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

July 2, 1992

Docket No. 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: BROWNS FERRY NUCLEAR PLANT UNIT 2 - ISSUANCE OF AMENDMENT (TAC NO. M83351) (TS 317T)

The Commission has issued the enclosed Amendment No.202 to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant (BFN) Unit 2. This amendment responds to the application by the Tennessee Valley Authority (TVA) dated May 13, 1992.

The amendment revises Technical Specifications Table 3.2.C and Technical Specification 3.5.K/4.5.K to allow continued operation when the Rod Block Monitor (RBM) is inoperable provided that thermal margin defined by the minimum critical power ratio (MCPR) is within specified limits. Technical Specification Bases 3.2 is also changed, summarizing the basis for the revised specifications.

These changes were requested because BFN Unit 2 has experienced an unusually large number of failures of Local Power Range Monitor (LPRM) instruments which threatens the continued operability of the RBM. The RBM operability issue will be resolved when LPRM problems are addressed during the upcoming BFN Unit 2 refueling outage. Therefore, the amendment is a temporary change, and expires at the end of the current fuel cycle.

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A copy of the staff's safety evaluation of this amendment is provided as Enclosure 2. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Joseph F. Williams, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.202 to
   License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
See next page

citation error on clipped page corrected 7/2/92

OFC	PDII-4/LA	PDII-4/PM	PDII-4/PM	OGC	PDII-4/D
NAME	MSanders	JWilliams:as	TRoss		FHebdon
DATE	6/25/92	6/25/92	6/26/92	7/1/92	6/25/92

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Distribution

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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. 202
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 May 13, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 202 , 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. This amendment is temporary and expires at the end of the current fuel cycle (Cycle 6).

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: July 2, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 202

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.

REMOVE	INSERT
3.2/4.2-26	3.2/4.2-26
3.2/4.2-27	3.2/4.2-27
--	3.2/4.2-27a*
--	3.2/4.2-27b*
3.2/4.2-67	3.2/4.2-67**
3.2.4.2-68	3.2/4.2-68
3.5/4.5-18	3.5/4.5-18**
3.5/4.5-19	3.5/4.5-19

*Spillover pages
**Overleaf pages

NOTES FOR TABLE 3.2.C

1. The minimum number of OPERABLE channels for each trip function is detailed for the STARTUP and RUN positions of the reactor mode selector switch. The SRM, IRM, and APRM (STARTUP mode), blocks need not be OPERABLE in "RUN" mode, and the APRM (flow biased) rod blocks need not be OPERABLE in "STARTUP" mode.

   With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt).

3. IRM downscale is bypassed when it is on its lowest range.

4. SRMs A and C downscale functions are bypassed when IRMs A, C, E, and G are above range 2. SRMs B and D downscale function is bypassed when IRMs B, D, F, and H are above range 2.

   SRM detector not in startup position is bypassed when the count rate is $\geq$100 CPS or the above condition is satisfied.

5. During repair or calibration of equipment, not more than one SRM or RBM channel nor more than two APRM or IRM channels may be bypassed. Bypassed channels are not counted as OPERABLE channels to meet the minimum OPERABLE channel requirements. Refer to section 3.10.B for SRM requirements during core alterations.

6. IRM channels A, E, C, G all in range 8 or above bypasses SRM channels A and C functions.

   IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions.

*7. The following operational restraints apply to the RBM only.

   a. Both RBM channels are bypassed when reactor power is $\leq$30 percent or when a peripheral (edge) control rod is selected.

   b. The RBM need not be OPERABLE in the "startup" position of the reactor mode selector switch.

   c. Two RBM channels are provided and only one of these may be bypassed with the console selector. The other channel may also be defeated only if the conditions of "e" or "f" are met. If the inoperable channel cannot be restored within 24 hours, and the conditions of "e" or "f" are not met, the inoperable channel shall be placed in the tripped condition within one hour.

*The provisions of Note 7e and 7f are applicable during unit 2 cycle 6 only.

7. (Continued)

   d. With both RBM channels inoperable, and the conditions of "e" or "f" not met, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

   *e. The RBM need not be OPERABLE when reactor power is ≥90 percent and MCPR is ≥1.40.

   *f. The RBM need not be OPERABLE when reactor power is <90 percent and MCPR is ≥1.70.

8. This function is bypassed when the mode switch is placed in RUN.

9. This function is only active when the mode switch is in RUN. This function is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

10. The inoperative trips are produced by the following functions:

    a. SRM and IRM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Power supply voltage low.

       (3) Circuit boards not in circuit.

    b. APRM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Less than 14 LPRM inputs.

       (3) Circuit boards not in circuit.

    c. RBM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Circuit boards not in circuit.

       (3) RBM fails to null.

       (4) Less than required number of LPRM inputs for rod selected.

11. Detector traverse is adjusted to 114 ± 2 inches, placing the detector lower position 24 inches below the lower core plate.

*The provisions of Note 7e and 7f are applicable during unit 2 cycle 6 only.

