June 20, 2005

Mr. Karl Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AN AMENDMENT REGARDING DELETION OF THE LOW PRESSURE COOLANT INJECTION MOTOR-GENERATOR SETS (TAC NO. MC3822) (TS 427)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 254 to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. This amendment is in response to your application dated July 8, 2004, as supplemented on April 15, 2005.

The amendment removes the requirement to maintain an automatic transfer capability for the power supply to the Low Pressure Coolant Injection inboard injection and recirculation pump discharge valves.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Margaret H. Chernoff, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management

Docket No. 50-259

Enclosures: 1. Amendment No. 254 to
License No. DPR-33
2. Safety Evaluation

cc w/enclosures: See next page

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Package: ML051580060 TS: ML051730453
ADAMS Accession No.: ML051580047 NRR-058

OFFICE	PDII-2/PM	PDII-2/LA	DE/EEIB	DSSA/SRXB	OGC	PDII-2/SC
NAME	MChernoff	BClayton	RJenkins	FAkstulewicz	JHull	MMarshall
DATE	6/14/05	6/14/05	1/21/05	12 /22 /04	6 /14 /05	6/20/05

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Date: June 20, 2005

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 254
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 July 8, 2004, as supplemented on April 15, 2005, 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-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 254, 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael L. Marshall, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 20, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 254

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3.5-7	3.5-7
3.8-33	3.8-33
3.8-34	3.8-34
3.8-35	3.8-35
3.8-36	3.8-36
3.8-37	3.8-37

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 254 TO FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 INTRODUCTION

By letter dated July 8, 2004 (ADAMS Accession No. ML0042020415), as supplemented on April 15, 2005 (ADAMS Accession No. ML051050580), the Tennessee Valley Authority (the licensee) requested changes to the Browns Ferry Nuclear Plant (BFN), Unit 1, Technical Specifications (TSs). The requested change removes the requirement to maintain an automatic transfer capability for the power supply to the Low Pressure Coolant Injection (LPCI) inboard injection and recirculation pump discharge valves. The automatic transfer of the power supply for the LPCI inboard injection and recirculation pump discharge valves, which are powered from 480V Reactor Motor Operated Valve (RMOV) Boards D and E, is not required to satisfy regulatory requirements. References to the RMOV Boards D and E will be removed from the TSs.

The April 15, 2005, letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The regulations Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46 contain the acceptance criteria for emergency core cooling systems (ECCSs) for light-water nuclear power reactors. ECCSs must be designed so that the calculated cooling performance following postulated loss-of-coolant accidents (LOCAs) conforms to the criteria specified in 10 CFR 50.46(b). One of the criteria is that ECCSs shall be designed to ensure peak clad temperature does not exceed 2200 E Fahrenheit. Appendix K to Part 50 contains the regulations applicable to ECCS evaluation models.

The BFN, Unit 1 licensing basis for ECCS protection systems is described in Updated Final Safety Analysis Report (UFSAR) Section 8.9, “Safety Systems Independence Criteria and Bases for Electrical Cable Installation,“ and Section 7.4, “Emergency Core Cooling Control and Instrumentation.” Appendix A of the BFN, Unit 1 UFSAR provides a discussion of how the design of BFN, Unit 1 conformed to the Atomic Energy Commission Draft General Design Criteria. Section 8.9 of the UFSAR states that “electrical circuits associated with redundant or counterpart divisions, components, or subsystems of electrical systems important to safety are separated from each other by means of spacing or barriers or analysis to demonstrate functional redundancy." These systems are designed to meet the proposed criteria of Institute of Electrical and Electronics Engineers Standard "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE-279-1971).

3.0 TECHNICAL EVALUATION

3.1 Description of Affected Systems

3.1.1 Emergency Core Cooling System

The BFN, Unit 1 ECCS consists of the High Pressure Coolant Injection (HPCI) system, the Automatic Depressurization System (ADS), the Core Spray (CS) system and the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) System.

