ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATIONS REVISIONS

BROWNS FERRY NUCLEAR PLANT

UNIT 2

(TVA BFN TS 272)

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PDR ADOCK 05000260
P PDC

LIST OF ILLUSTRATIONS

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

3.5 Core and Containment Cooling Systems

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

3.5.M.3. (Cont'd)

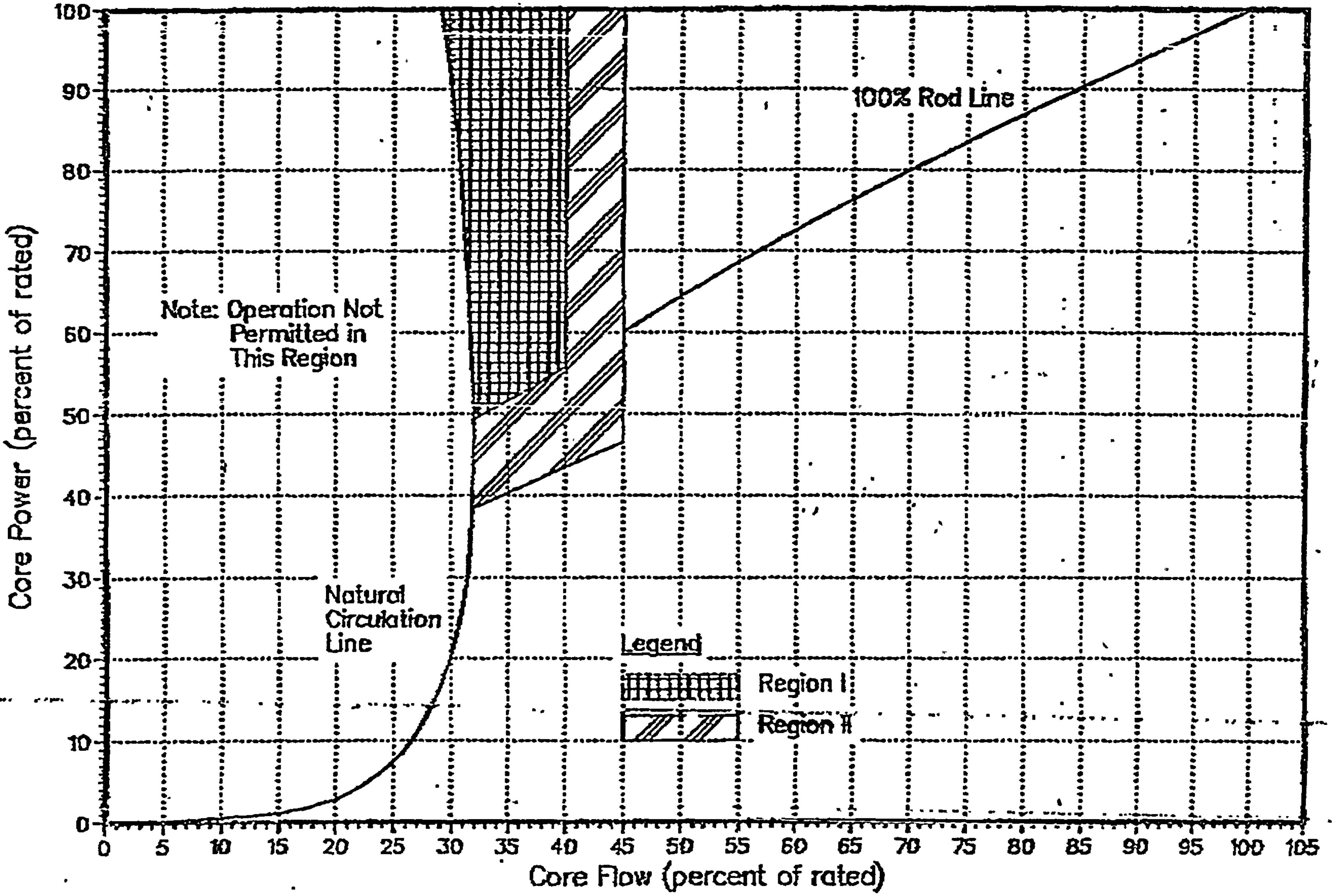
a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and

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.

