May 5, 1993

Docket Nos. 50-260

Dr. Mark O. Medford, Vice President
Nuclear Assurance, Licensing & Fuels
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS REGARDING REACTOR WATER CLEANUP SYSTEM TEMPERATURE SWITCHES (TAC NO. M85253) (TS 329)

The Commission has issued the enclosed Amendment No.213 to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant (BFN), Unit 2. This amendment is in response to your application dated December 23, 1992, requesting changes to the BFN Unit 2 Technical Specifications, adding requirements for new temperature switches intended to mitigate effects of postulated Reactor Water Cleanup System pipe breaks.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No.213 to
   License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDII-4LA	PDII-4PM	PDII-4PM	OGC	PDII-4D
NAME	MSanders MS	TRoss TR	JWilliams:as	C Marco	FHebdon
DATE	4/15/93	4/22/93	4/22/93	4/23/93	5/5/93

DOCUMENT NAME:g:\bfn/ts329.amd

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AMENDMENT NO.213 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260

DATED: May 5, 1993

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 213
License No. DPR-52

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated December 23, 1992 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 5, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 213

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

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

REMOVE	INSERT
3.2/4.2-11a	3.2/4.2-11a
3.2/4.2-42	3.2/4.2-42**
3.2/4.2-43	3.2/4.2-43
3.7/4.7-35	3.7/4.7-35

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel Reactor Water Cleanup System Main Steam Valve Vault (TIS-069-834A-D)	$\leq$ 201.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor
2	Instrument Channel Reactor Water Cleanup System Pipe Trench (TIS-069-835A-D)	$\leq$ 135.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor
2	Instrument Channel Reactor Water Cleanup System Pump Room 2A (TIS-069-836A-D)	$\leq$ 152.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor
2	Instrument Channel Reactor Water Cleanup System Pump Room 2B (TIS-069-837A-D)	$\leq$ 152.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor
2	Instrument Channel Reactor Water Cleanup System Heat Exchanger Room (TIS-069-838A-D)	$\leq$ 143.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor
2	Instrument Channel Reactor Water Cleanup System Heat Exchanger Room (TIS-069-839A-D)	$\leq$ 170.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor

TABLE 4.2.A (Cont'd)
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Group 1 (Initiating) Logic	Checked during channel functional test. No further test required.(11)	N/A	N/A
Group 1 (Actuation) Logic	Once/operating cycle (21)	N/A	N/A
Group 2 (Initiating) Logic	Checked during channel functional test. No further test required.	N/A	N/A
Group 2 (RHR Isolation-Actuation) Logic	Once/operating cycle (21)	N/A	N/A
Group 8 (Tip-Actuation) Logic	Once/operating cycle (21)	N/A	N/A
Group 2 (Drywell Sump Drains-Actuation) Logic	Once/operating cycle (21)	N/A	N/A
Group 2 (Reactor Building and Refueling floor, and Drywell Vent and Purge-Actuation) Logic	Once/operating cycle (21)	N/A	N/A
Group 3 (Initiating) Logic	Checked during channel functional test. No further test required.	N/A	N/A
Group 3 (Actuation) Logic	Once/operating cycle (21)	N/A	N/A
Group 6 Logic	Once/operating cycle (18)	N/A	N/A
Group 8 (Initiating) Logic	Checked during channel functional test. No further test required.	N/A	N/A
Reactor Building Isolation (refueling floor) Logic	Once/6 months (18)	(6)	N/A
Reactor Building Isolation (reactor zone) Logic	Once/6 months (18)	(6)	N/A

TABLE 4.2.A (Cont'd)
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
SGTS Train A Logic	Once/6 months (19)	N/A	N/A
SGTS Train B Logic	Once/6 months (19)	N/A	N/A
SGTS Train C Logic	Once/6 months (19)	N/A	N/A
Instrument Channel – Reactor Water Cleanup System Main Steam Valve Vault (TIS-069-834A-D)	(1)(27)	4 months	N/A
Instrument Channel – Reactor Water Cleanup System Pipe Trench (TIS-069-835A-D)	(1)(27)	4 months	N/A
Instrument Channel – Reactor Water Cleanup System Pump Room 2A (TIS-069-836A-D)	(1)(27)	4 months	N/A
Instrument Channel Reactor Water Cleanup System Pump Room 2B (TIS-069-837A-D)	(1)(27)	4 months	N/A
Instrument Channel Reactor Water Cleanup System Heat Exchanger Room (TIS-069-838A-D)	(1)(27)	4 months	N/A
Instrument Channel – Reactor Water Cleanup System Heat Exchanger Room (TIS-069-839A-D)	(1)(27)	4 months	N/A

in the system, isolation is provided by high temperature in the cleanup system area. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain OPERABLE and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - (Deleted)

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 213 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 December 23, 1992, the Tennessee Valley Authority (the licensee) proposed changes to the technical specifications for the Browns Ferry Nuclear Plant (BFN), Unit 2. The proposed changes add requirements for new temperature switches to be installed in the Reactor Water Cleanup (RWCU) System heat exchanger room. The switches are intended to detect a postulated RWCU pipe break, and provide a signal to isolate that system to prevent excessive discharge of high energy fluid from the reactor coolant system and ensure that vital plant equipment does not experience an environment which could compromise its function.

