ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, and 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-353S1
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

INDEX

I. DESCRIPTION OF THE PROPOSED CHANGE

This proposed change to BFN TS consists, broadly, of two groups of changes related to the below activities:

A. Replacement of the power range portion of the existing Neutron Monitoring System (NMS) with a General Electric (GE) digital Nuclear Measurement Analysis and Control Power Range Power Range Neutron Monitor (NUMAC PRNM) retrofit design. TS changes related to this activity are consistent with the recommendations of Reference 2.

B. Implementation of Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) Technical Specification (ARTS) improvements and operation in an expanded core power/flow domain, the Maximum Extended Load Line Limit (MELLL) region. TS changes related to this activity are consistent with the recommendations of Reference 3.

This change package is based on TS-362, the proposed general revision of custom BFN TS to Improved Standard TS (ISTS) format submitted for NRC review on September 6, 1996 (Reference 13). TS-362 is based on NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/4)."

BFN's proposed ISTS changes are provided below. Applicable TS page numbers and sections are identified. Each change listed below is associated with the **[PRNM]** activity or with **[ARTS/MELLL]**, as marked.

1. Page 1.1-4, Section 1.1: **[ARTS/MELLL]**

   The definition of "MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)" is deleted.

2. Pages 3.2-5 and 3.2-6, Section 3.2.4: **[ARTS/MELLL]**

   Section 3.2.4, the Limiting Condition for Operation (LCO) and Surveillance Requirements (SRs) entitled "Average Power Range Monitor (APRM) Gain and Setpoints," is deleted in its entirety.

3. Page 3.3-1, Section 3.3.1.1: **[PRNM]**

   New notes are added for Required Action A.2 and for Condition B. Before the change, the affected parts of the table on page 3.3-1 read as follows:

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
		OR		
		A.2	Place associated trip system in trip.	12 hours
B.	One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
		OR		
		B.2	Place one trip system in trip.	6 hours

After the change, the affected parts of the table on page 3.3-1 will read as follows:

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
		OR		
		A.2	---------NOTE--------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. ---------------------- Place associated trip system in trip.	12 hours
B.	--------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. ------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
		OR		
		B.2	Place one trip system in trip.	6 hours

4. Page 3.3-3, Section 3.3.1.1: **[ARTS/MELLL]**

SR 3.3.1.1.2 is revised to delete the APRM gain adjustment required by LCO 3.2.4. Before the change, SR 3.3.1.1.2 reads as follows:

SURVEILLANCE	FREQUENCY
SR 3.3.1.1.2 ------------------NOTE------------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints" while operating at ≥ 25% RTP.	7 days

After the change, SR 3.3.1.1.2 will read as follows:

SURVEILLANCE	FREQUENCY
SR 3.3.1.1.2 ------------------NOTE------------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	7 days

5. Page 3.3-4, Section 3.3.1.1: **[PRNM]**

SR 3.3.1.1.9 is revised to delete the reference to Function 2.a. The section currently reads as:

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.9	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Functions 1 and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	92 days

After revision the SR will read as:

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.9	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	92 days

6. Page 3.3-5, Section 3.3.1.1: **[PRNM]**

For SR 3.3.1.1.11, delete the surveillance description and the frequency requirement.

7. Page 3.3-5, Section 3.3.1.1: **[PRNM]**

For SR 3.3.1.1.13, add new notes excluding the neutron detectors and including requirements for Function 2.a. After the change SR 3.3.1.1.13 will read as follows:

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	18 months

8. Page 3.3-5, Section 3.3.1.1: **[PRNM]**

In BFN Unit 1 TS, a new CHANNEL FUNCTIONAL TEST surveillance (SR 3.3.1.1.16) with frequency of 184 days is added. In BFN Unit 2 and Unit 3 TS, a new note related to requirements for Function 2.a. is added to the existing SR 3.3.1.1.16 After the change, SR 3.3.1.1.16 will be identical in BFN Unit 1, Unit 2, and Unit 3 TS and will read as follows:

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.16	------------------NOTES------------------ For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	184 days

9. Pages 3.3-6 and 3.3-7, Table 3.3.1.1-1: **[PRNM]**

The APRM Functions in Table 3.3.1.1-1 are revised extensively. The "Downscale" trip Function is deleted. A new "2-Out-Of-4 Voter" Function is added. A new footnote is added, applicable to the "Required Channels Per Trip System." The Required Channels Per Trip System is changed from "2" to "3" for the existing APRM functions. Surveillance Requirements

are revised. All of these changes are related to the PRNM modification and are consistent with the recommendations of Reference 2.

Page 3.3-6, Table 3.3.1.1-1: **[ARTS/MELLL]**

The Allowable Value (flow-biased function) for APRM Function 2.b is changed. This change is consistent with the recommendations of Reference 3.

Before the change, the affected parts of Table 3.3.1.1-1 read as follows:

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
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
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
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA

After the change, the affected parts of Table 3.3.1.1-1 and new footnote "b" will read as follows:

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
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. 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
d. Inop	1,2	3[(b)]	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA

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

10. Page 3.3-17, Section 3.3.2.1: [PRNM]

    The Frequency for SR 3.3.2.1.1 is changed from 92 days to 184 days.

11. Page 3.3-18, Section 3.3.2.1: [PRNM]

    The Frequency for SR 3.3.2.1.4 is changed from 92 days to 18 months.

12. Page 3.3-19, Section 3.3.2.1: **[ARTS/MELLL]**

A new Surveillance Requirement, SR 3.3.2.1.8 is added. This new SRs will read as follows:

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.8	-------------------NOTE------------------- Neutron detectors are excluded. -- Verify the RBM: a. Low Power Range -- Upscale Function is not bypassed when THERMAL POWER is ≥ 28% and ≤ 63% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 63% and ≤ 83% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 83% RTP.	18 months

13. Page 3.3-20, Table 3.3.2.1-1: **[ARTS/MELLL]**

Revise the Rod Block Monitor portion of Table 3.3.2.1-1 and associated notes to reflect the change from flow-biased to power-dependent RBM limits.

Before the change, affected portions of Table 3.3.2.1-1 and associated footnotes read as follows:

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale (Flow Biased)	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(e)
b. Inop	(a),(b)	2	SR 3.3.2.1.1	NA
c. Downscale	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	≥ 3% RTP

(a) THERMAL POWER ≥ 90% and MCPR < 1.40.

(b) THERMAL POWER ≥ 29% and < 90% and MCPR < 1.70.

....

(e) Less than or equal to the Allowable Value Specified in the COLR.

After the change, affected portions of Table 3.3.2.1-1 and associated footnotes will read as follows:

1. Rod Block Monitor				
a. Low Power Range — Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range — Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range — Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. Inop	(g),(h)	2	SR 3.3.2.1.1	NA
e. Downscale	(g),(h)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(i)

(a) THERMAL POWER ≥ 28% and ≤ 63% RTP and MCPR < 1.75.

(b) THERMAL POWER > 63% and ≤ 83% RTP and MCPR < 1.75.

....

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 83% and < 90% RTP and MCPR < 1.75.

(g) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(h) THERMAL POWER ≥ 28% AND < 90% RTP and MCPR < 1.75.

(i) Greater than or equal to the Allowable Valuc specified in the COLR.

14. Page 3.4-3, Section 3.4.1 **[ARTS/MELLL]**

In the Frequency for SR 3.4.1.2, the core flow value is changed from 45% to 50%.

15. Page 3.4-4, Section 3.4.1 **[ARTS/MELLL]**

Figure 3.4.1-1 is modified to expand the Region II area to include the power/flow map area between 45% and 50% core flow, and above the 108% rod line. Also, the figure is reformatted to improve readability.

16. Page 3.10-20, Section 3.10.8: **[PRNM]**

In LCO 3.10.8.a, a reference to item 2.d of Table 3.3.1.1-1 is added to correlate the LCO to the previous changes in the subject table.

17. Page 3.10-22, Section 3.10.8: **[PRNM]**

For SR 3.10.8.1, a reference to item 2.d of Table 3.3.1.1-1 is added to correlate the SR to the previous changes in the subject table.

18. Page B 3.2-1, Section B 3.2.1: **[ARTS/MELLL]**

A new reference, Reference 7, is added in the first paragraph of "Applicable Safety Analyses."

In the second paragraph of "Applicable Safety Analysis," the last sentence is deleted, and the following discussion is added:

APLHGR limits are developed as a function of exposure and various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Reference 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Reference 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the

transient response to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.

19. Page B 3.2-2, Section B 3.2.1: **[ARTS/MELLL]**

    The following is added at the end of the existing discussion of the LCO:

    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 limit.

20. Page B 3.2-3, Section B 3.2.1: **[ARTS/MELLL]**

    The following new references are added in the References section:

    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.

21. Pages B 3.2-4 & -5, Section B 3.2.2: **[ARTS/MELLL]**

    A new reference, Reference 8, is added in the first sentence of "Applicable Safety Analyses."

    The second paragraph of "Applicable Safety Analysis," is deleted, and the following discussion is added:

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 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 following sentence is added to the existing description of the LCO for Minimum Critical Power Ratio (MCPR):

The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

22. Page B 3.2-8, Section 3.2.2: **[ARTS/MELLL]**

The following new references are added in the References section:

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.

23. Pages B 3.2-12 through B 3.2-17: **[ARTS/MELLL]**

Section B 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoints, is deleted in its entirety.

24. Page B 3.3-6, Section B 3.3.1.1: **[PRNM]**

The Bases section entitled "Average Power Range Monitor" is rewritten both to provide appropriate descriptions for the new PRNM equipment and to consolidate the existing Bases information. The following is added as an initial discussion of the APRMs:

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any two unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs to each RPS trip system logic channel (A1, A2, B1, and B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM functions, at least twenty (20) LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

25. Pages B 3.3-6 & -7, Section B 3.3.1.1: **[PRNM]**

In the Bases description of APRM Function 2.a, the following text is deleted (moved to the discussion described under item 24, above):

The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power change. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.

In the Bases description of APRM Function 2.a, the following text is deleted (no longer applicable for the new PRNM design):

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux-High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

26. Pages B 3.3-8 & -9, Section B 3.3.1.1: **[PRNM]**

In the Bases description of APRM Function 2.b, the second paragraph is extensively changed both to delete no longer applicable information and to provide a new discussion of the core flow signals provided to the APRMs. The revised paragraph will read as follows:

Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part

of the APRM channel OPERABILITY requirements for this function.

27. Pages B 3.3-9 & -10, Section B 3.3.1.1: **[PRNM]**

    In the Bases description of APRM Function 2.c, the following text is deleted (moved to the discussion described under item 24, above):

    The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases.

    In the Bases description of APRM Function 2.c, the following text is deleted (no longer applicable for the new PRNM design):

    The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux-High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

28. Page B 3.3-11, Section B 3.3.1.1: **[PRNM]**

    The existing Bases description of APRM Function 2.d, Average Power Range Monitor -- Downscale, is deleted in its entirety.

29. Pages B 3.3-11 & -12, Section B 3.3.1.1: **[PRNM]**

    The existing Bases description of "Average Power Range Monitor -- Inop," APRM Function 2.e, is re-numbered as APRM Function 2.d. The re-numbered section is substantially re-written.

The first paragraph of this section is revised to read as follows:

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions. This Function (Inop) provides assurance that a minimum number of APRMs are OPERABLE. For any APRM channel, any time its mode switch is in any position other than "Operate," an APRM module is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The second paragraph of this section is deleted (no longer applicable for the new PRNM design).

30. Page B 3.3-12, Section B 3.3.1.1: **[PRNM]**

A new description of the 2-Out-Of-4 Voter Function is added, APRM Function 2.e. The new description will read as follows:

<u>2.e. 2-Out-Of-4 Voter</u>

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

31. BFN Unit 1 TS Pages B 3.3-21 & -22; BFN Units 2 & 3 TS Pages B 3.3-22 & -23, Section B 3.3.1.1: **[PRNM]**

Under the Bases discussion of Actions A.1 and A.2, a new reference, Reference 12, is added.

In addition, the following discussion of a new note is added as follows:

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

32. BFN Unit 1 TS Pages B 3.3-22 & -23; BFN Units 2 & 3 TS Page B 3.3-23, Section B 3.3.1.1: **[PRNM]**

Under the Bases discussion of Actions B.1 and B.2, a new reference, Reference 12, is added in two places. In addition, the following discussion of a new note is added:

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, and 2.d, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

33. BFN Unit 1 TS Pages B 3.3-25 & -26; BFN Units 2 & 3 TS Page B 3.3-26, Section B 3.3.1.1: **[ARTS/MELLL]**

Under the Bases discussion of SR 3.3.1.1.2, the following text is deleted:

LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD.

34. BFN Unit 1 TS Page B 3.3-28, Section B 3.3.1.1: **[PRNM]**

This change is made in BFN Unit 1 TS only.

The Bases title discussion of SR 3.3.1.1.8 and SR 3.3.1.1.12 is relabeled to include discussion of SR 3.3.1.1.16.

After this change, the SR Bases discussion titles are the same in BFN Unit 1, Unit 2, and Unit 3 TS.

35. BFN Unit 1 TS Page B 3.3-28; BFN Units 2 & 3 TS Page B 3.3-29, Section B 3.3.1.1: **[PRNM]**

The following discussion of SR 3.3.1.1.16 for APRM Functions is added into the applicable Bases section:

The 184 day frequency of SR 3.3.1.1.16 for the APRM Functions supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing -- applicable to Function 2.b, only), the 2-out-of-4 voter channels, and the interface connections into the RPS trip systems from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions is based on the reliability analysis of Reference 2. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.14.) A Note for

SR 3.3.1.1.16 is provided that requires the APRM Function 2.a SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

36. BFN Unit 1 TS Page B 3.3-29; BFN Units 2 & 3 TS Page B 3.3-30, Section B 3.3.1.1: **[PRNM]**

    In the Bases discussion of SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13, in three places the existing descriptions of notes for SR 3.3.1.1.9 are revised to include applicability for SR 3.3.1.1.13.

37. BFN Unit 1 TS Page B 3.3-30; BFN Units 2 & 3 TS Pages B 3.3-30 & -31, Section B 3.3.1.1: **[PRNM]**

    The description of SR 3.3.1.1.11 is deleted in its entirety. This SR number is designated as "Deleted."

38. BFN Unit 1 TS Page B 3.3-30; BFN Units 2 & 3 TS Page B 3.3-31, Section B 3.3.1.1: **[PRNM]**

    The following description for testing of APRM Function 2.e is added to the existing description of requirements for SR 3.3.1.1.14:

    The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

39. Page B 3.3-32, Section B 3.3.1.1: **[PRNM]**

    The following reference is added to the existing list of references:

    12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

40. Page B 3.3-42, Section B 3.3.2.1: **[PRNM]**

In the Background section, the following text is deleted:

A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system.

The deleted text is replaced with the following:

A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate or high) is enabled.

41. Pages B 3.3-43 & -44, Section B 3.3.2.1: **[ARTS/MELLL]**

The section entitled "Applicable Safety Analyses, LCO, and Applicability" is changed as follows:

The following sentences are deleted:

Note that the RBM setpoint is flow-biased until implementation of ARTS improvements described in Reference 3. However, the generic RWE analysis in Reference 3 is currently applicable to establish required conditions of RBM OPERABILITY.

The deleted sentences are replaced with the following:

The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

The phrase "for the associated power range" is added to the first sentence of the third paragraph in this section. After the change, the sentence will read as follows:

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that

no single instrument failure can preclude a rod block from this Function.

In the first sentence of the fifth paragraph in this section, the limit above which the RBM is assumed to mitigate the consequences of an Rod Withdraw Error (RWE) event is changed from "≥ 29% Rated Thermal Power (RTP)" to "≥ 28% RTP." Also, in the same paragraph, the referenced MCPR values of 1.70 and 1.40 are changed to 1.75 and 1.44 respectively.

42. Page B 3.3-49, Section B 3.3.2.1: **[PRNM]**

    In the description of SR 3.3.2.1.1, the Frequency is changed from "92 days" to "184 days." The related reference is changed from "Ref. 8" to "Ref. 11."

43. Page B 3.3-50, Section B 3.3.2.1: **[PRNM]**

    In the description of SR 3.3.2.1.4, the assumed calibration interval is changed from "a 184 day" to "an 18 month."

44. Page B 3.3-51, Section B 3.3.2.1: **[ARTS/MELLL]**

    A description of new SR 3.3.2.1.8 is added. The new description will read as follows:

    <u>SR 3.3.2.1.8</u>

    The RBM Setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1 and the COLR, each within a specific power range. The powers at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately

tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

45. Page B 3.3-52, Section B 3.3.2.1: **[PRNM]**

The following new reference is added to the References section:

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

46. Pages B 3.4-3 and B 3.4-7, Section B 3.4.1: **[ARTS/MELLL]**

In the Applicable Safety Analyses and SR 3.4.1.2 bases, a 50% core flow is substituted for the current 45% core flow value.

47. Page B 3.10-34, Section B 3.10.8: **[PRNM]**

In the Bases for LCO 3.10.8, a new reference to Function 2.d in Table 3.3.1.1-1 is added.

48. Page B 3.10-37, Section B 3.10.8: **[PRNM]**

A new reference to Function 2.d in Table 3.3.1.1-1 is added to the Bases discussion of SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3.

II. <u>REASON FOR THE PROPOSED CHANGE</u>

The proposed TS changes are related either to the replacement of the existing power range neutron monitoring equipment or to implementation of ARTS/MELLL improvements. In the discussion below, the reasons for the PRNM-related changes are discussed first.

As stated in Reference 10, BFN is planning to replace the power range monitor portion of the NMS with a GE digital NUMAC PRNM retrofit system. The new equipment will include capability for an automatic Oscillation Power Range Monitor (OPRM) trip to detect and suppress possible thermal hydraulic instabilities in the reactor. The new OPRM trip function, when enabled, will implement the Boiling Water Reactor Owners Group (BWROG) defined "Stability Option III" alternative. The OPRM portion of the system will be

operated in the "indicate only" mode during each unit's first cycle of operation, and OPRM stability trip function will not be enabled until the next cycle of operation. Therefore, this proposed change in TS does not include revisions to incorporate the Stability Option III automatic trip function. All other PRNM functions will be operable, and this package provides the required TS changes for these functions.

The planned modification involves replacement of all of the existing power range monitor electronics with new NUMAC digital PRNM hardware. The current equipment is mounted in a 5-bay panel in the main control room of each reactor unit. The modification removes and replaces virtually all of the existing power range monitor equipment within the confines of the main control room panels, but with minor exceptions, leaves the plant level cabling and interfaces undisturbed.

All power range monitor functions are maintained, including LPRM detector signal processing, LPRM averaging, APRM trips, and RBM logic and interlocks. Recirculation flow signal processing, previously accomplished using separate hardware within the power range monitor control panels, is integrated into the APRM chassis in the new PRNM system.

The six existing APRM channels in the current system are replaced with four APRM channels, each using 1/4 of the total LPRM detectors. The APRM function is retained, but four 2-out-of-4 trip output voters are added to the input to the RPS, two in each RPS trip system. The trip outputs from all four APRM channels are sent to each voter so that each of the inputs to the RPS is a voted result of all four APRM channels. The number of recirculation flow instrument inputs to the APRMs is increased from two total-flow instrument loops (four transmitters) to four total-flow instrument loops (eight transmitters), permitting one recirculation total-flow instrument loop to be assigned to each APRM channel.

The reasons for the various individual proposed TS changes associated with the PRNM replacement are as follows:

In LCO 3.3.1.1, Table 3.3.1.1-1, the required minimum number of operable instrument channels for the APRM high and inoperable scram trip functions is changed from 2 to 3 because the new configuration will have 4 total APRM channels combined in a 2-out-of-4 logic. In the proposed configuration, a minimum of 3 of the 4 channels is required operable to meet single failure criteria for the RPS trips

initiated by APRMs. Note "b" is added to Table 3.3.1.1-1 to highlight the fact that, in the new configuration, each APRM instrument channel provides input to both RPS trip systems.

In LCO 3.3.1.1, Table 3.3.1.1-1, the requirement for an APRM downscale scram trip function is deleted. The APRM downscale scram trip is not credited with performing any safety function, and deletion of this APRM function is justified in Reference 2.

In LCO 3.3.1.1, Table 3.3.1.1-1, a new 2-out-of-4 voter function is added. The 2-out-of-4 voter function requires a minimum of 2 operable instrument channels per RPS trip system. This requirement is consistent with the proposed new hardware configuration. There are 2 voters per RPS trip system, and requiring 2 voters operable in each of the two RPS trip systems ensures that single failure criteria is met. Because operability of the voters is required whenever any other APRM trip function is required, the applicable modes for voter operability are MODE 1 and MODE 2. Inoperability of one or more voters results in entry to Condition A, Condition B, or Condition C, as appropriate. Failure to complete the required actions within the allowable completion times requires that the reactor be in MODE 3 (where APRM operability is not required) within 12 hours.

In LCO 3.3.1.1, Table 3.3.1.1-1, the following changes are made to surveillance requirements for APRM Functions 2.a, 2.b, 2.c and 2.d (previously numbered "2.e"), and surveillance requirements for the new APRM Function 2.e (2-Out-Of-4 Voter) are added.

For APRM Function 2.a, Neutron Flux - High, Setdown, the following changes are made: The Channel Functional Test (SR 3.3.1.1.3) with 7-day frequency is deleted; in its place a Channel Functional Test (SR 3.3.1.1.16) with a 184-day frequency is added. The Channel Calibration (SR 3.3.1.1.9) with a 92-day frequency and the corresponding reference in SR 3.3.1.1.9 are deleted; in their place the Channel Calibration (SR 3.3.1.1.13) with an 18-month frequency is added. The Logic System Functional Test (SR 3.3.1.1.14) with an 18-month frequency is deleted; testing of the logic for Function 2.a will be included as part of the Logic System Functional Test (SR 3.3.1.1.14) of the 2-Out-Of-4 Voter (new APRM Function 2.e). These changes in surveillance frequency are supported by the reliability analysis presented in Reference 2 and are consistent with the recommendations of Reference 2.

