameters will allow the operators to determine if reactor depressurization is needed in order to prevent the drywell pressure and temperature limits from being exceeded. Since drywell pressure and drywell temperature limits will not be challenged following vessel depressurization and subsequent low pressure coolant injection (LPCI), there is no need to monitor these parameters when HPCI and RCIC are not credited for operation.

Suction for the HPCI or RCIC system can be taken from either the condensate storage tank or the suppression pool. The condensate storage tank is the preferred water supply for the HPCI system and RCIC system. The redundant supply of water for the condensate storage tank is from the torus. For the HPCI system, level switches 3-LS-073-0056A and 3-LS-073-0056B monitor the water level in the condensate storage tank header and automatically transfer the HPCI system suction supply to the torus upon sensing a low level in the condensate storage tank header. Level switches 3-LS-073-0056A and 3-LS-073-0056B, and HPCI torus supply valves 3-FCV-073-0026 and

3-FCV-073-0027, are available for fires in areas where the HPCI system is required.

For the RCIC system, transfer of the suction supply from the condensate storage tank to the torus requires manual transfer. However, with the plant in a normal configuration, there will be a minimum of 135,000 gallons of water in the condensate storage tank header. With the RCIC system having a capacity of 600 gpm, the 135,000 gallons of water will be sufficient for about 4 hours of RCIC system operation. For safe shutdown of Unit 3, RCIC operation is only credited for 2 hours of operation. In addition, torus suction valves 3-FCV-071-0017 and 3-FCV-071-0018 remain available for those fire areas where the RCIC system is credited for operation. The manual actions to transfer the RCIC system from the condensate storage tank to the torus can be performed from outside the control room. Procedures documenting these actions will be verified by inspection prior to Unit 3 restart.

The licensee's process monitoring meets the requirements of Section III.L of Appendix R, and is, therefore, acceptable.

3.4.6 Support Functions

The adequacy of support systems and equipment was previously evaluated by the staff in its SE of December 8, 1988. Later, in its SE of November 3, 1989, the staff indicated that the licensee had reevaluated the RHR pump seal temperature rating and, on the basis of this evaluation, upgraded the rating from 160°F to 215°F. The maximum suppression pool temperature was determined to be less than 200°F. Therefore, the RHR pump seal coolers are not needed to support post-fire safe shutdown operations.

During the NRC inspection of July 17-21, 1995, the licensee provided technical justification for not requiring the RHR pump seal coolers. The documentation provided by the licensee indicated that the pump seal materials can tolerate a temperature of 400°F continuously. If the seals cavitate, the seal leakage that would occur is only 25 gpm as opposed to the pump capacity of 10,800 gpm. The inspectors reviewed the licensee's documentation and found that it adequately addressed the issue of not requiring the RHR pump seal coolers.

The licensee's support functions satisfy the requirements of Section III.L of Appendix R and are, therefore, acceptable.

3.5 Manual Operator Actions, Safe Shutdown Procedures, and Manpower

In the SE of December 8, 1988, the staff stated that for a fire in any location except fire area 16, the licensee will have to perform manual operations inside the control room and outside the control room at a number of local shutdown stations. The local stations include the 4 kV bus tie board, 4 kV and 480V shutdown boards, 480V and DG auxiliary boards, 250V DC battery, distribution and reactor motor operated valve (RMOV) boards, DG 125V DC distribution boards, backup control panels 25-32, Unit 3 DG building panels (25-270 A&B), yard panels (25-246 A&B) and intake pumping station (fire area 25). In the event of a fire in fire area 16, all operator actions are performed outside the main control room, except manual scram and reactor isolation.

For combined Unit 2 and Unit 3 operation, in the event of a fire in either unit that meets the entry conditions of the SSIs, the licensee will shut down both units. In the event that either unit had already been in hot or cold shutdown prior to the fire, there would be no difference in the principal operator actions required to achieve safe shutdown of the operating unit.

