March 6, 1984

Docket Nos. 50-260/296

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 Nos. 88 and 61 to Facility Operating License Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit 2 and 3. These amendments are in response to your application dated June 2, 1983 (TVA BFNP TS 188).

The amendments change the Technical Specifications to allow operation of Browns Ferry Units 2 and 3 with increased core flow during the remainder of Cycle 5 for each unit.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

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

Enclosures:
1. Amendment No. 88 to License No. DPR-52
2. Amendment No. 61 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	SNorris	JMTaylor	ACRS (10)
NRC PDR	RClark	TBarnhart (8)	OPA, CMiles
Local PDR	OELD	WJones	RDiggs
ORB#2 Reading	SECY	DBrinkman	NSIC
HDenton	LJHarmon	WHazelton	Gray File
DEisenhut	ELJordan	WKoo	Extra - 5

DL:ORB#2	DL:ORB#2	DL:ORB#2	OELD	DL:ADL-OR
SNorris:ajs	RClark	DVassallo	R. Tedesco	GLainas
02/17/84	02/17/84	02/21/84	02/28/84	02/ /84

Mr. Hugh G. Parris
Tennessee Valley Authority
Browns Ferry Nuclear Plant, Units 1, 2 and 3

cc:

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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. 88
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 June 2, 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 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. 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 6, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

73

75

172a

173

2. The marginal lines on these pages denote the area being changed.

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)	≤1×10^5 counts/sec.
2(1) (6)	SRM Downscale (4) (8)	≥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.