For APRM Function 2.b, Flow Biased Simulated Thermal Power - High, the following changes are made: The Channel Functional Test (SR 3.3.1.1.8) with 92-day frequency is deleted; in its place the Channel Functional Test (SR 3.3.1.1.16) with 184-day frequency is added. The Channel Calibration (SR 3.3.1.1.9) with 92-day frequency and the corresponding reference in SR 3.3.1.1.9 are deleted; in their place the Channel Calibration (SR 3.3.1.1.13) with 18-month frequency is added. The flow signal calibration (SR 3.3.1.1.11) with 18-month frequency is deleted as a separate item; the flow signal calibration will be included as part of SR 3.3.1.1.13. The Logic System Functional Test (SR 3.3.1.1.14) with 18-month frequency is deleted; testing of the logic for Function 2.b will be included as part of the Logic System Functional Test (SR 3.3.1.1.14) of the 2-Out-Of-4 Voter (new APRM Function 2.e). These changes in surveillance frequency are supported by the reliability analysis presented in Reference 2 and are consistent with the recommendations of Reference 2.

For APRM Function 2.c, Neutron Flux - High, the following changes are made: The Channel Functional Test (SR 3.3.1.1.8) with 92-day frequency is deleted; in its place the Channel Functional Test (SR 3.3.1.1.16) with 184-day frequency is added. The Channel Calibration (SR 3.3.1.1.9) with 92-day frequency and the corresponding reference in SR 3.3.1.1.9 are deleted; in their place the Channel Calibration (SR 3.3.1.1.13) with 18-month frequency is added. The Logic System Functional Test (SR 3.3.1.1.14) with 18-month frequency is deleted; testing of the logic for Function 2.c will be included as part of the Logic System Functional Test (SR 3.3.1.1.14) of the 2-Out-Of-4 Voter (new APRM Function 2.e). These changes in surveillance frequency are supported by the reliability analysis presented in Reference 2 and are consistent with the recommendations of Reference 2.

For re-numbered APRM Function 2.d, Inop, the following changes are made: The calibration of local power range monitors (SR 3.3.1.1.7) is deleted; this calibration remains a requirement of APRM Functions 2.a, 2.b and 2.c, where the local power range monitors provide direct inputs to the process signals monitored by the APRM trip functions. The Channel Functional Test (SR 3.3.1.1.8) with 92-day frequency is deleted; in its place the Channel Functional Test (SR 3.3.1.1.16) with 184-day frequency is added. The Logic System Functional Test (SR 3.3.1.1.14) with 18-month frequency is deleted; testing of the logic for Function 2.d will be included as part of the Logic

System Functional Test (SR 3.3.1.1.14) of the 2-Out-Of-4 Voter (new APRM Function 2.e). These changes in surveillance frequency are supported by the reliability analysis presented in Reference 2 and are consistent with the recommendations of Reference 2.

For new APRM Function 2.e, 2-Out-Of-4 Voter, the following surveillance requirements are specified: A Channel Check (SR 3.3.1.1.1) with frequency of 24-hours is specified; this is consistent with the established Channel Check frequency for the other APRM Functions. A Logic System Functional Test (SR 3.3.1.1.14) with frequency of 18-months is specified; this test will include testing the logic of APRM Functions 2.a, 2.b, 2.c and 2.d together with the logic of APRM Function 2.e. A Channel Functional Test (SR 3.3.1.1.16) with frequency of 184-days is specified. These surveillance requirements and frequencies are supported by the reliability analysis presented in Reference 2 and are consistent with the recommendations of Reference 2.

In LCO 3.3.1.1, Table 3.3.1.1-1, the Allowable Value of APRM Function 2.b is changed. This change, part of the ARTS/MELLL changes, is discussed later in this section.

In the Actions for LCO 3.3.1.1 a new note is added to state that Required Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, or 2.d. A similar note is also added to Condition B. As described in a revised Bases discussion, neither Required Action A.2 nor Condition B is applicable for APRM Functions 2.a, 2.b, 2.c, or 2.d. Required Action A.2 is not applicable because in the new configuration inoperability of one APRM channel affects both RPS trip systems. Thus, for an inoperable APRM channel, Required Action A.1 must be satisfied and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Condition B also is not applicable because inoperability of more than one required APRM channel results in loss of trip capability; thus, in this circumstance entry is required into Condition C, as well as into Condition A for each channel.

In the table of SRs for LCO 3.3.1.1, the note in SR 3.3.1.1.9 is revised to remove the APRM Function 2.a reference which is now addressed in SR 3.3.1.1.13. Also, the description of SR 3.3.1.1.11 and the frequency are deleted. This change reflects inclusion of the recirculation flow loop calibrations as part of the overall Channel Calibration (SR 3.3.1.1.13) for APRM Function 2.b.

In the table of SRs for LCO 3.3.1.1, new notes are added to SR 3.3.1.1.13. These notes exclude neutron detectors from the Channel Calibration and allow 12 hours to complete the requirement for APRM Function 2.a when entering MODE 2 from MODE 1. As discussed in the Bases section, the neutron detectors are excluded from Channel Calibration because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Because testing of APRM Function 2.a cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links, the note provides 12 hours in which to bring current Channel Calibration for APRM Function 2.a when entering MODE 2 from MODE 1; the twelve hours is based on operating experience and consideration of providing a reasonable time in which to complete the SR.

In BFN Unit 1 TS, a new Channel Functional Test (SR 3.3.1.1.16) with 184-day frequency is added. In BFN Unit 2 and Unit 3 TS in the table of Surveillance Requirements for LCO 3.3.1.1, a new note is added to the existing Channel Functional Test (SR 3.3.1.1.16) with 184-day frequency. After these changes the BFN Unit 1, Unit 2, and Unit 3 surveillance requirements are identical. The new note allows 12 hours to complete the requirement for APRM Function 2.a when entering MODE 2 from MODE 1. The reason for the note is as described in the preceding paragraph for SR 3.3.1.1.13.

In the SRs for LCO 3.3.2.1, the frequency of SR 3.3.2.1.1 (Rod Block Monitor Channel Functional Test) is changed from 92 days to 184 days. This change in surveillance frequency is supported by the reliability analysis presented in Reference 2 and is consistent with the recommendations of Reference 2.

In the SRs for LCO 3.3.2.1, the frequency of SR 3.3.2.1.4 (Rod Block Monitor Channel Calibration) is changed from 92 days to 18 months. This change in surveillance frequency is supported by the reliability analysis presented in Reference 2 and is consistent with the recommendations of Reference 2.

(Note: The remainder of the changes to TS Section 3.3.2.1 are related to ARTS/MELLL implementation and are discussed separately, below.)

In LCO 3.10.8.a and SR 3.10.8.1 relating to RPS instrumentation required during shutdown margin tests, a reference to Function 2.d (now the "Inop" function) is added to stay consistent with the previous changes to Table 3.3.1.1-1. The existing 2.e reference in LCO 3.10.8.a and SR 3.10.8.1 which previously referenced the "Inop" function now corresponds to the new "2-Out-of-4 Voter" function in Table 3.3.1.1-1.

In the following paragraphs the reasons for the ARTS/MELLL-related TS changes are discussed. These changes are proposed concurrent with the NUMAC PRNM related changes because equipment design, interface and setpoint modifications required to implement these proposed TS changes will be included as part of the NUMAC PRNM design.

Implementation of the ARTS improvements requires physical modification of the RBM system. The proposed modification changes the RBM trip setpoints from flow-biased to power-biased values and reconfigures the LPRM inputs to the RBMs. These proposed changes are intended to eliminate limitations of the current RBM system, which was designed in the mid-1960s. Since that time there have been significant advances in the fields of two-phase heat transfer and electronics. The current RBM signals do not always correlate well with thermal margin changes during control rod withdrawal, and the system performs its function only at the expense of significant operational penalties due to excessive conservatism inherent in the design of the system. The modified RBM system will provide improved correlation of RBM response with changes in fuel thermal margin and will enhance operator confidence in the system by reducing the frequency of nonessential rod blocks. Also, the proposed changes will upgrade the performance of the RBM system and will provide new RBM setpoint and operability requirements such that the Rod Withdrawal Error will not be the limiting transient.

In addition to changes in the RBM system's configuration, setpoints and operability requirements, ARTS improvements eliminate the current TS requirement to lower (setdown) the flow-biased APRM scram and rod block trips when the MFLPD exceeds the Fraction of Rated Power (FRP). To support elimination of this requirement, as well as to support the change to power-biased RBM setpoints, new power-dependent and flow-dependent fuel thermal limits are

proposed to be implemented. The proposed replacement of the current APRM trip setdown requirement by more meaningful power- and flow-dependent thermal limits eliminates a need for manual setpoint adjustments and is anticipated to enhance administration of thermal limits compliance.

The proposed expansion of allowable operation to the MELLL region provides enhanced ability to achieve and maintain operation at rated power. Because rated power can be maintained with recirculation flow adjustments over a wider flow range, less frequent control rod adjustments are required to compensate for reactivity depletion, and the need for power reductions to perform control rod withdrawal is decreased. The plant will be able to operate longer at rated power, will have more flexibility to schedule load reductions and will be able to operate in a more safe, efficient, and economical manner.

The reasons for the various individual TS changes associated with ARTS/MELLL implementation are as follows:

TS Section 1.1 is revised to delete the definition of "Maximum Fraction of Limiting Power Density (MFLPD)." This definition is no longer needed because LCO 3.2.4, APRM Gain and Setpoints, and SRs which use this definition are deleted.

LCO 3.2.4, APRM Gain and Setpoints, and related SRs are deleted as justified by the evaluation in Section 5.3 of Reference 3. Deletion of this requirement is supported by implementation of new power- and flow-dependent limits for Average Planar Linear Heat Generation Rate (APLHGR) and Minimum Critical Power Ratio (MCPR). The proposed deletion of this LCO eliminates the need to make APRM setpoint or gain adjustments based on peaking factors. Eliminating this requirement reduces administrative and manpower burdens and eliminates the risks of spurious trips associated with the previously required APRM adjustments.

In LCO 3.3.1.1, Table 3.3.1.1-1, the Allowable Value for APRM Function 2.b, Flow Biased Simulated Thermal Power - High, is changed from "$\leq$ 0.58W + 62% RTP and $\leq$ 120% RTP" to "$\leq$ 0.66W + 71% RTP and $\leq$ 120% RTP." This change supports operation in the MELLL region by providing adequate operating margin between boundaries of the MELLL region and the flow-biased APRM scram. This Allowable Value is based on the Analytical Limit of "$\leq$ 0.66W +73% RTP and $\leq$ 122% RTP" as documented in References 3 and 12. The Allowable Value is calculated from the Analytical Limit using the

plant-specific setpoint methodology of Reference 12. The APRM flow-biased rod block setpoint in the Core Operating Limits Report (COLR) is similarly revised as part of the proposed change.

SR 3.3.1.1.2 is revised to delete reference to APRM gain adjustments required by LCO 3.2.4. This change reconciles the wording of the SR with deletion of LCO 3.2.4 described above.

In LCO 3.3.2.1, Table 3.3.2.1-1, the RBM requirements are revised. Three power-dependent upscale functions and associated setpoints are added: Low Power Range - Upscale, Intermediate Power Range - Upscale, and High Power Range - Upscale. Applicable existing footnotes are revised and new footnotes are added to provide appropriate definitions of Applicable Modes Or Other Specified Conditions. The RBM portion of revised Table 3.3.2.1-1 is consistent with the same table in BWR/4 ISTS (NUREG-1433, Revision 1), except for changes in footnote numbering and placement of the RBM setpoint Allowable Values in the COLR. An adjustment to the MCPR values in footnotes a, b, f, g, and h was made to account the use of an MCPR Safety Limit (SL) of 1.10 in ISTS SL 2.1.1.2 compared to the SL of 1.07 used in Reference 3. The applicable ranges for the RBM Functions are plant-specific values based on the analyses in References 3 and 12.

Figure 3.4.1-1, Thermal Power Versus Core Flow Stability Regions, is modified to expand Region II to include the power/flow map segment between 45% and 50% core flow, and above the 108% rod line. This change is made to preserve the BFN commitment to use the BWROG Guidelines for Stability Interim Corrective Actions. The BWROG Guidelines include this added Region II segment for plants operating under the expanded MELLL power/flow map. In SR 3.4.1.2, the 45% core flow value is changed to 50% to correspond to the change in Figure 3.4.1-1.

Bases Sections B 3.2.1, B 3.2.2, B 3.2.4, B 3.3.1.1, B 3.3.2.1, B 3.4.1, and B 3.10.8 have been revised to reflect the TS changes described above. For the PRNM related TS changes, the revised Bases incorporates the recommended wording provided in Reference 2. For the ARTS/MELLL related TS changes, the revised Bases incorporates the recommended wording provided in the generic BWR/4 ISTS (NUREG-1433, Revision 1), which are modeled for plants on which the ARTS improvement have been incorporated.

III. SAFETY ANALYSIS

This Section discusses the safety analyses associated with replacement of the existing power range neutron monitoring system with a GE digital NUMAC PRNM retrofit design. GE Licensing Topical NEDC-32410P provides detailed descriptions, discussion, bases, and data applicable to the GE NUMAC PRNM retrofit designs. NRC has reviewed the subject Topical Report, (issued as NEDC-32410P-A, Reference 2) and issued a Safety Evaluation Report (SER) (Reference 11) which indicated the acceptability of the generic design and accompanying TS changes. In the SER, NRC requested that licensees address six plant specific issues to take credit for the evaluations provided in Topical Report. Attachment 1 of this enclosure provides TVA's responses to the actions listed in the SER.

A safety evaluation of the proposed modifications and TS changes is summarized below:

For the functions addressed by the proposed TS change, the NUMAC PRNM has the same design basis requirements as the original power range neutron monitoring system. The original system was designed to meet IEEE 279-1971 compliance; therefore the requirements of this standard apply to the replacement design. In addition, USNRC Regulatory Guide 1.152 -1985 is applied as a requirement, and Reference 2 includes a "compliance matrix" that correlates the requirements of the Regulatory Guide to the NUMAC PRNM implementing program. Section 4 of Reference 2 further discusses the design bases and regulatory requirements applicable for the NUMAC PRNM system.

All previous APRM upscale scram trips are retained in the new design. The proposed design and related LCOs permit one APRM instrument channel to be bypassed at any time for maintenance or testing while retaining the ability to withstand single failure of one of the remaining instrument channels. However, because of requirements to meet single failure criteria, bypass of any 2-out-of-4 voter is not permitted. LCO action times and required actions for fewer than the required number of APRM trip functions or 2-out-of-4 voter functions are consistent with what has been previously approved for the BWR/4 ISTS.

The proposed change deletes the previously required APRM downscale scram trip in the Run mode. No postulated event takes credit for this downscale trip, and eliminating the logic for the trip reduces the potential for spurious

scrams in testing, maintenance or operation. This change was recommended in Reference 2.

The proposed TS change extends required surveillance intervals of the APRM and RBM equipment to the maximum periods supported by Reference 2. This reduction in surveillance frequency is supported by the increased reliability and the extensive self-test capability of the new hardware. Extending APRM surveillance intervals reduces the potential for spurious trips while testing is being performed, thus enhancing the overall reliability of plant operations.

The following discusses the proposed changes associated with implementing ARTS improvements and with expanding operation to the MELLL region of the power/flow map. The NRC has previously approved implementation of ARTS/MELLL changes at other BWRs (References 4, 5, 6 and 7) and has also approved expanding BFN's original operating region to the Extended Load Line Limit region (References 8 and 9).

Reference 3 documents the results of analyses and evaluations performed for BFN by GE to support the proposed ARTS/MELLL changes. Appendix A of Reference 3 discusses major features of the modified RBM system, and Section 10 provides an in-depth discussion of the RBM system evaluation and requirements to support the ARTS improvement. Sections 4 and 5 provide a description of APRM improvements and a detailed discussion of the new power- and flow-dependent thermal limits which support elimination of the previous APRM setpoint setdown requirements. Reference 3 also provides documentation of extensive analyses of operation in the MELLL region performed for BFN based on the Unit 2, Cycle 8 fuel core loading. Specific allowable setpoints were generated in a separate GE calculation for BFN (Reference 12) and will be documented in the COLR report for individual fuel cycles. As discussed in Reference 3, the analyses performed typically yield generic limits which will be applicable to future core reloads. Appropriate portions of these analyses will be reconfirmed in cycle-specific core reload analyses.

The evaluations documented in Reference 3 to justify the safety of operation in the MELLL region consist of two segments. One segment is that which is not fuel dependent. The other segment is that which is fuel dependent, and therefore, fuel cycle dependent. In general, the limiting anticipated operational occurrences (AOOs) MCPR calculation and the reactor vessel overpressure protection analysis are

fuel cycle dependent. These analyses as presented in Reference 3 are based on the BFN Unit 2, Cycle 8, core loading at the current rated core thermal power of 3293 megawatts thermal (MWt). For non-fuel dependent evaluations such as containment responses, an uprated power level of 3458 MWt (105% of the current rated core thermal power) is used. The non-fuel dependent evaluations are based on hardware design, geometries, and system performance which are similar among the BFN units. Thus, these non-fuel dependent evaluation are generically applicable to BFN Units 1, 2, and 3 for MELLL region operation. As noted previously, appropriate portions of these analyses will be reconfirmed in the cycle-specific reload analyses. Changes to setpoints, if necessary, will be presented in the cycle specific COLRs in the same manner currently in place for cycle-specific limit changes.

Evaluations of the specific changes associated with ARTS/MELLL implementation are summarized below:

The proposed TS change revises the flow-biased APRM scram setpoint Allowable Value from "≤ 0.58W + 62% RTP and ≤ 120% RTP" to "≤ 0.66W + 71% RTP and ≤ 120% RTP.". The flow-biased APRM scram setpoint maximum (clamped) Allowable Value of 120% does not change. In addition, the flow-biased APRM rod block setpoints documented in the COLR will be changed. These changes incorporate new setpoints for the flow-biased APRM scram and rod block functions based on the MELLL Analytical Limits documented in Reference 3 and the setpoint calculations of Reference 12, which are based on incorporation of the new NUMAC PRNM hardware.

For original plant operation with the maximum load line limited to the rated rod line, the setpoint for the flow-biased APRM scram line was ≤ 0.66W + 54% RTP. With the first expansion of the power/flow map to allow operation up to the 108% rod line (references 8 and 9), the flow-biased APRM flux scram line was modified to ≤ 0.58W + 62% RTP. With the proposed expansion of the power/flow map to include the MELLL region depicted in Figure 2-1 of Reference 3, the upper boundary of the analyzed operating domain is further extended to the 121% rod line. The proposed change in flow-biased APRM setpoints maximizes plant operating flexibility, restores the slopes of the flow-biased APRM scram and rod block setpoints to their original design basis values, and restores the original design basis operating margin between the maximum extended load line and the APRM flow-biased scram setpoint.

The purpose of the flow-biased APRM rod block is to block control rod withdrawal when core power exceeds rated conditions and approaches the scram level. Should operation continue in a manner such that the power/flow condition exceeds that specified by the APRM rod block setpoint, the flow-biased APRM scram trip setpoint would initiate a scram.

The proposed TS change revises LCO 3.3.2.1, Table 3.3.2.1-1 to reflect the change from flow-biased to power-dependent RBM setpoints. The revisions to Table 3.3.2.1-1 are consistent with the presentation in BWR/4 ISTS, Revision 1. The RBM system is designed specifically to mitigate the consequences of the RWE event and is not assumed to be available to mitigate any other AOOs. The current RBM system configuration is described in detail in Section 10 of Reference 3. The modified RBM system configuration is also described in Section 10 and Appendix A of Reference 3. The modified RBM system uses advances in electronics to enhance instrumentation accuracy and improve the signal to thermal margin correlation. The RBM modifications incorporate power-dependent setpoints and provide a system response which more accurately reflects the actual margin to the safety limit at various power conditions.

Coincident with the modification of the RBM system, analyses were performed which generically bound the consequences of the RWE event. These analyses established boundaries of power level and operating MCPR value. Outside these values, no RWE event can result in exceeding the MCPR safety limit or jeopardizing the fuel thermal-mechanical design limits. Inside these boundaries credit for the actions of the RBM is taken to limit the consequences of an RWE event. This approach to analysis of the RWE event included determining appropriate MCPR requirements and corresponding RBM power-dependent setpoints for the modified RBM system for current fuel designs. By appropriate selection of the setpoints, the RWE will not be the limiting event and will not determine the operating limit MCPR. Appropriate RBM setpoints are selected based upon cycle-specific MCPR operating limit values (as determined from non-RWE events). The RBM setpoints are, thus, reload-design dependent and are documented in the COLR.

The TS change revises several footnotes in Table 3.3.2.1-1 to specify RTP and MCPR conditions where operability of the RBM system is not required. Section 10.5 of Reference 3 documents that with RTP $\geq$ 90% and operating MCPR $\geq$ 1.40, or with RTP < 90% and operating MCPR $\geq$ 1.70, withdrawal of any

single control rod from the full-in to the full-out position will not result in violation of the MCPR safety limit. The proposed ISTS change adds additional MCPR margin (resulting in MCPR values of 1.44 and 1.75) to account for increases for specific reload core analyses over the base 1.07 MCPR Safety Limit used in Reference 3 and as used in SL 2.1.1.2. Thus, under these upper limit conditions, the RBM system is not required to function in order to assure that an RWE has acceptable results.