Six operators are the maximum required to perform the manual actions outside the control room to shutdown both units. The licensee stated that a minimum of seven operators will be available at all times. The manual actions required prior to abandoning the control room are the same for Unit 3 and for Unit 2. Specifically, the operators are required to scram the reactor and isolate the reactor. In addition, the licensee verified that reactor scram can be performed outside the control room by opening the supply breakers to reactor protection system (RPS) MG sets 3A and 3B. These supply breakers are located on the 480V RMOV board 3A (fire area 13) and the 480V RMOV board 3B (fire area 12), respectively.

For each fire area or zone, the manual actions to be taken, the time by which the actions must be performed, and the location they are to be performed, whether in the plant or in the control room, have been identified by the licensee in the manual operator calculation, ND-Q0999-920116. None of the manual operator actions in the plant or in the control room constitutes a repair as defined by the NRC staff.

The manual actions to be performed for combined Unit 2 and Unit 3 operation involve the same type of actions currently performed on Unit 2. These actions include positioning valves, operating switches, and opening breakers. The time period for performing these actions is consistent with those established for Unit 2. During the July 17-21, 1995, inspection, the licensee stated that the operations personnel had completed the walkdown in the 34 fire areas/zones for both units, which included the verification of the ability to perform actions both outside and inside the control rooms. The comments generated by the walkdowns have been incorporated into the Units 2 and 3 SSIs.

With respect to the SSI for fire area 16, the licensee performed a timed walkdown of the required actions. The actions were evaluated for feasibility and included the adequacy of emergency lighting, labeling, accessibility, logical grouping and sequencing for the operators, and time restraints. The licensee considered modifications that were not field complete in the evaluation. The licensee concluded that the SSI was satisfactorily completed.

There are no manual actions required to be performed in any fire area which will require entry into the fire area prior to the extinguishment of the fire. No entry is made, either to perform a manual action or for ingress/egress purposes, into a fire area where the fire originated prior to the fire being extinguished by the fire brigade. There are a number of fire areas that the operators must reenter to perform manual actions after the fire is extinguished. In no case must reentry into a fire-affected area be made prior to 1 hour after detection of the fire. Specifically, in the event of fire in fire area 16, ingress and egress may be required 1 hour after the fire has been detected to open the doors of battery board rooms 1, 2, and 3. Similarly, for fire area 23 (Unit 3 4kV shutdown board rooms 3EC and 3ED), ingress and egress may be required 1 hour after the fire has been detected for access to 4kV shutdown board room 3EA. For all other areas, operator entry will not be necessary for at least 2 hours for ingress and egress purposes only.

Prior to Unit 3 restart, the existing Unit 2 SSIs will be replaced with the combined Unit 2 and Unit 3 SSIs. This one set of instructions will provide the required actions for safe shutdown whether the site is operating one or both units.

3.5.1 Operator Actions

In the case of either one or both units in operation, the principal manual operator actions, are:

1. Immediately initiate manual scram and reactor vessel isolation in the main control room.

2. Immediately initiate and verify operation of the HPCI and RCIC systems in the main control room. These actions are required for fires in fire areas where HPCI and RCIC systems are available. Actions are taken outside of the fire-affected area.

3. Assure reactor scram by opening the supply breakers to the RPS MG Sets and assure closure of the MSIVs by transferring control of the MSIVs from the main control room (10 minutes). These actions are required for fires in fire area 16 only. Actions are taken outside of fire area 16.

4. Transfer the diesel generators and 4kV shutdown boards associated with the required EECW pumps, and the required EECW pumps themselves to local control (10 minutes). Actions are required for fires in fire area 16 only. Actions are taken outside of fire area 16.

5. Close the HPCI steam supply shutoff valve to prevent the intrusion of water into the main steam lines (10 minutes). Action is required for fires in fire areas where HPCI is not available. Action is taken outside of fire-affected area.