BFN UNIT 2

3.5/4.5-22a

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



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. <u>References</u>

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.

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.

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.

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

4.6.F. Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.

2. No additional surveillance required.

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)

the reactor 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.

3.6.F.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

3.6.G Structural Integrity

3.6.G.1 (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.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

4.6.G.2 (Cont'd)

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	-	TCS-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 Recirculation Pump Operation.

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 Structural Integrity

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.

ENCLOSURE 2

DESCRIPTION AND JUSTIFICATION
BROWNS FERRY NUCLEAR PLANT (BFN)

Reason for Change

BFN unit 2 technical specifications Sections 3.5/4.5-M are being added and section 3.6.F is being revised to incorporate surveillance requirements and Limiting Conditions for Operation (LCO) for reactor core thermal-hydraulic stability. These changes are being proposed to support the BFN unit 2 fuel reload technical specification submitted August 26, 1988 (TS 254) and to also implement the requirements of NRC Bulletin 88-07, Supplement 1.

Background

General Design Criteria (GDC) 12 requires that reactor power oscillations either be (1) prevented or (2) detected and suppressed. The stability licensing basis for U.S. Boiling Water Reactors (BWRs) has been either that oscillations will not occur in allowable operating regions (as demonstrated by decay ratio calculations) or that oscillations can be detected and suppressed by reactor operators before protection limits are exceeded. In the past, BFN demonstrated compliance with GDC 12 by performing decay ratio analyses for each reload core to show that core thermal-hydraulic oscillations would not occur in allowable reactor operating regions.

Recent instability events at LaSalle and Vermont Yankee have led to concerns relative to the capability of currently approved analytical methods to adequately predict when instabilities will occur. These events as well as analyses performed by General Electric (GE) and NRC contractors indicate that instabilities may occur which result in regional oscillations and local power peaking greater than previously analyzed for in-phase core oscillations. In addition, preliminary calculations performed by GE indicate that under some operating conditions, the safety limit Minimum Critical Power Ratio (MCPR) may be violated during regional power oscillations.

The BWR Owners Group (BWROG) is currently working with GE and NRC to develop a long-term resolution to stability concerns. In November 1988, interim corrective actions to address stability concerns were issued by GE and subsequently adopted by the BWROG. NRC Bulletin 88-07, Supplement 1 (December 30, 1988), requires all BWR licensees to implement the GE recommendations. In addition, the bulletin requires some BWRs (BFN included) to initiate a manual scram following two recirculation pump trips when the reactor is in the RUN mode.

All BWR licensees have or are currently revising procedures to implement the requirements of the NRC bulletin. In discussions with BFN Site Licensing, NRC has indicated that in addition to procedural changes, BFN technical specifications must be modified to address stability concerns before restart of BFN unit 2.

Description and Justification for the Proposed Change

1. Add LCO 3.5.M to read as follows:

   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:

      a. Immediately initiate action and exit the region within 2 hours by inserting controls rods or by increasing core flow (starting a recirculation pump to exit the region is not 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.

Justification for Proposed Change 3.5.M

The NRC bulletin (and the GE interim corrective actions); define regions of concern on the power/flow map referred to as Regions A, B, and C. Region A includes operating conditions above the 100-percent rod line with core flow less than 40 percent of rated flow. Region B includes operating conditions between the 80- and 100-percent rod lines with core flow less than 40 percent of rated flow. Region C includes operating conditions above the 80-percent rod line with core flow between 40 and 45 percent of rated flow. The regions defined by GE cover the high power/low flow corner of the operating domain where stability margins are the lowest. The GE recommended region boundaries are based on plant operating experience, special stability tests, and analytical studies.

