UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 236
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 June 2, 1997, as supplemented November 19, 1998, 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-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. 236, 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

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 23, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 236

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

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

Remove	Insert
2.0-1	2.0-1
3.3-6	3.3-6
3.4-1	3.4-1
3.4-2	3.4-2
B 3.2-2	B 3.2-2
B 3.2-5	B 3.2-5
B 3.2-7	B 3.2-7
B 3.2-10	B 3.2-10
B 3.4-4	B 3.4-4
B 3.4-5	B 3.4-5
---------	B 3.4-5a
B 3.4-7	B 3.4-7
B 3.4-8	B 3.4-8
B 3.4-10	B 3.4-10
B 3.4-14	B 3.4-14

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 0.58 W + 62% RTP and ≤ 120% RTP(b)
c. Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) [0.58 W + 62% - 0.58 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

Two recirculation loops with matched flows shall be in operation with core flow as a function of THERMAL POWER outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.

OR

One recirculation loop may be in operation with core flow as a function of THERMAL POWER outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1 and provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor Upscale (Flow Biased)), Allowable Value of Table 3.3.2.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor operation with core flow as a function of THERMAL POWER inside of Region I of Figure 3.4.1-1.	A.1 Place mode switch in the shutdown position.	Immediately
B. Reactor operation with core flow as a function of THERMAL POWER inside of Region II of Figure 3.4.1-1.	B.1 Place mode switch in the shutdown position. AND B.2 Exit Region II.	Immediately upon discovery of thermal hydraulic instability 2 hours
C. Requirements of the LCO not met for reasons other than A or B.	C.1 Satisfy the requirements of the LCO.	24 hours
D. Required Action and associated Completion Time of Conditions B or C not met. OR No recirculation loops in operation while in MODE 2.	D.1 Be in MODE 3.	12 hours
E. No recirculation loops in operation while in MODE 1.	E.1 Place mode switch in the shutdown position.	Immediately

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 5. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, an APLHGR multiplier is applied to the APLHGR limit (Ref. 5 and Ref. 7). The multiplier is documented in the COLR. This multiplier is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe heatup during a LOCA.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 7). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appendix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

BASES (continued)

APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, and 8. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained. Flow dependent correction factor for MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The flow dependent correction factor is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System. The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).
LCO	The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 5. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, an APLHGR multiplier is applied to the APLHGR limit (Ref. 5 and Ref. 7). The multiplier is documented in the COLR. This multiplier is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe heatup during a LOCA.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 7). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appendix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, and 8. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

Flow dependent correction factor for MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The flow dependent correction factor is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Refs. 7 and 8).

The transient analyses of Chapter 14 of the FSAR have also been performed for single recirculation loop operation (Ref. 7) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument and RBM setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR setpoints for single loop operation are specified in the COLR. The APRM Flow Biased Simulated Thermal Power-High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" and the RBM Flow Biased Upscale setpoint is in the COLR as referenced by LCO 3.3.2.1, "Control Rod Block Instrumentation."

Safety analyses performed for FSAR Chapter 14 implicitly assume core conditions are stable. However, at the high power/low flow corner of the power/flow map, an increased probability for limit cycle oscillations exists (Ref. 3) depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow). Generic evaluations indicate that when regional power oscillations become detectable on the APRMs, the safety margin may be insufficient under some operating conditions to ensure actions taken to respond to the APRMs signals would prevent violation of the MCPR Safety Limit (Ref. 4). NRC Generic Letter 86-02 (Ref. 5) addressed

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BASES

APPLICABLE SAFETY ANALYSES (continued)

stability calculation methodology and stated that due to uncertainties, 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and 12 could not be met using analytic procedures on a BWR 4 design. However, Reference 5 concluded that operating limitations which provide for the detection (by monitoring neutron flux noise levels) and suppression of flux oscillations in operating regions of potential instability consistent with the recommendations of Reference 3 are acceptable to demonstrate compliance with GDC 10 and 12. The NRC concluded that regions of potential instability could occur at calculated decay ratios of 0.8 or greater by the General Electric methodology.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This decay ratio also helps ensure sufficient margin to an instability occurrence is maintained. The generic region has been determined to be bounded by the 80% rod line and the 45% core flow line. BFN conservatively implements this generic region with the "Operation Not Permitted" Region and Regions I and II of Figure 3.4.1-1. This conforms to Reference 3 recommendations. Operation is permitted in Region II provided neutron flux noise levels are verified to be within limits. The reactor mode switch must be placed in the shutdown position (an immediate scram is required) if Region I is entered.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Refs. 7 and 8).