6. Assure closure of the MSRVs by transferring control of the MSRVs from the main control room (10 minutes). Actions are required for fires in fire areas 16 and 19 only. Actions are taken outside of fire areas 16 and 19.

7. Initiate and assure the operation of the RCIC system by transferring to local control (20 minutes). Action is required for fires in fire area 16 only. Action is taken outside of fire area 16.

8. Initiate a manual reactor vessel depressurization by opening three MSRVs. Cooldown is controlled by the operators at the rate of 100°F/hour when either HPCI or RCIC is available. Depressurization must occur within 20 minutes for fires in fire areas where neither HPCI nor RCIC is available. Action is taken outside of the fire-affected area.

9. Initiate LPCI injection for reactor coolant inventory control. Depressurization must occur within 20 minutes for fires in fire areas where neither HPCI nor RCIC is available. Action is taken outside of the fire-affected area.

10. Transfer the required diesel generators and associated 4kV shutdown boards to local control.

11. Transfer the required 480V shutdown boards, the 480V RMOV boards, the 250V RMOV boards, and the diesel generator auxiliary boards to local control and/or their alternate power feeds.

12. Initiate and assure isolation of the RWCU system (30 minutes).

13. Realign the 250V DC battery chargers to assure a long term DC power supply (60 minutes).

14. Initiate a manual reactor vessel depressurization by opening three MSRVs.

15. Initiate LPCI injection for reactor coolant inventory control.

16. Isolate potential nitrogen flow divergence paths to the drywell and torus, and connect the containment atmospheric dilution (CAD) system to the dry control air (DCA) system to provide a long term MSRV air (nitrogen) supply.

17. Initiate the RHRSW system, including the manual alignment of the RHRSW discharge valves, to remove decay heat from the torus.

18. Initiate cooling of the control building via their associated air-handling units and water chillers.

19. Initiate cooling of the Unit 2 and Unit 3 shutdown board rooms via their associated air-conditioning units.

20. Establish portable ventilation in the battery board rooms to maintain acceptable room temperatures.

21. Establish portable ventilation in the control building hallway to maintain acceptable temperatures.

22. Close and assure that the main steam drain lines are closed in order to maintain vessel inventory.

23. Establish portable ventilation in the Unit 1 shutdown board rooms to maintain acceptable room temperatures.

24. Close the supply line from the condensate storage tank to preclude overfilling the torus (for those fire areas where HPCI/RCIC systems are not available).

With regard to step 12 (initiate/assure isolation of the RWCU system in 30 minutes), the licensee has a modification concerning the isolation of the RWCU system high-low pressure interface boundary. New manual actions will be required as a result of this modification. Since the new actions may impact the manual actions and required time frames described in step 12, this will be verified by inspection prior to Unit 3 restart.

In the SEs of November 3, 1989, and December 8, 1988, the staff addressed the local actions to be performed and the local stations at which actions would have to be taken for a fire in fire area 16 for Unit 2 operation only. During the July 17-21, 1995, inspection, the inspectors walked down the SSIs. The results of the walkdown and followup of the SSIs are documented in Inspection Report No. 50-296/95-37.

3.6 Seventy Two Hour Cold Shutdown Requirement and Repairs

The licensee stated that in the event of fire in any plant area, cold shutdown conditions can be achieved by Units 2 and 3 within 72 hours. Additionally, for combined Unit 2 and Unit 3 operation none of the manual operator actions in the plant or in the control room constitutes a repair activity. The licensee's approach satisfies the requirements of Appendix R to 10 CFR Part 50 and is, therefore, acceptable.

3.7 Associated Circuits

The licensee has evaluated the potential impact of fire damage on associated circuits of concern including common power source, spurious actuation, and common enclosure.