The TS change deletes the requirements for flow-biased APRM scram and rod block setpoint setdown, or APRM gain adjustment under conditions when MFLPD is greater than the Fraction of RTP (LCO 3.2.4); and, in lieu of these requirements, implements flow- and power-dependent operating thermal limits for APLHGR and MCPR. Related Bases sections are revised to reflect deletion of LCO 3.2.4 and to include discussion of this new treatment of thermal limits. Specifically, the change eliminates the requirement for setdown of the flow-biased APRM scram and rod block trip setpoints when the MFLPD is greater than the Fraction of RRP and substitutes adjustments to the rated MCPR and APLHGR operating limits that are flow- and power-dependent. Analyses documented in Reference 3 demonstrate that with the setpoint setdown requirement eliminated and flow- and power-dependent thermal limits implemented, 1) MCPR safety limit will not be violated as a result of any AOOs, 2) all fuel thermal-mechanical design bases will remain within the licensing limits described in the GE generic fuel licensing report GESTAR-II, and 3) peak cladding temperature and maximum cladding oxidation fraction following a Loss of Coolant Accident (LOCA) will remain within the limits defined in 10 CFR 50.46.

Figure 3.4.1-1, Thermal Power Versus Core Flow Stability Regions, is modified to expand Region II to include the power/flow map segment between 45% and 50% core flow, and above the 108% rod line. This change is made to maintain compatibility with the improved BWROG Guidelines for Stability Interim Corrective Actions for plants operating under a MELLL expanded power/flow map. Prior to implementation of MELLL, the existing APRM rod blocks and load line limitations physically restricted reactor entry in this area. With the MELLL expanded power/flow map and increased rod block lines, it becomes possible to operate in this area. The BWROG Guidelines, however, classify this area as a Region II restricted area, and to stay consistent with BFN's commitments to implement the BWROG Stability Guidelines, the TS Figure is modified to include this power/flow map region as Region II area. SR 3.4.1.2 is also modified to increase the SR domain from 45% to 50%

core flow to match the change in the Figure 3.4.1-1 core flow value.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3 in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   PRNM Changes: These proposed TS changes are associated with the NUMAC PRNM retrofit design. The proposed TS changes involve modification of the LCOs and SRs for equipment designed to mitigate events which result in power increase transients. For the APRM system mitigative action is to block control rod withdrawal or initiate a reactor scram which terminates the power increase when setpoints are exceeded. For the RBM system mitigative action is to block continuous control rod withdrawal prior to exceeding the MCPR safety limit during a postulated RWE. The worst case failure of either the APRM or the RBM systems is failure to initiate mitigative action (failure to scram or block rod withdrawal). Failure to initiate mitigative action will not increase the probability of an accident. Thus, the proposed changes do not increase the probability of an accident previously evaluated.

   For the APRM and the RBM systems, the NUMAC PRNM design, together with revised operability requirements (LCOs) and revised testing requirements (SRs), results in equipment which continues to perform the same mitigation functions conditions with reliability equal to or greater than the equipment which it replaces. Because there is no change in mitigation functions and because reliability of the functions is maintained, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

   ARTS/MELLL Changes: These proposed changes are associated with implementation of the ARTS/MELLL analyses. The proposed changes permit expansion of the

current allowable power/flow operating region and will apply a new methodology for assuring that fuel thermal and mechanical design limits are satisfied. Reference 3 evaluates operation in the MELLL region with assumed implementation of the ARTS changes. The conclusion of Reference 3 is that for all events and parameters considered, there is adequate design margin for operation in the MELLL region. Because operation in the MELLL region maintains adequate design margin, the proposed changes do not increase the probability of an accident previously evaluated.

In support of operation in the MELLL region, the proposed change modifies flow-biased APRM scram and rod block setpoints and implements new RBM power-biased setpoints. No direct credit for the flow-biased APRM scram or APRM flow-biased rod block is taken in mitigation of any design basis event, although it affords an additional margin to thermal limits for events that result in slow power increases such as loss of feedwater heaters. Reference 3 includes a reanalysis of applicable events and concludes design margins are not degraded by the proposed changes.

The proposed changes to the RBM system potentially impact mitigation of the RWE. However, per discussion in Reference 3, the proposed RBM changes, will assure that the RWE is not a limiting event and that the RBM continues to enforce rod blocks under appropriate conditions.

Therefore, the proposed changes do not increase the probability or the consequences of an accident previously evaluated.

2. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed PRNM and ARTS/MELLL changes involve modification and replacement of the existing power range neutron monitoring equipment, modification of the setpoints and operational requirements for the APRM and RBM systems, implementation of a new methodology for administering compliance with fuel thermal limits, and operation in an extended power/flow domain. These proposed changes do not modify the basic functional requirements of the affected equipment, create any new system interfaces or interactions, nor create any new system failure modes or sequence of events that could

lead to an accident. The worst case failure of the affected equipment is failure to perform a mitigation action, and failure of this mitigative equipment does not create the possibility of a new or different kind of accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed amendment does not involve a significant reduction in a margin of safety.**

PRNM Changes: These proposed TS changes are associated with the NUMAC PRNM retrofit design. The NUMAC PRNM change does not impact reactor operating parameters or the functional requirements of the power range neutron monitoring system. The replacement equipment continues to provide information, enforce control rod blocks, and initiate reactor scrams under appropriate specified conditions. The proposed change does not revise any safety margin requirements. The replacement APRM/RBM equipment has improved channel trip accuracy compared to the current system, and meets or exceeds system requirements previously assumed in setpoint analysis. Thus, the ability of the new equipment to enforce compliance with margins of safety equals or exceeds the ability of the equipment which it replaces. Therefore, the proposed changes do not involve a reduction in a margin of safety.

ARTS/MELLL Changes: These proposed changes are associated with implementation of recommendations presented in the ARTS/MELLL analyses (Reference 3). Operation in the MELLL region does not affect the ability of the plant safety-related trips or equipment to perform their functions, nor does it cause any significant increase in offsite radiation doses resulting from any analyzed event. Analyses documented in Reference 3 demonstrate that, for operation in the MELLL region, adequate margin to design limits is maintained. Implementation of the ARTS improvements provides flow- and power-dependent thermal limits which maintain existing margins of safety in normal operation, anticipated operational occurrences, and accident events. Implementation of power-biased RBM setpoints improves the margin of safety in a postulated RWE by assuring that the RWE is not a limiting event. Thus, the proposed changes do not involve a reduction in a margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or an increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

VI. REFERENCES

1. TVA letter to NRC, dated June 2, 1995, Browns Ferry Nuclear Plant (BFN), - Units 1, 2, and Unit 3 - Technical Specification (TS) 353 - Power Range Neutron Monitor (PRNM) Upgrade With Implementation of Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) TS (ARTS) Improvements and Maximum Extended Load Line Limit (MELLL) Analyses.

2. Licensing Topical Report, Nuclear Measurement Analysis And Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function, Volumes 1 and 2, NEDC-32410P-A, October 1995. Including applicable parts of NEDC-32410P, Supplement 1, May 1996.

3. GE Report, Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3, NEDC-32433P

4. Letter from NRC to Carolina Power & Light Company, dated October 23, 1990, Issuance of Amendment No. 147 to Facility Operating License No. DPR-71 Regarding Maximum Extended Operating Domain.

5. Letter from NRC to Carolina Power & Light Company, dated October 12, 1989, Issuance of Amendment No. 168 to Facility Operating License No. DPR-62 Regarding Maximum Extended Operating Domain.

6. Letter from NRC to Detroit Edison Company, dated May 15, 1991, Amendment No. 69 to Facility Operating License No. NPF-43.

7. Letter from NRC to PECO Energy Company, dated August 10, 1994, Expanded Operating Domain (ARTS/MELLLA) Technical Specifications, Peach Bottom Atomic Power Station, Unit 2.

8. Letter from NRC to TVA, dated December 18, 1990, Issuance of Amendment (TAC No. 76934) (TS 285) [Extended Load Line Limit Analysis - Amendment 181 to BFN Unit 2 Technical Specifications]

9. Letter from NRC to TVA, dated February 24, 1995, Issuance of Technical Specification Amendment for the Browns Ferry Nuclear Plant Units 1, 2 and 3 (TAC Nos. M89251, M89252, and M89253) (TS 339) [Extended Load Line Limit and Revised Rod Block Monitor Operability Requirements (Units 1 and 3); Deletion of Specific Values.... (Units 1, 2, and 3)]

10. Letter from TVA to NRC, dated July 10, 1996, Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Installation Schedule for the Stability Long-Term Solution For NRC Generic Letter (GL) 94-02

11. Letter from NRC to General Electric, dated September 5, 1995, Acceptance of Licensing Topical Report NEDC-32410P, Nuclear Measurement Analysis and Control Power Range Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function (TAC No. M90616)

12. GE Calculation, APRM Neutron Flux, Flow-Biased and Scram Clamp and Rod Block, and RBM Neutron Flux Downscale, Power and Trip Setdown Calculations, ARTS/MELLL(NUMAC) ---- Current-Rated Condition for Tennessee Valley Authority, Browns Ferry Nuclear Plant, EDE-28-0990, Revision 1, Supplement F, August 1995

13. Letter from TVA to NRC, dated September 6, 1996, Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specification (TS) Change TS-362 - Request to Convert Current TS to Improved Standard TS (ISTS) Consistent with NUREG-1433, Revision 1

ENCLOSURE 1

ATTACHMENT 1

Plant-Specific Information Required for NUMAC PRNM Retrofit

TS-353S1

Plant-Specific Information Required for NUMAC PRNM Retrofit

The following information is provided to address the six Plant-Specific Actions listed in Section 5 of the NRC Safety Evaluation Report (SER) (Reference 11 of Enclosure 1) for NEDC-32410P-A.

Plant Specific Actions

1) Confirm the applicability of NEDC-32410, including clarifications and reconciled differences between the specific plant design and the licensing topical report (LTR) design descriptions.

   Response

   Design descriptions in Licensing Topical Report NEDC-32410P-A are directly applicable both for the existing power range monitoring system and for the proposed Power Range Neutron Monitoring System (PRNMS) modification at Browns Ferry Nuclear Plant (BFN). Specific sections of NEDC-32410P-A which describe the current and the proposed BFN design configurations are identified as follows: (Applicable LTR sections and paragraph numbers are listed.)

	Current	Proposed
APRM Configuration	2.3.3.1.1-2	2.3.3.1.2-2
RBM Configuration	2.3.3.2.1-1	2.3.3.2.2-1
Recirculation Flow Channels	2.3.3.3.1-1	2.3.3.3.2-2
Rod Control System Interface	2.3.3.4.1-1	2.3.3.4.2-1
ARTS	2.3.3.5.1-2	2.3.3.5.2-1
Operator's Panel Interface	2.3.3.6.1-1	2.3.3.6.2-1

2) Confirm the applicability of BWROG topical reports that address PRNMS and associated instability functions, setpoints and margins.

   Response

   The PRNMS modification for BFN implements the instability detection algorithms described in Boiling Water Reactor Owners Group (BWROG) topical reports NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," June 1991, and NEDO-31960 Supplement 1, "BWR Owners' Group Long

Term Stability Solutions Licensing Methodology," March 1992. The final Oscillation Power Range Monitor (OPRM) setpoints and margins will be confirmed in accordance with the methodology described in BWROG topical report NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1995. Any action required as a result of the NRC's review of NEDO-32465-A will be addressed as part of a separate OPRM TS change submittal to be provided to the NRC for review prior to activation of the OPRM trip function in the PRNMS.

3) Provide plant-specific revised Technical Specifications (TS) for the PRNMS functions consistent with NEDC-32410, Appendix H.

Response

BFN's proposed TS change TS-353S1 contains two groups of changes. One group of changes is related to the proposed PRNMS modification; these changes are labeled **"[PRNM]"** in Enclosure 1, Section I. The other group of changes is related to the proposed, concurrent implementation of ARTS/MELLL improvements; these changes are labeled **"[ARTS/MELLL]"** in Enclosure 1, Section I. The TS-353S1 changes are based on the TS-362 submittal which is currently under NRC review. TS-362 converts BFN's custom TS to the BWR/4 ISTS format, consistent with NUREG-1433, Revision 1.

The PRNMS-related changes proposed in TS-353S1 are very similar to the examples presented in NEDC-32410P-A, Appendix H, Section H.1.1, "Example of Changes for Improved Standard Tech Specs." Specific differences are tabulated in this attachment's Table 1, "Comparison of TS-353S1 with NEDC-32410P-A."

Table 2 provides an itemized summary of the ARTS/MELLL-related changes and identifies page numbers of NEDC-32433P, April 1995, related to the changes. The NRC SER did not specifically request a comparison of the ARTS/MELLL TS changes. Since, however, in some cases, PRNMS changes and ARTS/MELLL changes are associated, TVA is providing this comparison to assist NRC in their review.

4) Confirm that the plant-specific environmental conditions are enveloped by the PRNM equipment environmental qualification values.

Response

BFN's design change process and implementing procedures require documentation or demonstration that the environmental conditions at the mounted location of safety-related components are within the environmental qualification envelope of those components. Thus, as part of the normal design change process, BFN's environmental conditions, at applicable locations, will be confirmed to be within the envelope of the PRNMS equipment environmental qualification values. Specific parameters of the PRNMS environmental qualification envelope are discussed as follows:

Temperature: The PRNM control room electronics is qualified for continuous operation in the temperature range 5 to 50 °C (41 to 122 °F); these are the same temperatures to which the existing power range monitoring equipment is qualified. Normal control room temperature is 76 °F, and maximum abnormal temperature is 104 °F. Thus, allowing for temperature differences between the ambient room and the mounting panels plus reasonable heat loads expected of the PRNM equipment (less than for the existing equipment), the PRNM control room electronics will be well within the qualified range.

Humidity: The PRNM control room electronics is qualified for continuous operation in a humidity range of 10% to 90%. Normal control room relative humidity is in the range of 40% to 60%, which is well within the range for which the PRNM equipment is qualified.

Pressure: The PRNM control room electronics is qualified for continuous operation in a pressure range of 13 to 16 pounds per square inch absolute (-0.4 to +1.7 pounds per square inch gauge). Normal control room pressure is maintained slightly higher than atmospheric pressure, between 0.125" and approximately 0.5" of water (gauge). This is within the qualified pressure range.

Radiation: The PRNM control room electronics is qualified for continuous operation with a Dose Rate ≤ 0.001 Rads/hr and Total Integrated Dose (TID) ≤ 1000 Rads. Control room dose rates and TID are within the qualified range.

Seismic Acceleration: Calculations confirm that the maximum seismic accelerations at the mounting locations of the PRNM equipment do not exceed the qualification limits of that equipment.

Electromagnetic Interference: TVA has previously provided information to the NRC by letter dated December 23, 1993, Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3 -- Reply to NRC Request to Provide Results of BFN Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) Tests and On-Site Surveys for the Reactor Building Ventilation Radiation Monitoring System (TAC Nos. M84161, M84162 and M84163). Based on the previously provided test results, the BFN environment satisfies the conditions and limitations defined in EPRI TI-102323 and is within the qualification envelope of the PRNM equipment.

5) Confirm that administrative controls are provided for manually bypassing APRM/OPRM channels or protective functions, and for controlling access to the panel and the APRM/OPRM channel bypass switch.

Response

The level and method of control of bypass of the APRM/OPRM channels is the same as for the current APRM channels. At BFN the APRM/OPRM channel bypass switch is located in the Main Control Room and is under the direct control and supervision of the

licensed Unit Operator. The bypass switch cannot be accessed without the knowledge and permission of the Unit Operator. The Unit Operator's knowledge of TS requirements for operability of the APRM/OPRM channels, together with Conduct of Operations procedures and PRNMS operating or surveillance instructions, provides adequate administrative controls for manually bypassing APRM/OPRM channels or protective functions, and for controlling access to the panel and the APRM/OPRM channel bypass switch.

6) Confirm that any changes to the plant operator's panel have received human factors reviews per plant-specific procedures.

Response

BFN's design change process and implementing procedures require completion of a Human Factors Engineering (HFE) Process Checklist and performance of an HFE review of changes to the plant operator's panel. An HFE review, per applicable procedures, of the proposed changes to the operator's panel will be performed, and documentation of that review will be included in the final design change package(s) for the PRNMS.

Table 1
Comparison of TS-353S1 with NEDC-32410P-A, Appendix H

Enclosure 1 Section I Item #	NEDC-32410P-A Appendix H Page #	Comparison of TS-353S1 with NEDC-32410P-A
3	H-3	TS-353S1 implements the changes as shown in the LTR.
5	H-7	BFN SR 3.3.1.1.9 is revised to be specific to IRMs since a separate APRM calibration SR has been created as revised SR 3.3.1.1.13.
6	H-5	TS-353S1 implements the change shown in the LTR for SR 3.3.1.1.3. In BFN TS the comparable surveillance requirement is numbered SR 3.3.1.1.11.
7	H-7	TS-353S1 revises BFN TS SR 3.3.1.1.13 to match the wording of SR 3.3.1.1.13 as presented in the LTR for this CHANNEL CALIBRATION with 18-month frequency.
8	H-7	TS-353S1 revises BFN TS SR 3.3.1.1.16 to match the wording of SR 3.3.1.1.11 as presented in the LTR for this CHANNEL FUNCTIONAL TEST with 184-day frequency.
9	H-9 & 10	TS-353S1 revises the APRM requirements of Table 3.3.1.1-1 to match the requirements as presented in the LTR. Numbering of the BFN SRs and footnote differs from the LTR example; but the content of the requirements is identical. BFN chooses to retain the existing APRM Function descriptions, rather than revise them to the wording of the LTR.
10	H-12	TS-353S1 implements the change as shown in the LTR.
11	H-13	TS-353S1 revises BFN SR 3.3.2.1.4 to match the wording of SR 3.3.2.1.7 as presented in the LTR for this CHANNEL CALIBRATION with 18-month frequency.
16	N/A	LCO 3.10.8.a is revised to correlate with the revisions to Table 3.3.1.1-1
17	N/A	SR 3.10.8.1 is revised to correlate with the revisions to Table 3.3.1.1-1
24	H-15	TS-353S1 revises the APRM Bases description to match the wording as presented in the LTR.
25	H-15 & 16	TS-353S1 revises the Bases description of APRM Function 2.a to match the wording as presented in the LTR, except that the existing name of this function is retained.
26	H-17 & 18	TS-353S1 revises the Bases description of APRM Function 2.b to match the wording as presented in the LTR, except that the existing name of this function is retained.
27	H-19	TS-353S1 revises the Bases description of APRM Function 2.c to match the wording as presented in the LTR, except that the existing na[illegible] of this function is retained.
28	H-20 & 21	TS-353S1 implements the changes a[illegible]wn in the LTR.
29	H-21	TS-353S1 implements the changes a[illegible]own in the LTR.
30	H-22	TS-353S1 implements the changes [illegible]own in the LTR.
31	H-23	TS-353S1 implements the changes [illegible]own in the LTR, except for a difference in the numbering of the reference.

Table 1 (continued)
Comparison of TS-353S1 with NEDC-32410P-A, Appendix H

Enclosure 1 Section I Item #	NEDC-32410P-A Appendix H Page #	Comparison of TS-353S1 with NEDC-32410P-A
32	H-24 & 25	TS-353S1 implements the changes as shown in the LTR, except for a difference in the numbering of the reference.
34	N/A	TS-353S1 changes the title of this Bases section in Unit 1 TS to make it identical to the Unit 2 and Unit 3 TS.
35	H-30	TS-353S1 implements changes comparable to what is shown in the LTR Bases discussion of SR 3.3.1.1.11. Differences are due to differences in numbering of the SRs and in the Bases discussion formats.
36	N/A	TS-353S1 revises this Bases discussion to make it applicable for both SR 3.3.1.1.9 and SR 3.3.1.1.13.
37	H-27	TS-353S1 deletes the discussion of SR 3.3.1.1.11; this is the same as deletion of the Bases discussion of SR 3.3.1.1.3 in the LTR.
38	H-32	TS-353S1 revises the Bases discussion of SR 3.3.1.1.14 to incorporate the changes shown in the LTR for SR 3.3.1.1.15. Differences are due to differences in numbering of the SRs.
39	H-34	TS-353S1 revises the list of references to include the changes as shown in the LTR. Differences are due to differences in numbering of the references.
40	H-35	TS-353S1 implements the changes as shown in the LTR.
42	H-36	TS-353S1 implements the changes as shown in the LTR. Differences are due to differences in numbering of the references.
43	H-37	TS-353S1 revises the RBM calibration interval to 18 months. This is not shown as a change in the LTR because the LTR example specification already included the 18 month interval.
45	H-39	TS-353S1 incorporates the reference shown in the LTR, differences are due to different numbering of the references.
47	N/A	The Bases for LCO 3.10.8 are revised to correlate with the revisions to Table 3.3.1.1-1
48	N/A	The Bases for SR 3.10.8.1 are revised to correlate with the revisions to Table 3.3.1.1-1

Table 2
Comparison of TS-353S1 with BWR/4 ISTS, NUREG-1433, Revision 1 and with Recommendations of NEDC-32433P

Enclosure 1 Section I Item #	BWR/4 ISTS, Revision 1 Section #	Comparison of TS-353S1 with BWR/4 ISTS, NUREG-1433, Revision 1 and with Recommendations of NEDC-32433P, MELLL and ARTS Improvement Program Analysis for Browns Ferry Units 1, 2, and 3
1	1.1	The definition of MFLPD is indicated as "optional" in the BWR/4 ISTS. This definition is deleted by TS-353S1 because it is no longer required in BFN TS due to ARTS/MELLL Change.
2	3.2.4	LCO 3.2.4 for Average Power Range Monitor (APRM) Gain and Setpoints is indicated as optional in the BWR/4 ISTS. This section is deleted by TS-353S1, consistent with the recommendation of NEDC-32433P, Section 11, Item (1).
4	SR 3.3.1.1.2	BWR/4 ISTS indicates that the deleted phrase is optional. TS-353S1 makes this change consistent with deletion of LCO 3.2.4, as described above.
9	Table 3.3.1.1-1	TS-353S1 revises the APRM flow-biased scram setpoint consistent with the recommendations of NEDC-32422P, Section 11, Item (7).
12	SR 3.3.2.1.8	TS-353S1 adds a new surveillance requirement to support the change to power-dependent RBM setpoints. This SR is the same as SR 3.3.2.1.4 in the BWR/4 ISTS. This change is consistent with NEDC-32422P, Section 11, Items (8) and (9).
13	Table 3.3.2.1-1	TS-353S1 revises the RBM requirements similar to the requirements in Table 3.3.2.1-1 of BWR/4 ISTS. This change is consistent with NEDC-32422P, Section 11, Items (8) and (9). RBM setpoints values continue to be documented in the COLR. An adjustment to the MCPR values in footnotes a, b, f, g, and h was made to account for an MCPR Safety Limit (SL) of 1.10 in ISTS SL 2.1.1.2 versus that used in NEDC-32422P (1.07).
14	N/A	In the FREQUENCY column for SR 3.4.1.2, the core flow is changed from 45% to 50% to match revised Figure 3.4.1-1. See the next item for discussion.
15	N/A	The Power/Flow stability map is expanded to add the map segment between 45% and 50% core flow, and above the 108% rod line. ISTS does not have an equivalent figure. The revision is made to maintain current commitments to BWROG Stability Monitoring Guidelines pending activation of the OPRM. This change is consistent with the NEDC-32422P discussion in Section 9.2.
18	B 3.2.1	TS-353S1 revises the Bases discussion of APLHGR limits to reflect the change to power and flow dependent limits. The revised wording the same as the discussion in BWR/4 ISTS. This change is consistent with NEDC-32433P, Section 11, Items (4), (5), and (10).
19	B 3.2.1	TS-353S1 revises the Bases discussion of APLHGR LCO to reflect the change to power and flow dependent limits. The revised wording the same as the discussion in BWR/4 ISTS.
20	B 3.2.1	TS-353S1 revises the references in the APLHGR Bases section to include the new ARTS/MELLL analysis and include additional references identified in BWR/4 ISTS.
21	B 3.2.2	TS-353S1 revises the Bases discussion of MCPR operating limits and LCO to reflect the change to power and flow dependent limits. The revised wording is the same as the discussion in BWR/4 ISTS. This change is consistent with NEDC-32422P, Section 11, Items (2), (3) and (10).