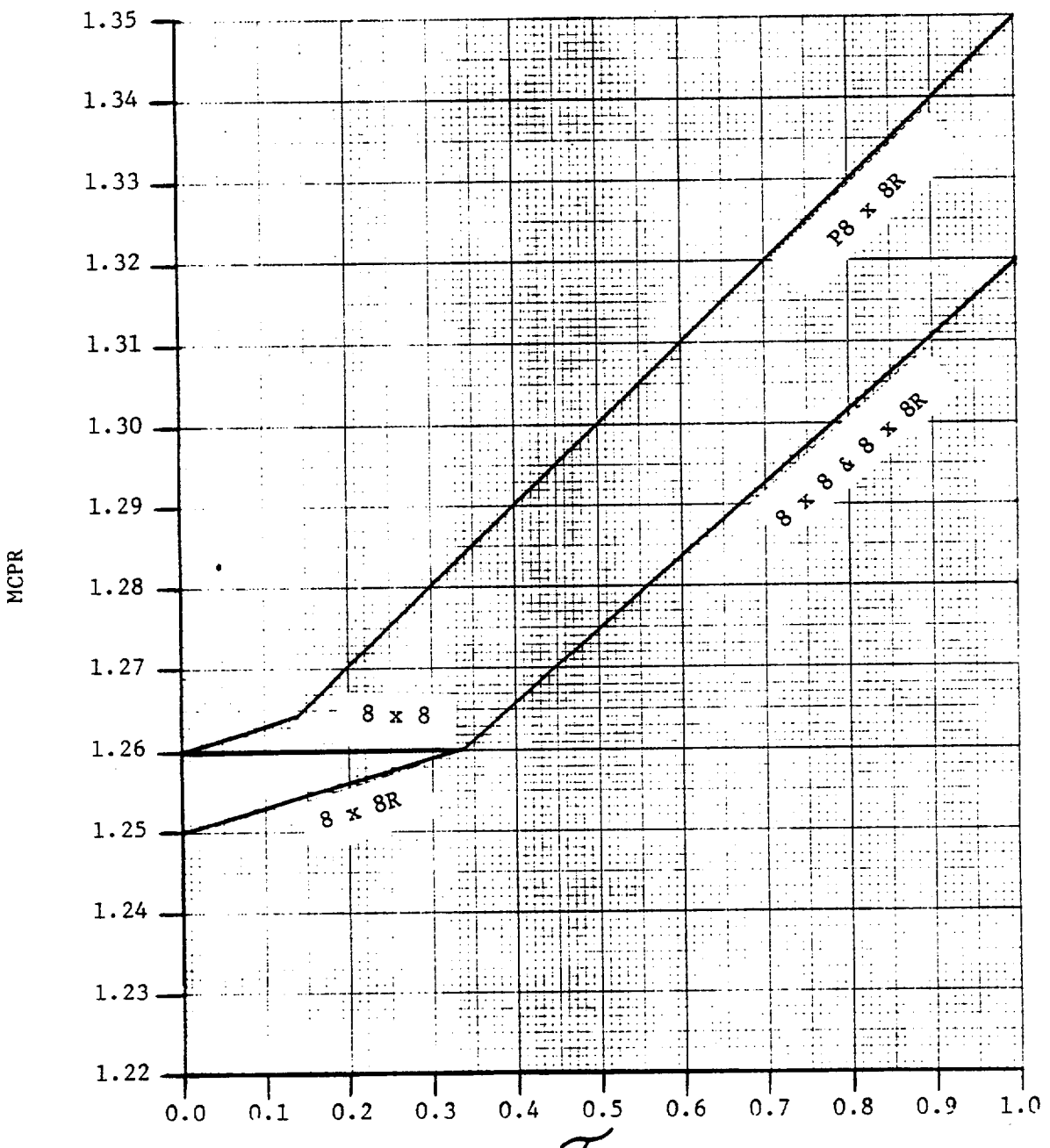


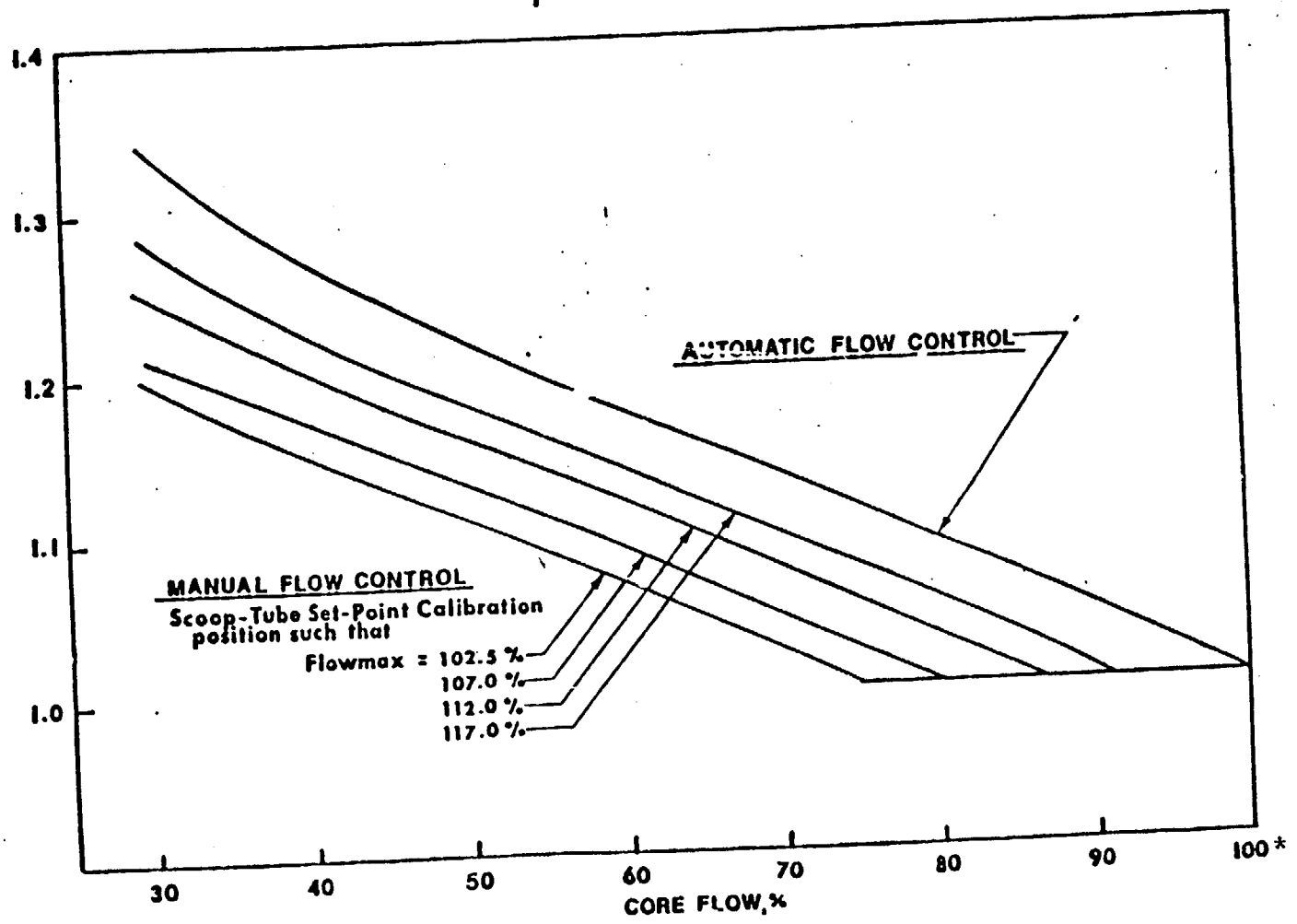
Figure 3.5.K-1

MCPR LIMITS

BROWNS FERRY NUCLEAR PLANT

FIGURE 3.5.2

K_f FACTOR



*K_f = 1.0 for core flow $\geq$100%.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 2, 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 Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 61, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 6, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

76

78

182b

183

2. The marginal lines on these pages denote the area being changed.

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+60% (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 10^6 counts/sec.
2(1) (6)	SRM Downscale (4) (8)	≥ 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 ± 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.

