Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

November 10, 2003

TVA-BFN-TS-430

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-259
Tennessee Valley Authority)

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE 430 - POWER RANGE NEUTRON MONITOR UPGRADE WITH IMPLEMENTATION OF AVERAGE POWER RANGE MONITOR AND ROD BLOCK MONITOR TECHNICAL SPECIFICATION IMPROVEMENTS AND MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSES

Pursuant to 10 CFR 50.90, Tennessee Valley Authority (TVA) is submitting a request for a TS change (TS-430) to license DPR-33 for BFN Unit 1. The proposed TS change includes the necessary TS revisions for the planned replacement of the power range monitoring portion of the existing Neutron Monitoring System with a digital upgrade. This portion of the application is based on Licensing Topical Report NEDC-32410P-A, Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitor (PRNM) Retrofit Plus Option III Stability Trip Functions, which was formally accepted for use in licensing applications as documented in an NRC Safety Evaluation Report, dated September 5, 1995.

These proposed TS changes are needed for TVA's planned implementation of the 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). The ARTS/MELLL changes enhance operating flexibility and efficiency by implementing RBM design improvements, incorporating APRM/RBM Technical Specification improvements, and expanding the current allowable operating

A001

domain to the MELLL region of the power/flow chart. NEDC-32433P, Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2, and 3, is the basis document for this portion of the submittal.

Additionally, implementation of the PRNM system will also allow TVA to proceed with implementation of the long-term stability solution. Prior to enabling the Oscillation Power Range Monitor (OPRM) trip function, related TS changes will be submitted.

These proposed changes have previously been approved and implemented on Browns Ferry Units 2 and 3 (References 1 and 2). Minor differences between the proposed Unit 1 changes and the Units 2 and 3 precedent are discussed in Section 3.4 of Enclosure 1. As also discussed in that section, the technical analysis submitted for this Unit 1 TS change incorporates the same elements previously submitted in support of the previous TS changes for Units 2 and 3.

The proposed TS changes are necessary to support the restart of Unit 1 and improve the fidelity with Units 2 and 3. In addition, TVA plans to make subsequent changes to portions of these TS in order to implement both the extended power uprate and a 24-month fuel cycle for Unit 1 prior to restart. Therefore, TVA requests that the amendment be approved by December 1, 2004.

TVA has determined that there are no significant hazards considerations associated with the proposed amendment and that the amendment qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and attachments to the Alabama State Department of Public Health.

Enclosure 1 provides TVA's evaluation of the TS change. Enclosure 2 provides mark-ups of the proposed change to the TS and TS Bases pages. Enclosure 3 provides retyped TS and TS Bases pages.

There are no regulatory commitments associated with this submittal. If you have any questions about this amendment, please contact me at (256)729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 10, 2003.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures:

1. TVA Evaluation of Proposed Change

2. Proposed changes to the Technical Specifications and the Technical Specification Bases (mark-ups)

3. Proposed changes to the Technical Specifications and the Technical Specification Bases (retyped)

References:

1. NRC letter, J.F. Williams to O.D. Kingsley, dated September 11, 1997, "Issuance of Amendment - Browns Ferry Nuclear Plant Unit 2 (TAC No. M92504) (TS 353).

2. NRC letter, A.W. DeAgazio to J.A. Scalice, dated September 3, 1998, "Amendment No. 213 to Facility Operating License No. DPR-68: Power Range Neutron Monitor Upgrade with Implementation of Average Power Range Monitor and Rod Block Monitor Technical Specification Improvements and Maximum Extended Load Line Limit Analyses - Technical Specification Change TS-353 (TAC No. M92505)

Enclosure
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ENCLOSURE 1

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

TECHNICAL SPECIFICATION (TS) CHANGE 430 -
POWER RANGE NEUTRON MONITOR UPGRADE WITH IMPLEMENTATION
OF AVERAGE POWER RANGE MONITOR AND ROD BLOCK MONITOR
TECHNICAL SPECIFICATION IMPROVEMENTS AND MAXIMUM
EXTENDED LOAD LINE LIMIT ANALYSES

TVA EVALUATION OF PROPOSED CHANGE

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1.0 DESCRIPTION

This letter requests an amendment to license DPR-33 for BFN Unit 1. The proposed TS change accomplishes two objectives:

1. 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 Neutron Monitor (NUMAC PRNM) retrofit design.

2. 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.

