October 5, 1989

Docket No. 50-260

Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: TECHNICAL SPECIFICATION CHANGES INVOLVING THERMAL-HYDRAULIC STABILITY, SECTION 3.5/4.5-M (TAC 73435) (TS 272) - BROWNS FERRY NUCLEAR PLANT, UNIT 2

The Commission has issued the enclosed Amendment No. 174, to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. This amendment is in response to your application dated June 20, 1989.

This amendment adds Sections 3.5/4.5-M and incorporates Limiting Conditions for Operation and Surveillance Requirements addressing reactor core thermal-hydraulic stability issues identified in NRC Bulletin 88-07, Supplement 1.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 174 to License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
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AMENDMENT NO. 174 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
DATED: October 5, 1989

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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. 174
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 20, 1989, 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. 174, 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 5, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 174

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 pages* are provided to maintain document completeness.

REMOVE	INSERT
i*	i*
ii	ii
vii	vii*
viii	viii
3.5/4.5-20	3.5/4.5-20
-	3.5/4.5-20a
3.5/4.5-21	3.5/4.5-21*
3.5/4.5-21a	3.5/4.5-21a*
3.5/4.5-21b	-
3.5/4.5-22	3.5/4.5-22*
3.5/4.5-22a	3.5/4.5-22a
3.5/4.5-32	3.5/4.5-32
3.5/4.5-33	3.5/4.5-33*
3.6/4.6-11	3.6/4.6-11*
3.6/4.6-12	3.6/4.6-12
3.6/4.6-13	3.6/4.6-13
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LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is $\geq$ 25% of rated, the ration of FRP/CMFLPD shall be $\geq$ 1.0, or the APRM scram and rod block setpoint equations listed in Sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

$$S_{RB} \leq (0.66W + 42\%) \left(\frac{FRP}{CMFLPD}\right)$$

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq$ 25% of rated thermal power within 4 hours.

M. Core Thermal-Hydraulic Stability

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.

2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.

3. If Region II of Figure 3.5.M-1 is entered:

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq$ 25% of rated thermal power.

M. Core Thermal-Hydraulic Stability

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:

   a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and

   b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5 <u>Core and Containment Cooling Systems</u>	4.5 <u>Core and Containment Cooling Systems</u>

3.5.M.3. (Cont'd)

a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is <u>not</u> an appropriate action), and

b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB284L QUAD+

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.2
30,000	10.8
35,000	10.2
40,000	9.5
45,000	8.8

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB265H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB284Z

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.2
5,000	11.7
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.1
30,000	10.4
35,000	9.8
40,000	9.1
45,000	8.5

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.2
30,000	10.8
35,000	10.2
40,000	9.5