Table 2 (continued)
Comparison of TS-353S1 with BWR/4 ISTS, NUREG-1433, Revision 1 and with Recommendations of NEDC-32433P

Enclosure 1 Section I Item #	BWR/4 ISTS, Revision 1 Section #	Comparison of TS-353S1 with BWR/4 ISTS, NUREG-1433, Revision 1 and with Recommendations of NEDC-32433P, MELLL and ARTS Improvement Program Analysis for Browns Ferry Units 1, 2, and 3
22	B 3.2.2	TS-353S1 revises the references in the MCPR Bases section to include the new ARTS/MELLL analysis and includes an additional reference identified in BWR/4 ISTS.
23	B 3.2.4	TS-353S1 deletes the Bases discussion of LCO 3.2.4 because the LCO is deleted. This is consistent with NEDC-32422P, Section 11, Items (1) and (10).
33	B 3.3.1.1	TS-353S1 deletes a Bases reference to LCO 3.2.4 which is deleted. This is consistent with NEDC-32433P, Section 11, Items (1) and (10).
41	B 3.3.2.1	TS-353S1 revises the Bases discussion of the RBM to reflect implementation of ARTS improvements. When revised, the Bases discussion is similar to the discussion in BWR/4 ISTS.
44	B 3.3.2.1	TS-353S1 revises the Bases discussion of the RBM SR 3.3.2.1.8 to reflect implementation of ARTS improvements. When revised, the Bases discussion is similar to the comparable discussion in BWR/4 ISTS. This change is consistant with NEDC-32433P, Section 11, Item (10).
46	N/A	In the Bases discussion for the power/flow stability map, the core flow value is changed from 45% to 50% to corrospond with changes to SR 3.4.1.2 and Figure 3.4.1-1. The power/flow map Region II stability monitoring area expands to meet BWROG Guidelines for MELLL plants. This change is consistent with the discussion in Section 9.2 of NEDC-32433P. BWR/4 ISTS do not have a corresponding Bases section.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY (TVA)

BROWNS FERRY NUCLEAR PLANT (BFN)

UNITS 1, 2, and 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-353S1

MARKED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
1.1-4 3.2-5 3.2-6 3.3-1 3.3-3 through 3.3-7 3.3-17 through 3.3-20 3.4-3 3.4-4 3.10-20 3.10.22 B 3.2-1 through B 3.2-5 B 3.2-8 B 3.2-12 through B 3.2-17 B 3.3-6 through B 3.3-12 B 3.3-21 through B 3.3-23 B 3.3-25 B 3.3-26 B 3.3-28 through B 3.3-30 B 3.3-32 B 3.3-42 through B 3.3-44 B 3.3-49 through B 3.3-52 B 3.4-3 B 3.4-7 B 3.10-34 B 3.10-37	1.1-4 3.2-5 3.2-6 3.3-1 3.3-3 through 3.3-7 3.3-17 through 3.3-20 3.4-3 3.4-4 3.10-20 3.10.22 B 3.2-1 through B 3.2-5 B 3.2-8 B 3.2-12 through B 3.2-17 B 3.3-6 through B 3.3-12 B 3.3-22 B 3.3-23 B 3.3-26 B 3.3-29 through B 3.3-32 B 3.3-42 through B 3.3-44 B 3.3-49 through B 3.3-52 B 3.4-3 B 3.4-7 B 3.10-34 B 3.10-37	1.1-4 3.2-5 3.2-6 3.3-1 3.3-3 through 3.3-7 3.3-17 through 3.3-20 3.4-3 3.4-4 3.10-20 3.10-22 B 3.2-1 through B 3.2-5 B 3.2-8 B 3.2-12 through B 3.2-17 B 3.3-6 through B 3.3-12 B 3.3-22 B 3.3-23 B 3.3-26 B 3.3-29 through B 3.3-32 B 3.3-42 through B 3.3-44 B 3.3-49 through B 3.3-52 B 3.4-3 B 3.4-7 B 3.10-34 B 3.10-37

II. MARKED PAGES

See attached.

TS - 353S1

UNIT 1

MARKED UP PAGES

TABLE OF CONTENTS

1.1 Definitions (continued)

LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
~~MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)~~	~~The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP; or

b. Each required APRM setpoint specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

Delete Page

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-------------------NOTE------------------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements. -- Verify MFLPD is within limits.	Once within 12 hours after ≥ 25% RTP **AND** 24 hours thereafter
SR 3.2.4.2	-------------------NOTE------------------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements. -- Verify APRM setpoints or gains are adjusted for the calculated MFLPD.	12 hours

Delete Page

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable. ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. ------------------------	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. ------------------------ 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTES-------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP ~~plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints"~~ while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	-----------------NOTE----------------- Only required to be met during entry into MODE 2 from MODE 1. -------------------------------------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 effective full power hours
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	-----------------NOTES----------------- 1. Neutron detectors are excluded. 2. For Function~~s~~ 1 ~~and 2~~ not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -------------------------------------- Perform CHANNEL CALIBRATION.	92 days

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	Deleted ~~(Not Used.)~~ 2 ~~Adjust the channel to conform to a calibrated flow signal.~~ e	~~18 months~~ e
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.13	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	18 months
SR 3.3.1.1.16	NOTE: For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. Perform CHANNEL FUNCTIONAL TEST	184 days

single line

NOTES
1. Neutron detectors are excluded.
2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.

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) ~~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 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b) ~~2~~	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.14~~	≤ 0.66W + 71% RTP ~~0.58 W + 62% RTP~~ and ≤ 120% RTP
c. Neutron Flux - High	1	3(b) ~~2~~	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 ~~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) Each APRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
~~d. Downscale~~ e	~~1~~	~~2~~	~~F~~	~~SR 3.3.1.1.7~~ ~~SR 3.3.1.1.8~~ ~~SR 3.3.1.1.14~~	~~≥ 3% RTP~~
d ~~e.~~ Inop	1,2	3(b) ~~2~~ e	G	~~SR 3.3.1.1.7~~ e SR 3.3.1.1.~~8~~ 16 ~~SR 3.3.1.1.14~~ e	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(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.

e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	N/A

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more Reactor Mode Switch - Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal.	Immediately
	<u>AND</u>	
	E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE	FREQUENCY
SR 3.3.2.1.1 Perform CHANNEL FUNCTIONAL TEST.	184 ~~92~~ days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	18 months ~~92 days~~ e
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	18 months
SR 3.3.2.1.6	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.7	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM
SR 3.3.2.1.8	-----NOTE----- Neutron detectors are excluded. ---------- Verify the RBM: a. Low Power Range – Upscale Function is not bypassed when THERMAL POWER is ≥ 28% and ≤ 63% RTP. b. Intermediate Power Range – Upscale Function is not bypassed when THERMAL POWER is > 63% and ≤ 83% RTP. c. High Power Range – Upscale Function is not bypassed when THERMAL POWER is > 83% RTP.	18 months

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
~~a. Upscale (Flow Biased)~~	~~(a),(b)~~	~~2~~	~~SR 3.3.2.1.1~~ ~~SR 3.3.2.1.4~~	~~(e)~~
d. ~~b.~~ Inop	(g),(h) ~~(a),(b)~~	2	SR 3.3.2.1.1	NA
e. ~~c.~~ Downscale	(g),(h) ~~(a),(b)~~	2	SR 3.3.2.1.1 SR 3.3.2.1.4	~~≥ 3% RTP~~ (i)
2. Rod Worth Minimizer	$1^{(c)}$, $2^{(c)}$	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch - Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ~~≥ 90% RTP and MCPR ≥ 1.44~~ ≥ 28% and ≤ 63% RTP and MCPR < 1.75.

(b) THERMAL POWER ~~≥ 29% and < 90% RTP~~ > 63% and ≤ 83% RTP and MCPR < ~~1.70~~ 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 83% and < 90% RTP and MCPR < 1.75.

(g) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(h) THERMAL POWER ≥ 28% and < 90% RTP and MCPR < 1.75.

(i) Greater than or equal to the Allowable Value specified in the COLR.

a. Low Power Range - Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range - Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	--------------------NOTE------------------ Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
SR 3.4.1.2	Verify the reactor is outside of Region I and II of Figure 3.4.1-1.	Immediately after any increase > 5% RTP while initial core flow is < ~~45%~~ of rated. 50% AND Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated.

Replace with
improved Figure

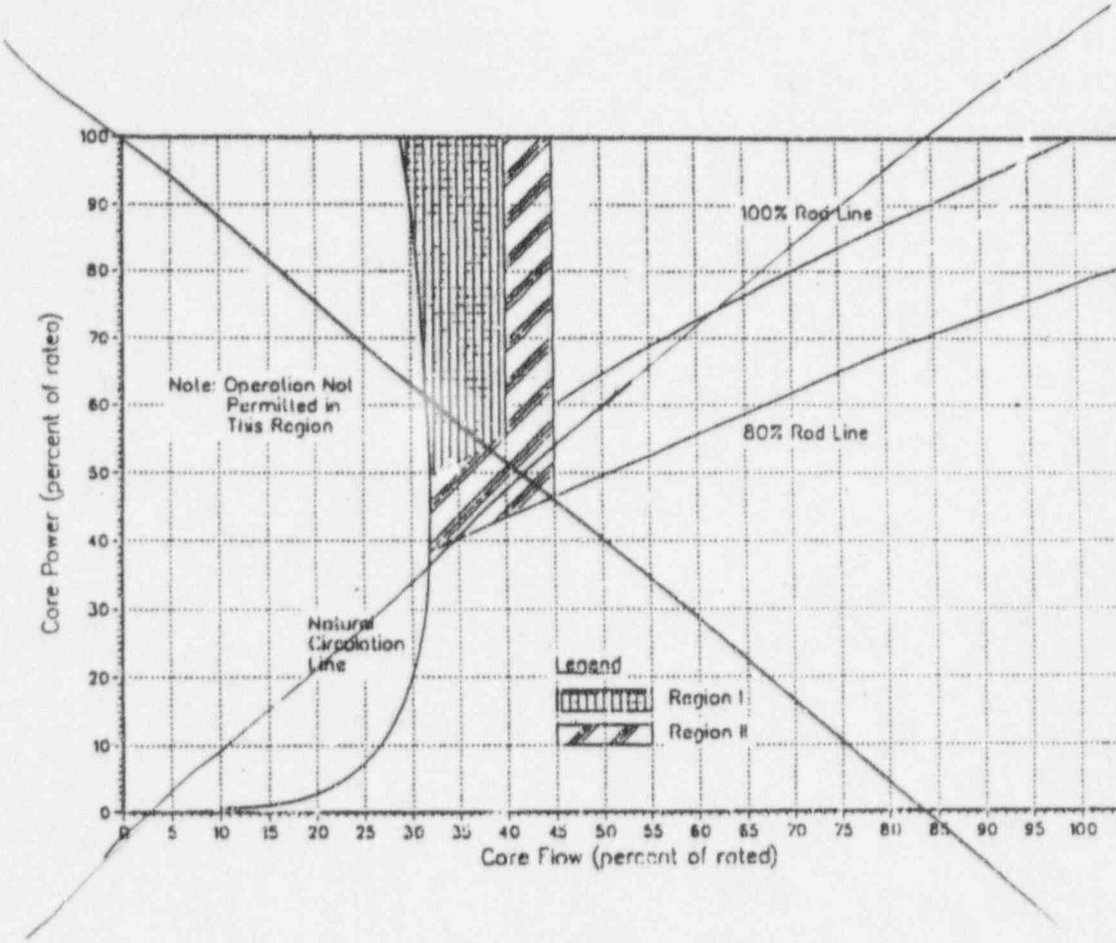

Figure 3.4.1-1

THERMAL POWER VERSUS CORE FLOW
STABILITY REGIONS

Replacement Figure

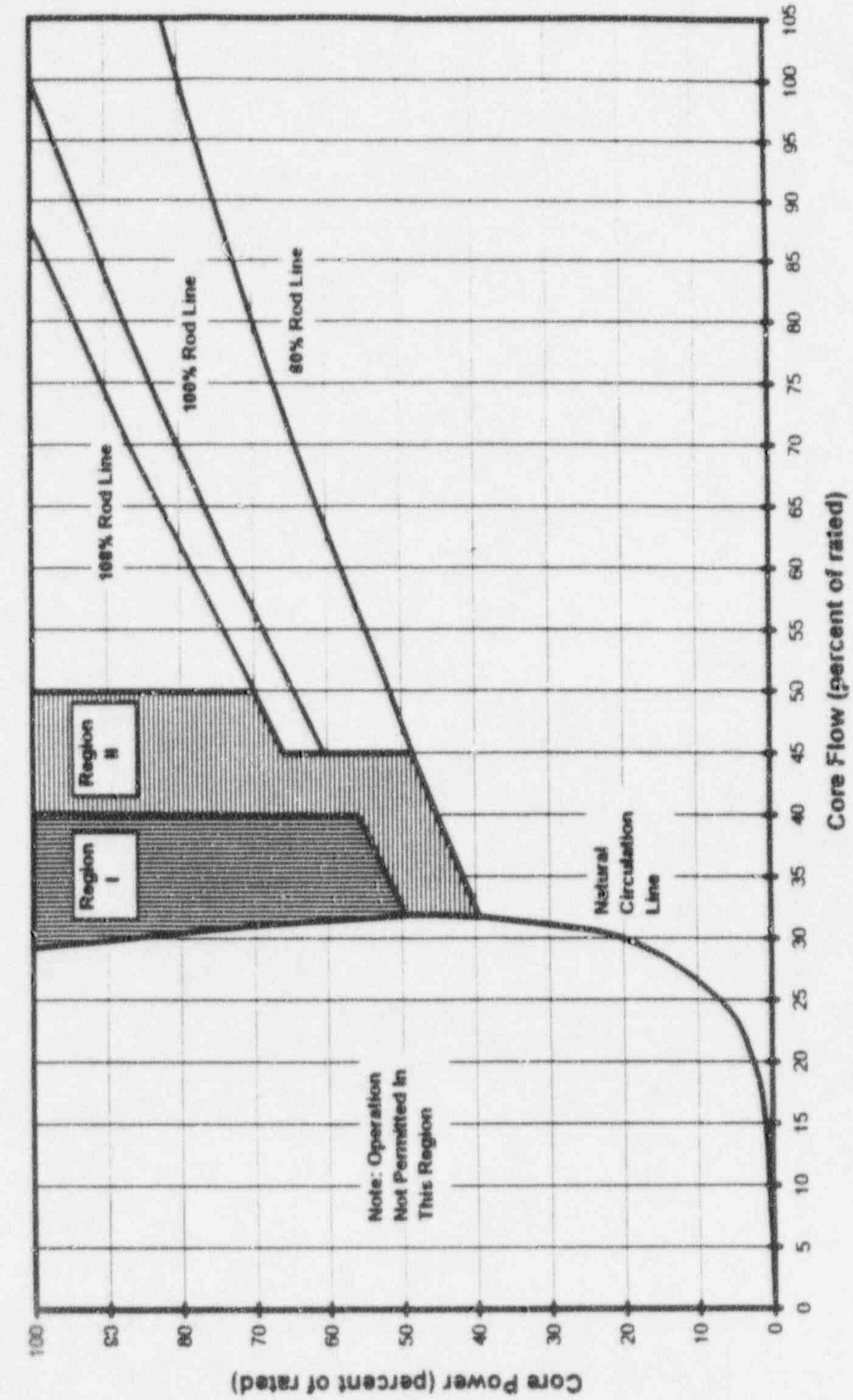

Figure 3.4.1-1

THERMAL POWER VERSUS
CORE FLOW STABILITY
REGIONS

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Functions 2.a , 2.d and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence (BPWS) requirements of SR 3.3.2.1.7 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of BPWS control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.e of Table 3.3.1.1-1. 2.d,	According to the applicable SRs
SR 3.10.8.2	------------------NOTE------------------ Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	------------------NOTE------------------ Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

<u>TABLE OF CONTENTS</u>

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, ~~and~~ 4, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. ~~APLHGR limits are developed as a function of exposure and fuel bundle type.~~

Insert A

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

(continued)

INSERT A:

APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Reference 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Reference 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.

BASES

APPLICABLE SAFETY ANALYSES (continued)

conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

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.

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 $\leq$ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

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.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-11 "General Electric Standard Application for Reactor Fuel," November 1995.
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 1, February 1996.
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.

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

, and 8.

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, and 5. 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~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.~~

Insert B

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.

The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the

(continued)

INSERT B:

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.

BASES

REFERENCES (continued)

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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection System Failure Modes" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

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The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{MFLPD}{FRTP} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow.

(continued)

BASES

BACKGROUND (continued)

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses presented in References 1 and 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

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The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR and MCPR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR and MCPR) occurs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

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The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. 4).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

(continued)

BASES

LCO (continued)

c. Increasing APRM gains to cause the APRM to read ≥ 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

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For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

(continued)

BASES

ACTIONS

A.1 (continued)

may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

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B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared with FRTP, or APRM gains or setpoint, to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.2.4.1 and SR 3.2.4.2</u> (continued) The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to FRP. When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23. 2. FSAR, Chapter 14.. 3. FSAR, Chapter 3. 4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux - High (continued)

unexpected reactivity excursions. In MODE 1, the APRM System and the RBM provide protection against control rod withdrawal error events and the IRMs are not required.

1.b. Intermediate Range Monitor - Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor - Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux - High Function is required.

Insert B

Average Power Range Monitor

2.a. Average Power Range Monitor Neutron Flux - High, Setdown

~~The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.~~ For operation at

(continued)

INSERT C:

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs to each RPS trip system logic channel (A1, A2, B1, or B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM functions, at least twenty (20) LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>2.a. Average Power Range Monitor Neutron Flux—High, Setdown</u> (continued)

low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux – High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux – High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux – High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux – High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux – High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

~~The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux – High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux – High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux – High, Setdown (continued)

In MODE 1, the Average Power Range Monitor Neutron Flux – High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power – High

The Average Power Range Monitor Flow Biased Simulated Thermal Power – High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux – High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power – High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux – High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power – High Function setpoint is exceeded.

~~The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

~~Average Power Range Monitor Flow Biased Simulated Thermal Power - High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~ e

15 Each APRM channel receives a total drive flow signal representative of total core flow. uses one

~~The total drive flow signals are generated by two flow units, one of which supplies signals to the trip system A APRMs, while the other one supplies signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power - High channel requires an input from its associated OPERABLE flow unit.~~ e

the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this function.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux - High

~~The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases.~~ e The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor Fixed Neutron Flux — High (continued)

trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux — High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux — High Function to terminate the CRDA.

~~The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux — High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~ e

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux — High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux — High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux — High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux — High Function is not required in MODE 2.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~2.d. Average Power Range Monitor — Downscale~~

~~This signal ensures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to the APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associated Intermediate Range Monitor Neutron Flux — High or Inop signal generates a trip signal. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.~~

~~The APRM System is divided into two groups of channels with three inputs into each trip system. The system is designed to allow one channel in each trip system to be bypassed. Four channels of Average Power Range Monitor — Downscale with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. The Intermediate Range Monitor Neutron Flux — High and Inop Functions are also part of the OPERABILITY of the Average Power Range Monitor — Downscale Function (i.e., if either of these IRM Functions cannot send a signal to the Average Power Range Monitor — Downscale Function, the associated Average Power Range Monitor — Downscale channel is considered inoperable).~~

~~The Allowable Value is based upon ensuring that the APRMs are in the linear scale range when transfers are made between APRMs and IRMs.~~

~~This Function is required to be OPERABLE in MODE 1 since this is when the APRMs are the primary indicators of reactor power.~~

d

2.~~e~~. Average Power Range Monitor — Inop

Function (Inop)

This ~~signal~~ provides assurance that a minimum number of APRMs are OPERABLE. ~~Anytime an APRM mode switch is moved to any position other than "Operate," an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs (< 14), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be~~

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d

2.~~e~~. Average Power Range Monitor—Inop (continued) A

e

~~inoperable without resulting in an RPS trip signal.~~ This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

~~Four channels of Average Power Range Monitor—Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.~~

For any APRM channel, any time its mode switch is in any position other than "Operate," an APRM module is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

Insert D

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux—High signal, not the Reactor Vessel Steam Dome Pressure—High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure—High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to

(continued)

INSERT D:

2.e. 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	12. RPS Channel Test Switches (continued) any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

and 12

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 9) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

would result in a full scram), Condition D must be entered and its Required Action taken.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 9 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

or 12

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in Reference 9, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

Or 12

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, and 2.d, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. ~~LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated~~ e

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

MELLLA. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 9).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

<u>SR 3.3.1.1.5 and SR 3.3.1.1.6</u> (continued)

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

<u>SR 3.3.1.1.7</u>

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 effective full power hours Frequency is based on operating experience with LPRM sensitivity changes.