The transient analyses of Chapter 14 of the FSAR have also been performed for single recirculation loop operation (Ref. 7) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument and RBM setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR setpoints for single loop operation are specified in the COLR. The APRM Flow Biased Simulated Thermal Power-High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" and the RBM Flow Biased Upscale setpoint is in the COLR as referenced by LCO 3.3.2.1, "Control Rod Block Instrumentation."

Safety analyses performed for FSAR Chapter 14 implicitly assume core conditions are stable. However, at the high power/low flow corner of the power/flow map, an increased probability for limit cycle oscillations exists (Ref. 3) depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow). Generic evaluations indicate that when regional power oscillations become detectable on the APRMs, the safety margin may be insufficient under some operating conditions to ensure actions taken to respond to the APRMs signals would prevent violation of the MCPR Safety Limit (Ref. 4). NRC Generic Letter 86-02 (Ref. 5) addressed

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BASES

APPLICABLE SAFETY ANALYSES (continued)

stability calculation methodology and stated that due to uncertainties, 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and 12 could not be met using analytic procedures on a BWR 4 design. However, Reference 5 concluded that operating limitations which provide for the detection (by monitoring neutron flux noise levels) and suppression of flux oscillations in operating regions of potential instability consistent with the recommendations of Reference 3 are acceptable to demonstrate compliance with GDC 10 and 12. The NRC concluded that regions of potential instability could occur at calculated decay ratios of 0.8 or greater by the General Electric methodology.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This decay ratio also helps ensure sufficient margin to an instability occurrence is maintained. The generic region has been determined to be bounded by the 80% rod line and the 45% core flow line. BFN conservatively implements this generic region with the "Operation Not Permitted" Region and Regions I and II of Figure 3.4.1-1. This conforms to Reference 3 recommendations. Operation is permitted in Region II provided neutron flux noise levels are verified to be within limits. The reactor mode switch must be placed in the shutdown position (an immediate scram is required) if Region I is entered.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

(continued)

BASES (continued)

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR Limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, 'MINIMUM CRITICAL POWER RATIO (MCPR)"), APRM Flow Biased Simulated Thermal Power-High Setpoint (LCO 3.3.1.1), and RBM Flow Biased Upscale Setpoint (LCO 3.3.2.1) may be applied to allow continued operation consistent with the assumptions of References 7 and 8. In addition, the core flow expressed as a function of THERMAL POWER must be outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

1. A sustained increase in APRM and/or LPRM peak-to-peak signal noise level, reaching two or more times its initial level at reduced core flow conditions. Any noticeable increase in noise level warrants closer monitoring of the LPRM signals.

   The increased noise occurs with a characteristic period of less than 3 seconds.

2. LPRM and or APRM upscale and/or downscale annunciators that alarm with a characteristic period of less than 3 seconds.

C.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

(continued)

BASES

ACTIONS

C.1 (continued)

The 24 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

D.1

With no recirculation loops in operation while in MODE 2 or the Required Action and associated Completion Time of Condition B or C not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

This SR ensures the reactor THERMAL POWER and core flow are within appropriate parameter limits to prevent uncontrolled power oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal hydraulic instability. Figure 3.4.1-1 is based on guidance provided in Reference 3, which is used to respond to operation in these conditions. Performance immediately after any increase of more than 5% RTP while initial core flow is < 45% of rated and immediately after any decrease of more than 10% rated core flow while initial thermal power is > 40% of rated is adequate to detect power oscillations that could lead to thermal hydraulic instability.

REFERENCES

1. FSAR, Section 14.6.3.
2. FSAR, Section 4.3.5.
3. GE Service Information Letter No. 380, "BWR Core Thermal Hydraulic Stability," Revision 1, February 10, 1984.
4. NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors (BWRs)," Supplement 1, December 30, 1988.
5. NRC Generic Letter 86-02, "Technical Resolution of Generic Issue B-19, Thermal Hydraulic Stability," January 22, 1986.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.
8. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Each recirculation loop must satisfy one of the performance criteria provided. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while baselining new "established patterns," engineering judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow and loop flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump flow and loop flow versus pump speed relationship must be verified.