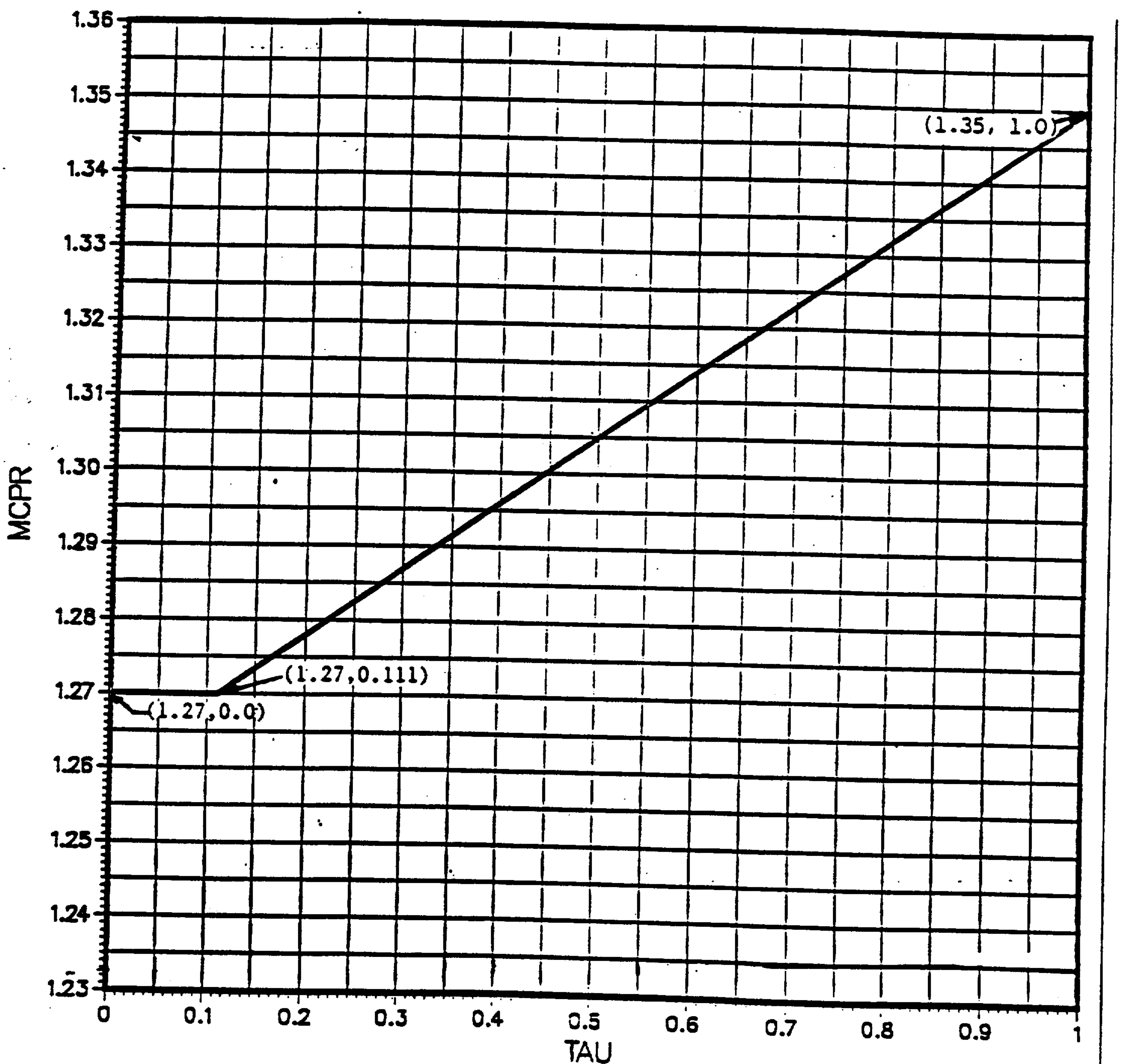
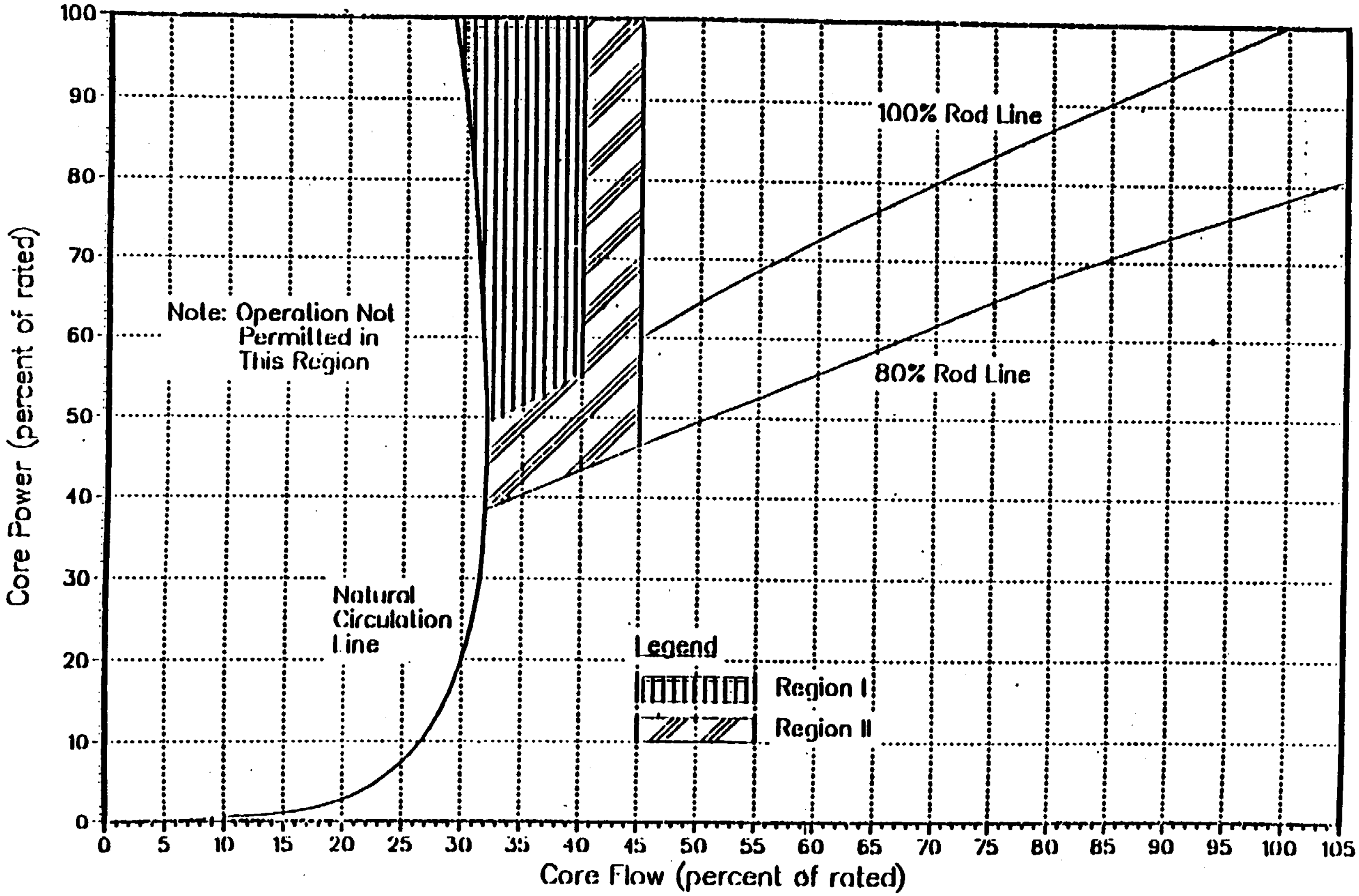