Justification for Proposed Change 3.5.M (Cont'd)

Region I of the proposed technical specification change corresponds to Region A as defined in the GE interim corrective actions. Most oscillations have occurred during testing and operation at or above the 100-percent rod line with core flow near natural circulation. This behavior is consistent with analyses which predict reduced stability margin with increasing power or decreasing flow. Region I bounds the majority of events and tests where core oscillations have been observed in GE BWRs. This region represents the least stable conditions on the power/flow map and is therefore considered an excluded region in which normal operation is not allowed. Because operating experience has demonstrated that oscillations may rapidly develop in this region, operator actions are required to prevent the initiation of core oscillations in the event Region I is entered.

Region II of the proposed technical specification change includes both Region B and C defined in the GE interim corrective actions. Region B of the interim corrective actions is also considered to be an excluded region (i.e., no intentional entry) because of the relatively low core flow. Even though the probability of core oscillations is lower in Region B than in Region A, several events and tests have demonstrated that oscillations can occur in this region for certain operating conditions. However, because the power level is lower, the margin to fuel safety limits is greater in Region B. Region C of the interim corrective actions is defined as a buffer zone to the excluded regions. Although no oscillations have been reported in this region, the possibility of core oscillations in this region cannot be ruled out. Therefore, the interim corrective actions allow operation in Region C only for control rod withdrawals during startup requiring fuel preconditioning. When operation does occur in Region C, the operator should be aware of the possibility of core oscillations and procedures should ensure that adequate surveillance of nuclear instrumentation is performed. The proposed BFN technical specification change will combine Regions B and C into Region II and will conservatively apply Region B restrictions to Region C.

The potential for core thermal-hydraulic oscillations to occur when operating outside of Regions I and II is very small and therefore special restrictions are not required outside of these regions.

The region boundaries for the interim corrective actions were developed based on plant operating and test experience and analysis of GE fuel designs. The regions were chosen to generically apply to all licensed GE fuel designs and operating domains (e.g., Extended Load Line, Single Loop Operations). The BFN unit 2, cycle 6 core will contain four Westinghouse (W) QUAD+ lead test assemblies. The presence of four QUAD+ bundles (about 0.5 percent of the core) will not significantly affect the thermal-hydraulic stability characteristics of the core. In addition, the QUAD+ fuel design contains several design features which make the bundle more stable than the GE 8 by 8 fuel design (reference 1). Therefore, the regions identified in the GE interim corrective actions are appropriate for QUAD+ operation in BFN unit 2, cycle 6.

Justification for Proposed Change 3.5.M (Cont'd)

Proposed change 3.5.M.1 restricts normal operation within guidelines of the power-flow map to conditions outside of Regions I and II. The excluded regions represent the least stable conditions for the plant. Regions I and II are usually entered as a result of plant transients (recirculation pump trip) and are not part of the normal operating domain. All events (including test experience) that have resulted in core oscillations have occurred in either Region I or II. Intentional operation is not allowed in these regions in order to minimize the probability of encountering core oscillations and potentially challenging fuel safety limits.

Proposed change 3.5.M.2 requires the operator to manually scram the reactor if Region I is entered. Because BFN does not have a flow-biased unfiltered neutron flux scram, automatic scram protection is not provided until APRM oscillations reach a peak magnitude of 120 percent of rated power. Because of partial cancellation of out-of-phase LPRM signals during regional oscillations, local neutron flux can be significantly higher than indicated by the APRM signal. Preliminary calculations by GE indicate that during operation in Region I, the safety limit MCPR (SLMCPR) may be violated in some situations when APRM oscillations are approximately 15-45 percent peak-to-peak (reference 2). Because stability margins are the lowest in Region I, the potential exists for oscillations to rapidly increase in magnitude once they initiate. During transients which cause entry into Region I, the operator may not have sufficient time to manually insert control rods or increase core flow to suppress oscillations before they reach an unacceptable magnitude. The prompt action of manually scramming the reactor if Region I is entered will ensure adequate protection of the SLMCPR.

Proposed change 3.5.M.3 requires the operator to take immediate action to exit Region II if entered inadvertently. Because core thermal-hydraulic stability is very sensitive to core power and flow, stability margins are greater in Region II than in Region I. The increased margin means that the probability of core oscillations is less and that oscillations will not increase in magnitude as rapidly as in Region I. Also, because of the lower power and/or increased core flow in Region II, the margin to the SLMCPR will be larger than in Region I. Because of the increased stability and SLMCPR margins, the operator will have more time to suppress oscillation in Region II before the SLMCPR is violated.