NOTES FOR TABLE 3.2.C (Cont'd)

12. This function may be bypassed in the SHUTDOWN or REFUEL mode. If this function is inoperable at a time when OPERABILITY is required the channel shall be tripped or administrative controls shall be immediately imposed to prevent control rod withdrawal.

13. RBM upscale flow-biased setpoint clipped at 106 percent rated reactor power.

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flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established nominal setting of three times normal background and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm with a nominal setpoint of 1.5 x normal full-power background is provided also.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system. Each trip system consists of two elements. Each channel contains one temperature switch located in the pump room and three temperature switches located in the torus area. The RCIC high flow and high area temperature sensing instrument channels are arranged in the same manner as the HPCI system.

The HPCI high steam flow trip setting of 90 psid and the RCIC high steam flow trip setting of 450" H_2O have been selected such that the trip setting is high enough to prevent spurious tripping during pump startup but low enough to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

The HPCI and RCIC steam line space temperature switch trip settings are high enough to prevent spurious isolation due to normal temperature excursions in the vicinity of the steam supply piping. Additionally, these trip settings ensure that the primary containment isolation steam supply valves isolate a break within an acceptable time period to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

High temperature at the Reactor Water Cleanup (RWCU) System in the main steam valve vault, RWCU pump room 2A, RWCU pump room 2B, RWCU heat exchanger room or in the space near the pipe trench containing RWCU piping could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

A General Electric study, GE-NE-770-06-0392 shows for the unit 2 cycle 6 core that if the initial MCPR is as specified in item 7e or 7f of Table 3.2.C, then no single rod withdrawal error can cause the MCPR to decrease below the MCPR safety limit. When core operating conditions have been verified to be within the limits of items 7e or 7f of Table 3.2.C, the RBM is not required. When the RBM is required, the minimum instrument channel requirements apply. These requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.I Average Planar Linear Heat Generation Rate	4.5.I Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
During steady-state power operation, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1, 2, 3, and 4. If at any time during operation it is determined by normal surveillance that the limiting value for MAPLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.
J. Linear Heat Generation Rate (LHGR)	J. Linear Heat Generation Rate (LHGR)
During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 kW/ft. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	The LHGR shall be checked daily during reactor fuel operation at ≥ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

3.5.K Minimum Critical Power Ratio (MCPR)

Except when the provisions of Note 7 of Table 3.2.C are being employed due to the inoperability of the Rod Block Monitor, the minimum critical power ratio (MCPR) as a function of scram time and core flow, shall be equal to or greater than shown in Figure 3.5.K-1 multiplied by the K_f shown in Figure 3.5.2, where:

$\tau = 0$ or $\dfrac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}$, whichever is greater

$\tau_A = 0.90$ sec (Specification 3.3.C.1 scram time limit to 20% insertion from fully withdrawn)

$$\tau_B = 0.710+1.65 \left[\frac{N}{n}\right]^{\frac{1}{2}} (0.053) \text{ [Ref.2]}$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} \tau_i}{n}$$

n = number of surveillance rod tests performed to date in cycle (including BOC test).

τ_i = Scram time to 20% insertion from fully withdrawn of the i^{th} rod.

N = total number of active rods measured in Specification 4.3.C.1 at BOC.

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.

2. Except as provided by Note 7 of Table 3.2.C, the MCPR limit shall be determined for each fuel type 8X8, 8X8R, P8X8R, from Figure 3.5.K-1, respectively, using:

   a. τ = 0.0 prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1.

   b. τ as defined in Specification 3.5.K following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

   The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

UNITED STATES
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WASHINGTON, D.C. 20555

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 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 May 13, 1992, the Tennessee Valley Authority (the licensee) submitted a request to change the Browns Ferry Nuclear Plant (BFN) Unit 2 Technical Specifications. The proposed changes revise Technical Specification (TS) Table 3.2.C, TS 3.5.K, and TS 4.5.K.2, modifying the operability requirements for the Rod Block Monitor (RBM) system if sufficient thermal margin, as measured by the Minimum Critical Power Ratio (MCPR), can be maintained. These changes were requested because BFN Unit 2 has experienced an unusually large number of failures of Local Power Range Monitor (LPRM) instruments which threaten the continued operability of the RBM. The RBM operability issue will be resolved when LPRM problems are addressed during the upcoming BFN Unit 2 refueling outage. Therefore, the amendment is a temporary change, and expires at the end of the current fuel cycle (Cycle 6).

To support its request, TVA submitted the proposed TS changes, a description and evaluation of the physical and analytical changes, and a General Electric Company (GE) proprietary report (GE-NE-770-06-0392) on the operability requirements for a revised RBM operational analyses for BFN Unit 2 Cycle 6. The changes requested by TVA are similar to those requested by other utilities, such as Hatch 1 and 2, Monticello, and Fermi 2.

2.0 EVALUATION

2.1 Rod Block Monitor System

The Rod Block Monitor System is used to prevent violation of fuel thermal-hydraulic limits in the event of inadvertent continuous withdrawal of a control rod. When a control rod is selected for withdrawal, the surrounding Local Power Range Monitor (LPRM) strings are selected. The RBM system monitors their response to the withdrawal, and will block the withdrawal if that response exceeds certain limits.