Figure 3.5.K-1
MCPR Limits for P8 X 8R/8 X 8R/ QUAD+

BFN Unit 2

Figure 3.5.M-1
BFN Power/Flow Stability Regions



of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

During regional oscillations, the safety limit MCPR is not approached until APRM oscillations are 30 percent peak-to-peak or larger in magnitude. In addition, periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2, NEDO - 24088-1 and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE - 24011-P-A and Addenda.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

LIMITING CONDITIONS FOR OPERATION

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the COLD SHUTDOWN CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.D. Relief Valves

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

   b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

   c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.6.E. Jet Pumps 2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.
3.6.F Recirculation Pump Operation 1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.	4.6.F. Recirculation Pump Operation 1. Recirculation pump speeds shall be checked and logged at least once per day.
2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.	2. No additional surveillance required.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval, restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature of the reactor	3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

   a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds as listed to provide additional protection against pipe whip, which could damage auxiliary and control systems.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.G Structural Integrity	4.6.G Structural Integrity
3.6.G.1 (Cont'd)	4.6.G.2 (Cont'd)
b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.	Feedwater - GFW-9, KFW-13 GFW-12, GFW-26, KFW-31, GFW-29, KFW-39, GFW-15, KFW-38, and GFW-32 Main Steam - GMS-6, KMS-24 GMS-32, KMS-104, GMS-15 and GMS-24 RHR - DSRHR-4, DSRHR-7, DSRHR-6 Core Spray - TSC-407, TSC-423, TSCS-408, and TSC-424 Reactor Cleanup - DSRWC-4, DSRWC-3 DSRWC-6, DSRWC-5 HPCI - THPCI - 70 THPCI - 70A THPCI - 71 THPCI - 72

