March 14, 1983

Docket No. 50-259

Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

The Commission has issued the enclosed Amendment No. 88 to Facility License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. This amendment changes the Technical Specifications in response to your request of February 1, 1983 (TVA BFNP TS 184).

The amendment changes the Technical Specifications to allow operation of Browns Ferry Unit 1 with increased core flow during the remainder of Cycle 5.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Richard J. Clark, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 88 to DPR-33
2. Safety Evaluation'
3. Notice

cc w/enclosures
See next page

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SURNAME	SNorris	RClark:bob	DVassallo	GLainas	J.R. GRAY		
DATE	3/10/83	3/09/83	3/10/83	3/10/83	3/11/83		

Mr. Hugh G. Parris

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
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 February 1, 1983, 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 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. 88, 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 the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Mooney for

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 14, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages:

   73
   75
   172b
   173

2. The marginal lines on these pages indicate the revised area.

Amendment No. 44, 47, 83 88

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum No. Operable Per Trip Sys (5)	Function	Trip Level Setting
2(1)	APRM Upscale (Flow Bias)	≤0.66W+42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	≤12%
2(1)	APRM Downscale (9)	≥3%
2(1)	APRM Inoperative	(10b)
1(7)	RBM Upscale (Flow Bias)	≤ 0.66W+40% (2) (13)
1(7)	RBM Downscale (9)	≥ 3%
1(7)	RBM Inoperative	(10c)
3(1)	IRM Upscale (8)	≤ 108/125 of full scale
3(1)	IRM Downscale (3) (8)	≥ 5/125 of full scale
3(1)	IRM Detector not in Startup Position (8)	(11)
3(1)	IRM Inoperative (8)	(10a)
2(1) (6)	SRM Upscale (8)	≤ 1 x 10^5 counts/sec.
2(1) (6)	SRM Downscale (4) (6)	≥ 3 counts/sec.
2(1) (6)	SRM Detector not in Startup Position (4) (8)	(11)
2(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	≤ 10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤ 115% recirculation flow
1(1)	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-85-61A and PS-85-61B)	147 psig turbine first-stage pressure
1(12)	Scram Discharge Tank Water Level High	≤25 gal.

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 $\pm$ 2 inches, placing the detector lower position 24 inches below the lower core plate.

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% rated reactor power.

