May 9, 2005

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

SUBJECT: BROWN FERRY NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT REGARDING REVISION TO LOW PRESSURE EMERGENCY CORE COOLING SYSTEM ALLOWED OUTAGE TIME (TAC NO. MC6824) (TS-452)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 294 to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. This amendment is in response to your application dated April 26, 2005, as supplemented on April 29 and May 3, 2005. In your submittal, you requested that the proposed change be considered on an emergency basis.

The proposed amendment revises the completion time for the action associated with an inoperable low pressure Emergency Core Cooling System injection/spray system to 14 days on a one-time basis.

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. 50-260

Enclosures: 1. Amendment No. 294 to License No. DPR-52
2. Safety Evaluation

cc w/enclosures: See next page

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Package: ML051290271 Tech Spec: ML051290383
ADAMS Accession No.ML051290264 NRR-058

OFFICE	PDII-2/PM	PDII-2/PM	PDII-2/LA	DSSA/RSX/SC	DSSA/SPSB/SC	OGC	PDII-2/SC
NAME	MChernoff	SLewis	BClayton	FAkstulewicz Memo dated	SWong for MReinkhart	AHodgdon	MMarshall
DATE	5/9/05	5/9/05	5/9/05	5/2/05	5/9/05	5/9/05	5/9/05

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 294
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 April 26, 2005, as supplemented on April 29 and May 2, 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-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. 294, 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 7 days from the date of issuance. This amendment expires on June 1, 2005.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Change to the Technical Specifications

Date of Issuance: May 9, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 294

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

REMOVE	INSERT
3.5-1	3.5-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 294 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

By letter dated April 26, 2005, as supplemented on April 29 and May 2, 2005, the Tennessee Valley Authority (the licensee) submitted a request for a change to the Browns Ferry Nuclear Plant (BFN), Unit 2, Technical Specifications (TSs). The requested change would revise the completion time (CT) for the actions associated with an inoperable Emergency Core Cooling System (ECCS) injection/spray system to 14 days. In its April 26, 2005, letter, the licensee requested that this change be considered on an emergency basis in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.91(a)(5).

The April 29 and May 2, 2005, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The licensee requested a revision to the CT associated with Condition A of TS 3.5.1. TS 3.5.1 requires that each ECCS injection/spray subsystem and the Automatic Depressurization System function of six safety/relief valves be Operable in Mode 1 and under certain conditions in Modes 2 and 3. If one low pressure ECCS injection/spray subsystem or one low pressure coolant injection (LPCI) pump in both LPCI subsystems is inoperable, the required action is to restore the subsystem(s) to operable status within 7 days. The requested revision was to extend the CT for this condition to 14 days. The NRC staff reviewed this change as a one-time change to the CT.

Browns Ferry was constructed prior to promulgation of the 10 CFR Part 50, Appendix A, General Design Criteria (GDC). However, GDC 35, “Emergency Core Cooling,” GDC 36, “Inspection of Emergency Core Cooling System,” and GDC 37, “Testing of Emergency Core Cooling System,” are applicable to this proposed change. Because this proposed change only affects the CT and does not involve a change to the design or method of operation of the ECCS, conformance to the GDC criteria is not altered by this change.

ENCLOSURE

The regulatory requirements related to the contents of TSs are contained in 10 CFR 50.36. Pursuant to 10 CFR 50.36, TSs are required to contain (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls. This change request involves a change to a limiting condition for operation.

The regulations in 10 CFR 50.46 provide acceptance criteria for ECCS for light-water nuclear power reactors. ECCS must be designed so that its calculated cooling performance following postulated loss-of-coolant-accidents (LOCAs) conforms to the specified criteria. One of the specified criteria is that the calculated maximum fuel element cladding temperature shall not exceed 2200 degrees Fahrenheit (EF).

Two Regulatory Guides (RGs) provide guidance for using risk information to evaluate changes to TS Allowed Outage Times (AOTs) and Surveillance Intervals in order to assess the impact of such proposed changes on the risk associated with plant operation. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides guidance to the U.S. Nuclear Regulatory Commission (NRC) staff for the use of Probabilistic Risk Assessment to support decisions to modify an individual plant's licensing basis. RG 1.177, "An Approach for Using Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," provides guidance for the review of licensee-initiated TS change requests when the results of risk analyses are used to help justify TS changes.

The regulations in 10 CFR 50.91 describe the procedures the Commission will follow for an application requesting an amendment to an operating license. Specifically, 10 CFR 50.91(a)(5) describes the process that is followed if the Commission determines that an emergency exists.