Tests and operating experience have demonstrated that the insertion of control rods or the increase of core flow will rapidly dampen core thermal-hydraulic oscillations and move the plant into a region of increased stability margin. At reactor conditions where core oscillations begin, the insertion of a few control rod notches or a 1-2 percent increase in core flow will effectively suppress the oscillations (reference 2). When control rod insertion is used to exit the region, a predefined set of control rods will generally be used by the operator to ensure an expedient reduction in core thermal power. If one or more recirculation pumps are operational, increasing core flow is an acceptable alternative to inserting control rods and is generally simpler to perform. However, starting a recirculation pump to exit the region is not an appropriate action since it can lead to sudden reactivity insertions and initiate core oscillations. Also, starting a circulation pump can potentially distract the operators attention away from the detection of potential oscillations while in the region.

Justification for Proposed Change 3.5.M (Cont'd)

The actions described above will minimize the probability of core thermal-hydraulic oscillations occurring following a transient which places the plant in Region II.

The presence of core thermal-hydraulic oscillations is an indication of the loss of control of the reactor, and if not mitigated, might rapidly lead to conditions which violate the SLMCPR. Experience has shown that oscillations can grow rapidly to high levels. Even though most events have been terminated by operator actions, it cannot be demonstrated that a manual control rod insertion or a core flow increase will always be rapid enough to prevent exceeding the SLMCPR. Therefore, if oscillations are detected, a rapid power reduction (i.e., scram) is the appropriate method of mitigation. This will ensure that oscillations are rapidly suppressed.

To avoid unnecessary challenges to the reactor protection system, a scram should be initiated only when there is clear evidence of core oscillations. APRM oscillations which exceed 10 percent peak-to-peak are clear indications of core thermal hydraulic instability. LPRM oscillations which exceed 30 percent peak-to-peak are approximately equivalent to APRM oscillations of 10 percent peak-to-peak during regional oscillations (reference 2) and are also clear indications of core instability. Periodic upscale or downscale LPRM alarms may be indicators of core thermal hydraulic oscillations. However, LPRM alarms alone should not be used to initiate a reactor scram because they only provide an indirect indication of oscillations and may be indicators of other conditions or equipment failures. If any LPRM alarms are received, the APRM's and individual LPRM's should be immediately evaluated to confirm the presence of oscillations.

Based on GE analyses (reference 2), the SLMCPR is not approached during regional oscillations until APRM oscillations are greater than approximately 30 percent peak-to-peak. In addition, periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit (reference 2). Therefore, initiating a manual scram on evidence of core instability as described above is sufficient to ensure that the SLMCPR will not be violated.

2. Add surveillance requirement 4.5.M to read as follows:

   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.

Justification for Proposed Change 4.5.M

The above surveillance requirements are being added to verify that the Reactor is operating in the proper region (acceptable region) when reactor power is increased greater than 5% rated thermal power (RTP) with initial reactor core flow less than 45% or a decrease of 10% core flow while initial RTP is greater than 40 percent.

3. Add figure 3.5.M-1 (See attached)

Justification for Proposed Change Figure 3.5.M-1

Figure 3.5.M-1, "BFN Power/Flow Stability Regions," provides the operators a clear illustration as to what conditions of reactor core power versus core flow are acceptable or unacceptable. Based on this table and the appropriate LCO 3.5.M, an operator can identify what action needs to be implemented to exit an unacceptable region.

4. Add Bases 3.5.M to read as follows:

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.

Justification for Bases Section 3.5.M

This section is being updated to provide consistency and provide additional information supporting the reasoning for adding Limiting Condition for Operation and Surveillance Instruction 3.5.M and 4.5.M.

5. Existing LCO 3.6.F.3 reads:

Steady-state 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...

Change existing LCO 3.6.F.3 to read as follows:

"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...