The proposed amendment is necessary to support the restart of Unit 1 and improves the fidelity with Units 2 and 3. In addition, TVA plans to make subsequent changes to portions of these TS in order to implement both the extended power uprate and a 24-month fuel cycle for Unit 1 prior to restart. Therefore, TVA requests that the amendment be approved by December 1, 2004.

2.0 PROPOSED CHANGE

As mentioned above, the proposed TS change accomplishes two objectives:

1. Replacement of the power range portion of the existing NMS GE digital NUMAC PRNM retrofit design. As part of the planned modification, the number of APRM instrument channels will be reduced from six to four. The Local Power Range Monitor (LPRM) inputs to the APRMs will also be reconfigured. The four APRM instrument channels will be combined in four 2-out-of-4 trip logic channels which provide input to the Reactor Protection System (RPS) trip channels. The number of recirculation flow instrument channels associated with the APRMs will be increased from two total flow channels (four transmitters) to four total flow channels (eight transmitters). TS changes related to this activity are consistent with the recommendations of NEDC-32410P-A (Reference 1).

2. Implementation of APRM and RBM TS improvements and operation in an expanded core power/flow domain, the MELLL region. RBM modifications and APRM setpoint changes required to implement the proposed ARTS/MELLL operation are included in the NUMAC PRNM design. The proposed expanded operating region above the rated (design) power/flow control line is bounded by the rated (100%) power line and the power/flow control line which passes through the 100% power/75% core flow point (approximately the 121% rod line). TS changes related to this activity are consistent with the recommendations contained in NEDC-32433P (Reference 2).

The proposed changes to the BFN Unit 1 TS are provided below. Applicable TS page numbers and sections are identified. Each change listed below is associated with the Group 1 [PRNM] activity, with the Group 2 [ARTS/MELLL] activity, or [Administrative] as marked.

1. Page i, Table of Contents [ARTS/MELLL]

   The listing of Section 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoints, has been deleted.

2. Page 1.1-5, Section 1.1 [ARTS/MELLL]

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

3. Pages 3.2-7 and 3.2-8, 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.

4. 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
		<u>OR</u>		
		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
		<u>OR</u>		
		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

5. 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

6. 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

7. 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.

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

For SR 3.3.1.1.13, add a new note excluding the neutron detectors. After the change SR 3.3.1.1.13 will read as follows:

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	------------------NOTE------------------ Neutron detectors are excluded. Perform CHANNEL CALIBRATION.	18 months

9. Page 3.3-5, Section 3.3.1.1 [PRNM]

A new CHANNEL FUNCTIONAL TEST surveillance (SR 3.3.1.1.16) with frequency of 184 days is added. After the change, SR 3.3.1.1.16 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

10. 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. The applicable Surveillance Requirements are revised.

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, along with the current Note b (which is being renumber to Note c).

Before the two changes described above, 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(b)
c. Neutron Flux — High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120% RTP
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

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

After the two changes, 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)
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.
(c) [0.66 W + 71% - 0.66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

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

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

12. 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.

13. 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 SR 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 ≥ 27% and ≤ 62% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 62% and ≤ 82% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 82% RTP.	18 months

14. Page 3.3-20, Table 3.3.2.1-1 [ARTS/MELLL]

The RBM portion of Table 3.3.2.1-1 and associated notes are revised 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 < 1.44.

(b) THERMAL POWER ≥ 29% and < 90% and MCPR < 1.75.

....

(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 ≥ 27% and ≤ 62% RTP and MCPR less than the value specified in the COLR.

(b) THERMAL POWER > 62% and ≤ 82% RTP and MCPR less than the value specified in the COLR.

....

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

(f) THERMAL POWER > 82% and < 90% RTP and MCPR less than the value specified in the COLR.

(g) THERMAL POWER ≥ 90% RTP and MCPR less than the value specified in the COLR.

(h) THERMAL POWER ≥ 27% AND < 90% RTP and MCPR less than the value specified in the COLR.

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

15. 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%.

16. 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.

17. Page 3.10-22, 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.

18. Page 3.10-24, 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.

<u>Changes to the Technical Specification Bases:</u>

19. Page B i, Table of Contents **[ARTS/MELLL]**

The listing of Section 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," has been deleted.

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

Two changes are made as described below:

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

B) 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 the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Ref. 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 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.

21. Pages B 3.2-2 and B 3.2-5, Section B 3.2.1 **[Administrative]**

The current Reference 7 and the citations in the text to this reference are changed to Reference 10 for consistency with the numbering in the current Units 2 and 3 TS Bases. New References 7 through 9 are added as discussed below.