2.0 EVALUATION

The NRC staff reviewed two aspects of the proposed change. First, the staff considered the adequacy of the instrumentation design to fulfill its design function. The staff also evaluated the high energy line break (HELB) analysis results to ensure that an appropriate instrument setpoint was proposed. The two evaluations are presented below.

2.1 Adequacy of Instrumentation Design

The safety function of the RWCU heat exchanger room temperature switches is to automatically isolate the system from the reactor coolant pressure boundary following detection of high temperature in the areas surrounding the RWCU system piping and components. This isolation prevents excessive discharge of high energy fluid which could create a severe operating environment for safety equipment. The reliability of this instrumentation is assured by providing sufficient redundancy and independence such that no single failure results in the loss of its protective function. The instrumentation is designed to permit periodic testing when the reactor is in operation. The licensee indicates that these new temperature detection loops consist of environmentally qualified resistance temperature detectors (RTDs) and Class 1E qualified analog trip units (ATUs). This instrumentation will perform the same function as the existing temperature switches around the other areas of RWCU piping and will be included in the Technical Specifications. The four switches will be configured similarly to the presently installed switches in the RWCU leak detection system instrumentation. Two instruments are assigned to both Division I and Division II to ensure physical and electrical independence, and are arranged in a one-out-of-two taken twice logic. This arrangement will ensure the safety function of the RWCU system temperature instrumentation in case of a single failure of an RTD, an ATU, or the power supply, and will minimize spurious isolation.

The instrument trip level setting is based upon the licensee's HELB analysis discussed below. This setpoint is selected high enough to prevent spurious trips, and includes provision for instrument drift and other inaccuracies. The applicable technical specification required Limiting Condition for Operation and Surveillance Requirements are exactly the same as those for the existing temperature instrumentation in the RWCU leak detection system. The staff finds that the proposed instrumentation meets applicable design requirements, and that the proposed technical specifications provide appropriate testing and operational limits.

2.2 Adequacy of Instrumentation Setpoint

In its HELB analysis, the licensee postulated a double-ended RWCU system line break at the exit of containment isolation valve (FCV-69-2) in the heat exchanger room. Fluid mass and energy discharged from the broken piping was calculated by the staff-approved RELAP5 computer code. The MONSTER computer code, developed by the licensee and based on the staff-approved CONTEMPT4 and COMPARE computer codes, used the mass and energy calculated by RELAP5 to predict the temperature response in the RWCU heat exchanger room and other related compartments. The licensee uses the calculated temperatures as part of the required performance envelope for environmental qualification of safety equipment.

The licensee calculated a peak temperature (214°F) in the RWCU heat exchanger room four seconds following the initiation of the RWCU pipe break. The licensee selected 170.0°F as the setpoint for the new temperature detection loops. The licensee indicated that this setpoint ($\leq$ 170°F) is high enough to ensure that spurious trips would be prevented during normal operation, and low enough to provide timely detection of the break and isolation of the RWCU system lines to and from the reactor. Thus, excessive loss of reactor coolant resulting from a RWCU system line break in the reactor building will be prevented.

The staff has not approved or accepted the MONSTER computer code as a tool for compartment pressure/temperature response analysis. Rather than reviewing and approving the MONSTER computer code for the purpose of accepting the proposed technical specification change, the staff performed an independent calculation to verify the compartment temperature responses predicted by the licensee for a postulated HELB in the RWCU heat exchanger room. The staff did not perform an independent calculation of the mass and energy release, and did not review details of the RELAP5 model used in the licensee's analysis.

Data required for the staff's calculation was provided at a meeting on April 1, 1993. At this meeting, the licensee also provided bounding temperature profiles used for equipment qualification. The staff's calculated compartment temperatures are comparable to those calculated by the licensee. Predicted temperatures are within the qualification envelopes provided by the licensee. Therefore, the proposed temperature switch setpoint of 170°F provides timely isolation of a postulated RWCU pipe break. The staff accepts the proposed setpoint.

3.0 SUMMARY

The staff has found that the proposed instrument design meets appropriate requirements, and that the proposed technical specifications, including the temperature switch setpoint, provide adequate assurance that safety equipment will not be exposed to an excessively severe environment in the event of a RWCU pipe break. Therefore, the proposed changes are acceptable.

4.0 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.

5.0 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 and Bases. 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 (58 FR 16232). 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.

6.0 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: D. Shum and I. Ahmed

Date: May 5, 1993