<u>SR 3.3.1.1.8, ~~and~~ SR 3.3.1.1.12</u> and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing – applicable to Function 2.b only), the 2-out-of-4 voter channels, and the interface connections into the RPS trip systems from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions is based on the reliability analysis of Reference 12. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.14.) A Note for SR 3.3.1.1.16 is provided that requires the APRM Function 2.a SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

<u>SR 3.3.1.1.9, SR 3.3.1.1.10, and SR 3.3.1.1.13</u>

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel

(continued)

BASES

13

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.8 also includes calibrating the associated recirculation loop flow channel. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 also includes physical inspection and actuation of the switches.

and SR 3.3.1.1.13

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

and SR 3.3.1.1.13

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.11 (~~Not used.~~ Deleted.)

e

~~The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint are appropriately compared to a calibrated flow signal and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow.~~

~~The Frequency of 18 months is based on system design considerations which do not support flow unit bypass during operation. Thus, this calibration is performed during refueling outages.~~

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

(continued)

BASES

REFERENCES (continued)

9. NEDC-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. MED-32-0286, "Technical Specification Improvement Analysis for Browns Ferry Nuclear Plant, Unit 2," October 1995.

12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch - Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. ~~A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system.~~ A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate or high) is enabled. If the APRM is indicating less than the low power setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

(continued)

BASES

BACKGROUND (continued)

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. ~~Note that the RBM setpoint is flow-biased until implementation of ARTS improvements described in Reference 3. However, the generic RWE analysis in Reference 3 is currently applicable to establish required conditions for RBM OPERABILITY.~~

The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

e

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

28%

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 2[illegible]% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.70, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.40, no RWE event will result in exceeding the MCPR

1.75

1.44

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform a channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of ~~92~~ 18M days is based on reliability analyses (Ref. ~~8~~ 11).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. This test is performed as soon as possible after the applicable conditions are entered. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

1 hour after THERMAL POWER is reduced to ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and THERMAL POWER reduction to ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

an 18 month

The Frequency is based upon the assumption of ~~a 184 day~~ calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch – Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch – Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.7

Insert E

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.

(continued)

INSERT E:

SR 3.3.2.1.8

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1 and the COLR, each within a specific power range. The powers at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

BASES

REFERENCES (continued)

3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.

4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).

5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (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 40% core flow line. BFN
50%
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

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

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.

50%

2. FSAR, Section 4.3.5.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a and 2.e) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.e of Table 3.3.1.1-1

,2.d

(continued)

BASES

ACTIONS

B.1 (continued)

MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

,2.d

LCO 3.3.1.1, Functions 2.a and 2.e, made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met. However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These Surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior

(continued)

TS - 353S1

UNIT 2

MARKED UP PAGES

TABLE OF CONTENTS

1.1 Definitions (continued)

LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
~~MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)~~	~~The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP; or

b. Each required APRM setpoint specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.

Delete Page

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

Delete Page

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-------------------NOTE------------------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements. -- Verify MFLPD is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter
SR 3.2.4.2	-------------------NOTE------------------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements. -- Verify APRM setpoints or gains are adjusted for the calculated MFLPD.	12 hours

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR -----NOTE----- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. -------------- A.2 Place associated trip system in trip.	12 hours 12 hours
-----NOTE----- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. -------------- B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP ~~plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints"~~ while operating at ≥ 25% RTP.	 7 days
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	 7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 effective full power hours
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function~~s~~ 1 ~~and 2.a~~, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	92 days

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	Deleted ~~(Not used.)~~ ~~Adjust the channel to conform to a calibrated flow signal.~~	~~18 months~~
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.13	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	18 months
SR 3.3.1.1.16	Perform CHANNEL FUNCTIONAL TEST.	184 days

-------------------------NOTE-------------------------
For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.

-------------------------NOTES-------------------------
1. Neutron detectors are excluded.
2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.

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) ~~2~~	G	SR 3.3.1.1.1 ~~SR 3.3.1.1.[illegible]~~ SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.~~[illegible]~~ 13 SR 3.3.1.1.~~[illegible]~~ 16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b) ~~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~~ 13 SR 3.3.1.1.~~9~~ 16 ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.14~~	≤ ~~0.58 W + 62% RTP~~ 0.66 W + 71% RTP and ≤ 120% RTP
c. Neutron Flux - High	1	3(b) ~~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~~ 13 SR 3.3.1.1.~~9~~ 16 ~~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) Each APRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
~~c. Downscale~~ e	~~1~~	~~2~~	~~F~~	~~SR 3.3.1.1.7~~ ~~SR 3.3.1.1.8~~ ~~SR 3.3.1.1.14~~	~~≥ 5% RTP~~
d ~~c.~~ Inop	1,2	$3^{(b)}$ ~~2~~ e	G	~~SR [illegible]~~ e SR 3.3.1.1.16 -16 ~~SR [illegible]~~ e	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	N/A
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	$5^{(a)}$	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	$5^{(a)}$	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(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.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One or more Reactor Mode Switch – Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal. AND E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE	FREQUENCY
SR 3.3.2.1.1 Perform CHANNEL FUNCTIONAL TEST.	184 ~~92~~ days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	18 months ~~92 days~~
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	18 months
SR 3.3.2.1.6	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.7	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

single line

SR 3.3.2.1.8	------------------------------NOTE------------------------------ Neutron detectors are excluded. -- Verify the RBM: a. Low Power Range -- Upscale Function is not bypassed when THERMAL POWER is ≥ 28% and ≤ 63% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 63% and ≤ 83% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 83% RTP.	18 months

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
~~a. Upscale (Flow Biased)~~	~~(a),(b)~~	~~2~~	~~SR 3.3.2.1.1~~ ~~SR 3.3.2.1.4~~	~~(c)~~ e
d. ~~b.~~ e Inop	(g),(h) ~~(a),(b)~~ e	2	SR 3.3.2.1.1	NA
e. ~~c.~~ e Downscale	(g),(h) ~~(a),(b)~~ e	2	SR 3.3.2.1.1 SR 3.3.2.1.4	~~≥ 3% RTP~~ e (i)
2. Rod Worth Minimizer	$1^{(c)}$, $2^{(c)}$	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch - Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ~~≥ 90% RTP and MCPR < 1.40.~~ e ≥ 28% and ≤ 63% RTP and MCPR < 1.75.

(b) THERMAL POWER ~~≥ 29% and < 90% RTP~~ > 63% and ≤ 83% RTP and MCPR < ~~1.70.~~ 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 83% and < 90% RTP and MCPR < 1.75.

(g) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(h) THERMAL POWER ≥ 28% and < 90% RTP and MCPR < 1.75.

(i) Greater than or equal to the Allowable Value specified in the COLR.

a. Low Power Range -- Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intemediate Power Range -- Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range -- Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	--------------------NOTE------------------ Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	 24 hours
SR 3.4.1.2	Verify the reactor is outside of Region I and II of Figure 3.4.1-1.	Immediately after any increase > 5% RTP while initial core flow is < ~~45%~~ of rated. 50% AND Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated.

Replace with Improved Figure

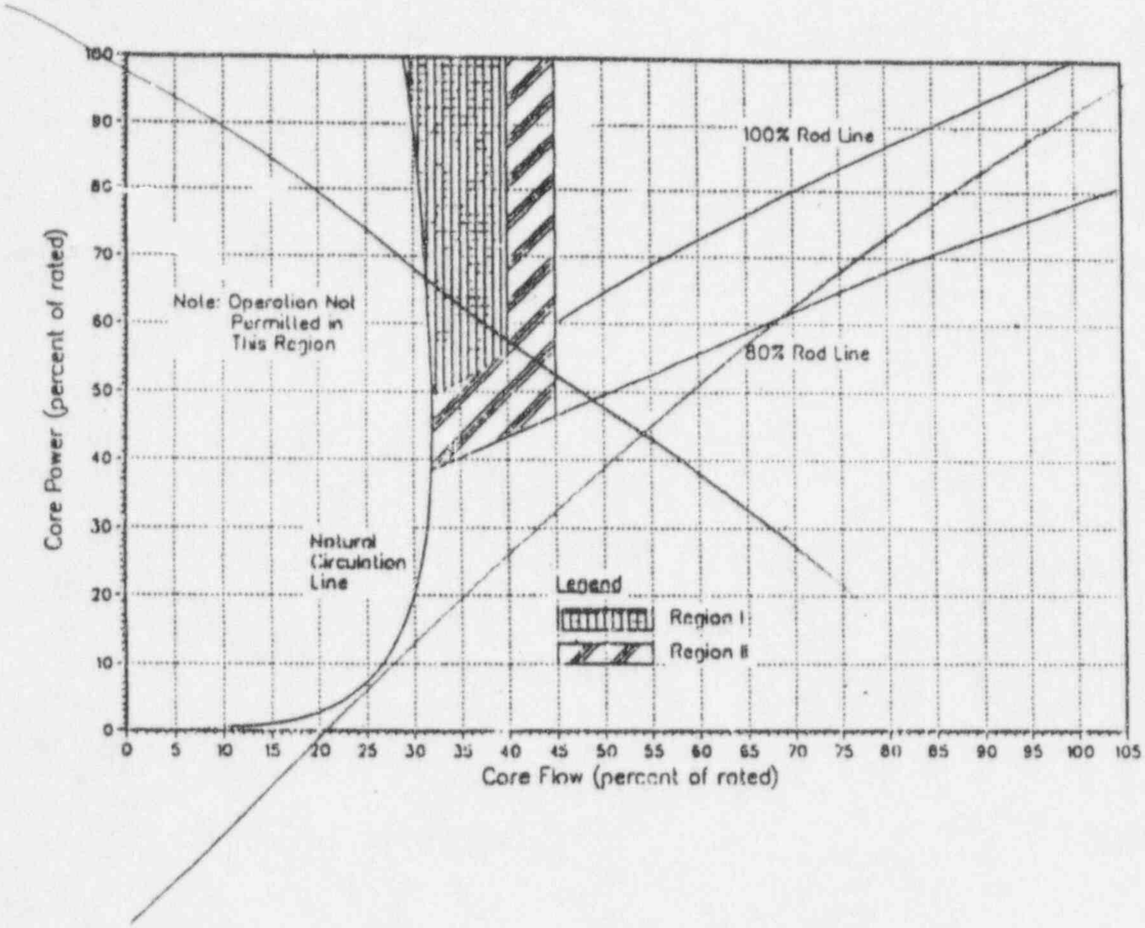

Figure 3.4.1-1

THERMAL POWER VERSUS CORE FLOW
STABILITY REGIONS

Replacement Figure

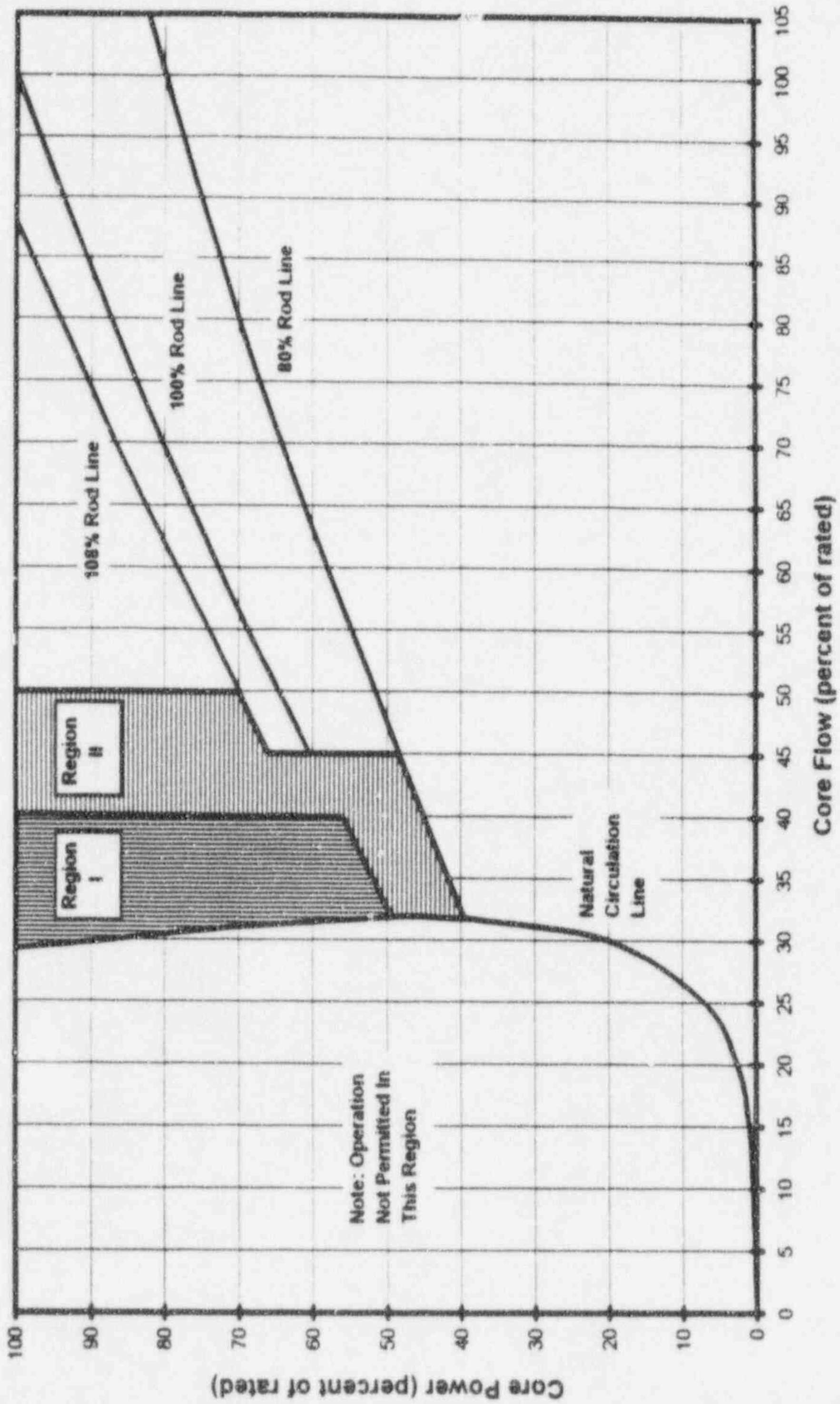

Figure 3.4.1-1

THERMAL POWER VERSUS
CORE FLOW STABILITY
REGIONS

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test – Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Functions 2.a ,2.d and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence (BPWS) requirements of SR 3.3.2.1.7 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of BPWS control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.e of Table 3.3.1.1-1. 2.d,	According to the applicable SRs
SR 3.10.8.2	-------------------NOTE------------------- Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	-------------------NOTE------------------- Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

TABLE OF CONTENTS

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, ~~and~~ 4, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. ~~APLHGR limits are developed as a function of exposure and fuel bundle type.~~

Insert A

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

(continued)

INSERT A:

APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Reference 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Reference 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.

BASES

APPLICABLE SAFETY ANALYSES (continued)

conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

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.

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.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-11 "General Electric Standard Application for Reactor Fuel," November 1995.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appen ix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 1, February 1996.
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.

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

, and 8.

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, and 5. 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insert B

~~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.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the

The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

(continued)

INSERT B:

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.

BASES

REFERENCES (continued)

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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection System Failure Modes" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

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The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

(continued)

BASES

BACKGROUND (continued)

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses presented in References 1 and 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

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APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR and MCPR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR and MCPR) occurs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

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The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. 4).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

(continued)

BASES

LCO (continued)

c. Increasing APRM gains to cause the APRM to read ≥ 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

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MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

(continued)

BASES

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ACTIONS

A.1 (continued)

may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared with FRTP, or APRM gains or setpoint, to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to FRP. When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
2. FSAR, Chapter 14.
3. FSAR, Chapter 3.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux — High (continued)

unexpected reactivity excursions. In MODE 1, the APRM System and the RBM provide protection against control rod withdrawal error events and the IRMs are not required.

1.b. Intermediate Range Monitor — Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor - Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux - High Function is required.

Insert C

Average Power Range Monitor

2.a. Average Power Range Monitor Neutron Flux — High, Setdown

~~The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.~~ For operation at

(continued)

INSERT C:

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs to each RPS trip system logic channel (A1, A2, B1, or B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM functions, at least twenty (20) LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High, Setdown (continued)

low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

~~The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux - High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux — High, Setdown (continued)

In MODE 1, the Average Power Range Monitor Neutron Flux — High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power — High

The Average Power Range Monitor Flow Biased Simulated Thermal Power — High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux — High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power — High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux — High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power — High Function setpoint is exceeded.

~~The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

~~Average Power Range Monitor Flow Biased Simulated Thermal Power - High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~ e Each APRM channel ~~receives a~~ uses one e total drive flow signal representative of total core flow. 15 The total drive flow signals ~~are~~ generated by ~~two flow units, one of which supplies signals to the trip system A APRMs, while the other one supplies signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power - High channel requires an input from its associated OPERABLE flow unit.~~ e

the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this function.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux - High

~~The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases.~~ e The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>2.c. Average Power Range Monitor Fixed Neutron Flux — High</u> (continued)

trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux — High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux — High Function to terminate the CRDA.

~~The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux — High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux — High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux — High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux — High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux — High Function is not required in MODE 2.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.d. Average Power Range Monitor — Downscale

This signal ensures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to the APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associated Intermediate Range Monitor Neutron Flux — High or Inop signal generates a trip signal. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The APRM System is divided into two groups of channels with three inputs into each trip system. The system is designed to allow one channel in each trip system to be bypassed. Four channels of Average Power Range Monitor — Downscale with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. The Intermediate Range Monitor Neutron Flux — High and Inop Functions are also part of the OPERABILITY of the Average Power Range Monitor — Downscale Function (i.e., if either of these IRM Functions cannot send a signal to the Average Power Range Monitor — Downscale Function, the associated Average Power Range Monitor — Downscale channel is considered inoperable).

The Allowable Value is based upon ensuring that the APRMs are in the linear scale range when transfers are made between APRMs and IRMs.

This Function is required to be OPERABLE in MODE 1 since this is when the APRMs are the primary indicators of reactor power.

e

d

2.~~e~~. Average Power Range Monitor — Inop

Function (Inop)

This ~~signal~~ provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than "Operate," an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs (< 14), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be

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Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d

2.~~e~~. Average Power Range Monitor—Inop (continued)

e 9

~~Inoperable without resulting in an RPS trip signal.~~ This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

e

~~Four channels of Average Power Range Monitor—Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.~~

For any APRM channel, any time its mode switch is in any position other than "Operate," an APRM module is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

Insert D

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux—High signal, not the Reactor Vessel Steam Dome Pressure—High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure—High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to

(continued)

INSERT D:

2.e. 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

BASES

ACTIONS (continued)

compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 9 and 12) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 9 for the 12 hour
(5) (or 12)

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

or 12

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in Reference 9, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, and 2.d, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. ~~LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD.~~ The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 effective full power hours Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day frequency of SR 3.3.1.1.16 for the APRM Functions supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing -- applicable to Function 2.b only), the 2-out-of-4 voter channels, and the interface connections into the RPS trip systems from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions is based on the reliability analysis of Reference 12. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.14.) A Note for SR 3.3.1.1.16 is provided that requires the APRM Function 2.a SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The 184 day Frequency of SR 3.3.1.1.16 for the scram pilot air header low pressure trip function is based on the functional reliability previously demonstrated by this function, the need for minimizing the radiation exposure associated with the functional testing of this function, and the increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by the more frequent functional testing.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel

(continued)

13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.8 also includes calibrating the associated recirculation loop flow channel. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 also includes physical inspection and actuation of the switches.

and SR 3.3.1.1.13

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

and SR 3.3.1.1.13

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

(Deleted ~~Not used.~~)

~~The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint are appropriately compared to a~~

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.1.1.11 (continued)~~

~~The Frequency of 18 months is based on system design considerations which do not support flow unit bypass during operation. Thus, this calibration is performed during refueling outages.~~

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

REFERENCES

1. FSAR, Section 7.2.

2. FSAR, Chapter 14.

3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.

4. FSAR, Appendix N.

5. FSAR, Section 14.6.2.

6. FSAR, Section 6.5.

7. FSAR, Section 14.5.

8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.

9. NEDC-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. MED-32-0286, "Technical Specification Improvement Analysis for Browns Ferry Nuclear Plant, Unit 2," October 1995.

12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch – Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. ~~A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system.~~ A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate or high) is enabled. If the APRM is indicating less than the low power setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

(continued)

BASES

BACKGROUND (continued)

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. ~~Note that the RBM setpoint is flow-biased until implementation of ARTS improvements described in Reference 3. However, the generic RWE analysis in Reference 3 is currently applicable to establish required conditions for RBM OPERABILITY.~~

The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value, for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints deriveo in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ ~~27%~~ 28% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ ~~1.70~~ 1.75 no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ ~~1.40~~ 1.44, no RWE event will result in exceeding the MCPR

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform a channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 92 days is based on reliability analyses (Ref. 8).