3.0 TECHNICAL EVALUATION

3.1 ECCS Configuration

The BFN 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 at 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 Core Spray 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 low level and drywell high pressure exist 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 for 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) in the second loop discharge into recirculation loop-A. On receipt of an initiation signal following a recirculation 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 the other two RHR pumps (one LPCI loop) is directed to the broken recirculation loop. At present, there is no loop selection logic for LPCI injection.

Only one RHR subsystem with one pump and one heat exchanger will be inoperable during the extended outage time (7 days to 14 days). The remaining three RHR pumps and the three heat exchangers are operable. Also, the HPCI system, Core Spray and ADS are unaffected. Inoperability of one single RHR pump is an analyzed condition in terms of recovering from a LOCA.

There is a cross-tie between the RHR service water system and the RHR system, which can be used as a last resort system to flood the reactor and the containment for long-term cooling. Also, there are RHR system cross-tie connections between Units 2 and 1 and also between Unit 2 and Unit 3.

Even though RHR system redundancy is reduced due to the inoperability of one RHR subsystem, there are sufficient systems available for core cooling.

The following single failures were considered in the LOCA analysis as shown in Final Safety Analysis Report Table 6.5-3:

(a) Battery
(b) Opposite Unit False LOCA Signal
(c) LPCI injection Valve
(d) Diesel Generator
(e) HPCI

For all the above assumed failures, except HPCI failure, three or fewer LPCI pumps are required to mitigate a LOCA. For the HPCI failure, during a recirculation suction break, all four LPCI pumps are required to mitigate a LOCA. However, the probability of HPCI failure during the extended outage time of 14 days is low. For the risk assessment of this proposed change refer to section 3.2 of this Safety Evaluation.
The most limiting single failure for BFN is the Battery failure during the Recirculation Suction Break. The operation of the ECCS for that limiting event is unchanged by the proposed amendment.

The BFN Unit 2 core contains GE-13, GE-14 and ATRIUM-10 fuel types. The licensee evaluated the plant response to a large break LOCA using staff approved evaluation models SAFER and GESTR. The Peak Cladding Temperature for the most limiting case, battery failure with recirculation break, for GE-14 fuel is 1810 EF, which is less than the acceptance criterion of 2200 EF specified in 10 CFR 50.46 and hence is acceptable. TVA also calculated the peak cladding temperature for ATRIUM-10 fuel using NRC staff approved evaluation models. The peak cladding temperature for the Framatome fuel was calculated for extended power uprate operation of 3952 Mwt. The calculation is bounding for the current licensed power level of 3458 Mwt. The calculated peak cladding temperature is 2007 EF, which is less than the acceptance criterion of 2200 EF specified in 10 CFR 50.46 and hence is acceptable.

The CT for the Required Action for Condition A when one low pressure ECCS injection/spray subsystem inoperable is changed from 7 days to 14 days. The proposed one-time change is acceptable as explained above.

3.2 Risk Insights

3.2.1 Risk Assessment Evaluation

In evaluating the risk information submitted by the licensee, the NRC staff followed the three-tiered approach documented in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

Under the first tier, the NRC staff determines if the proposed change is consistent with the NRC's Safety Goal Policy Statement, as documented in RG 1.174. Specifically, the first tier objective is to ensure that the plant risk does not increase unacceptably during the period the equipment is taken out of service.

The second tier addresses the need to preclude potentially high-risk plant configurations that could result if additional equipment, not associated with the proposed change, is taken out of service during the proposed 7-day additional AOT extension.

The third tier addresses the establishment of a configuration risk management program for identifying risk-significant configurations resulting from maintenance or other operational activities, and taking appropriate compensatory measures to avoid such configurations.

3.2.2 Basis and Quality of Risk Assessment

The licensee used its Probabilistic Risk Assessment (PRA) model and appropriate conservative assumptions to assess the risk increase associated with operation at power for a period of 7 additional days without an operable Unit 2 train "A" RHR system. The risk consideration included maintaining defense-in-depth and quantifying risk to determine the change in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) as a result of the proposed 7-day AOT extension for the "A" RHR. Also, the licensee is maintaining the continuous on-line risk management program to control the performance of other risk-significant tasks during the extended AOT period with consideration of specific compensatory measures listed in the submittal to minimize risk. The dominant accident sequences contributing to the assessed risk increase include the occurrence of conditions due to the unavailability of and demand for the use of the "A" RHR subsystem. Current TS allows one train of the RHR system to be out of service for 7 days.