In the submittal for BFN Unit 2, TVA proposed changes to the RBM system that are generally identical to those changes previously reviewed and approved for other facilities, including changes to the instrumentation system and the new approaches, analyses and setpoints.

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2.2 MCPR Requirements

The function of the RBM system is to assist the operator in safe plant operation in the power range by:

a. initiating a rod block to prevent violation of the fuel integrity safety criteria during withdrawal of a single control rod, and

b. provide a signal to permit operator evaluation of the change in the local relative power during control rod movement.

The probability of an administrative error in selection of rods for withdrawal is not increased, because the proposed amendment only changes the range of the allowable values for the initial MCPR during rod withdrawal. The proposed amendment does not revise the administrative limitations on the selection and withdrawal of rods. GE's analyses showed that, even if the RBM did not intervene, the MCPR would not decrease below the allowable safety limit if the operator withdrew one or more rods erroneously while operating the core within the MCPR limits proposed by this amendment. GE also showed that the proposed initial MCPR requirements during rod withdrawal are bounded by previously analyzed limits and thus do not violate the thermal margin requirements based on other analyzed transients.

The data base as described in the GE report was used to determine the MCPR operating limits with the condition that no rod withdrawal error could lead to exceeding MCPR safety limits. The results of the analysis demonstrated that additional limits on thermal mechanical margin were not required. The analysis also showed that the following limiting MCPR values would provide the required margin for full withdrawal of any control rod:

a. MCPR greater than or equal to 1.40 with the reactor power greater than or equal to 90 percent of rated thermal power, and

b. MCPR greater than or equal to 1.70 with the reactor power less than 90 percent of rated thermal power.

Thus, when the operating MCPR is within these limits, the RBM is allowed to be bypassed completely because it is not required to be operable. When the operating MCPR is below these values, the plant is operating with a "limiting control rod pattern," and the RBM system must be operable.

We have reviewed these changes and analyses for the RBM and have concluded that the analyses, methods used, criteria, and setpoints are acceptable.

2.3 Technical Specification Changes

Implementing the hardware changes and revised analyses described above requires changes in the BFN Unit 2 Technical Specifications. These changes allow the RBM to be inoperable when the MCPR is within specified limits, and will expire at the end of the current fuel cycle. These changes are discussed below.

Limiting Condition for Operation Table 3.2.C

Notes 7.e. and 7.f. are added to provide the thermal margin limits that permit the RBM to be inoperable. These notes read as follows:

> 7.e. "The RBM need not be OPERABLE when reactor power is greater or equal to 90 percent and the MCPR is greater than or equal to 1.40."
>
> 7.f. "The RBM need not be OPERABLE when reactor power is less than 90 percent and the MCPR is greater than or equal to 1.70."

Notes 7.c. and 7.d. are revised to include references to the new notes 7.e. and 7.f.

Note 7.a. is revised by adding the word "edge" to clarify the meaning of "peripheral control rod." This clarification is acceptable.

These changes to Table 3.2.C and the associated bases are based on the GE study, GE-NE-770-06-0392, for BFN Unit 2 Cycle 6. The study shows that if the initial MCPR is as specified in item 7.e. and 7.f. of Table 3.2.C, then no single rod withdrawal error can cause the MCPR to decrease below the MCPR safety limit. Also, when the core operating conditions have been verified to be within the limits of items 7.e. and 7.f. of Table 3.2.C, the RBM is not required. When the RBM is required, the minimum instrument channel requirements apply. These changes also include requirements for sufficient instrumentation to ensure that the single failure criteria are met.

Limiting Condition for Operation 3.5.K and Surveillance Requirement 4.5.k.2

The change to section 3.5.K stipulates that except when the provisions of note 7 to Table 3.2.C are being employed due to the inoperability of the Rod Block Monitor, the minimum critical power ratio (MCPR) as a function of scram time and core flow, shall be equal to or greater than that shown in TS Figure 3.5.k-1 multiplied by the K_f shown in TS Figure 3.5.2.

The change to section 4.5.K.2 stipulates that except as provided by note 7 of Table 3.2.C, the MCPR safety limit shall be determined for each fuel type.

The changes to the TS allow control rod withdrawal operations appropriate for proper core management at times when thermal margin is sufficient to obviate the need for the RBM. The staff reviewed the analyses provided by TVA and found that the proposed changes are safe, because when no RBM channels are operable, control rods can be withdrawn only during those conditions in which the MCPR is high enough that the RBM need not intervene.

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 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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 21833). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51.22(c)(9). Pursuant to 10 CFR Part 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The NRC staff has reviewed the reports submitted by TVA for the continued operation of BFN Unit 2 Cycle 6 and concludes that the appropriate material was submitted for Technical Specification changes pertaining to the inoperability of the RBM system. GE's study has shown that these new thermal limits can be met. Therefore, we conclude that the requested TS changes satisfy the staff's positions and requirements in these areas.

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 Contributor: A. Attard

Date: July 2, 1992