184

11

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. This test is performed as soon as possible after the applicable conditions are entered. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

1 hour after THERMAL POWER is reduced to ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and THERMAL POWER reduction to ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

The Frequency is based upon the assumption of an 18 month ~~a 184 day~~ calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch - Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch - Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.7

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

Insert E

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.

(continued)

INSERT E:

SR 3.3.2.1.8

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1 and the COLR, each within a specific power range. The powers at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

BASES

REFERENCES (continued)

3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.

4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).

5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (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
50%.
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

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

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.

50%

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a ,2.d and 2.e) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.e of Table 3.3.1.1-1

(continued)

BASES

ACTIONS

B.1 (continued)

MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

,2.d

LCO 3.3.1.1, Functions 2.a and 2.e, made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met. However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These Surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior

(continued)

TS - 353S1

UNIT 3

MARKED UP PAGES

TABLE OF CONTENTS

1.1 Definitions (continued)

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

~~MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)~~

~~The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~

MINIMUM CRITICAL POWER RATIO (MCPR)

The MCPR shall be the smallest critical power ratio (CPR) that exists in the core. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

MODE

A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

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3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP; or

b. Each required APRM setpoint specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------------------NOTE------------------ Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements. -- Verify MFLPD is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter
SR 3.2.4.2	------------------NOTE------------------ Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements. -- Verify APRM setpoints or gains are adjusted for the calculated MFLPD.	12 hours

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable. ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. ---------------------------	A.1 Place channel in trip. OR ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. --------------------------- A.2 Place associated trip system in trip.	12 hours 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP ~~plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints"~~ while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	 7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 effective full power hours
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1 and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	 92 days

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	Deleted ~~(Not used.)~~ ~~Adjust the channel to conform to a calibrated flow signal.~~	~~18 months~~
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.13	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	18 months
SR 3.3.1.1.16	Perform CHANNEL FUNCTIONAL TEST.	184 days

-------------------------NOTE-------------------------
For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.

-------------------------NOTES-------------------------
1. Neutron detectors are excluded.
2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.

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) ~~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.~~8~~ 13 SR 3.3.1.1.~~10~~ 16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b) ~~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~~ 13 SR 3.3.1.1.~~10~~ 16 ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.14~~	≤ ~~0.58 W + 62% RTP~~ 0.66 W + 71% RTP and ≤ 120% RTP
c. Neutron Flux - High	1	3(b) ~~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~~ 13 SR 3.3.1.1.~~10~~ 16 ~~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) Each APRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
~~c. Downscale~~	~~1~~	~~2~~	~~F~~	~~SR 3.3.1.1.7~~ ~~SR 3.3.1.1.8~~ ~~SR 3.3.1.1.16~~	~~≥ 3% RTP~~
d ~~c~~. Inop	1,2	3(b) ~~2~~	G	~~SR 3.3.1.1.11~~ SR 3.3.1.1.8 ~~SR 3.3.1.1.16~~	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(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.

e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	N/A

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch – Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTES-------------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	18 months ~~92 days~~
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	18 months
SR 3.3.2.1.6	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.7	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

single line

SR 3.3.2.1.8	-----NOTE----- Neutron detectors are excluded. ---------- Verify the RBM: a. Low Power Range -- Upscale Function is not bypassed when THERMAL POWER is ≥ 28% and ≤ 63% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 63% and ≤ 83% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 83% RTP.	18 months

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
~~a. Upscale (Flow Biased)~~	~~(a),(b)~~	~~2~~	~~SR 3.3.2.1.1~~ ~~SR 3.3.2.1.4~~	~~(e)~~ e
a. Low Power Range -- Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intemediate Power Range -- Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range -- Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. ~~b.~~ e Inop	(g),(h) ~~(a),(b)~~ e	2	SR 3.3.2.1.1	NA
e. ~~c.~~ e Downscale	(g),(h) ~~(a),(b)~~ e	2	SR 3.3.2.1.1 SR 3.3.2.1.4	~~≥ 3% RTP~~ e (i)
2. Rod Worth Minimizer	$1^{(c)}, 2^{(c)}$	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch - Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ~~≥ 90% RTP and MCPR < 1.40.~~ e ≥ 28% and ≤ 63% RTP and MCPR < 1.75.

(b) THERMAL POWER ~~≥ 29% and < 90% RTP~~ > 63% and ≤ 83% RTP and MCPR < ~~1.70.~~ 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 83% and < 90% RTP and MCPR < 1.75.

(g) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(h) THERMAL POWER ≥ 28% and < 90% RTP and MCPR < 1.75.

(i) Greater than or equal to the Allowable Value specified in the COLR.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	--------------------NOTE-------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. --- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
SR 3.4.1.2	Verify the reactor is outside of Region I and II of Figure 3.4.1-1.	Immediately after any increase > 5% RTP while initial core flow is < ~~45%~~ 50% of rated. AND Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated.

Replace with
Improved Figure

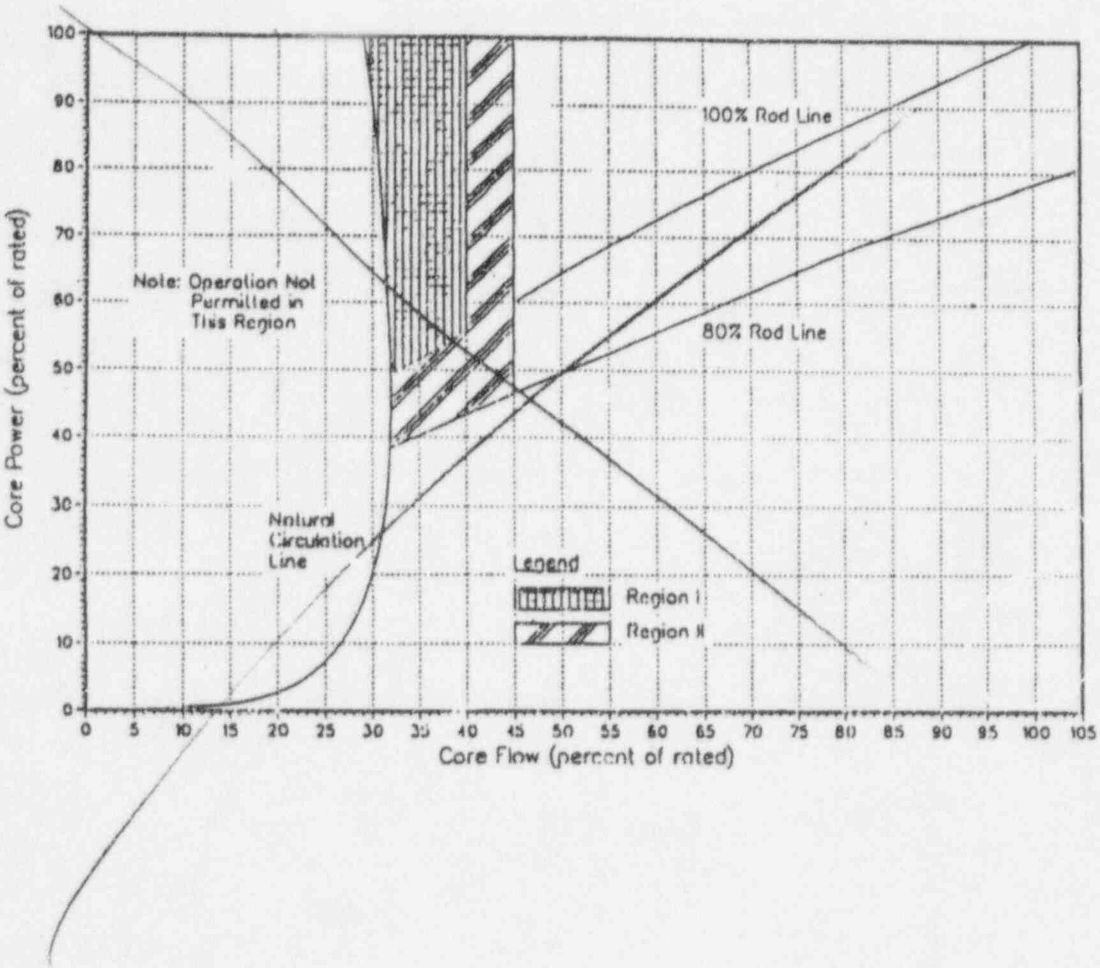

Figure 3.4.1-1

THERMAL POWER VERSUS CORE FLOW
STABILITY REGIONS

Improved
Figure

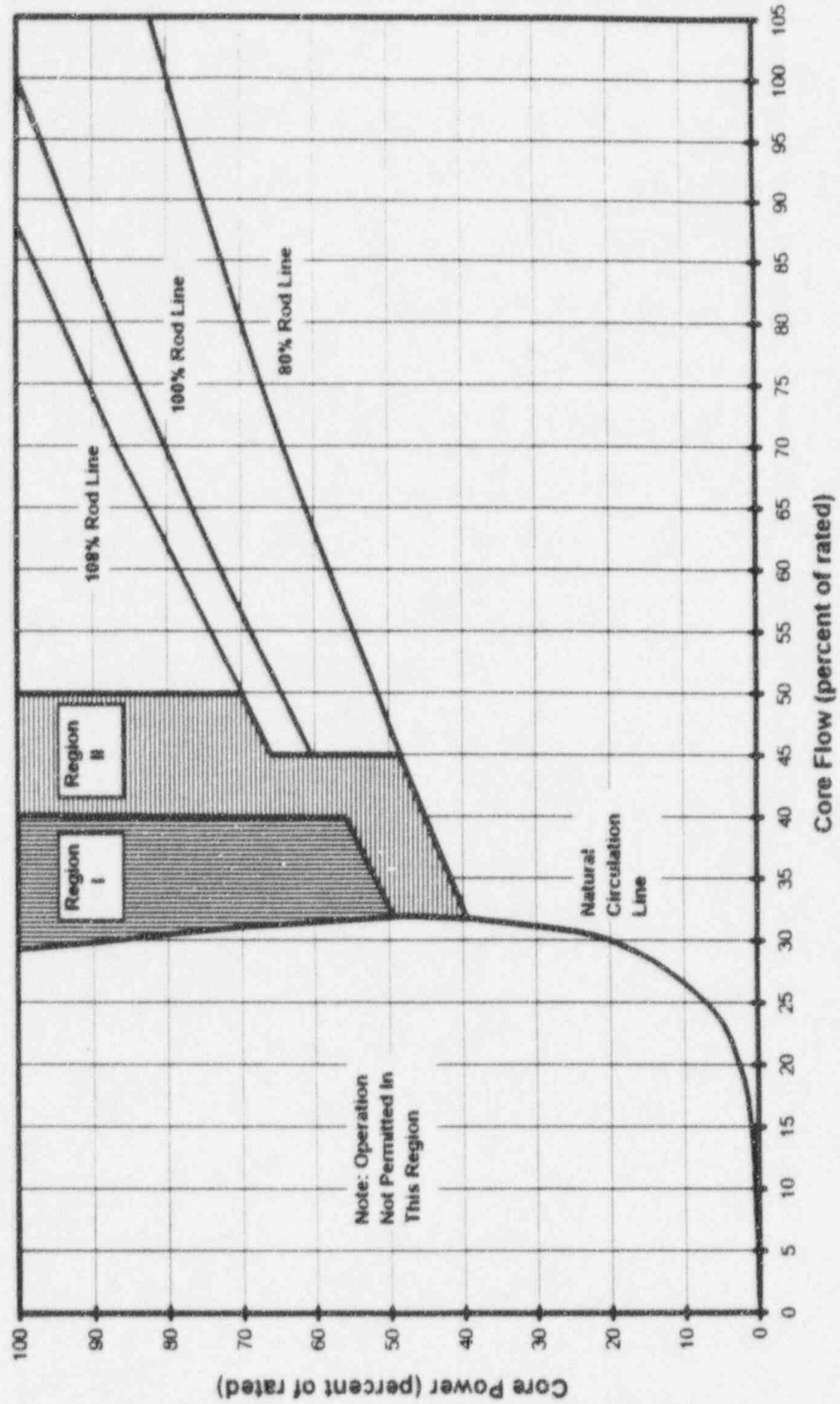

Figure 3.4.1-1

THERMAL POWER VERSUS
CORE FLOW STABILITY
REGIONS

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Functions 2.a ,2.d and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence (BPWS) requirements of SR 3.3.2.1.7 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of BPWS control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.e of Table 3.3.1.1-1. 2.d,	According to the applicable SRs
SR 3.10.8.2	-------------------NOTE------------------- Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	-------------------NOTE------------------- Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

TABLE OF CONTENTS

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, and 4, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. Insert A ~~APLHGR limits are developed as a function of exposure and fuel bundle type.~~

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

(continued)

INSERT A:

APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Reference 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Reference 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.

BASES

APPLICABLE SAFETY ANALYSES (continued)

conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

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.

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.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

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.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-11 "General Electric Standard Application for Reactor Fuel," November 1995.
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 1, February 1996.
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.

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, ~~and~~ 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~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.~~

e

Insert B

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.

The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the

INSERT B:

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.

BASES

REFERENCES (continued)

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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection System Failure Modes" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{MFLPD}{FRTP} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

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(continued)

BASES

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BACKGROUND (continued)

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses presented in References 1 and 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR and MCPR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR and MCPR) occurs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. 4).

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LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

(continued)

BASES

LCO (continued)

c. Increasing APRM gains to cause the APRM to read ≥ 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

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For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

(continued)

BASES

ACTIONS

A.1 (continued)

may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

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B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared with FRTP, or APRM gains or setpoint, to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to FRP. When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
2. FSAR, Chapter 14.
3. FSAR, Chapter 3.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

Delete Page

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux — High (continued)

unexpected reactivity excursions. In MODE 1, the APRM System and the RBM provide protection against control rod withdrawal error events and the IRMs are not required.

1.b. Intermediate Range Monitor — Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor - Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux - High Function is required.

Insert C

Average Power Range Monitor

2.a. Average Power Range Monitor Neutron Flux — High, Setdown

~~The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.~~ For operation at

(continued)

INSERT C:

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs to each RPS trip system logic channel (A1, A2, B1, or B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM functions, at least twenty (20) LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High, Setdown (continued)

low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

~~The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux - High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux -- High, Setdown (continued)

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux - High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux - High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function setpoint is exceeded.

~~The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

e

~~Average Power Range Monitor Flow Biased Simulated Thermal Power - High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~ Each APRM channel ~~receives a~~ uses one total drive flow signal representative of total core flow. The total drive flow signal ~~are~~ is generated by ~~two flow units, one of which supplies signals to the trip system A APRMs, while the other one supplies signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power - High channel requires an input from its associated OPERABLE flow unit.~~ the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this function.

e

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux - High

e

~~The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases.~~ The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor Fixed Neutron Flux – High (continued)

trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux – High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux – High Function to terminate the CRDA.

~~The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux – High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux – High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux – High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux – High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux – High Function is not required in MODE 2.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e

2.d. Average Power Range Monitor - Downscale

This signal ensures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to the APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associated Intermediate Range Monitor Neutron Flux - High or Inop signal generates a trip signal. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The APRM System is divided into two groups of channels with three inputs into each trip system. The system is designed to allow one channel in each trip system to be bypassed. Four channels of Average Power Range Monitor - Downscale with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. The Intermediate Range Monitor Neutron Flux - High and Inop Functions are also part of the OPERABILITY of the Average Power Range Monitor - Downscale Function (i.e., if either of these IRM Functions cannot send a signal to the Average Power Range Monitor - Downscale Function, the associated Average Power Range Monitor - Downscale channel is considered inoperable).

The Allowable Value is based upon ensuring that the APRMs are in the linear scale range when transfers are made between APRMs and IRMs.

This Function is required to be OPERABLE in MODE 1 since this is when the APRMs are the primary indicators of reactor power.

d e

2.e. Average Power Range Monitor - Inop

Function (Inop) e

This signal provides assurance that a minimum number of APRMs are OPERABLE.

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions.

e

Anytime an APRM mode switch is moved to any position other than "Operate," an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs (< 14), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.~~c~~d. Average Power Range Monitor — Inop (continued)

~~inoperable without resulting in an RPS trip signal.~~ This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

~~Four channels of Average Power Range Monitor — Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.~~

For any APRM channel, any time its mode switch is in any position other than "Operate," an APRM module is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

Insert D

3. Reactor Vessel Steam Dome Pressure — High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure — High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux — High signal, not the Reactor Vessel Steam Dome Pressure — High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure — High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure — High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to

(continued)

INSERT D:

2.e. 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

BASES

ACTIONS (continued)

compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 9 and 12) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in References 9 or 12 for the 12 hour

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

or 12

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in Reference 9, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, and 2.d, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. ~~LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD.~~ The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 effective full power hours Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing -- applicable to Function 2.b only), the 2-out-of-4 voter channels, and the interface connections into the RPS trip systems from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions is based on the reliability analysis of Reference 12. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.14.) A Note for SR 3.3.1.1.16 is provided that requires the APRM Function 2.a SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The 184 day Frequency of SR 3.3.1.1.16 for the scram pilot air header low pressure trip function is based on the functional reliability previously demonstrated by this function, the need for minimizing the radiation exposure associated with the functional testing of this function, and the increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by the more frequent functional testing.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel

(continued)

BASES

13

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.~~9~~ also includes calibrating the associated recirculation loop flow channel. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 also includes physical inspection and actuation of the switches.

and SR 3.3.1.1.13

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

and SR 3.3.1.1.13

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

Deleted (~~Not used.~~)

~~The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint are appropriately compared to a~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

e

~~SR 3.3.1.1.11 (continued)~~

~~The Frequency of 18 months is based on system design considerations which do not support flow unit bypass during operation. Thus, this calibration is performed during refueling outages.~~

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

REFERENCES

1. FSAR, Section 7.2.
2. FSAR, Chapter 14.
3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
4. FSAR, Appendix N.
5. FSAR, Section 14.6.2.
6. FSAR, Section 6.5.
7. FSAR, Section 14.5.
8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
9. NEDC-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
11. MED-32-0286, "Technical Specification Improvement Analysis for Browns Ferry Nuclear Plant, Unit 2," October 1995.
12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch - Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. ~~A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system.~~ A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate or high) is enabled. If the APRM is indicating less than the low power setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

(continued)

BASES

BACKGROUND (continued)

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. ~~Note that the RBM setpoint is flow-biased until implementation of ARTS improvements described in Reference 3. However, the generic RWE analysis in Reference 3 is currently applicable to establish required conditions for RBM OPERABILITY.~~

The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

28%

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ [illegible]% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.70, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.40, no RWE event will result in exceeding the MCPR

1.75

1.44

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintainscontrol rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform a channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of ~~92~~ 184 days is based on reliability analyses (Ref. ~~8~~ 11).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. This test is performed as soon as possible after the applicable conditions are entered. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until

(continued)

BASES

SURVEILLANCE REQUIREMENTS

<u>SR 3.3.2.1.2 and SR 3.3.2.1.3</u> (continued)

1 hour after THERMAL POWER is reduced to ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and THERMAL POWER reduction to ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

<u>SR 3.3.2.1.4</u>

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

The Frequency is based upon the assumption of ~~a 184 day~~ an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

<u>SR 3.3.2.1.5</u>

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch - Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch - Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.7

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

Insert E

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.

(continued)

INSERT E:

SR 3.3.2.1.8

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1 and the COLR, each within a specific power range. The powers at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

BASES

REFERENCES (continued)

3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.

4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).

5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (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
50% 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

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

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. 50%

2. FSAR, Section 4.3.5.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a and 2.e) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.e of Table 3.3.1.1-1

,2.d

(continued)

BASES

ACTIONS

B.1 (continued)

MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

2.d

LCO 3.3.1.1, Functions 2.a and 2.e, made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met. However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These Surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior

(continued)

ENCLOSURE 3
TENNESSEE VALLEY AUTHORITY (TVA)
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, and 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-353S1
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
1.1-4	1.1-4	1.1-4
3.2-5	3.2-5	3.2-5
3.3-1	3.3-1	3.3-1
3.3.2	3.3.2	3.3.2
3.3.2a	3.3.2a	3.3.2a
3.3-3	3.3-3	3.3-3
through	through	through
3.3-7	3.3-7	3.3-7
3.3-17	3.3-17	3.3-17
through	through	through
3.3-20	3.3-20	3.3-20
3.4-3	3.4-3	3.4-3
3.4-4	3.4-4	3.4-4
3.10-20	3.10-20	3.10-20
3.10.22	3.10.22	3.10.22
B 3.2-1	B 3.2-1	B 3.2-1
through	through	through
B 3.2-3	B 3.2-3	B 3.2-3
B 3.2-3a	B 3.2-3a	B 3.2-3a
B 3.2-4	B 3.2-4	B 3.2-4
B 3.2-5	B 3.2-5	B 3.2-5
B 3.2-8	B 3.2-8	B 3.2-8
B 3.2-12	B 3.2-12	B 3.2-12
B 3.3-6	B 3.3-6	B 3.3-6
through	through	through
B 3.3-32	B 3.3-32	B 3.3-32
B 3.3-42	B 3.3-32a	B 3.3-32a
through	B 3.3-42	B 3.3-42
B 3.3-44	through	through
B 3.3-49	B 3.3-52	B 3.3-44
through	B 3.3-52a	B 3.3-49
B 3.3-52	B 3.4-3	through
B 3.3-52a	B 3.4-7	B 3.3-52
B 3.4-3	B 3.10-34	B 3.3-52a
B 3.4-7	B 3.10-35	B 3.4-3
B 3.10-34	B 3.10-37	B 3.4-7
B 3.10-35		B 3.10-34
B 3.10-37		B 3.10-35
		B 3.10-37

II. REVISED PAGES

See attached.