The HPCI System is provided to assure that the reactor is adequately cooled to limit fuel cladding temperature during a LOCA and it consists of a steam driven pump discharging a flow rate of 4500 gallons per minute (gpm). The HPCI starts automatically when the reactor pressure vessel (RPV) level is low or drywell pressure high. Startup of the HPCI is completely independent of AC power.

The ADS uses six of the main steam safety relief valves and they are used to reduce the reactor pressure so that the low pressure ECCS subsystems such as the CS system and the LPCI mode of the RHR system can be used to mitigate a LOCA. The ADS starts automatically if one of the LPCI pumps or two CS pumps are running and the RPV level is low for 360 seconds or RPV level is low and drywell pressure is high for 150 seconds. All the safety relief valves discharge into the suppression pool.

The CS system provides the protection to the core for the large pipe break in the plant connected to the reactor coolant pressure boundary. There are two independent loops (5600 gpm flow per two pumps) each consisting of two 50-percent capacity centrifugal motor driven pumps discharging water into the core through a core spray sparger above the core. CS pumps A and C are in one loop and pumps B and D are in the second loop. The system is started automatically on reactor low water level or drywell high pressure. Both the pumps in the loop must operate for adequate spray cooling in the reactor during a LOCA.

LPCI is an operating mode of the RHR and this low pressure system is also used to mitigate the large break LOCA. The initiation logic is the same as the CS system. During LPCI operation, the four RHR pumps take suction from the suppression pool and discharge to the reactor vessel into the core region through both of the reactor recirculation loops. Two pumps LPCI-A and LPCI-C in one loop discharge into recirculation loop-B and two pumps LPCI-B and LPCI-D pumps in the second loop discharge into recirculation loop-A. On receipt of an initiation signal following a postulated recirculation line break, both LPCI injection valves are signaled to open (when the low pressure permissive is satisfied), both recirculation discharge valves are signaled to close when the reactor pressure decreases to 230 pounds per square inch gauge, and the LPCI flow from two RHR pumps (one LPCI loop) is directed to the unbroken recirculation loop. Flow from other two RHR pumps (one LPCI loop) is directed to the assumed broken recirculation loop. At present, there is no loop selection logic for LPCI injection.

3.1.2 480V Electrical Distribution System

There are two Unit 1 480V Shutdown Boards, A and B, which are normally powered from 4kV Shutdown Boards A and C, respectively. The 480V Shutdown Boards A and B feed the 480V RMOV Boards A and B (safety related), and RMOV Board C (nonsafety related). The 480V Shutdown Boards A and B also feed the 480V RMOV Boards D and E (safety related) through LPCI M-G sets.

The Unit 1 480V RMOV Board D provides Division I power to the following loads (backup power by Diesel Generator (DG) A):

FCV-68-79,	Recirculation B Pump Discharge Valve
FCV-74-7,	RHR (LPCI) Pumps A and C Minimum Flow Bypass Valve
FCV-74-53,	RHR (LPCI) Pumps A and C Injection Valve; and
FCV-74-59,	RHR Test Valve

Similarly, Unit 1 480V RMOV Board E provides Division II power to the following loads (backup power by DG C):

FCV-68-3,	Recirculation Pump A Discharge Valve
FCV-74-30,	RHR (LPCI) Pumps B and D Minimum Flow Bypass Valve
FCV-74-67,	RHR (LPCI) Pumps B and D Injection Valve; and
FCV-74-73,	RHR Test Valve

Presently, the 480V RMOV Boards D and E power supplies are designed to automatically transfer to the alternate source (the opposite division 480V Shutdown Board) upon detection of an undervoltage condition from the normal feed.