The NRC staff evaluated the quality of the PRA models, major assumptions, and data used in the risk assessment, and found them acceptable for this application. The licensee risk model included all functions that have a risk achievement worth greater than 2.0, and is being updated every other refueling outage. This evaluation compared the applicable findings from the NRC staff's review of the licensee's PRA with the NRC's Standardized Plant Analysis Risk Model (SPAR), Version 3.2, employing NRC PRA quantification tool, SAPHIRE Version 7 and NRC Manual Chapter 0609, Appendix H for LERF, as well as findings from similar evaluations of similar plants.

3.2.3 Risk Impact of the Proposed Change (Tier 1)

An acceptable approach to risk-informed decisionmaking is to show that the proposed change to the design basis meets several key principles. One of these principles is to show that the proposed change results in a small, but acceptable, increase in risk in terms of CDF and LERF, and is consistent with the NRC's Safety Goal Policy Statement. Acceptance guidelines for meeting this principle are presented in RG 1.174. The licensee used its PRA model to calculate risk increases due to the CT extension of 7 days, during which the other three RHR pumps would be available. Both the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP) were assessed. These quantities are a measure of the increase in probability of core damage and large early release, respectively, during a single outage that would last for the entire duration allowed by the proposed change. Based on the one-time extension of 7 days, the incremental changes are summarized in the following table:

	Baseline CDF	Incremental Change in CCDP	Baseline LERF	Incremental Change in ICLERP
Prior to AOT Extension	1.3E-06/yr		2.99E-07/yr	
Increase due to 7-days AOT extension		2.44E-08		2.13E-08
New Baseline CDF	**1.32-06/yr**		**3.12E-07/yr**	
Increase with 7-day AOT Extension (using NRC SPAR 3.2 Model)		**9.4E-09**		**9.4E-09**

New Baseline CDF after 7-day AOT extension based on annualized ICCDP (using NRC SPAR Model)	1.31E-06/yr		3.08E-07/yr	
Acceptance Criteria		1.0E-06		1.0E-07

The LERF is calculated employing NRC Inspection Manual Chapter 0609, Significance Determination Process Appendix H with the CDF-LERF conversion factor of 1.0. This ratio (LERF-to-CDF) is larger than 0.3, compared with the licensee results. The rationale for using such a large ratio is based on the unique containment design feature, as the BWR Mark I containment is more susceptible to containment liner melt-through, and is conservative in determining the risk acceptability of the proposed one-time CT extension.

During the proposed extension period, the baseline total CDF and LERF have been increased by 9.4E-09/yr due to the incremental changes in ICCDP and ICLERP respectively, resulting from the one time 7-day extension of the AOT under the TS 3.5.1. The new baseline values for this proposed change under TS 3.5.1 would be minimal for both CDF and LERF, and the risk increases under the proposed AOT extension are well within the acceptable range.

The transitional and shutdown risks are qualitatively evaluated with respect to at-power configuration risk. The evaluation basis includes potential relative risk contributors as well as relative functional attributes by the "A" RHR pump for various plant configurations. The RHR system is in standby in Mode 1 operation, but is a vital system when transitioning to and operating in Mode 3. Thus, the staff views the risk associated with plant transitional operation leading to shutdown as being relatively higher than the risk during at-power operation without the "A" RHR pump and associated support systems. Without one of the low pressure injection and spray pumps, the shutdown risk is viewed to be higher than that of the Mode 1 operation, during which there will be alternate injection paths with relatively high coolant pressure. Thus, the staff views the integral ICCDP due to the unavailable "A" RHR pump and its support systems to be smaller during at-power operation as compared with the integral ICCDP of transitional and shutdown configuration risk without the "A" RHR pump.

In conclusion, a one time 7-day extension of the CT associated with Condition A of TS 5.3.1 at power to perform appropriate maintenance work would be more desirable than to perform a transition to and maintenance at hot shutdown with the "A" RHR pump inoperable.

3.2.4 Avoidance of High Risk Plant Configurations (Tier 2)

The licensee's PRA will identify and estimate major risk contributors of plant configurations, contributing event sequences, and associated cutsets. Potential major risk contributors include plant equipment failures, human errors and common cause failures. Insights from the risk assessment would be used in identifying and monitoring the plant configurations or conditions that may lead to significant risk increases during the CT extension. The NRC staff finds that the proposed precautions, as well as the proposed compensatory measures, identified in the licensee's submittal are adequate for preventing plant configurations or conditions that may

increase risk significantly. In conclusion, there is reasonable assurance that high risk plant configuration will not occur during the proposed 7-day extension period.