3.7.1 Circuits Associated by Common Power Source

For common power source associated circuits, the licensee evaluated all circuits supplied from a power source (switchgear, motor control centers, and load centers) that also powers equipment required for post-fire safe shutdown. For the identified circuits, the coordination of electrical protection devices (fuses, circuit breakers, or relays) was verified to ensure that a fire induced fault on a branch circuit of a required supply will be cleared by at least one branch circuit protective device prior to fault current propagating to cause a trip of any upstream feeder breaker to the supply.

The licensee's method of protection for the common power supply associated circuit concern satisfies Appendix R requirements and is, therefore, acceptable.

3.7.2 High Impedance Faults

In the SE of December 8, 1988, the staff stated that the licensee's methodology for addressing high impedance faults was based on procedural direction for operators to strip all non-safe shutdown loads from the affected power source. In the SE of November 3, 1989, the staff stated that based on the licensee's determination that sufficient margin exists between power supply feed and load breakers to preclude multiple high impedance faults from affecting the shutdown capability, the previously approved method of resolving this concern (i.e., manual operator actions to shed nonessential loads) is no longer necessary.

The licensee restated this approach (ensuring adequacy of main breaker margin) in the SSA that is the subject of this SE, which stated that the occurrence of high impedance faults that may be initiated as a result of fire damage to connected load circuits was considered during the evaluation of common power source associated circuits, and based on the results of this evaluation, it was concluded that sufficient margin exists in the trip point setting of the main breakers of potentially affected supplies to preclude a trip as a result of such faults. However, the licensee did not describe the basic assumptions used in performing the analysis of this concern. Specifically, the staff could not determine if the evaluation was performed in accordance with the criteria specified in NRC Generic Letter (GL) 86-10, which requires consideration of multiple, simultaneous, faults.

By letter dated October 5, 1995, the licensee submitted additional information regarding the high impedance faults. BFN is divided into 34 fire zones. Depending on the specific location of the fire, either HPCI, RCIC, or LPCI will be used to achieve and maintain shutdown conditions in the reactor. For each system, the licensee has performed an analysis to ensure that time-dependent loads will not be lost due to multiple high impedance faults (MHIF).

3.7.2.1 HPCI and RCIC Systems

The licensee stated that neither the HPCI nor the RCIC systems power time-critical loads. Should MHIFs result in the loss of a required source of electrical power associated with these systems, the operator would have approximately 20 minutes to clear the affected board, re-establish the required electrical alignment, and re-initiate the HPCI or RCIC system. Therefore, for those areas where HPCI or RCIC is available, the licensee has determined that there is adequate time to restore any boards or panels lost due to a MHIF current condition. All operator actions necessary to restore the operation of affected power sources will be governed by written procedures.

3.7.2.2 LPCI System

Unlike the HPCI and RCIC systems, the time required to restore the operability of power supplies associated with the LPCI system may be critical. To address this concern, the licensee has performed additional analyses for the 14 fire areas/zones requiring LPCI for safe shutdown. This analysis considered MHIFs on all required and non-required loads routed in the fire area, and verified that the total MHIF current from these loads will not affect the board/panel feeder breaker at the 60 second trip rating. This approach (i.e., use of the 60 second trip current rating) reduces the number of manual actions that may otherwise be required, and is not used in the evaluation of power sources associated with time critical loads. For power supplies which power time-critical loads, the licensee intends to reduce the electrical loading on specific boards and panels (i.e., non-essential load shedding) in the event of a fire.

The licensee's approach for mitigating the loss of required equipment due to MHIFs satisfies the requirements of Appendix R, Sections III.G.2 and III.G.3, in a manner that is consistent with the guidance provided in GL 86-10, and is, therefore, acceptable.

3.7.3 Spurious Signals

Circuits whose fire-induced spurious actuation could affect the safe shutdown capability were identified and evaluated by the licensee. This evaluation included circuits of components whose inadvertent operation could defeat the safe shutdown capability or cause an unacceptable loss of reactor coolant inventory. When spurious actuation circuits of concern were identified in areas other than the control building, the licensee treated them as required circuits and provided a level of protection equivalent to that required by Section III.G.2 of Appendix R. In the event of fire in the control building, spurious actuations of concern will be mitigated by the alternative shutdown capability to manually control the shutdown systems from outside the main control room. Once manual control has been transferred from the control room to the local panels, any undesirable spurious operation may be corrected.