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

The following is added at the end of the first sentence 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.

23. Page B 3.2-5, 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.

24. Pages B 3.2-7 and 3.2-10, Section B 3.2.2 **[Administrative]**

The current Reference 8 and the citations in the text to this reference are changed to Reference 10 for consistency with the numbering in the current Units 2 and 3 TS Bases. New References 8 and 9 are added as discussed below.

25. Pages B 3.2-7, Section B 3.2.2 **[ARTS/MELLL]**

Three changes are made as described below:

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

B) 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.

C) 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.

26. Page B 3.2-10, 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.

27. Pages B 3.2-15 through B 3.2-22 [ARTS/MELLL]

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

28. Page B 3.3-9, 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.

29. Pages B 3.3-9 & -10, Section B 3.3.1.1 [PRNM]

Three changes are made as discussed below:

A) In the Bases description of APRM Function 2.a, the following text is deleted (moved to the discussion described 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.

B) In the Bases description of APRM Function 2.a, the description for Function 2.a is changed from "Neutron Flux - High, Setdown" to "Neutron Flux - High, (Setdown)".

C) 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.

30. Pages B 3.3-12, 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.

31. Pages B 3.3-13 and B 3.3-14, Section B 3.3.1.1 [PRNM]

Three changes are made in the Bases description of APRM Function 2.c as described below:

A) The following text is deleted (moved to the discussion described above):

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

B) 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.

C) The description for Function 2.a is changed from "Neutron Flux - High, Setdown" to "Neutron Flux - High, (Setdown)".

32. Page B 3.3-15, 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.

33. Pages B 3.3-16, Section B 3.3.1.1 [PRNM]

Three changes are made as described below:

A) 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.

B) The first paragraph of this section is replaced with two new paragraphs 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 the 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 unbypassed 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.

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

34. Page B 3.3-16, Section B 3.3.1.1 [PRNM]

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

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.

35. Page B 3.3-30, Section B 3.3.1.1 [PRNM]

Two changes are made as described below:

A) Under the Bases discussion of Actions A.1 and A.2, RPS Channel Test Switches, a new reference, Reference 12, is added.

B) 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.

36. Pages B 3.3-31 and B 3.3-32, Section B 3.3.1.1 **[PRNM]**

Three changes are made as described below:

A) Under the Bases discussion of Actions B.1 and B.2, a new reference, Reference 12, is added in two places.

B) 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.

C) Under the Bases discussion of Action C.1, the reference to the APRM function in the discussion of the typical one-out-of-two taken twice logic is deleted.

37. Page B 3.3-36, Section B 3.3.1.1 **[ARTS/MELLL]**

Under the Bases discussion of SR 3.3.1.1.2, the following text is deleted in the middle of the first paragraph:

> 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.

38. Page B 3.3-37, Section B 3.3.1.1 [PRNM]

Under the Bases discussion of SR 3.3.1.1.3, the reference to APRM functions has been deleted.

39. Pages B 3.3-39 and B 3.3-40, Section B 3.3.1.1 [PRNM]

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

40. Page B 3.3-40, 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.

41. Pages B 3.3-40 and B 3.3-41, Section B 3.3.1.1 [PRNM]

Three changes are made in the Bases discussion of SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 as discussed below:

A) In the first paragraph, the following sentence is added:

> For the APRM Simulated Thermal Power-High Function, SR 3.3.1.1.13 also includes calibrating the associated recirculation loop flow channel.

B) The description of notes was revised to reference the applicability of SR 3.3.1.1.13.

C) The description has been revised to reflect the deletion of the second Note in SR 3.3.1.1.13.

42. Page B 3.3-42, 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."

43. Page B 3.3-42, 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.

44. Page B 3.3-44, 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.

45. Page B 3.3-46, Section B 3.3.1.2 **[PRNM]**

In the Bases description of APRM Function 2.a, the description for Function 2.a is changed from "Neutron Flux - High, Setdown" to "Neutron Flux - High, (Setdown)".

46. Pages B 3.3-57 and 3.3-58, 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.

47. Pages B 3.3-59 and 3.3-60, Section B 3.3.2.1 **[ARTS/MELLL]**

In the section entitled "Applicable Safety Analyses, LCO, and Applicability, Section 1. Rod Block Monitor", four changes are made as discussed below:

A) 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.

B) 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.

C) In the first sentence of the fifth paragraph in this section, the limit above which the RBM is assumed to mitigate the consequences of a Rod Withdraw Error (RWE) event is changed from "≥ 29% Rated Thermal Power (RTP)" to "≥ 27% RTP."