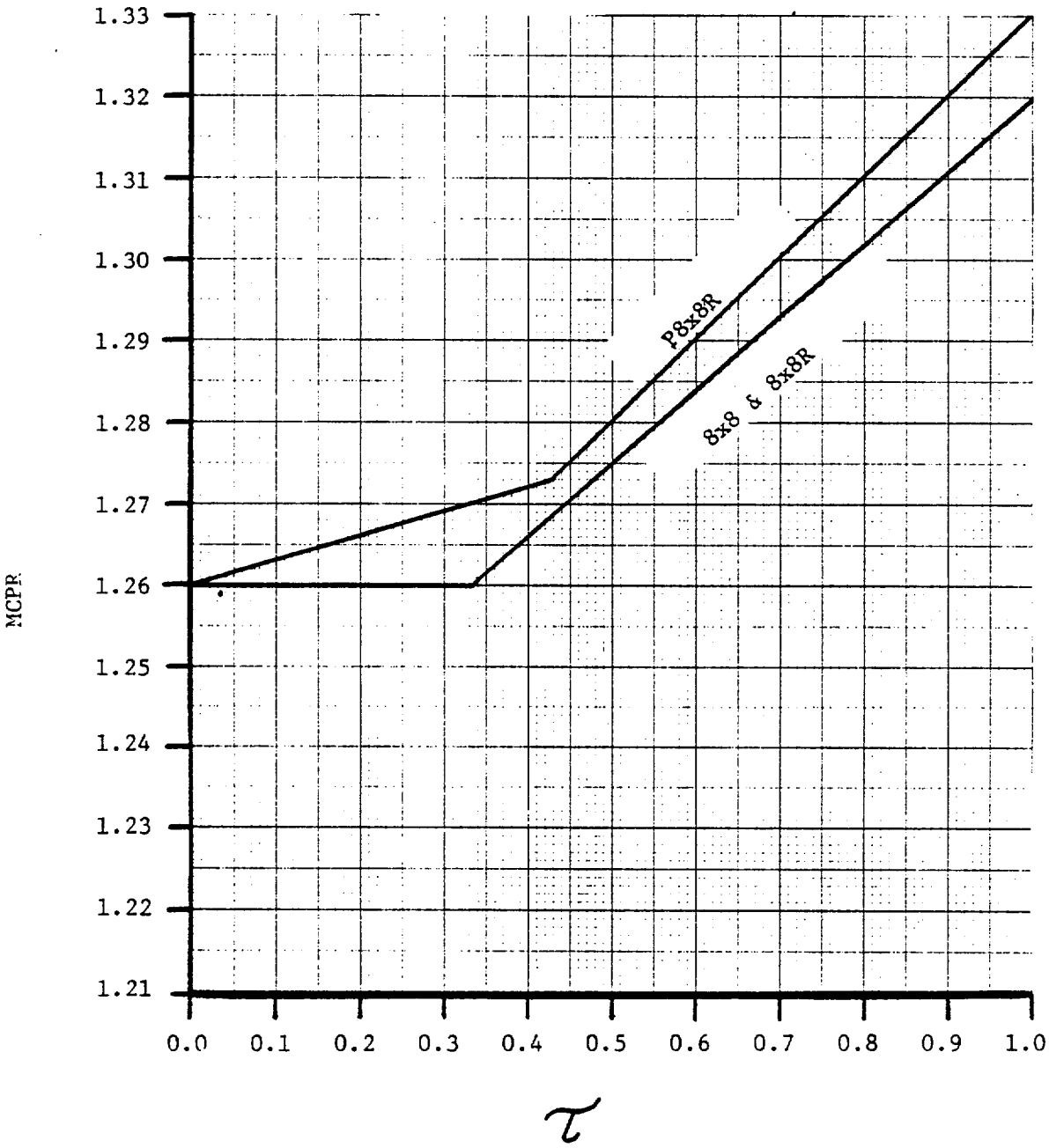


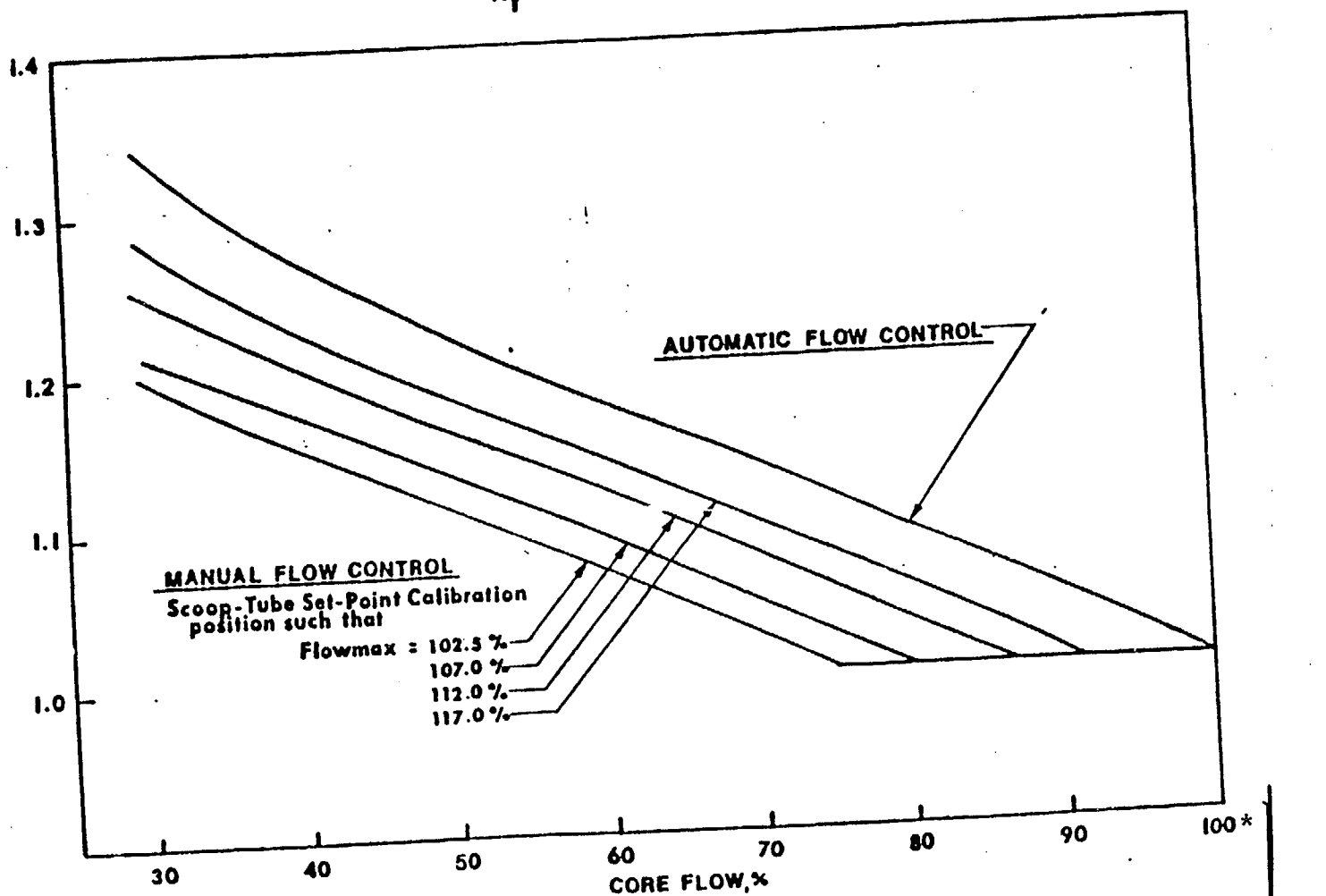
Figure 3.5.K-1
MCPR LIMITS*

*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 $\geq$ 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-52

AND AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated June 2, 1983 (TVA BFNP TS 188) (Reference 1), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License Nos. DPR-52 and DPR-68 for Browns Ferry Nuclear Plant, Units 2 and 3. The proposed amendments would revise the Technical Specifications to permit the licensee to increase the coolant flow through the reactor core during coastdown operations (reactor coastdown conditions occur at the end of a reactor cycle, prior to fuel reloading, at which time reactor power has to be reduced due to fuel burnout). Increasing the reactor coolant flow would reduce the amount of power reduction required of Browns Ferry Units 2 and 3 during the end-of-cycle coastdown operations. A similar amendment has been previously approved for Unit 1. (Amendment No. 88 to Facility Operating License No. DPR-33 issued March 14, 1983.)