There are two independent loops of RHR (LPCI) equipment. The automatic transfer capability for 480V RMOV Boards D and E has been designed to ensure that the LPCI injection occurred in both loops with at least one pump in each loop. If one loop's LPCI injection valve and the associated reactor recirculation loop discharge valve lost power, the RMOV board would automatically transfer to the opposite division's power supply to ensure operation of the valves. In the original scheme, the automatic transfer could propagate an electrical fault to both divisions of power supply. As a result, M-G sets were included in the existing design for both the normal and alternate power supplies in order to provide electrical isolation between the associated 480V Shutdown Board and the RMOV Board.

3.2 <u>Description of Proposed Changes</u>

In the proposed change, due primarily to the obsolescence of equipment, the Unit 1 LPCI Motor-Generator (M-G) sets and the associated 480V RMOV Boards D and E will be removed from service prior to restart of Unit 1. The loads, presently fed from 480V RMOV Boards D and E, will be transferred to Division I and Division II 480V RMOV Boards A and B respectively.

The proposed TS changes are briefly described as follows:

TS Surveillance Requirement 3.5.1.12, which requires “Verify automatic transfer of the power supply from the normal source to the alternate source for each LPCI subsystem inboard injection valve and each recirculation pump discharge valve,” every 18 months will be deleted.

The electrical loads currently supplied power from 480V RMOV Boards D and E can be relocated onto other safety-related RMOV Boards A and B. Therefore, the references to RMOV Boards D and E will be deleted. The remaining items will be renumbered in TS 3.8.7.

3.3 Technical Evaluation

With the deletion of Boards D and E, the automatic transfer capability for the power supply to the LPCI injection valves and recirculation pump discharge valves will also be deleted. The RMOV Boards A and B (to which the loads are transferred) do not have automatic transfer capability. The impact on the number of the Emergency Core Cooling Subsystems available under various contingencies due to the loss of automatic transfer capability is discussed in Section 3.3.1 of this Safety Evaluation.

3.3.1 Impact on ECCS

The ECCS subsystems are designed to limit peak clad temperature over the complete spectrum of possible break sizes, including the design basis break. The design basis break is defined as the complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel (i.e., one of the recirculation loop pipes) with displacement of the piping so that blowdown occurs through both ends.

Performance of the ECCS for a complete spectrum of pipe break sizes and postulated single failures has been analyzed. The limiting postulated failures evaluated in the LOCA analyses are:

- Failure of a unit battery board, assuming offsite power available and not available;
- Spurious accident signal from another unit;
- Failure of a LPCI injection valve;
- Failure of a DG, assuming offsite power available and not available; and
- Failure of HPCI.

The proposed change does not affect the available equipment for the following single failures:

- Failure of a unit battery board, assuming offsite power available and not available;
- Spurious accident signal from another unit;
- Failure of a LPCI injection valve; and
- Failure of HPCI.

The most limiting single failure for BFN, Unit 1 is the battery failure during the recirculation suction line break. The operation of the ECCS for that limiting event is unchanged by the proposed amendment.

The proposed change impacts the available equipment upon loss of a DG. During a LOCA (suction or discharge line break), without offsite power available, the loss of a DG as the single failure will cause loss of power to either 480V RMOV Boards A and D or B and E. After the proposed change is implemented, the loads previously powered from either RMOV Boards D or E will not automatically transfer to receive power. Therefore, there will be one less LPCI pump actually available for injection into the vessel.

For the case of the recirculation suction line break concurrent with the loss of a DG, instead of three RHR pumps operating and injecting water through two LPCI loops, the revised licensing bases will have two RHR pumps operating and injecting water through one LPCI loop.

At present for the DG failure during a recirculation suction line break, HPCI, ADS, one CS and three LPCI pumps will be available. The DG failure will take out one LPCI pump and one CS pump. The power supply to the LPCI injection and recirculation discharge valve in one loop is also lost. The current LOCA analysis takes credit only for two LPCI pumps into one loop, one CS, HPCI and ADS. Because the minimum equipment requirements of the current licensing bases are still met, the proposed change is acceptable.