The licensee has also analyzed the potential spurious signals that could adversely affect the availability of onsite power when required for the case when off-site power is available and the case when off-site power is not available.

The methodology for identifying potential fire-induced spurious operations was to first identify those components which could adversely impact safe shutdown of the reactor. These are the components whose spurious operation could result in either inventory loss from the vessel, or flow divergence or flow blockage in the inventory make-up or decay heat removal systems. Once these potentially spurious components have been identified, the cables associated with each component were identified, along with their routing. For each fire zone and area, those cables and components whose spurious actuation could adversely affect the safe shutdown of the reactor were identified and an appropriate method of resolution was implemented.

The methodology for mitigating potential fire-induced spurious operations was based on the following approach: (1) to first eliminate the potential of the spurious operation by way of rerouting the impacted cables or performing a circuit modification, where the spurious operation could not be tolerated (e.g., removal of power and placement of the power cable in a dedicated conduit for the RHR system suction valve), or (2) specify manual operator actions to be performed to assure correct system alignments were achieved.

The licensee's method of protection for the spurious signals associated circuit concern satisfies Appendix R requirements, and is, therefore, acceptable.

3.7.4 High/Low Pressure Interfaces

The licensee has identified all high/low pressure interface valves. These valves are associated with the RHR, RWCU, core spray, HPCI, RCIC, control rod drive (CRD), feedwater, recirculation sampling, and main steam systems. As discussed in the SE of December 1988, the only interfaces requiring corrective action are RHR suction valves FCV-74-47 and FCV-74-48 and RWCU valves FCV-69-01, FCV-69-02, FCV-69-16 and FCV-69-17.

For the RHR interface, motive power is removed and breaker for FCV-74-47 tagged out during normal power operation. To prevent fire-induced hot shorts, the power cable for FCV-74-47 is routed in dedicated conduits. The lack of a hot short source in the dedicated conduit provided assurance that the associated safe shutdown component would not spuriously operate and is, therefore, acceptable.

In the SE of December 8, 1988, the staff noted that the RWCU discharge lines share a normally-open valve on the common discharge portion of the lines, and single normally-closed valves on the branch connections to the main condenser and radwaste system downstream of the common valve. The staff stated that the licensee proposed to mitigate the spurious operation of downstream valves by removing motive power from a normally-closed upstream valve during normal plant operation, thus locking it in its normally closed position. Additionally, at 8 hours after the manual scram, the two downstream valves (i.e., downstream of the common valve) will be manually closed at the applicable local stations. The staff found this approach to be acceptable.

On the basis of the results of additional evaluations of the RWCU interface however, the licensee has determined that isolating the RWCU system had been erroneously assumed to be capable of preventing high temperature water from failing low temperature piping in the event of fire in fire area 16 or fire zones 2-4 and 3-3. This lack of isolation capability could result in piping temperatures downstream of the nonregenerative heat exchanger in excess of the piping design temperature, thereby potentially rupturing the piping. To resolve this concern the licensee has issued a design change to install a pneumatically actuated isolation valve (3-FCV-69-94) downstream of the RWCU nonregenerative heat exchangers. Control air vent valves allowing the operators to manually close the new isolation valve are to be installed in the reactor building near panels 3-25-2 and 3-25-36. The location of these control air vent valves ensures that the RWCU system can be isolated during a fire in any fire area. The new valve will also automatically close on RWCU system high temperature when actuated by a fusible plug. The fusible plug is a threaded pipe end filled with a low temperature alloy eutectic material which is attached to the RWCU pipe upstream of the new valve, 3-FCV-69-94. The eutectic material will melt on RWCU process flow high temperature thereby venting the control air line which closes RWCU isolation valve 3-FCV-69-94. Position indication for the new isolation valve, 3-FCV-69-94, is provided on El. 593. The indicating lights are mounted on a new electrical junction box (3-ECAB-69-94) installed on the wall adjacent to 3-VTV-32-5102. A new control air pressure indicator (3-PI-32-5100) is also installed adjacent to the manual vent valve on El. 593 to provide the operators with an indication of control air supply pressure to the new isolation valve. Existing Appendix R emergency lights are located in the vicinity of panels 3-25-2 (for new manual vent valve 3-VTV-32-5102) and 3-25-36 (for new manual vent valve 3-VTV-32-5103). This ensures that there is sufficient illumination to allow the operators to perform the required Appendix R manual actions to open the vent valves when needed.