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 256
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, 1997, as supplemented November 19, 1998, 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. 256, 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

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 256

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

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

Remove	Insert	Note
2.0-1	2.0-1	
3.3-7	3.3-7	Applicable after Cycle 11only
-------	3.3-7a	Applicable for Cycle 10 only
3.4-1	3.4-1	
3.4-2	3.4-2	
B 3.2-3	B 3.2-3	
---------	B 3.2-3a	
B 3.2-5	B 3.2-5	
B 3.2-6	B 3.2-6	
B 3.2-7	B 3.2-7	
B 3.2-11	B 3.2-11	
B 3.4-4	B 3.4-4	
B 3.4-5	B 3.4-5	Applicable after Cycle 11 only
B 3.4-5	B 3.4-5(1)	Applicable for Cycle 10 only
---------	B 3.4-5a	
B 3.4-7	B 3.4-7	
B 3.4-8	B 3.4-8	
B 3.4-10	B 3.4-10	
B 3.4-14	B 3.4-14	

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 66% RTP and ≤ 120% RTP(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(c) [.66 W + 66% - .66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Note: This page is applicable after commencing Cycle 11 operation.

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 71% RTP and ≤ 120% RTP(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(c) [.66 W + 71% - .66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Note: This page is applicable only for Cycle 10 operation.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

Two recirculation loops with matched flows shall be in operation with core flow as a function of THERMAL POWER outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.

<u>OR</u>

One recirculation loop may be in operation with core flow as a function of THERMAL POWER outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1 and provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor operation with core flow as a function of THERMAL POWER inside of Region I of Figure 3.4.1-1.	A.1	Place mode switch in the shutdown position.	Immediately
B. Reactor operation with core flow as a function of THERMAL POWER inside of Region II of Figure 3.4.1-1.	B.1	Place mode switch in the shutdown position.	Immediately upon discovery of thermal hydraulic instability
	<u>AND</u>		
	B.2	Exit Region II.	2 hours
C. Requirements of the LCO not met for reasons other than A or B.	C.1	Satisfy the requirements of the LCO.	24 hours
D. Required Action and associated Completion Time of Conditions B or C not met. <u>OR</u> No recirculation loops in operation while in MODE 2.	D.1	Be in MODE 3.	12 hours
E. No recirculation loops in operation while in MODE 1.	E.1	Place mode switch in the shutdown position.	Immediately

BASES

APPLICABLE SAFETY ANALYSES (continued)

rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR..

For single recirculation loop operation, an APLHGR multiplier is applied to the APLHGR limit (Ref. 5 and Ref. 10). The multiplier is documented in the COLR. This multiplier is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe heatup during a LOCA.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For operation at other than 100% power and 100% recirculation flow conditions, the APLHGR operating limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 10). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR.

(continued)

BASES (continued)

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 4) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appendix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 8, and 10. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Reference 8). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Reference 6) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined by the one dimensional transient code (Reference 9). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly. (Refs. 7 and 8).

The transient analyses of Chapter 14 of the FSAR have also been performed for single recirculation loop operation (Ref. 7) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR setpoints for single loop operation are specified in the COLR. The APRM Flow Biased Simulated Thermal Power-High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Safety analyses performed for FSAR Chapter 14 implicitly assume core conditions are stable. However, at the high power/low flow corner of the power/flow map, an increased probability for limit cycle oscillations exists (Ref. 3) depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow). Generic evaluations indicate that when regional power oscillations become detectable on the APRMs, the safety margin may be insufficient under some operating conditions to ensure actions taken to respond to the APRMs signals would prevent violation of the MCPR Safety Limit (Ref. 4). NRC Generic Letter 86-02 (Ref. 5) addressed stability calculation methodology and stated that due to uncertainties, 10 CFR 50, Appendix A, General Design Criteria

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(GDC) 10 and 12 could not be met using analytic procedures on a BWR 4 design. However, Reference 5 concluded that operating limitations which provide for the detection (by monitoring neutron flux noise levels) and suppression of flux oscillations in operating regions of potential instability consistent with the recommendations of Reference 3 are acceptable to demonstrate compliance with GDC 10 and 12. The NRC concluded that regions of potential instability could occur at calculated decay ratios of 0.8 or greater by the General Electric methodology.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This decay ratio also helps ensure sufficient margin to an instability occurrence is maintained. The generic region has been determined to be bounded by the 76.2% rod line and the 50% core flow line. BFN conservatively implements this generic region with the "Operation Not Permitted" Region and Regions I and II of Figure 3.4.1-1. This conforms to Reference 3 recommendations. Operation is permitted in Region II provided neutron flux noise levels are verified to be within limits. The reactor mode switch must be placed in the shutdown position (an immediate scram is required) if Region I is entered.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

(continued)

Note: This page is applicable after commencing Cycle 11 operation.