3.6.E/4.6.E (Cont'd)

resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F <u>Recirculation Pump Operation</u>

Operation without forced recirculation is permitted for up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be operable while in the RUN mode provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G <u>Structural Integrity</u>

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

3.6.G/4.6.G (Cont'd)

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)

7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

In a letter and enclosures from M. Ray, Tennessee Valley Authority (TVA), to the NRC, dated June 20, 1989 (Reference 1), TVA proposed Technical Specifications (TS) changes for Browns Ferry Nuclear Plant, Unit 2 (BFN2). The proposed changes define regions on the operating power-flow map and operating restrictions on activities relating to those regions.

The proposed regions and restrictions are intended to avoid problems with thermal hydraulic instability, which have been a focus of NRC attention following the LaSalle instability event of March 1988. This attention has resulted in the issuance of NRC Bulletin 88-07 and Supplement 1 to that bulletin (References 2 and 3). These provide NRC action requests for utilities to provide operator training, instrumentation verification and operating procedures intended to minimize instability potential or consequences. The requested operating procedures of Supplement 1 are based on the General Electric (GE) Interim Recommendations for Stability Actions (IRSA). They are presented in the attachment to the supplement. These recommendations, along with other NRC staff requests presented in the supplement, constitute current NRC recommendations for BWR thermal hydraulic stability (THS) related operations. They are the result of calculations and reviews by the NRC, GE, the BWR Owners' Group and associated consultants. The bulletin supplement requested that licensees implement the IRSA (and other associated requests) by modifying relevant procedures. Modification of TS was not specifically requested since it is expected that long term solution implementation, to replace the interim recommendations, will begin within about a year. However, several licensees have modified their stability TS to correspond to the bulletin requests. Since BFN2 currently has no stability related TS, TVA proposed TS addressing Supplement 1 requests before restart of BFN2.

The proposed changes to the BFN2 TS are (1) addition of TS 3.5.M.1, 2 and 3, TS 4.5.M.1, Figure 3.5.M-1 and the addition of the associated Bases 3.5.M and (2) changes to TS 3.6.F.3 and 4.6.F.3, the addition of 3.6.F.4 and additions to the Bases for 3.6.F/4.6.F. There are also associated changes to the table of contents and list of illustrations.

2.0 EVALUATION

The IRSA specify three regions (A, B, C) on the power-flow map involving different degrees of allowed or prohibited operation. These are bounded by constant flow lines or control rod lines (lines of flow variation with all other reactor parameters, particularly control rod position, held constant). Region A is above the 100 percent rod line (intercepts 100 percent rated power at 100 percent rated flow) and below 40 percent flow. Region B is between the 80 and 100 percent rod lines and below 40 percent flow. Region C is above the 80 percent rod line and between 40 and 45 percent flow. Deliberate entry into Regions A and B is not permitted, and if it occurs, immediate exit is required. For a Group 2 plant (such as BFN2) immediate scram is required in Region A, while for region B control rod insertion or flow increase may be used to exit. Operations may be conducted in Region C, with suitable surveillance, if required during "startups" to prevent fuel damage. If during operations in B or C instability occurs, the reactor shall be scrammed, with evidence for instability coming from Average Power Range Monitor (APRM) oscillation greater than 10 percent or Local Power Range Monitor (LPRM) upscale or downscale alarms.