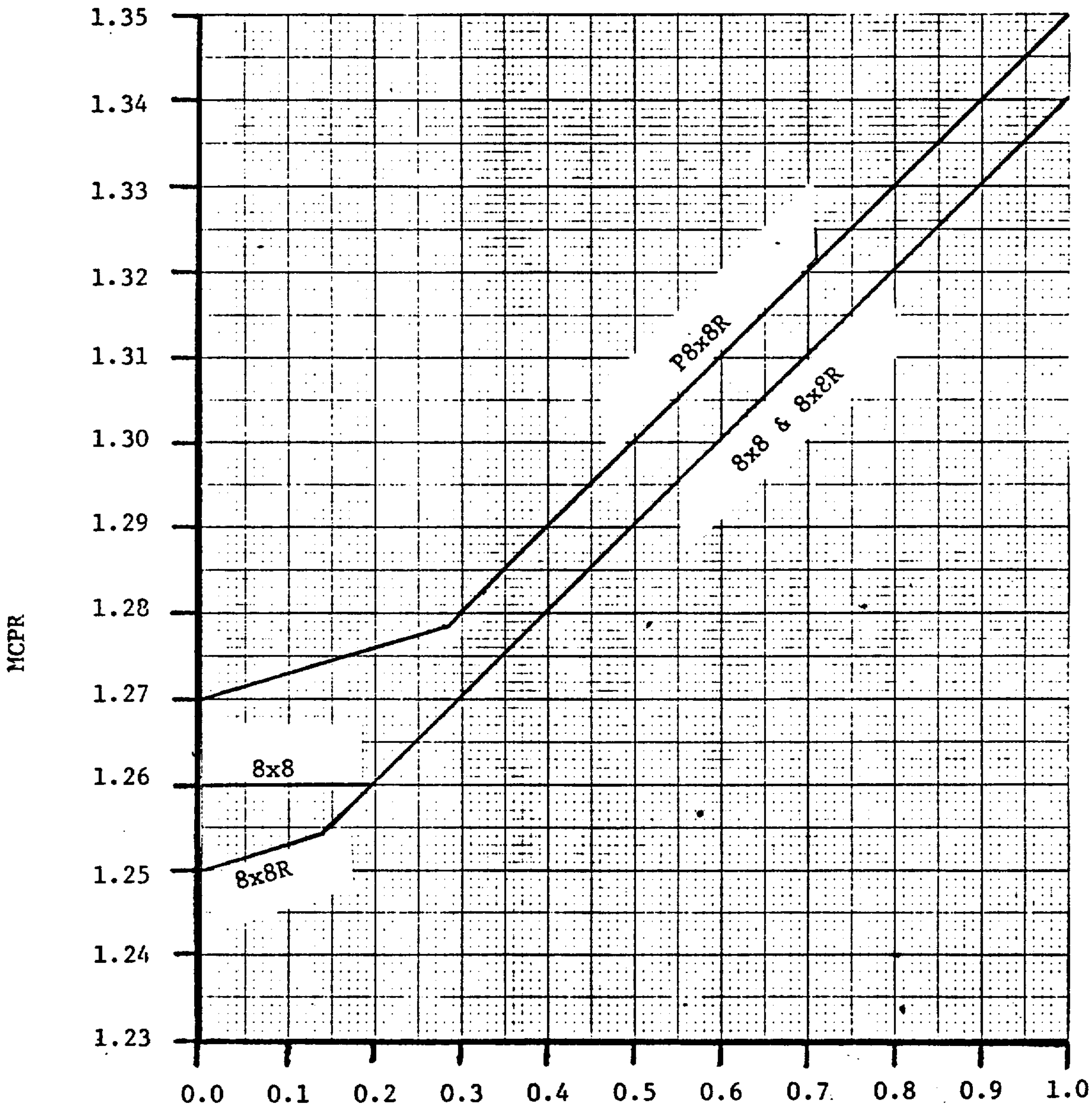


Figure 3.5.K-1

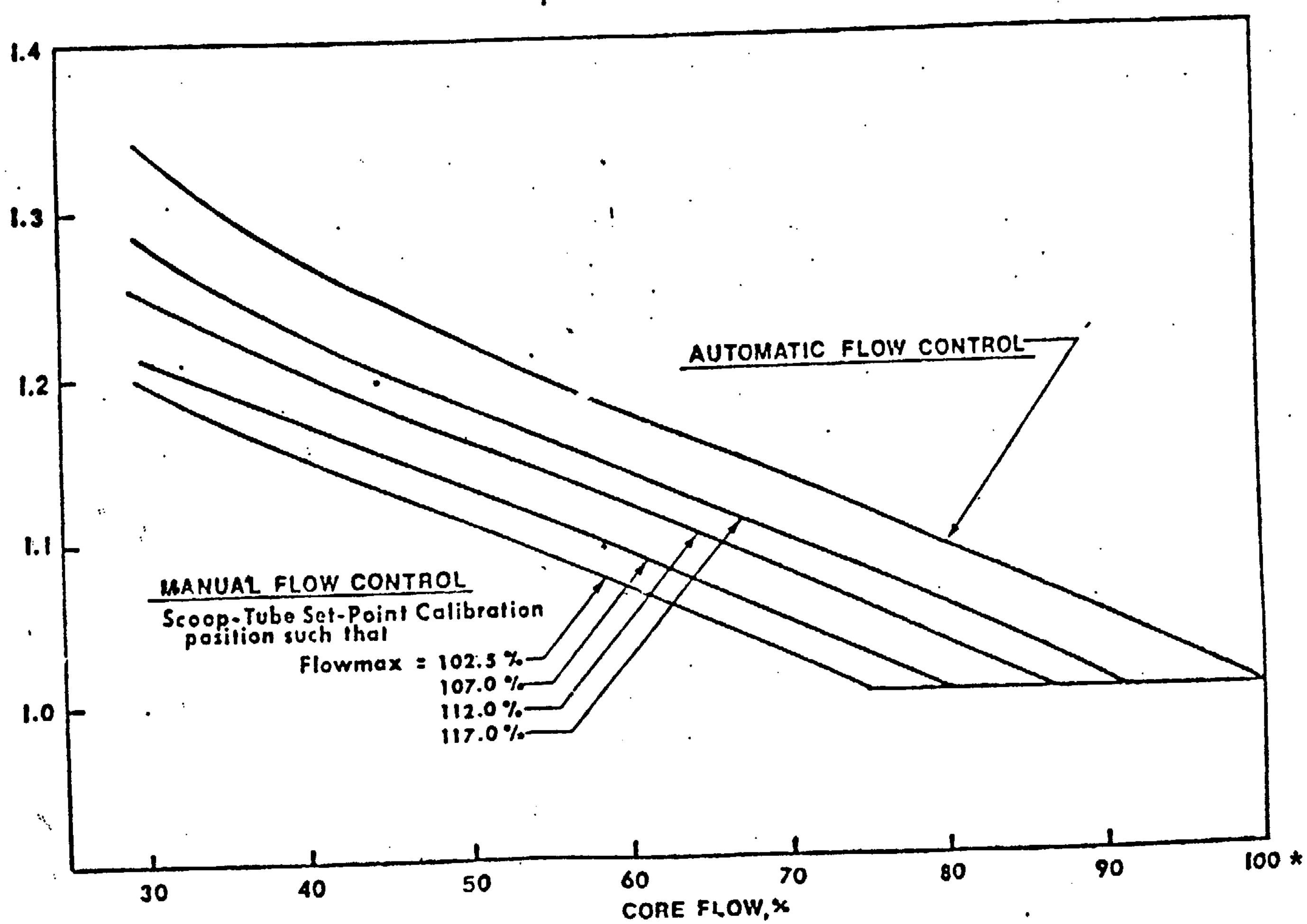
MCPR LIMITS*

*NOTE: Lead test assemblies are categorized as P8x8R bundles.

BROWNS FERRY NUCLEAR PLANT

FIGURE 3.5.2

K_f FACTOR



*K_f = 1.0 for core flow ≥ 100%.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-259

1.0 Introduction

By letter dated February 1, 1983, (TVA BFNP TS 184), the Tennessee Valley Authority (the licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. The proposed Technical Specifications would allow operation of Browns Ferry Unit 1 (BF-1) with increased core flow during the remainder of Cycle 5. In support of this application, the licensee submitted a safety evaluation performed by the General Electric Company (GE), (NEDO-22135, "Safety Review of Browns Ferry Nuclear Plant Unit No. 1 at Core Flow Conditions Above Rated Flow During Cycle 5").

2.0 Disucssion

BF-1 is presently in a coastdown mode of operation and is scheduled to be shutdown about mid April 1983 for a four to five-month refueling and maintenance outage. At present, the maximum attainable power is 992 MWe, about 93% of the normal electrical output, operating at 100% of rated flow.

The proposed changes to the Technical Specifications are to permit BF-1 to operate with core flows up to 105% of rated flow for the rest of the fuel cycle. The increased core flow would permit the unit to generate about 3% more power than would otherwise be attainable during the current coast-down mode of operation. This amendment does not authorize BF-1 to exceed the thermal power limit authorized by License No. DPR-33.

BF-1 is operating in a coastdown mode because of the delayed restart of Browns Ferry Unit 2 (BF-2). BF-2 shutdown on July 30, 1982 for refueling and major modifications (e.g., the Mark I torus modifications). BF-2 was originally scheduled to return to service by mid January 1983 and BF-1 was scheduled to shutdown about March 1, 1983. The projected startup date for BF-2 is now about mid March 1983. To avoid having two units down at the same time, the shutdown date for BF-1 has been postponed until mid-April 1983.