3.2.5 Risk-Informed Configuration Risk Management (Tier 3)

The intent of risk-informed configuration risk management is to ensure that plant safety is maintained and monitored. A formal commitment to maintain a configuration risk management program is necessary on the part of a utility prior to implementation of a risk-informed TS. This program can support the licensee's decisionmaking regarding the appropriate actions to control risk whenever a risk-informed TS Condition is entered. The NRC staff finds that the licensee has an adequate configuration management program.

4.0 EMERGENCY CIRCUMSTANCES

In its April 26, 2005, letter, the licensee requested that this amendment be treated as an emergency amendment. In accordance with 10 CFR 50.91(a)(5), the licensee provided information regarding why this emergency situation occurred and how it could not be avoided.

The licensee requested this amendment to provide additional flexibility for preventive or corrective maintenance and repair of the RHR system. Because of a current maintenance issue on RHR Heat Exchanger 2A, the licensee will have to isolate the associated RHR Pump. When the RHR Pump is isolated, the licensee will enter Condition A of TS 3.5.1. Currently, this Condition has a 7-day CT for restoring the affected system to operable status. If this CT is not met, the licensee would be required to proceed to Mode 3. The licensee could not reasonably have foreseen that this maintenance issue would arise or that the repair would require a period of time potentially in excess of the CT for the Required Action associated with Condition A of TS 3.5.1.

The Commission finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of the plant. The licensee has explained why the emergency situation occurred and why it could not be avoided. Therefore, this request was handled under the provisions of 10 CFR 50.91(a)(5).

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or,

(2) Create the possibility of a new or different kind of accident from any previously evaluated; or,

(3) Involve a significant reduction in a margin of safety.

The following analysis was provided by the licensee in its letter of April 26, 2005.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The low pressure ECCS subsystems are designed to reflood the reactor vessel after a design basis Loss of Coolant Accident (LOCA). The proposed 14 day AOT does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in the manner in which the ECCS provides plant protection or which create new modes of plant operation. In addition, a Probabilistic Safety Assessment (PSA) evaluation concluded that the risk contribution of the AOT extension is non-risk significant.

   The proposed request will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

   Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. There are no hardware changes nor are there any changes in the method by which any plant system performs a safety function. This request does not affect the normal method of plant operation.

   The proposed amendment does not introduce new equipment, which could create a new or different kind of accident.

   No new external threats, release pathways, or equipment failure modes are created. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this request. Therefore, the implementation of the proposed amendment will not create a possibility for an accident of a new or different type than those previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   No. BFN's ECCS is designed with sufficient redundancy such that a low pressure ECCS subsystem may be removed from service for maintenance or testing. The remaining subsystems are capable of providing water and removing heat loads to satisfy the UFSAR [Updated Final Safety Analysis Report] requirements for accident mitigation or unit safe shutdown.

A PSA evaluation concluded that the risk contribution of the AOT extension is non-risk significant.

There will be no change to the manner in which safety limits or limiting safety system settings are determined nor will there be any change to those plant systems necessary to assure the accomplishment of protection functions. There will be no change to post-LOCA peak clad temperatures. For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

The Commission agrees with the licensee's analysis and, thus, makes a final determination that the amendment does not involve a significant hazards consideration.

6.0 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.

7.0 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. 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 made a final no significant hazards finding with respect to this amendment. 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.

8.0 CONCLUSION

The Commission has concluded, based on the discussion provided in Sections 3.1 and 3.2 of this safety evaluation, 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Additionally, the Commission has concluded, based on the considerations discussed above, that (1) the amendment does not: (a) involve a significant increase in the probability or consequences of an accident previously evaluated; or, (b) create the possibility of a new or different kind of accident from any previously evaluated; or, (c) involve a significant reduction in a margin of safety and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (3) such activities will be conducted in compliance with the Commission's regulations, and (4) 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: J. Chung, NRR
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Date: May 9, 2005

SUBJECT: BROWN FERRY NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT REGARDING REVISION TO LOW PRESSURE EMERGENCY CORE COOLING SYSTEM ALLOWED OUTAGE TIME (TAC NO. MC6824) (TS-452)

Dated: May 9, 2005

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