TS - 353S1

UNIT 1

REVISED PAGES

TABLE OF CONTENTS

1.1 Definitions (continued)

LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 (Deleted)

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	<u>OR</u>		
	A.2	--------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. -------------------- Place associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. -------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTES-------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	 7 days
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	 7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 effective full power hours
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	92 days

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted)	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.13	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	 18 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	18 months
SR 3.3.1.1.16	------------------NOTE------------------ For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST	 184 days

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. 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.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level – Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level – High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(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.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch – Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	<u>AND</u>		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	18 months
SR 3.3.2.1.6	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.7	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM
SR 3.3.2.1.8	------------------NOTE------------------ Neutron detectors are excluded. -- Verify the RBM: a. Low Power Range -- Upscale Function is not bypassed when THERMAL POWER is ≥ 28% and ≤ 63% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 63% and ≤ 83% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 83% RTP.	18 months

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range -- Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range -- Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range -- Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. Inop	(g),(h)	2	SR 3.3.2.1.1	NA
e. Downscale	(g),(h)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(i)
2. Rod Worth Minimizer	1(c),2(c)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch – Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 28% and ≤ 63% RTP and MCPR < 1.75.

(b) THERMAL POWER > 63% and ≤ 83% RTP and MCPR < 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 83% and < 90% RTP and MCPR < 1.75.

(g) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(h) THERMAL POWER ≥ 28% and < 90% RTP and MCPR < 1.75.

(i) Greater than or equal to the Allowable Value specified in the COLR.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------ Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
SR 3.4.1.2	Verify the reactor is outside of Region I and II of Figure 3.4.1-1.	Immediately after any increase > 5% RTP while initial core flow is < 50% of rated. <u>AND</u> Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated.

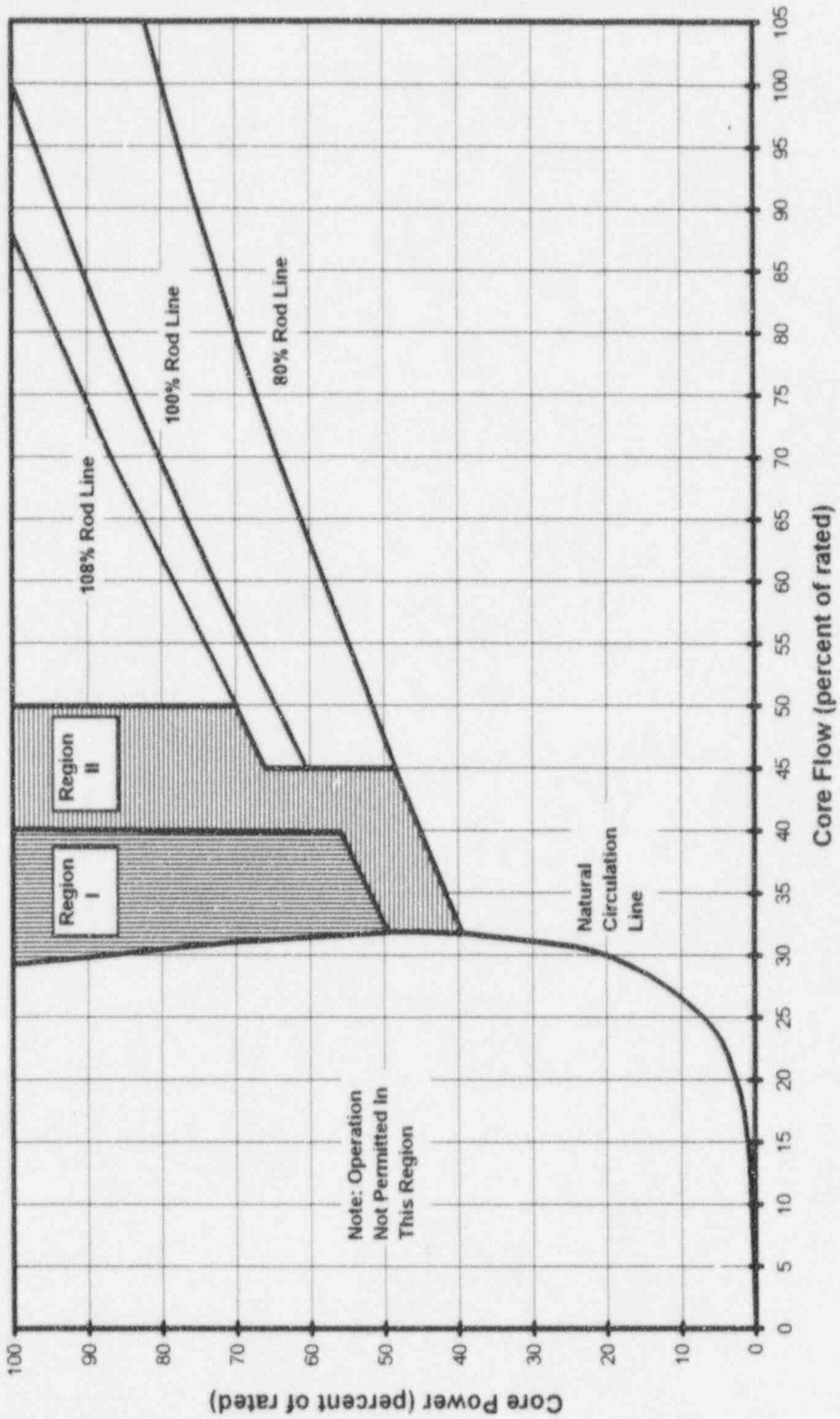

Figure 3.4.1-1

THERMAL POWER VERSUS
CORE FLOW STABILITY
REGIONS

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence (BPWS) requirements of SR 3.3.2.1.7 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of BPWS control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1.	According to the applicable SRs
SR 3.10.8.2	-------------------NOTE------------------- Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	-------------------NOTE------------------- Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

TABLE OF CONTENTS

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Reference 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Reference 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.

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.

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.

(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.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-11 "General Electric Standard Application for Reactor Fuel," November 1995.
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 1, February 1996.
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.

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, 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.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

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).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the

(continued)

BASES

REFERENCES (continued)

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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 (Deleted)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux—High (continued)

unexpected reactivity excursions. In MODE 1, the APRM System and the RBM provide protection against control rod withdrawal error events and the IRMs are not required.

1.b. Intermediate Range Monitor— Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor - Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux - High Function is required.

Average Power Range Monitor

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Average Power Range Monitor (continued)

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs to each RPS trip system logic channel (A1, A2, B1, or B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM functions, at least twenty (20) LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

2.a. Average Power Range Monitor Neutron Flux—High, Setdown

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux-- High, Setdown (continued)

position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux - High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux - High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function setpoint is exceeded.

Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this function.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux—High

The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO and APPLICABILITY	2.c. Average Power Range Monitor Fixed Neutron Flux—High (continued)

valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux - High Function to terminate the CRDA.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux - High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux - High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux - High Function is not required in MODE 2.

2.d. Average Power Range Monitor—Inop

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions. This Function (Inop) provides assurance that a minimum number of APRMs are OPERABLE. For any APRM channel, any time its mode switch is in any position other than "Operate," an APRM module is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is no Allowable Value for this Function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor—Inop (continued)

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

2.e. 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux—High signal, not the Reactor Vessel Steam Dome Pressure—High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure—High Allowable Value is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Vessel Steam Dome Pressure—High (continued)

chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level—Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low, Level 3 Function is assumed in the analysis of the recirculation line break (Ref. 6). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low, Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low, Level 3 Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value is selected to ensure that (a) during normal operation the steam dryer skirt is not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder), and (b) for transients involving

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level—Low, Level 3 (continued)

loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water - Low Low Low, Level 1 will not be required.

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

5. Main Steam Isolation Valve—Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 7 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow).

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve - Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Main Steam Isolation Valve—Closure (continued)

Valve - Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines must close in order for a scram to occur.

The Main Steam Isolation Valve - Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the Main Steam Isolation Valve - Closure Function, with eight channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

6. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure - High Function is a secondary scram signal to Reactor Vessel Water Level - Low, Level 3 for LOCA events inside the drywell. However, no credit is taken for a scram initiated from this Function for any of the DBAs analyzed in the FSAR. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Drywell Pressure—High (continued)

logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

7a, 7b. Scram Discharge Volume Water Level—High

The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. Should this volume fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volume is still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level - High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the FSAR. However, they are retained to ensure the RPS remains OPERABLE.

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two thermal probes for a total of eight level signals. The outputs of these devices are arranged so that there is a signal from a level switch and a thermal probe to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 8.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

Four channels of each type of Scram Discharge Volume Water Level - High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7a, 7b. Scram Discharge Volume Water Level—High (continued)

conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve - Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure transmitter is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Two channels of Reactor Mode Switch - Shutdown Position Function, with one channel in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch - Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Manual Scram

The Manual Scram push button channels provide signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is one Manual Scram push button channel for each of the two RPS manual scram logic channels. In order to cause a scram it is necessary that each channel in both manual scram trip systems be actuated.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of Manual Scram with one channel in each manual scram trip system are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

12. RPS Channel Test Switches

There are four RPS Channel Test Switches, one associated with each of the four automatic scram logic channels (A1, A2, B1, and B2). These keylock switches allow the operator to test the OPERABILITY of each individual logic channel without the necessity of using a scram function trip. When the RPS Channel Test Switch is placed in test, the associated scram logic channel is deenergized and OPERABILITY of the channel's scram contactors can be confirmed. The RPS Channel Test Switches are not specifically credited in the accident analysis. However, because the Manual Scram Function at Browns Ferry Nuclear Plant is not configured the same as the generic model in Reference 9, the RPS Channel Test Switches are included in the analysis in Reference 11. Reference 11 concludes that

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

12. RPS Channel Test Switches (continued)

the Surveillance Frequency extensions for RPS functions, described in Reference 9, are not affected by the difference in configuration since each automatic RPS channel has a test switch which is functionally the same as the manual scram switches in the generic model. Weekly testing of scram contactors is credited in Reference 9 with supporting the Surveillance Frequency extension of the RPS functions.

There is no Allowable Value for this Function since the channels are mechanically actuated solely on the position of the switches.

Four channels of the RPS Channel Test Switch Function with two channels in each trip system arranged in a one-out-of-two logic are available and required to be OPERABLE. The function is required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

allowable out of service time of 12 hours has been shown to be acceptable (Ref. 9 and 12) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and

(continued)

ACTIONS

B.1 and B.2 (continued)

one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in References 9 or 12 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in References 9 or 12, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, and 2.d, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability.

A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel

(continued)

BASES

ACTIONS

D.1 (continued)

and provides for transfer to the appropriate subsequent Condition.

E.1, F.1, and G.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Action E.1 is consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains RPS trip

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 9).

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 9. (The RPS Channel Test Switch Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 and SR 3.3.1.1.6 (continued)

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 effective full power hours Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing - applicable to Function 2.b only), the 2-out-of-4 voter channels, and the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16 (continued)

interface connections into the RPS trip systems from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions is based on the reliability analysis of Reference 12. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.14.) A Note for SR 3.3.1.1.16 is provided that requires the APRM Function 2.a SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10, and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.13 also includes calibrating the associated recirculation loop flow channel. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 also includes physical inspection and actuation of the switches.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10, and SR 3.3.1.1.13 (continued)

Note 1 to SR 3.3.1.1.9 and SR 3.3.1.1.13 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 and SR 3.3.1.1.13 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

(Deleted)

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.14 (continued)

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

This SR ensures that scrams initiated from the Turbine Stop Valve − Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure − Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve − Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure − Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve − Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure − Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

REFERENCES

1. FSAR, Section 7.2.
2. FSAR, Chapter 14.

(continued)

BASES

REFERENCES (continued)

3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
4. FSAR, Appendix N.
5. FSAR, Section 14.6.2.
6. FSAR, Section 6.5.
7. FSAR, Section 14.5.
8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
9. NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements, "July 23, 1993.
11. MED-32-0286, "Technical Specification Improvement Analysis for Browns Ferry Nuclear Plant, Unit 2," October 1995.
12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch – Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate or high) is enabled. If the APRM is indicating less than the low power setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

(continued)

BASES

BACKGROUND (continued)

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 28% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.75, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.44, no RWE event will result in exceeding the MCPR

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BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform a channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 184 days is based on reliability analyses (Ref. 11).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. This test is performed as soon as possible after the applicable conditions are entered. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour

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SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

after any control rod is withdrawn at ≤ 10% RTP in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is reduced to ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and THERMAL POWER reduction to ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.5 (continued)

is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch – Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch – Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.7

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.8

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1 and the COLR, each within a specific power range. The powers at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.
3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.
4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).
5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.
6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

(continued)

BASES

REFERENCES (continued)

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (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).

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.
LCO	As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical,

(continued)

BASES

LCO (continued)

RPS MODE 2 requirements for Functions 2.a and 2.e of Table 3.3.1.1-1 must be enforced and the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2), or must be verified by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). To provide additional protection against an inadvertent criticality, control rod withdrawals that do not conform to the banked position withdrawal sequence specified in LCO 3.1.6, "Rod Pattern Control," (i.e., out of sequence control rod withdrawals) must be made in the individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. Furthermore, since the control rod scram function with the RCS at atmospheric pressure relies solely on the CRD accumulator, it is essential that the CRD charging water header remain pressurized. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

ACTIONS

A.1

With one or more control rods discovered uncoupled during this Special Operation, a controlled insertion of each uncoupled control rod is required; either to attempt recoupling, or to preclude a control rod drop. This controlled insertion is preferred since, if the control rod

(continued)

BASES

ACTIONS

B.1 (continued)

MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e, made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met. However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These Surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior

(continued)

TS - 353S1

UNIT 2

REVISED PAGES

TABLE OF CONTENTS

1.1 Definitions (continued)

LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 (Deleted)

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required channels inoperable.	A.1	Place channel in trip.	12 hours
	<u>OR</u>		
	A.2	--------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. --------------------	
		Place associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. -------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	<u>OR</u>		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTES-------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	 7 days
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	 7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 effective full power hours
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	92 days

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted)	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.13	-----------------NOTES----------------- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --------------------------------------- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	18 months
SR 3.3.1.1.16	-----------------NOTE----------------- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --------------------------------------- Perform CHANNEL FUNCTIONAL TEST.	184 days

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. 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.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(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.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch - Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	<u>AND</u>		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	form CHANNEL FUNCTIONAL TEST.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. --------------------------------------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. --------------------------------------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	------------------NOTE------------------ Neutron detectors are excluded. --------------------------------------- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	18 months
SR 3.3.2.1.6	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. --------------------------------------- Perform CHANNEL FUNCTIONAL TEST.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.7	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM
SR 3.3.2.1.8	------------------NOTE------------------ Neutron detectors are excluded. -- Verify the RBM: a. Low Power Range -- Upscale Function is not bypassed when THERMAL POWER is ≥ 28% and ≤ 63% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 63% and ≤ 83% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 83% RTP.	18 months

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range -- Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range -- Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range -- Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. Inop	(g),(h)	2	SR 3.3.2.1.1	NA
e. Downscale	(g),(h)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(i)
2. Rod Worth Minimizer	1(c),2(c)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch - Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 28% and ≤ 63% RTP and MCPR < 1.75.

(b) THERMAL POWER > 63% and ≤ 83% RTP and MCPR < 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 83% and < 90% RTP and MCPR < 1.75.

(g) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(h) THERMAL POWER ≥ 28% and < 90% RTP and MCPR < 1.75.

(i) Greater than or equal to the Allowable Value specified in the COLR.

BASES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	------------------NOTE------------------ Not required to be performed until 24 hours after both recirculation loops are in operation. --- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
SR 3.4.1.2	Verify the reactor is outside of Region I and II of Figure 3.4.1-1.	Immediately after any increase > 5% RTP while initial core flow is < 50% of rated. <u>AND</u> Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated.

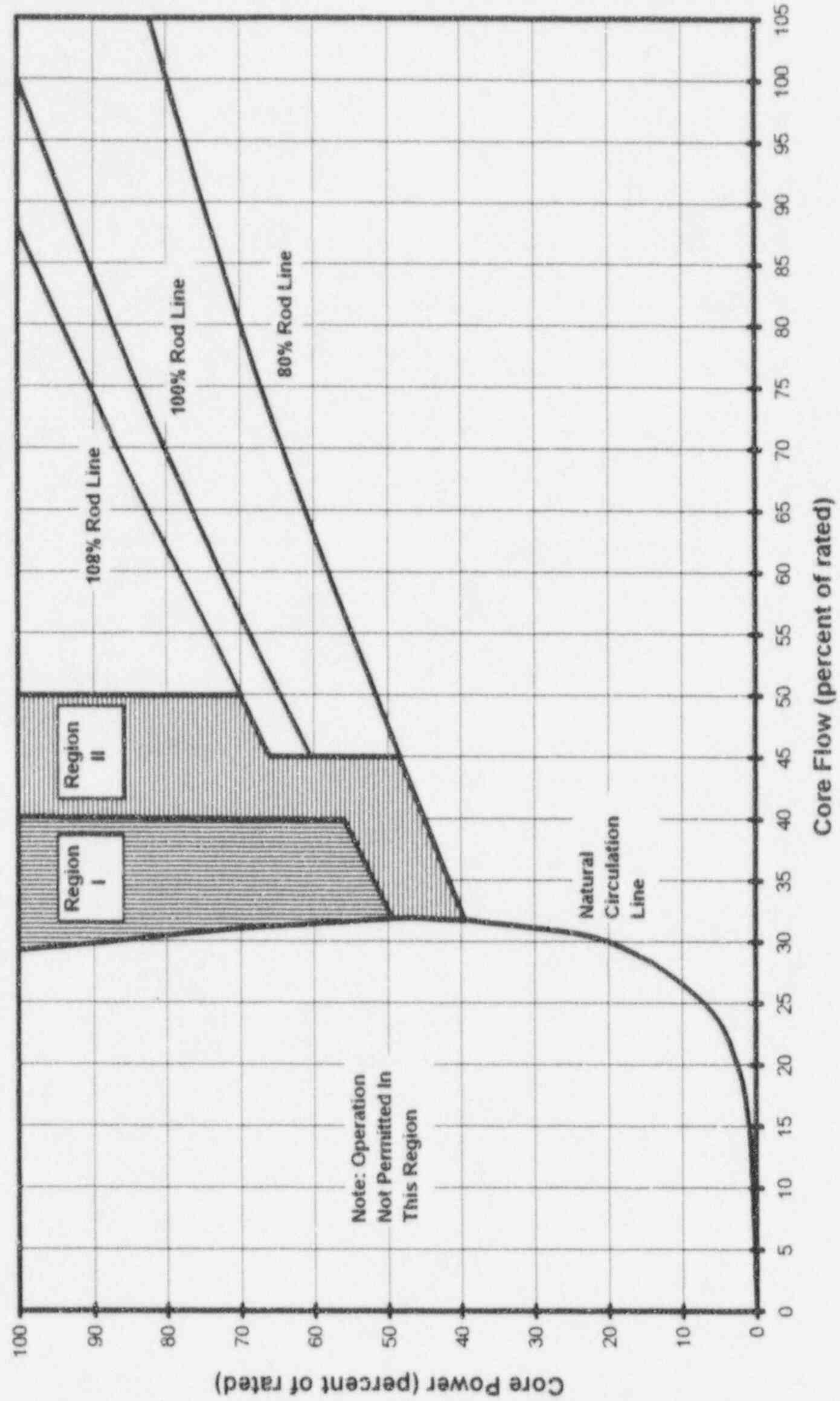

Figure 3.4.1-1

THERMAL POWER VERSUS
CORE FLOW STABILITY
REGIONS

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence (BPWS) requirements of SR 3.3.2.1.7 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of BPWS control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure $\geq$ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1.	According to the applicable SRs
SR 3.10.8.2	-------------------NOTE------------------- Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	-------------------NOTE------------------- Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

TABLE OF CONTENTS

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Reference 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Reference 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.

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.

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.

(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.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-11 "General Electric Standard Application for Reactor Fuel," November 1995.

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 1, February 1996.

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.

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, 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the

(continued)

BASES

REFERENCES (continued)

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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 (Deleted)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux—High (continued)

unexpected reactivity excursions. In MODE 1, the APRM System and the RBM provide protection against control rod withdrawal error events and the IRMs are not required.

1.b. Intermediate Range Monitor—Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor - Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux - High Function is required.

Average Power Range Monitor

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Average Power Range Monitor (continued)

indication of average reactor power from a few percent to greater than RTP.

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs to each RPS trip system logic channel (A1, A2, B1, or B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM functions, at least twenty (20) LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

2.a. Average Power Range Monitor Neutron Flux—High, Setdown

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, Setdown

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High, Setdown (continued)

Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux — High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux — High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power — High

The Average Power Range Monitor Flow Biased Simulated Thermal Power — High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux — High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power — High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux - High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function setpoint is exceeded.

Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this function.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.c. Average Power Range Monitor Fixed Neutron Flux—High

The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux - High Function to terminate the CRDA.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux - High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux - High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux - High Function is not required in MODE 2.

2.d. Average Power Range Monitor—Inop

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions. This Function (Inop) provides assurance that a minimum number of APRMs are OPERABLE. For any APRM channel, any time its mode switch is in any position other than "Operate," an APRM module is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor—Inop (continued)

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

2.e. 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux - High

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Vessel Steam Dome Pressure—High (continued)

signal, not the Reactor Vessel Steam Dome Pressure - High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure - High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level—Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low, Level 3 Function is assumed in the analysis of the recirculation line break (Ref. 6). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low, Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low, Level 3 Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level—Low, Level 3 (continued)

ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value is selected to ensure that (a) during normal operation the steam dryer skirt is not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder), and (b) for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water - Low Low Low, Level 1 will not be required.

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

5. Main Steam Isolation Valve—Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 7 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow).

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Main Steam Isolation Valve—Closure (continued)

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve - Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve - Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines must close in order for a scram to occur.

The Main Steam Isolation Valve - Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the Main Steam Isolation Valve - Closure Function, with eight channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

6. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure - High Function is a secondary scram signal to Reactor Vessel Water Level - Low, Level 3 for LOCA events inside the drywell. However, no credit is taken for a scram initiated from this Function for any of the DBAs analyzed in the FSAR. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Drywell Pressure—High (continued)

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

7a, 7b. Scram Discharge Volume Water Level—High

The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. Should this volume fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volume is still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level - High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the FSAR. However, they are retained to ensure the RPS remains OPERABLE.