3.0 Evaluation

3.1 Thermal and Hydraulic Design

The objective of the review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, and is not susceptible to thermal-hydraulic instability.

The review includes the following areas: (1) safety limit minimum critical power ratio (MCPR), (2) operating limit MCPR, (3) thermal-hydraulic stability, and (4) changes to Figures 3.5.K-1 and 3.5.2 of the Technical Specifications.

The licensee has submitted the analysis report for Cycle 5 operation at core flow conditions above rated flow (Ref. 2). This report relies on a generic document (Ref. 3), which has been reviewed and approved (Ref. 4) by the staff. We conclude that additional staff review of this portion of Reference 2 concerning the standard thermal-hydraulic design is not required for Cycle 5 operation at core flow conditions above rated flow since it has been previously reviewed and found acceptable. Discussion of the review concerning the thermal-hydraulic design for Cycle 5 operation follows:

3.1.1 Safety Limit MCPR

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8x8 fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The 1.07 SLMCPR is unchanged from the SLMCPR previously approved for Cycle 5. The basis for this safety limit is addressed in Reference 3.

3.1.2 Operating Limit MCPR

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). The ΔCPR values given in Table 2-1 of Reference 2 are plant specific values calculated by using the ODYN methods. The calculated ΔCPRs are adjusted to reflect either Option A or Option B ΔCPRs by employing the conversion method described in Reference 7. The MCPR values are determined by adding the adjusted ΔCPRs to the safety limit MCPR. Table 6.1 of Reference 2 presents both the cycle MCPR values for the non-pressurization and pressurization events. The maximum cycle MCPR values (Options A and B) in Table 6.1 are specified as the operating limit MCPRs and incorporated into the Technical Specifications. Since the approved method was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated operation transients, we conclude that these limits are acceptable.