BASES

APPLICABLE SAFETY ANALYSES (continued)

(GDC) 10 and 12 could not be met using analytic procedures on a BWR 4 design. However, Reference 5 concluded that operating limitations which provide for the detection (by monitoring neutron flux noise levels) and suppression of flux oscillations in operating regions of potential instability consistent with the recommendations of Reference 3 are acceptable to demonstrate compliance with GDC 10 and 12. The NRC concluded that regions of potential instability could occur at calculated decay ratios of 0.8 or greater by the General Electric methodology.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This decay ratio also helps ensure sufficient margin to an instability occurrence is maintained. The generic region has been determined to be bounded by the 80% rod line and the 50% core flow line. BFN conservatively implements this generic region with the "Operation Not Permitted" Region and Regions I and II of Figure 3.4.1-1. This conforms to Reference 3 recommendations. Operation is permitted in Region II provided neutron flux noise levels are verified to be within limits. The reactor mode switch must be placed in the shutdown position (an immediate scram is required) if Region I is entered.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

(continued)

Note: This page is applicable only for Cycle 10 operation.

BASES (continued)

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR Limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, 'MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power-High Setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the assumptions of References 7 and 8. In addition, the core flow expressed as a function of THERMAL POWER must be outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

1. A sustained increase in APRM and/or LPRM peak-to-peak signal noise level, reaching two or more times its initial level at reduced core flow conditions. Any noticeable increase in noise level warrants closer monitoring of the LPRM signals.

   The increased noise occurs with a characteristic period of less than 3 seconds.

2. LPRM and or APRM upscale and/or downscale annunciators that alarm with a characteristic period of less than 3 seconds.

C.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

(continued)

BASES

ACTIONS

C.1 (continued)

The 24 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

D.1

With no recirculation loops in operation while in MODE 2 or the Required Action and associated Completion Time of Condition B or C not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

This SR ensures the reactor THERMAL POWER and core flow are within appropriate parameter limits to prevent uncontrolled power oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal hydraulic instability. Figure 3.4.1-1 is based on guidance provided in Reference 3, which is used to respond to operation in these conditions. Performance immediately after any increase of more than 5% RTP while initial core flow is < 50% of rated and immediately after any decrease of more than 10% rated core flow while initial thermal power is > 40% of rated is adequate to detect power oscillations that could lead to thermal hydraulic instability.

REFERENCES

1. FSAR, Section 14.6.3.
2. FSAR, Section 4.3.5.
3. GE Service Information Letter No. 380, "BWR Core Thermal Hydraulic Stability," Revision 1, February 10, 1984.
4. NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors (BWRs)," Supplement 1, December 30, 1988.
5. NRC Generic Letter 86-02, "Technical Resolution of Generic Issue B-19, Thermal Hydraulic Stability," January 22, 1986.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.
8. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Each recirculation loop must satisfy one of the performance criteria provided. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while baselining new "established patterns," engineering judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow and loop flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump flow and loop flow versus pump speed relationship must be verified.

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
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, 1997, as supplemented November 19, 1998, 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. 216, 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

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 216

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

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

Remove	Insert
2.0-1	2.0-1
3.3-7	3.3-7
3.4-1	3.4-1
3.4-2	3.4-2
B 3.2-3	B 3.2-3
---------	B 3.2-3a
B 3.2-5	B 3.2-5
B 3.2-6	B 3.2-6
B 3.2-7	B 3.2-7
B 3.2-11	B 3.2-11
B 3.4-4	B 3.4-4
B 3.4-5	B 3.4-5
---------	B 3.4-5a
B 3.4-7	B 3.4-7
B 3.4-8	B 3.4-8
B 3.4-10	B 3.4-10
B 3.4-14	B 3.4-14

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall.be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 66% RTP and ≤ 120% RTP(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(c) [.66 W + 66% - .66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

Two recirculation loops with matched flows shall be in operation with core flow as a function of THERMAL POWER outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.

<u>OR</u>

One recirculation loop may be in operation with core flow as a function of THERMAL POWER outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1 and provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor operation with core flow as a function of THERMAL POWER inside of Region I of Figure 3.4.1-1.	A.1	Place mode switch in the shutdown position.	Immediately
B. Reactor operation with core flow as a function of THERMAL POWER inside of Region II of Figure 3.4.1-1.	B.1	Place mode switch in the shutdown position.	Immediately upon discovery of thermal hydraulic instability
	<u>AND</u>		
	B.2	Exit Region II.	2 hours
C. Requirements of the LCO not met for reasons other than A or B.	C.1	Satisfy the requirements of the LCO.	24 hours
D. Required Action and associated Completion Time of Conditions B or C not met. <u>OR</u> No recirculation loops in operation while in MODE 2.	D.1	Be in MODE 3.	12 hours
E. No recirculation loops in operation while in MODE 1.	E.1	Place mode switch in the shutdown position.	Immediately

BASES

APPLICABLE SAFETY ANALYSES (continued)

rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, an APLHGR multiplier is applied to the APLHGR limit (Ref. 5 and Ref. 10). The multiplier is documented in the COLR. This multiplier is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe heatup during a LOCA.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. For operation at other than 100% power and 100% recirculation flow conditions, the APLHGR operating limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 10). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR.