The proposed BFN2 TS conservatively implements these region designations and associated operational requirements by adding a new specification, TS 3/4.5.M, Core Thermal-Hydraulic Stability, and power-flow map, Figure 3.5.M-1. The regions designated in the figure are the same as in IRSA except that regions B and C are combined into a single Region II (and Region A is designated Region I). The IRSA operating restrictions of Region B are conservatively applied throughout Region II. There is no allowed operation such as is permitted by IRSA for, e.g., startup in Region C. TS 3.5.M.1, 2 and 3 specify that operation is not permitted in Regions I and II, and upon inadvertent entry, scram is required if in Region I, and immediate initiation of action to depart by control rod insertion or flow increase is required for Region II. While exiting Region II, scram is required if there are indications of instability as evidenced by APRM oscillations above 10 percent peak-to-peak of rated power or LPRM oscillations above 30 percent, and LPRM upscale or downscale alarms require immediate checks of APRM and LPRM readings. These requirements all meet or exceed the IRSA specifications and are acceptable TS for meeting the bulletin requests for implementing the interim recommendations. TS 4.5.M provides surveillance requirements for determining that operation is outside of Regions I and II when operating in the vicinity of these regions. They too are acceptable. The new Bases 3.5.M provides a reasonable discussion of the background, regions, operations and requirements for these specifications and is also acceptable.

Bulletin 88-07, Supplement 1 also requested that plants which do not have effective automatic scram protection for regional oscillations (Group 2 plants in the IRSA), should initiate a manual reactor scram when two recirculation pumps trip (or "no pumps operating") with the reactor in the RUN mode. BFN2 is a Group 2 plant, and the proposed addition of 3.6.F.4 to TS 3/4.6.F, Recirculation Pump Operation, is intended to comply with this request. It specifies that the reactor shall not be operated in the RUN mode with both recirculation pumps out-of-service, and an immediate manual scram is required, in the RUN mode, following a trip of both recirculation pumps. This is an acceptable implementation for two pump operation, and BFN2 TS do not permit extended single loop operation (SLO). However, SLO is permitted for a short time (24 hours) and it should therefore be noted that the Supplement 1 recirculation pump trip scram request as stated in "(or "no pumps operating")" intends scram upon the trip of the operating pump in SLO in the RUN mode. The licensee has committed to modify the Bases for 3.6.F to address this issue prior to Unit 2 restart. The staff finds this to be an acceptable approach.

There are also modifications to TS 3/4.6.F.3 which currently permits operation for up to 12 hours with both recirculation pumps out-of-service. The modifications permit such operation at power only while not in the RUN mode, i.e., permitted only at low power. This is an acceptable change.

The overall conclusion of the review is that the proposed TS changes and the material submitted to support the changes are acceptable. It should be noted, however, that the NRC staff, its consultants, the BWR Owners' Group (BWROG), GE and others are continuing the review of THS concerns. The BWROG is developing several long term solutions for the problem. It is expected that a selection will be announced by the end of 1989. Any new requirements resulting from the continuing generic review of THS concerns and BWROG long term solutions will be applicable to BF2 and may impact some of the operations, systems surveillance or TS found to be acceptable in this review.

We have reviewed the reports submitted by TVA for BFN2 proposing TS changes relating to THS requirements for power-flow map operating restraints and surveillance. Based on this review, we have concluded that appropriate documentation was submitted and the proposed power-flow action regions, surveillance and TS changes satisfy staff positions (NRC Bulletin 88-07 and Supplement 1), and requirements in these areas. Operation in the modes proposed for BFN2 is acceptable. This conclusion may be subject to future review based on results from the staff continuing generic review and conclusions on long term solutions.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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 nor environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 29414) on July 12, 1989 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

5.0 REFERENCES

1. Letter from M. Ray, TVA, to NRC, dated June 20, 1989, "Browns Ferry Nuclear Plant (BFN) - TVA BFN Technical Specification No. 272 - Thermal-Hydraulic Stability Section 3.5/4.5-M."

2. NRC Bulletin No. 88-07: Power Oscillations in Boiling Water Reactors (BWR), dated June 15, 1988.

3. NRC Bulletin No. 88-07, Supplement 1: Power Oscillations in Boiling Water Reactors (BWR)," dated December 30, 1988.

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