3.1.3 Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (Ref. 2) show that the maximum reactor core stability decay ratio in increased core flow operation during Cycle 5 is bounded by the Reload-4 licensing submittals which have been previously approved (Ref. 6). Therefore, we conclude that the thermal-hydraulic stability results are acceptable for increased core flow operation during Cycle 5.

3.1.4 Changes to Figures 3.5.K-1 and 3.5.2 of the Technical Specifications

Figure 3.5.K-1 of the Technical Specifications has been modified to include the operating limit MCPR for Cycle 5 extended flow operation. Using Option A, the operating limit MCPRs shall be 1.35 for P8X8R fuel, and 1.34 for 8X8 and 8X8R fuel types. Using Option B, the operating limit MCPR shall be 1.27, 1.26 and 1.25 for P8X8R, 8X8 and 8X8R fuel types respectively. Figure 3.5.2 has been changed to include a note to reflect that the K_f factor is equal to rated core flow.

3.1.5 Fuel Bundle Liftoff

GE re-evaluated the bundle liftoff margin for 105 percent core flow. The method used was described in a letter from R. Gridley (GE) to D. Eisenhut (NRC) dated July 11, 1977. The new analysis yielded a bundle liftoff margin of 132 lbs, which is 15 lbs less than the old analysis using 100 percent core flow. We conclude that this is a small variation and an adequate liftoff margin is maintained for the increased core flow during Cycle 5 operation.

3.2 Nuclear Design

The rod block monitor is programmed to block rod withdrawal when its output is 106 percent of full power. If the program were not changed, at 105 percent flow the block would occur at 109.3 percent of full power. This would result in a change in CPR of 0.31 for 8X8 fuel - an unacceptably high value. Accordingly the RBM upscale flow biased setpoint is clipped at 106 percent rated power. The change in CPR would then be 0.19 for this event for the 8X8 fuel. This is an acceptable procedure and result. Table 3.2.C of the Technical Specifications has been modified to show this change.

The rod drop accident is a low flow startup event that is not affected by the change in flow except for end-of-cycle operation where the initial conditions are slightly altered. However, end-of-cycle conditions are not limiting for this event and the previous analysis is still valid.

3.3 Summary of Evaluation

We find that approved thermal hydraulic methods have been used and the the results of analyses support the proposed MCPR limits, which avoid violation of the safety limit MCPR for design transients. We conclude that the changes approved by this amendment will not adversely affect the capability to operate BF-1 safely during Cycle 5 extended flow operation and that the proposed changes to Figures 3.5.K-1 and 3.5.2 of the Technical Specifications discussed above are acceptable.

Based on the discussion in Section 3 above we conclude that clipping the Rod Block Monitor at 106 percent of rated power will permit the plant to be operated within the limits shown on Figure 3.5.K.1. In summary we conclude that operating during the remainder of Cycle 5 with extended flow will not endanger the health and safety of the public.

4.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 14, 1983

Principal Contributors: W. Brooks, S. Sun, S. Wu

References:

1. Letter from L. Miller (TVA) and attachments to H. Denton (NRC) dated February 1, 1983.

2. NEDO-22135, "Safety Review of Browns Ferry Nuclear Plant Unit No. 1 at Core Flow Conditions above Rated Flow During Cycle 5," dated May 1982.

3. NEDO-24011-A-4, "General Electric Boiling Water Reactor Generic Reload Fuel Applications," January 1982.

4. Letter from D. G. Eisenhut (NRC) to R. Gridley (GE) dated May 12, 1978.

5. Y1003J01A19, Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 1, Reload No. 4 (Cycle 5) dated March 1981.

6. Letter from T. Ippolito (NRC) to H. Parris (TVA) dated September 15, 1981.

7. Letter from R. Buchholz (GE) to P. Check (NRC), Response to NRC Request for Information on ODYN Computer Model, September 5, 1980.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-259

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 88 to Facility Operating License No. DPR-33 issued to Tennessee Valley Authority (the licensee), which revised the Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 1 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment changes the Technical Specifications to permit operation of Browns Ferry, Unit 1 with increased core flow during the remainder of Cycle 5.

The application for this amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated February 1, 1983 (2) Amendment No. 88 to license No. DPR-33, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of March 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Vernon L Rooney

Vernon L. Rooney, Acting Chief
Operating Reactors Branch #2
Division of Licensing