During the July 17-21, 1995, inspection, the inspectors reviewed the modification package related to this design change and found that it provided an acceptable approach for assuring the integrity of the RWCU interface. However, the modification had not been completed at the time of the inspection.

3.7.5 Common Enclosure Associated Circuits

To address common enclosure associated circuits, the licensee has performed an evaluation of all circuits that may share a common enclosure (e.g., cable tray, conduit, panel or junction box) with an Appendix R required circuit. On the basis of this evaluation, the licensee determined that adequate electrical protective equipment is provided for nonessential circuits which share a common enclosure with cables of equipment required for safe shutdown. For the majority of cables, the licensee's evaluation determined that cables would not exceed the maximum short-circuit temperature for the type of cable being analyzed. However, the licensee's analysis also determined that the occurrence of high resistance (low current magnitude) faults on certain cables having cross-linked polyethylene insulation, may cause the temperature of the affected cable to exceed its insulation damage temperature limit (250 °C) but would remain within the auto-ignition temperature of the cable.

Based on the limited population of cables having cross-linked polyethylene insulation, the low probability of the high-resistance type of fault required to cause cable damage prior to actuation of the circuit breaker, and the ability of the cable to withstand this type of fault, if it were to occur, without exceeding the auto-ignition temperature of its insulation, the licensee's approach provides adequate assurance that fire initiated faults on nonessential cables will not result in secondary fires within the common enclosure or prevent required cables located within the enclosure from performing their intended function. On this basis, the licensee's methodology for evaluating the potential effect of fire damage on nonessential circuits which share a common enclosure with circuits of required equipment satisfies the requirements of Appendix R to 10 CFR Part 50 and is, therefore, acceptable.

3.8 Modifications

To ensure that the post-fire safe shutdown capability of Unit 3 is sufficient to support restart, the licensee committed to complete 111 modifications of circuits, cables, associated circuits, fire protection features, and miscellaneous items. Proposed modifications include design changes necessary to ensure the availability of at least four MSRVs in the event of fire in any area, rerouting of MSRV control circuits in dedicated conduits to prevent spurious actuations, preventing spurious opening of MSRVs by providing isolation capability for control cables of non-automatic depressurization MSRVs routed through the control building, installation of new analog trip units, rerouting of cables to ensure the availability of reactor level and pressure instrumentation, and cable modifications which involve rerouting of cables associated with some safe shutdown components. Except as discussed above, the staff did not review these modifications as part of this SE. Verification of the modifications is subject to future inspections.

4. INSPECTOR FOLLOWUP ITEMS

Verification of the following items is required prior to restart of Unit 3:

1. Verification that the next revision to the BFN-FPR reflects the results of the most recent calculations of maximum drywell temperature (Section 3.2.2).

2. Verification of the manual action to transfer the RCIC system suction from the condensate storage tank to the torus from outside the control room (Section 3.4.5).

3. Verification that new manual actions that may be required as a result of the RWCU system modifications have been considered in the licensee's listing of principal manual actions and time frames (Section 3.5.1).