For the case of the recirculation discharge line break, coupled with failure of a DG, instead of one RHR pump operating and injecting water through a LPCI loop, after implementation of this amendment, there will be no RHR pumps available for LPCI.

At present for the DG failure during a recirculation discharge line break, ADS, HPCI, one CS and no LPCI pumps will be available. The DG failure will take out the power supply to the LPCI injection valve in the unbroken loop and one CS pump. No credit is taken for a single CS pump. The remaining CS pumps in the other division are assumed to be operable. The remaining two LPCI pumps will be injecting through the broken discharge line break into the drywell bypassing the reactor. No LPCI pump will be available to mitigate the recirculation line break LOCA. The ADS and one CS will be available. The current LOCA analysis for Recirculation discharge break takes credit only for one CS loop (two CS pumps) and ADS.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the impact of the proposed changes on the capabilities of the ECCS at BFN, Unit 1. Because the minimum equipment requirements of the current licensing bases are still met and the requirements of 10 CFR 50.46 are still satisfied, the NRC staff has determined that the proposed change is acceptable.

3.3.2 Impact on Electrical Distribution System

The licensee confirmed that the following evaluations associated with the impact on the electrical distribution system due to the proposed TS changes have been performed:

- Electrical calculations have been revised to show that the additional loading on the 480V RMOV Boards 1A and 1B is acceptable and voltage at the loads are adequate.

- The cables will be routed to ensure adequate separation is maintained between redundant equipment to satisfy divisional and Appendix R requirements.

- The cables have been sized to meet the ampacity, voltage drop and short circuit requirements.

- Control power for the 480V RMOV Boards is provided by control power transformers within the compartments. Moving the loads will have no effect on control power availability.

- Heat load calculations have been revised. There are no adverse impacts.

- The Boards are seismic Class I and meet seismic equipment qualification requirements.

- A Failure Modes Evaluation was performed to evaluate potential failure of the DG supplying emergency power, the 480V RMOV Board, and individual component failures. It is concluded that there will be no new failure modes or adverse effects as a result of the proposed changes.

The NRC staff has reviewed the description of the proposed changes to the electrical distribution system, and based on the considerations listed above, has concluded that implementation of this change will not adversely impact the electrical distribution system. The NRC staff concluded that the proposed changes will not alter the conformance of the electrical distribution system to the applicable regulatory requirements.

4.0 SUMMARY

As discussed in the preceding sections of this safety evaluation, the NRC staff has reviewed the licensee's proposed elimination of the LPCI M-G sets and concluded that there will be no adverse impact on the analyzed performance of the ECCS, nor will there be an adverse impact on the electrical distribution system at BFN, Unit 1. Therefore, the NRC staff has concluded that the proposed TS changes are acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 64990). Accordingly, the amendment meets 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 amendment.

7.0 CONCLUSION

The Commission has concluded, based on 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: G. Thomas
V. Goel

Date: June 20, 2005

cc:
Mr. Ashok S. Bhatnagar, Senior Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Larry S. Bryant, General Manager
Nuclear Engineering
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Michael D. Skaggs
Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

General Counsel
Tennessee Valley Authority
ET 11A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. John C. Fornicola, Manager
Nuclear Assurance and Licensing
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Kurt L. Krueger, Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Jon R. Rupert, Vice President
Browns Ferry Unit 1 Restart
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Robert G. Jones
Browns Ferry Unit 1 Plant Restart Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Scott M. Shaeffer
Browns Ferry Unit 1 Project Engineer
Division of Reactor Projects, Branch 6
U.S. Nuclear Regulatory Commission
61 Forsyth Street, SW.
Suite 23T85
Atlanta, GA 30303-8931

Mr. Fredrick C. Mashburn
Senior Program Manager
Nuclear Licensing
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

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

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, AL 35611-6970

State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, AL 36130-3017

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