D) Also, in the fifth paragraph, the following text is deleted:

> 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 SL (Ref. 3).

The deleted sentences are replaced with:

> Analyses (Ref. 3) have shown that for specified initial MCPR values, the RBM is not required to be OPERABLE. These MCPR values are provided in the COLR for operations ≥ 90% RTP, and for operations ≥ 27% and < 90% RTP. For these power ranges with the initial MCPR ≥ the COLR value, no RWE event will result in exceeding the MCPR SL (Ref. 3).

48. Page B 3.3-67, 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."

49. Page B 3.3-68, 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 92 day" to "an 18 month."

50. Page B 3.3-70, 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:

SR 3.3.2.1.8

The RBM Setpoints are automatically varied as a function of power. Three Allowable Values are specified in 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.

51. Page B 3.3-71, 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.

52. Pages B 3.4-5 and B 3.4-10, Section B 3.4.1 [ARTS/MELLL]

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

53. Page B 3.10-43, Section B 3.10.8 [PRNM]

In the Bases for LCO 3.10.8, a new reference to Function 2.d is added.

54. Page B 3.10-47, Section B 3.10.8 [PRNM]

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

3.0 BACKGROUND

Provided in this section is the reason for this proposed change, a description of the modifications required to implement the proposed change, and related regulatory correspondence. Also included at the end of this section is a comparison of the proposed change, reason for change, background information, and technical analysis submitted in support of this proposed amendment with the information provided by TVA and approved by NRC for the Units 2 and 3 license amendments. Additional information regarding the design and function of these subsystems can be found in Updated Final Safety Analysis Report sections 3.7.7, "Thermal and Hydraulic Design - Evaluation" and 14.5, "Analyses of Abnormal Operational Transients - Uprated".

3.1 Reason for the Proposed Change

TVA is planning to replace the power range monitor portion of the Neutron Monitoring System on Unit 1 with a GE digital NUMAC PRNM retrofit system.

The new equipment will include capability for an automatic OPRM trip to detect and suppress possible thermal hydraulic instabilities in the reactor. This proposed change does not include revisions to incorporate the automatic trip function. Prior to enabling the OPRM trip function, related TS changes will be submitted. All other PRNM functions will be operable and this package provides the required TS changes for these functions.

The proposed ARTS 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.

3.2 Description of the Proposed Modifications

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. 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 Reactor Protection System 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.

3.3 Related Regulatory Correspondence

The proposed PRNM - ARTS/MELLL changes were previously submitted by TVA for all three Browns Ferry units (TS 353). NRC approved the changes for Units 2 and 3, but requested TVA withdraw the changes for Unit 1. NRC requested the withdrawal since TVA had no schedule for installation of the upgrade of the power range neutron monitor instrumentation and for the resumption of operation at Unit 1 (References 3-14).

The associated modifications were installed and operated on Units 2 and 3. TVA subsequently submitted and received NRC approval for operation of the OPRM Upscale trip function for Unit 2 (TS 354) (References 15 through 17) and Unit 3 (TS 398) (References 18 and 19).

On June 29, 2001, General Electric provided NRC with a Part 21 notification associated with the use of non-conservative parameters for the high peak bundle power-to-flow ratio in the generic regional mode DIVOM curve and for high core average power-to-flow ratio in the generic core wide DIVOM curve contained in NEDO-32465-A (Reference 20). The net effect of the Part 21 conditions the possible generation and implementation of non-conservative OPRM upscale trip setpoints. The OPRM system was declared inoperable for Units 2 and 3 on July 2, 2001, pending recalculation and implementation of corrected OPRM setpoints.

TVA requested and NRC approved (References 21 through 23) the deletion of Units 2 and 3 TS Required Action 3.3.1.1.I.2, which limited plant operation to 120 days in the event of the inoperability of the OPRM trip system.

3.4 Comparison with previous Technical Specification changes for Unit 2 and 3

TVA has compared the proposed change, reason for change, background information, and technical analysis submitted in support of this proposed amendment with the information provided by TVA and approved by NRC in TS 353 (References 3 through 12) for the adoption of the Units 2 and 3 PRNM - ARTS/MELLL TS. The comparison for each of these areas is provided below:

- The proposed changes to the Unit 1 TS and TS Bases are basically the same change as that proposed and approved for Units 2 and 3. Minor differences were noted:
    - Existing Note b in Table 3.3.1.1-1 was revised to show the new Allowable Value for APRM Function 2.b when reset for single loop operation. Single loop operation was not incorporated into the Units 2 and 3 TS until after the PRNM-ARTS/MELLL TS change.
    - Some existing footnotes and references were renumbered to be consistent with the current Units 2 and 3 TS (Changes 10, 21 and 24).