Information regarding these items was provided by the licensee to inspectors on October 19, 1995. This inspection is documented in Inspection Report 50-259, 50-260, and 50-296 95-60.

With regard to inspector follow-up item 1, inspection confirmed that the licensee has documented the required changes to address the maximum drywell temperature calculation in the BFN FPR. The licensee has drafted a FPR revision to incorporate the required changes, and is tracking these changes as an open item (Punchlist Item REC-0208) to ensure implementation before BFN Unit 3 restart. Inspection confirmed the changes are appropriate, and that the licensee's open item tracking provides reasonable assurance the required changes will be completed. Therefore, inspector follow-up item 1 is closed.

With regard to inspector follow-up item 2, the licensee provided information from procedure 2/3-SSI-16, which provides instructions for the manual actions to transfer the RCIC system suction to the torus. Inspectors confirmed the instructions accomplish this action outside the control room, as discussed in Section 3.4.5, above. Therefore, inspector follow-up item 2 is closed.

For inspector follow-up item 3, the licensee provided draft changes for procedure 2/3-SSI-3-3, specifying manual actions required for operation of components isolating high temperature piping from low temperature components. The licensee also provided draft changes to the BFN FPR addressing the BFN Unit 3 configuration. These items are being tracked as part of Punchlist Item REC-0195. Inspection confirmed the changes are appropriate, and that the licensee's open item tracking provides reasonable assurance the required changes will be completed. Therefore, inspector follow-up item 3 is closed.

5. CONCLUSIONS

On the basis of its review, the staff has concluded that the portions of the revised BFN-FPR for Units 2 and 3 that are addressed in this SE meet NRC fire protection requirements and guidance and are, therefore, acceptable. Verification of adequate implementation of the revised FPR requires inspection of the items discussed in Section 4.

Principal Contributors: Amarjit Singh and Joseph F. Williams

Dated: November 2, 1995

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 241 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 200 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter dated September 30, 1993, Tennessee Valley Authority (the licensee), submitted a request for amendments to the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 Operating Licenses. The proposed changes add a reference to the November 2, 1995 Safety Evaluation (Enclosure 4 of this document) of the BFN Appendix R Safe Shutdown Program. The licensee also proposes to modify the license for BFN Unit 3 by adding the definition of the Appendix R Safe Shutdown Program.

The Technical Specifications (TS) for BFN Units 1, 2, and 3 were revised to implement the guidance of Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirement from Technical Specifications." Since the BFN Appendix R Safe Shutdown Program was originally developed to support Unit 2 restart it did not take into account operation of Units 1 or 3; therefore, in order to support Unit 3 restart, the program was modified by a submittal dated December 20, 1994. This modified program was approved by the staff in a Safety Evaluation dated November 2, 1995. It is this Safety Evaluation that the licensee has requested to reference in the Operating Licenses of all three units.

2.0 EVALUATION

Sections 2.C.13, 2.C.14, and 2.C.7 of the Operating Licenses for BFN Units 1, 2, and 3, respectively, contain references to the dates of the Safety Evaluations approving the Fire Protection Program that the licensee will implement. Since a modified Appendix R Safe Shutdown Program has been evaluated by the staff, the license for each unit must be amended to reflect accurately the program that is to be implemented.

The Appendix R Safe Shutdown Program has been reviewed and approved by the staff, as documented in a safety evaluation dated November 2, 1995. This amendment to the operating licenses of BFN Units 1, 2, and 3 appropriately updates the licenses to reflect the issuance of this safety evaluation, and is therefore acceptable.

Enclosure 5

The staff also finds that the definition of the Appendix R Safe Shutdown Program is consistent with the staff's November 2, 1995 safety evaluation, and is also acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations the Alabama state official was notified of the proposed issuance of the amendment. The state official had no comments.

4.0 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 (59 FR 629). 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.

5.0 CONCLUSION

The Commission has concluded, based upon the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: George F. Wunder

Dated: November 2, 1995