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two thermal probes for a total of eight level signals. The outputs of these devices are arranged so that there is a signal from a level switch and a thermal probe to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 8.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7a, 7b. Scram Discharge Volume Water Level—High (continued)

Four channels of each type of Scram Discharge Volume Water Level - High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve - Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure transmitter is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Two channels of Reactor Mode Switch - Shutdown Position Function, with one channel in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch - Shutdown Position Function is required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

10. Reactor Mode Switch—Shutdown Position (continued)

OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

11. Manual Scram

The Manual Scram push button channels provide signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is one Manual Scram push button channel for each of the two RPS manual scram logic channels. In order to cause a scram it is necessary that each channel in both manual scram trip systems be actuated.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of Manual Scram with one channel in each manual scram trip system are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

12. RPS Channel Test Switches

There are four RPS Channel Test Switches, one associated with each of the four automatic scram logic channels (A1, A2, B1, and B2). These keylock switches allow the operator to test the OPERABILITY of each individual logic channel without the necessity of using a scram function trip. When the RPS Channel Test Switch is placed in test, the

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

12. RPS Channel Test Switches (continued)

associated scram logic channel is deenergized and OPERABILITY of the channel's scram contactors can be confirmed. The RPS Channel Test Switches are not specifically credited in the accident analysis. However, because the Manual Scram Function at Browns Ferry Nuclear Plant is not configured the same as the generic model in Reference 9, the RPS Channel Test Switches are included in the analysis in Reference 11. Reference 11 concludes that the Surveillance Frequency extensions for RPS functions, described in Reference 9, are not affected by the difference in configuration since each automatic RPS channel has a test switch which is functionally the same as the manual scram switches in the generic model. Weekly testing of scram contactors is credited in Reference 9 with supporting the Surveillance Frequency extension of the RPS functions.

There is no Allowable Value for this Function since the channels are mechanically actuated solely on the position of the switches.

Four channels of the RPS Channel Test Switch Function with two channels in each trip system arranged in a one-out-of-two logic are available and required to be OPERABLE. The function is required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

13. Low Scram Pilot Air Header Pressure

The Low Scram Pilot Air Header Pressure trip performs the same function as the high water level in the scram discharge instrument volume for fast fill events in which the high level instrument response time may not be adequate. A fast fill event is postulated for certain degraded control air events in which the scram outlet valves unseat enough to allow 5 gpm per drive leakage into the scram discharge volume but not enough to cause rod insertion.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Low Scram Pilot Air Header Pressure (continued)

Four channels of Low Scram Pilot Air Header Pressure Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 9 and 12) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in References 9 or 12 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

(continued)

ACTIONS

B.1 and B.2 (continued)

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in References 9 or 12, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, and 2.d, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

(continued)

BASES

ACTIONS (continued)

E.1, F.1, and G.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Action E.1 is consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains RPS trip

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

A CHANNEL NCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 9).

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 9. (The RPS Channel Test Switch Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 and SR 3.3.1.1.6 (continued)

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 effective full power hours Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing - applicable to Function 2.b only), the 2-out-of-4 voter channels, and the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16 (continued)

interface connections into the RPS trip systems from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions is based on the reliability analysis of Reference 12. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.14.) A Note for SR 3.3.1.1.16 is provided that requires the APRM Function 2.a SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The 184 day Frequency of SR 3.3.1.1.16 for the scram pilot air header low pressure trip function is based on the functional reliability previously demonstrated by this function, the need for minimizing the radiation exposure associated with the functional testing of this function, and the increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by the more frequent functional testing.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.13 also includes calibrating the associated recirculation loop flow channel. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 also includes physical inspection and actuation of the switches.

Note 1 to SR 3.3.1.1.9 and SR 3.3.1.1.13 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 and SR 3.3.1.1.13 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

The Frequency of 18 months is based on engineering judgment and reliability of the components.

REFERENCES

1. FSAR, Section 7.2.
2. FSAR, Chapter 14.
3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
4. FSAR, Appendix N.
5. FSAR, Section 14.6.2.
6. FSAR, Section 6.5.
7. FSAR, Section 14.5.
8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
9. NEDC-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
11. MED-32-0286, "Technical Specification Improvement Analysis for Browns Ferry Nuclear Plant, Unit 2," October 1995.
12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch – Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate or high) is enabled. If the APRM is indicating less than the low power setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

(continued)

BASES

BACKGROUND (continued)

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating $\geq$ 28% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR $\geq$ 1.75, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at $\geq$ 90% RTP with MCPR $\geq$ 1.44, no RWE event will result in exceeding the MCPR

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE.

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, and 7. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

The RWM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Since the RWM is designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is < 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Refs. 5 and 7). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Rod Worth Minimizer (continued)

consequences of a CRDA are acceptable, since the reactor will be subcritical.

3. Reactor Mode Switch—Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch – Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch – Shutdown Position Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels are required to be OPERABLE to ensure that no single channel failure will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

During shutdown conditions (MODE 3, 4, or 5), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock (LCO 3.9.2, "Refuel Position One-Rod-Out Interlock") provides the required control rod withdrawal blocks.

A.1

With one RBM channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result

(continued)

BASES

ACTIONS

A.1 (continued)

in no control rod block capability for the RBM. For this reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM during withdrawal of one or more of the first 12 rods was not performed in the last 12 months. These requirements minimize the number of reactor startups initiated with the RWM inoperable. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed, control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2.

(continued)

BASES

ACTIONS

C.1, C.2.1.1, C.2.1.2, and C.2.2 (continued)

Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff (e.g., a qualified shift technical advisor or reactor engineer). The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch – Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch – Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform a channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 184 days is based on reliability analyses (Ref. 11).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. This test is performed as soon as possible after the applicable conditions are entered. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is reduced to ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and THERMAL POWER reduction to ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch - Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch - Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.7

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

SR 3.3.2.1.8

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1 and the COLR, each within a specific power range. The powers at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.
3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.

(continued)

BASES

REFERENCES (continued)

4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).

5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (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)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.e of

(continued)

BASES

LCO (continued)

Table 3.3.1.1-1 must be enforced and the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2), or must be verified by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). To provide additional protection against an inadvertent criticality, control rod withdrawals that do not conform to the banked position withdrawal sequence specified in LCO 3.1.6, "Rod Pattern Control," (i.e., out of sequence control rod withdrawals) must be made in the individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. Furthermore, since the control rod scram function with the RCS at atmospheric pressure relies solely on the CRD accumulator, it is essential that the CRD charging water header remain pressurized. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

ACTIONS

A.1

With one or more control rods discovered uncoupled during this Special Operation, a controlled insertion of each uncoupled control rod is required; either to attempt recoupling, or to preclude a control rod drop. This controlled insertion is preferred since, if the control rod

(continued)

BASES

ACTIONS

B.1 (continued)

MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e, made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met. However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These Surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior

(continued)

TS - 353S1

UNIT 3

REVISED PAGES

TABLE OF CONTENTS

1.1 Definitions (continued)

LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 (Deleted)

3.3 INSTRUMENTATION

3.3.1.1 Reactor Prutection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 --------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. -------------------- Place associated trip system in trip.	12 hours 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, or 2.d. -------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1 OR B.2	Place channel in one trip system in trip. Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.1.1.2	NOTE Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	NOTE Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 effective full power hours
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	92 days

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted)	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.13	-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	18 months
SR 3.3.1.1.16	-----NOTE----- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL FUNCTIONAL TEST	184 days

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. 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.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.8	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(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.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch – Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	18 months
SR 3.3.2.1.6	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.7	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM
SR 3.3.2.1.8	------------------NOTE------------------ Neutron detectors are excluded. -- Verify the RBM: a. Low Power Range -- Upscale Function is not bypassed when THERMAL POWER is $\geq$ 28% and $\leq$ 63% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 63% and $\leq$ 83% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 83% RTP.	18 months

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range -- Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range -- Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range -- Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. Inop	(g),(h)	2	SR 3.3.2.1.1	NA
e. Downscale	(g),(h)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(i)
2. Rod Worth Minimizer	1(c),2(c)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch – Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 28% and ≤ 63% RTP and MCPR < 1.75.

(b) THERMAL POWER > 63% and ≤ 83% RTP and MCPR < 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 83% and < 90% RTP and MCPR < 1.75.

(g) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(h) THERMAL POWER ≥ 28% and < 90% RTP and MCPR < 1.75.

(i) Greater than or equal to the Allowable Value specified in the COLR.

BASES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	--------------------NOTE------------------ Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
SR 3.4.1.2	Verify the reactor is outside of Region I and II of Figure 3.4.1-1.	Immediately after any increase > 5% RTP while initial core flow is < 50% of rated. <u>AND</u> Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated.

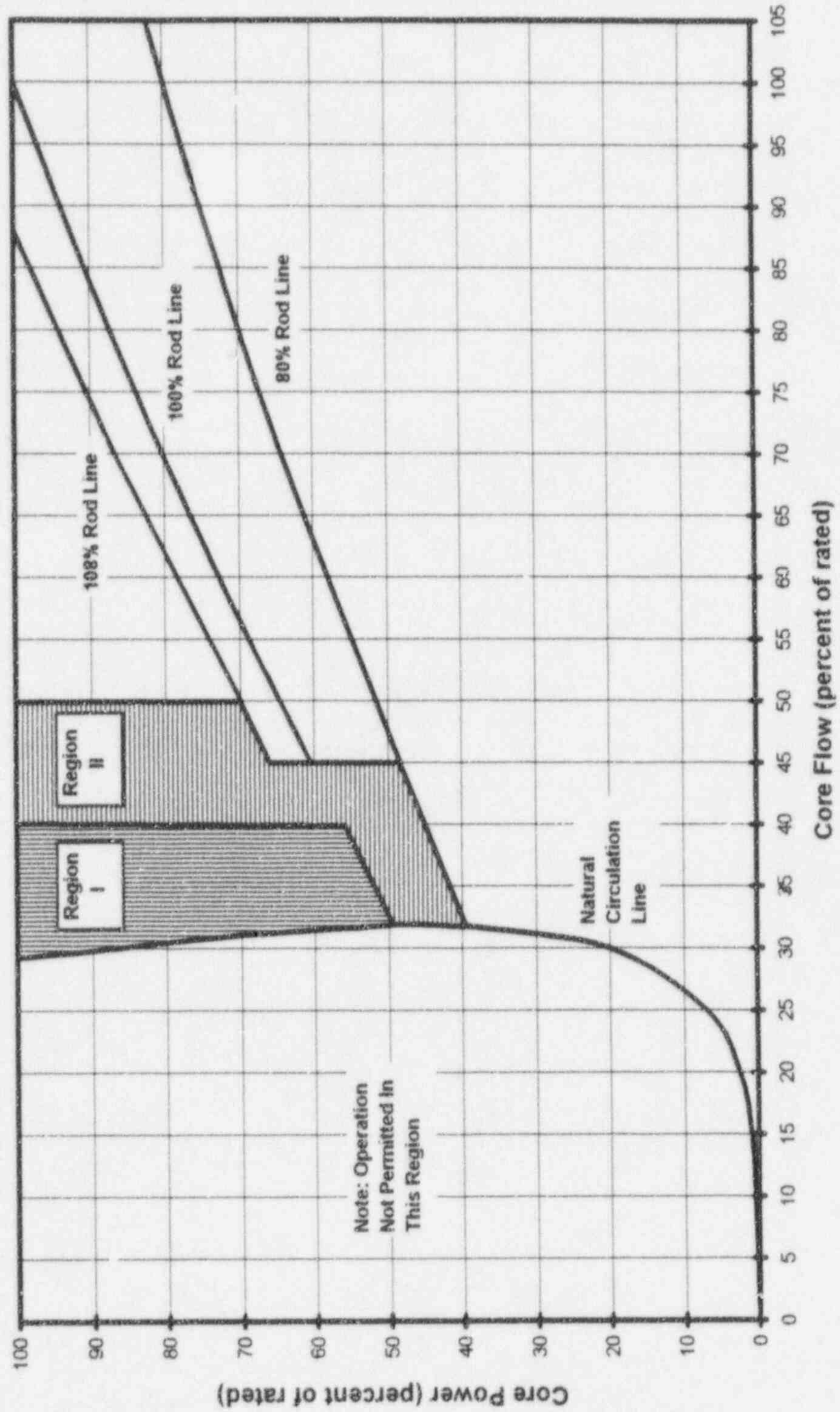

Figure 3.4.1-1

THERMAL POWER VERSUS
CORE FLOW STABILITY
REGIONS

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Functions 2.a, 2d, and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence (BPWS) requirements of SR 3.3.2.1.7 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of BPWS control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e ofTable 3.3.1.1-1.	According to the applicable SRs
SR 3.10.8.2	-------------------NOTE------------------- Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	-------------------NOTE------------------- Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

TABLE OF CONTENTS

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Reference 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Reference 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.

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.

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.

(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.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-11 "General Electric Standard Application for Reactor Fuel," November 1995.
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 1, February 1996.
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.

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, 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the

(continued)

BASES

REFERENCES (continued)

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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 (Deleted)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Intermediate Range Monitor Neutron Flux—High (continued)

unexpected reactivity excursions. In MODE 1, the APRM System and the RBM provide protection against control rod withdrawal error events and the IRMs are not required.

1.b. Intermediate Range Monitor—Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or when a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor - Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux - High Function is required.

Average Power Range Monitor

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Average Power Range Monitor (continued)

indication of average reactor power from a few percent to greater than RTP.

The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs to each RPS trip system logic channel (A1, A2, B1, or B2). Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM functions, at least twenty (20) LPRM inputs, with at least three (3) LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel.

2.a. Average Power Range Monitor Neutron Flux—High, Setdown

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, Setdown

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High, Setdown (continued)

Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux - High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux - High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function setpoint is exceeded.

Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this function.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.c. Average Power Range Monitor Fixed Neutron Flux—High

The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux - High Function to terminate the CRDA.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux - High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux - High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux - High Function is not required in MODE 2.

2.d. Average Power Range Monitor—Inop

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions. This Function (Inop) provides assurance that a minimum number of APRMs are OPERABLE. For any APRM channel, any time its mode switch is in any position other than "Operate," an APRM module is unplugged, or the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor—Inop (continued)

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

2.e. 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

There is no Allowable Value for this Function.

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux - High

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Vessel Steam Dome Pressure—High (continued)

signal, not the Reactor Vessel Steam Dome Pressure - High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure - High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level—Low, Level 3

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low, Level 3 Function is assumed in the analysis of the recirculation line break (Ref. 6). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low, Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low, Level 3 Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level—Low, Level 3 (continued)

ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value is selected to ensure that (a) during normal operation the steam dryer skirt is not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder), and (b) for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water - Low Low Low, Level 1 will not be required.

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

5. Main Steam Isolation Valve—Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 7 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow).

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Main Steam Isolation Valve—Closure (continued)

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve - Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve - Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines must close in order for a scram to occur.

The Main Steam Isolation Valve - Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the Main Steam Isolation Valve - Closure Function, with eight channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

6. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure - High Function is a secondary scram signal to Reactor Vessel Water Level - Low, Level 3 for LOCA events inside the drywell. However, no credit is taken for a scram initiated from this Function for any of the DBAs analyzed in the FSAR. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Drywell Pressure—High (continued)

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

7a, 7b. Scram Discharge Volume Water Level—High

The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. Should this volume fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volume is still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level - High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the FSAR. However, they are retained to ensure the RPS remains OPERABLE.

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two thermal probes for a total of eight level signals. The outputs of these devices are arranged so that there is a signal from a level switch and a thermal probe to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 8.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7a, 7b. Scram Discharge Volume Water Level—High (continued)

Four channels of each type of Scram Discharge Volume Water Level - High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve - Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure transmitter is associated with each control valve, and the signal from each transmitter is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Two channels of Reactor Mode Switch - Shutdown Position Function, with one channel in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch - Shutdown Position Function is required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

10. Reactor Mode Switch—Shutdown Position (continued)

OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

11. Manual Scram

The Manual Scram push button channels provide signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

There is one Manual Scram push button channel for each of the two RPS manual scram logic channels. In order to cause a scram it is necessary that each channel in both manual scram trip systems be actuated.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Two channels of Manual Scram with one channel in each manual scram trip system are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

12. RPS Channel Test Switches

There are four RPS Channel Test Switches, one associated with each of the four automatic scram logic channels (A1, A2, B1, and B2). These keylock switches allow the operator to test the OPERABILITY of each individual logic channel without the necessity of using a scram function trip. When the RPS Channel Test Switch is placed in test, the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

12. RPS Channel Test Switches (continued)

associated scram logic channel is deenergized and OPERABILITY of the channel's scram contactors can be confirmed. The RPS Channel Test Switches are not specifically credited in the accident analysis. However, because the Manual Scram Function at Browns Ferry Nuclear Plant is not configured the same as the generic model in Reference 9, the RPS Channel Test Switches are included in the analysis in Reference 11. Reference 11 concludes that the Surveillance Frequency extensions for RPS functions, described in Reference 9, are not affected by the difference in configuration since each automatic RPS channel has a test switch which is functionally the same as the manual scram switches in the generic model. Weekly testing of scram contactors is credited in Reference 9 with supporting the Surveillance Frequency extension of the RPS functions.

There is no Allowable Value for this Function since the channels are mechanically actuated solely on the position of the switches.

Four channels of the RPS Channel Test Switch Function with two channels in each trip system arranged in a one-out-of-two logic are available and required to be OPERABLE. The function is required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

13. Low Scram Pilot Air Header Pressure

The Low Scram Pilot Air Header Pressure trip performs the same function as the high water level in the scram discharge instrument volume for fast fill events in which the high level instrument response time may not be adequate. A fast fill event is postulated for certain degraded control air events in which the scram outlet valves unseat enough to allow 5 gpm per drive leakage into the scram discharge volume but not enough to cause rod insertion.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	13. Low Scram Pilot Air Header Pressure (continued)
	Four channels of Low Scram Pilot Air Header Pressure Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are tbe MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 9 and 12) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function's inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternatively, if it is not desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic, for any Function, would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in References 9 or 12 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels OPERABLE or in trip (or any combination) in one trip system.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Completing one of these Required Actions restores RPS to a reliability level equivalent to that evaluated in References 9 or 12, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, and 2.d. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM channels for which Condition B applies. For an inoperable APRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring operability) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required

(continued)

ACTIONS

B.1 and B.2 (continued)

Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, and 2.d, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

(continued)

BASES

ACTIONS (continued)

E.1, F.1, and G.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Action E.1 is consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains RPS trip

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continue)

capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary.

SR 3.3.1.1.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 9).

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 9. (The RPS Channel Test Switch Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 and SR 3.3.1.1.6 (continued)

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 effective full power hours Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing - applicable to Function 2.b only), the 2-out-of-4 voter channels, and the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16 (continued)

interface connections into the RPS trip systems from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.16 for the APRM Functions is based on the reliability analysis of Reference 12. (NOTE: The actual voting logic of the 2-out-of-4 Voter Function is tested as part of SR 3.3.1.1.14.) A Note for SR 3.3.1.1.16 is provided that requires the APRM Function 2.a SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The 184 day Frequency of SR 3.3.1.1.16 for the scram pilot air header low pressure trip function is based on the functional reliability previously demonstrated by this function, the need for minimizing the radiation exposure associated with the functional testing of this function, and the increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by the more frequent functional testing.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.13 also includes calibrating the associated recirculation loop flow channel. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 also includes physical inspection and actuation of the switches.

Note 1 to SR 3.3.1.1.9 and SR 3.3.1.1.13 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 and SR 3.3.1.1.13 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

(Deleted)

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SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

The Frequency of 18 months is based on engineering judgment and reliability of the components.

REFERENCES

1. FSAR, Section 7.2.
2. FSAR, Chapter 14.
3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
4. FSAR, Appendix N.
5. FSAR, Section 14.6.2.
6. FSAR, Section 6.5.
7. FSAR, Section 14.5.
8. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
9. NEDC-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
11. MED-32-0286, "Technical Specification Improvement Analysis for Browns Ferry Nuclear Plant, Unit 2," October 1995.
12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch – Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate or high) is enabled. If the APRM is indicating less than the low power setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

(continued)

BACKGROUND (continued)

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 28% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.75, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.44, no RWE event will result in exceeding the MCPR

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a second Note (Note 2) to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform a channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 184 days is based on reliability analyses (Ref. 11).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. This test is performed as soon as possible after the applicable conditions are entered. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

1 hour after THERMAL POWER is reduced to ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and THERMAL POWER reduction to ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

SR 3.3.2.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch - Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch - Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 18 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.7

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.8

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1 and the COLR, each within a specific power range. The powers at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.
3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.
4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).
5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

(continued)

BASES

REFERENCES (continued)

6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (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).

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

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.

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BASES

APPLICABLE SAFETY ANALYSES (continued)	CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.
LCO	As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.e of

(continued)

BASES

LCO (continued)

Table 3.3.1.1-1 must be enforced and the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2), or must be verified by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). To provide additional protection against an inadvertent criticality, control rod withdrawals that do not conform to the banked position withdrawal sequence specified in LCO 3.1.6, "Rod Pattern Control," (i.e., out of sequence control rod withdrawals) must be made in the individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. Furthermore, since the control rod scram function with the RCS at atmospheric pressure relies solely on the CRD accumulator, it is essential that the CRD charging water header remain pressurized. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

ACTIONS

A.1

With one or more control rods discovered uncoupled during this Special Operation, a controlled insertion of each uncoupled control rod is required; either to attempt recoupling, or to preclude a control rod drop. This controlled insertion is preferred since, if the control rod

(continued)

BASES

ACTIONS

B.1 (continued)

MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e, made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met. However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These Surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior

(continued)