- A sentence was added to the Bases description of Surveillance Requirements 3.3.1.1.9, 3.3.1.1.10 and 3.3.1.1.13 for consistency with the Units 2 and 3 TS (Change 41).

- In the Bases description of Surveillance Requirement 3.3.2.1.4, the Units 2 and 3 calibration intervals were originally 184 days. The current Unit 1 Calibration interval is 92 days. The Unit 1 interval is being changed to an 18-month frequency, which is consistent with that approved for Units 2 and 3 in TS 353 (Change 49).

• The underlying reason for the Unit 1 TS change is the same as that which was previously submitted for the Units 2 and 3 TS change:

  - Replacement of the power range portion of the existing NMS with a GE digital NUMAC PRNM retrofit design; and

  - Implementation of APRM and RBM TS improvements and operation in an expanded core power/flow domain, the MELLL region.

  In addition, TVA needs to maximize consistency between the Unit 1 and Units 2 and 3 TS, operations and maintenance practices prior to restarting Unit 1.

• The background information provided in support of the Unit 1 TS change incorporates the same elements previously submitted in support of the Units 2 and 3 TS change. Updates were as follows:

  - Section 3.3, Related Regulatory Correspondence, reflects the current operational status of the OPRM function for Browns Ferry Units 2 and 3 and other BWRs.

  - Enclosure 1, Table 1, the comparison with NEDC-32410P-A, was updated to reflect revised change numbering, incorporate information from TVA's responses to previous NRC requests for additional information, and provide a more detailed comparison.

- o Enclosure 1, Table 2 provides an itemized summary of the ARTS/MELLL-related changes and identifies the corresponding portions of NEDC 32433P, April 1995, related to the changes. Also included as part of this review, is a comparison with NUREG-1433, Standard Technical Specifications - General Electric Plants BWR/4. This comparison has been updated to reflect revised change numbering. In addition, the previous submittals evaluated the changes against NUREG-1433, Revision 1. This submittal evaluates the changes against NUREG-1433, Revision 2.

- The technical analysis submitted for this Unit 1 TS change incorporates the same elements previously submitted in support of the previous TS changes for Units 2 and 3. The Unit 1 TS change is being submitted assuming the current Unit 1 Licensed thermal power limit of 3,293 megawatts. This was also the Licensed thermal power limit of Units 2 and 3 when the Units 2 and 3 PRNM / ARTS-MELLL TS were submitted and approved (References 3-9). Unit 1 is currently in an extended outage. When Unit 1 restarts, TVA intends to operate the plant at extended power uprate conditions (i.e., 3,952 megawatts thermal power). Since Unit 1 will not operate at 3,293 megawatts and a core design to support operation at that power level will not be performed, Unit 1 specific analyses of operation in the MELLL region were not performed. Credit was taken for the analysis performed on Unit 2. The proposed TS amendment to justify the operation at extended power uprate conditions and the Unit 1 reload analysis will evaluate the impact on PRNM / ARTS-MELLL.

4.0 TECHNICAL ANALYSIS

4.1 Installation of PRNM

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 1) that included a Safety Evaluation Report (SER) (Reference 24) 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 1 includes a "compliance matrix" that correlates the requirements of the Regulatory Guide to the NUMAC PRNM implementing program. Section 4 of Reference 1 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 Standard TS.

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 1.

The proposed TS change extends required surveillance intervals of the APRM and RBM equipment to the maximum periods supported by Reference 1. 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.

4.2 Justification for Each Individual NUMAC PRNM Change

In the following paragraphs, the justifications for each individual PRNM-related TS changes are discussed:

Change 4 - 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.

Change 6 - 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.

Change 7 - SR 3.3.1.1.11 required a calibrated flow signal be used to verify the accuracy of the total loop drive flow signal for the APRMs Flow Biased Simulated Thermal Power - High function. Calibration is now performed under SR 3.3.1.1.13. Therefore, SR 3.3.1.1.11 is being deleted.

Change 8 - In the table of SRs for LCO 3.3.1.1, a new note is added to SR 3.3.1.1.13. The note excludes neutron detectors from the Channel Calibration. 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.