In support of this application, the licensee submitted safety evaluations prepared by the General Electric Company (GE), NEDO-22245 and NEDO-22149, "Safety Review of Browns Ferry Nuclear Plant Units 2 and 3 at Core Flow Conditions Above Rated Flow During Cycle 5," (Reference 2).

The proposed changes to the Technical Specifications are to permit Browns Ferry Units 2 and 3 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 coastdown mode of operation. These amendments do not authorize Browns Ferry Units 2 and 3 to exceed the thermal power limit authorized by License Nos. DPR-52 and DPR-68.

2.0 EVALUATION

2.1 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 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 (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 analysis reports for each unit for Cycle 5 operation at core flow conditions above rated flow (Reference 2). The reports rely on a generic document (Reference 3), which has been reviewed and approved (Reference 4) by the staff. Discussion of the review concerning the thermal-hydraulic design for Cycle 5 operation at core flow conditions above rated flow follows:

Safety Limit MCPR

The safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in Reference 3, the safety limit MCPR is 1.07. The same safety limit MCPR of 1.07 is used for the Browns Ferry Units 2 and 3 Cycle 5 operation.

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 6. 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 in Reference 3 was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated transients, we conclude that these limits are acceptable.

Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (Reference 2) show that the maximum reactor core stability decay ratio while operating with increased core flow during Cycle 5 is bounded by the Reload-4 licensing submittal(s). These were approved for Browns Ferry Unit 2 by Amendment No. 85 to Facility Operating License No. DPR-52 issued March 11, 1983 and for Browns Ferry Unit 3 by Amendment No. 51 to Facility Operating License No. DPR-68 issued March 29, 1982.

Therefore, we conclude that the thermal-hydraulic stability results are acceptable for increased core flow operation during Cycle 5.

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

Figures 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, for Unit 2 the operating limit MCPRs shall be 1.35 for P8X8R fuel, and 1.32 for 8X8 and 8X8R fuel; for Unit 3 the operating limit MCPRs shall be 1.33 for P8X8R fuel, and 1.32 for 8X8 and 8X8R fuel types. Using Option B, for Unit 2 the operating limit MCPRs shall be 1.26, 1.26 and 1.25 for P8X8R, 8X8 8X8R fuel types, respectively; for Unit 3 the operating limit MCPRs shall be 1.26 for P8X8R, 8X8 and 8X8R fuel types. Figure 3.5.2 has been changed to include a note to reflect that the K_f factor is equal to 1.0 for core flows greater than or equal to rated core flow.

The staff has reviewed the Technical Specification changes requested by the licensee. We find that for the determination of the OLMCPR, credit is assumed for operation of the highwater level (L8) trip and turbine bypass system. In this regard, we have concluded that this subject should be treated as a generic issue, and we plan to handle it in accordance with our internal procedures for dealing with such issues. We have also determined, based on preliminary analysis, that the risk of operating without Technical Specifications concerning surveillance of the highwater level turbine trip or turbine bypass systems until the generic issue is resolved is small. Accordingly, we find that the results of analyses are consistent with the proposed OLMCPRs and safety limit MCPR and conclude that the proposed OLMCPRs are acceptable for operation during the remainder of Cycle 5.

Fuel Bundle Liftoff

GE reevaluated 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.

2.2 Nuclear Design

The rod block monitor is programmed to block rod withdrawal when its output is 106 percent of full power (0.66 W + 40). 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 Specification 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 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.

2.3 Summary of Evaluation

We find thermal-hydraulic methods have been used which have been approved generically by Reference 4 and that the results of analyses support the proposed limit MCPRs, which avoid violation of the safety limit MCPR for design transients. We, therefore, conclude that the core flow increase beyond the rated flow will not adversely affect the capability to operate Browns Ferry Nuclear Plant, Units 2 and 3 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 2.2 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. The proposed Technical Specifications (Table 3.2.C) have been changed to require this clipping. We find this acceptable.

In summary, we conclude that operation during the remainder of Cycle 5 for Units 2 and 3 with extended flow will not endanger the health and safety of the public.

3.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that these amendments do 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 these amendments involve 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 these amendments.

4.0 CONCLUSIONS

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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

Dated: March 6, 1984

REFERENCES

1. Letter from L. Mills (TVA) and attachments to H. Denton (NRC) dated June 2, 1983.

2. NEDO-22245 and NEDO-22149, "Safety Review of Browns Ferry Nuclear Plant Units 2 (dated October 1982) and 3 (dated June 1983) at Core Flow Conditions above Rated Flow During Cycle 5."

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. V1003J01A40 and Y1003J01A19, Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Units 2 (dated November 1981) and 3 (dated March 1981), Reload No. 4 (Cycle 5).

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