(continued)

BASES (continued)

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 4) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appendix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 8, and 10. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Reference 8). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Reference 6) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined by the one dimensional transient code (Reference 9). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Refs. 7 and 8).

The transient analyses of Chapter 14 of the FSAR have also been performed for single recirculation loop operation (Ref. 7) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR setpoints for single loop operation are specified in the COLR. The APRM Flow Biased Simulated Thermal Power-High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Safety analyses performed for FSAR Chapter 14 implicitly assume core conditions are stable. However, at the high power/low flow corner of the power/flow map, an increased probability for limit cycle oscillations exists (Ref. 3) depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow). Generic evaluations indicate that when regional power oscillations become detectable on the APRMs, the safety margin may be insufficient under some operating conditions to ensure actions taken to respond to the APRMs signals would prevent violation of the MCPR Safety Limit (Ref. 4). NRC Generic Letter 86-02 (Ref. 5) addressed stability calculation methodology and stated that due to uncertainties, 10 CFR 50, Appendix A, General Design Criteria

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(GDC) 10 and 12 could not be met using analytic procedures on a BWR 4 design. However, Reference 5 concluded that operating limitations which provide for the detection (by monitoring neutron flux noise levels) and suppression of flux oscillations in operating regions of potential instability consistent with the recommendations of Reference 3 are acceptable to demonstrate compliance with GDC 10 and 12. The NRC concluded that regions of potential instability could occur at calculated decay ratios of 0.8 or greater by the General Electric methodology.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This decay ratio also helps ensure sufficient margin to an instability occurrence is maintained. The generic region has been determined to be bounded by the 76.2% rod line and the 45% core flow line. BFN conservatively implements this generic region with the "Operation Not Permitted" Region and Regions I and II of Figure 3.4.1-1. This conforms to Reference 3 recommendations. Operation is permitted in Region II provided neutron flux noise levels are verified to be within limits. The reactor mode switch must be placed in the shutdown position (an immediate scram is required) if Region I is entered.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

(continued)

BASES (continued)

LCO.

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR Limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, 'MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power-High Setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the assumptions of References 7 and 8. In addition, the core flow expressed as a function of THERMAL POWER must be outside Regions I and II and the Operation Not Permitted Region of Figure 3.4.1-1.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

1. A sustained increase in APRM and/or LPRM peak-to-peak signal noise level, reaching two or more times its initial level at reduced core flow conditions. Any noticeable increase in noise level warrants closer monitoring of the LPRM signals.

   The increased noise occurs with a characteristic period of less than 3 seconds.

2. LPRM and or APRM upscale and/or downscale annunciators that alarm with a characteristic period of less than 3 seconds.

C.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

(continued)

BASES

ACTIONS

C.1 (continued)

The 24 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

D.1

With no recirculation loops in operation while in MODE 2 or the Required Action and associated Completion Time of Condition B or C not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

This SR ensures the reactor THERMAL POWER and core flow are within appropriate parameter limits to prevent uncontrolled power oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal hydraulic instability. Figure 3.4.1-1 is based on guidance provided in Reference 3, which is used to respond to operation in these conditions. Performance immediately after any increase of more than 5% RTP while initial core flow is < 50% of rated and immediately after any decrease of more than 10% rated core flow while initial thermal power is > 40% of rated is adequate to detect power oscillations that could lead to thermal hydraulic instability.

REFERENCES

1. FSAR, Section 14.6.3.
2. FSAR, Section 4.3.5.
3. GE Service Information Letter No. 380, "BWR Core Thermal Hydraulic Stability," Revision 1, February 10, 1984.
4. NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors (BWRs)," Supplement 1, December 30, 1988.
5. NRC Generic Letter 86-02, "Technical Resolution of Generic Issue B-19, Thermal Hydraulic Stability," January 22, 1986.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.
8. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Each recirculation loop must satisfy one of the performance criteria provided. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while baselining new "established patterns," engineering judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow and loop flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump flow and loop flow versus pump speed relationship must be verified.

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