Justification for Proposed Change LCO 3.6.F.3

This change is being made to clarify that when the reactor mode switch is NOT in the RUN position, that both recirculation pumps may be out of service for up to 12 hours during REACTOR POWER OPERATION.

6. Existing SI 4.6.F.3 reads:

Before starting either recirculation pump during steady-state operation, check and log the loop discharge temperature and dome saturation temperature.

Change Existing SI 4.6.F.3 to read;

Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

Justification for Proposed Change 4.6.F.3

The word "reactor" is replacing the existing wording steady-state. This will provide consistency with the change made in LCO 3.6.F.3.

7. Add LCO 3.6.F.4 to read as follows:

"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."

Justification for LCO 3.6.F.4

Proposed change 3.6.F.4 requires the operator to manually scram the reactor following a trip of both recirculation pumps when the reactor is in the RUN mode. This action was not part of the GE interim corrective action recommendations but is a requirement of the NRC Bulletin. The reactor will enter either Region I or II following a recirculation pump trip from above the 80 percent rod line. Requiring a manual trip immediately following the loss of both recirculation pumps adds additional conservatism to ensure that thermal-hydraulic oscillations do not occur.

8. Revise Bases Section 3.6.F/4.6.F as follows:

"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 acceptable 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.

Justification for Revising Bases Section 3.6.F/4.6.F

This section is being updated to reflect the proposed change in LCO 3.6.F.4.

References

1. WCAP-10507, "QUAD+ Demonstration Assembly Report," Westinghouse Electric Corporation, March 1984.

2. General Electric Report prepared for the Boiling Water Reactors Owner's Group, "Fuel thermal Margin During Core Thermal Hydraulic Oscillations in a Boiling Water Reactor," March 1989.

3. Letter, D. N. Grace (BWROG) to A. Thadani (NRC), "Subject: NRC Bulletin 88-07 Supplement 1, Power Oscillations in Boiling Water Reactors," January 26, 1989.

[illegible]

ENCLOSURE 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

Description of Proposed Technical Specification Amendment

The proposed amendment would change the BFN technical specifications for unit 2 only. This amendment will add technical specification 3.5/4.5.M, Figure 3.5.M-1, "BFN Power/Flow Stability Regions," update bases section 3.5.M, revise section 3.6.F.3, add section 3.6.F.4, and update bases section 3.6.F/4.6.F. These changes will implement the requirements of NRC Bulletin 88-07, Supplement 1. These changes will define the reactor core regions of operation which are acceptable or unacceptable and provide require actions needed to exit operating in an unacceptable region.

Basis for Proposed No Significant Hazards Consideration Determination

NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from an accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

(1) This change does not involve a significant increase in the probability or consequences of an accident previously evaluated. Implementation of the proposed TS change decreases the probability of core thermal-hydraulic oscillations by precluding operating conditions where instabilities have occurred at other plants. In addition, the proposed change will provide additional assurance that core oscillations that do occur will be suppressed prior to exceeding fuel integrity limits. The proposed change does not have an adverse safety effect on any affected safety system nor are the assumptions of the safety analyses affected by restricting operation to outside of Regions I and II. Therefore, the proposed change reduces the probability and consequences of potential core oscillations and does not increase the probability or consequences of any other previously analyzed event.

(2) This proposed change does not create the possibility of a new or different kind of accident from any previously analyzed. Restricting operation to outside of Regions I and II does not create any new failure mechanisms. Plant procedures currently preclude normal operation in those regions. Emergency entry into a restricted region is permitted to protect plant safety equipment provided that the prescribed actions (i.e., scram or exit) for the region entered are performed. Operator actions to exit Region II will be performed in compliance with all plant procedures, fuel preconditioning restrictions, and technical specifications.

(3) This change does not involve a significant reduction in a margin to safety. The proposed changes are conservative in nature and provide increased assurance that the fuel safety limit MCPR will not be violated due to core oscillations. These changes are consistent with NRC and GE guidelines. The implementation of this tech spec will actually increase this margin of safety at BFN by not allowing the plant to operate in Regions I or II. If one of these Regions are entered specific operator actions are required which will place the plant in a more conservative and safe condition than current BFN Tech Specs required.