Change 9 - A new Channel Functional Test (SR 3.3.1.1.16) with 184-day frequency is added. The new note allows 12 hours to complete the requirement for APRM Function 2.a when entering MODE 2 from MODE 1. 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.

Change 10 - There are five groups of changes to LCO 3.3.1.1, Table 3.3.1.1-1:

A) 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.

B) 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 1.

C) 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.

D) 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. Specific discussions for each of the five functions are provided below:

- For APRM Function 2.a, Neutron Flux - High, (Setdown), the following changes are made:

    - o 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.

    - o 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.

    - o The Logic System Functional Test (SR 3.3.1.1.14) with an 18-month frequency is deleted.

  These changes in testing and surveillance frequency are supported by the reliability analysis presented in Reference 1 and are

consistent with the recommendations of Reference 1.

- For APRM Function 2.b, Flow Biased Simulated Thermal Power - High, the following changes are made:

  - o 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.

  - o 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 an 18-month frequency is added.

  - o The flow signal calibration (SR 3.3.1.1.11) with an 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.

  - o The Logic System Functional Test (SR 3.3.1.1.14) with an 18-month frequency is deleted.

  These changes in testing and surveillance frequency are supported by the reliability analysis presented in Reference 1 and are consistent with the recommendations of Reference 1.

- For APRM Function 2.c, Neutron Flux - High, the following changes are made:

  - o The Channel Functional Test (SR 3.3.1.1.8) with a 92-day frequency is deleted; in its place the 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 18-month frequency is deleted.

These changes in testing and surveillance frequency are supported by the reliability analysis presented in Reference 1 and are consistent with the recommendations of Reference 1.

- 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 a 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 an 18-month frequency is deleted.

  These changes in testing and surveillance frequency are supported by the reliability analysis presented in Reference 1 and are consistent with the recommendations of Reference 1.

- For new APRM Function 2.e, 2-Out-Of-4 Voter, the following surveillance requirements are specified:

- o A Channel Check (SR 3.3.1.1.1) with a frequency of 24-hours is specified; this is consistent with the established Channel Check frequency for the other APRM Functions.

- o A Logic System Functional Test (SR 3.3.1.1.14) with a frequency of 18-months is specified.

- o A Channel Functional Test (SR 3.3.1.1.16) with a frequency of 184-days is specified.

These surveillance requirements and frequencies are supported by the reliability analysis presented in Reference 1 and are consistent with the recommendations of Reference 1.

E) The Allowable Value of APRM Function 2.b is changed. This change, part of the ARTS/MELLL changes, is discussed later in this section.

Change 11 - 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 1 and is consistent with the recommendations of Reference 1.

Change 12 - 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 1 and is consistent with the recommendations of Reference 1.

Changes 17 and 18 - In LCO 3.10.8.a (Change 17) and SR 3.10.8.1 (Change 18) 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.

Bases Sections B 3.3.1.1, B 3.3.2.1, and B 3.10.8 have been revised to reflect the TS changes described above.

For the PRNM related TS changes, Table 1 provides a comparison of the proposed wording with the recommendations of NEDC-32410P-A and the standard wording for the BWR/4 Improved Standard Technical Specifications as documented in NUREG-1433, Revision 2, April 2001.

4.3 Implementation of ARTS and Operation in an Expanded Core Power/Flow Domain, the MELLL

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 25 through 28) and has also approved expanding BFN's original operating region to the Extended Load Line Limit region (References 29 and 30). NRC has also approved these specific proposed changes for Browns Ferry Units 2 and 3 (References 9 and 12).

Reference 2 (NEDC-32433P), which was docketed as part of Reference 31, documents the results of analyses and evaluations performed for BFN by GE to support the proposed ARTS/MELLL changes for Units 1, 2 and 3. Appendix A of Reference 2 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 2 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. Due to the similarities between the three units, unit and core reload specific analyses were determined not to be required. Specific allowable setpoints were generated in a separate GE calculation for BFN (Reference 32) and will be documented in the Core Operating Limits Report (COLR) report for individual fuel cycles. As discussed in Reference 2, 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 2 to justify the safety of operation in the MELLL region consist of two segments. One segment is not fuel dependent and the other 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 2 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 evaluations are generically applicable to BFN Unit 1 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.

4.4 Justification for Each Individual ARTS/MELLL Change

In the following paragraphs, the justifications for each individual ARTS/MELLL-related TS changes are discussed:

Changes 1 and 3 - LCO 3.2.4, APRM Gain and Setpoints, and related SRs are deleted as justified by the evaluation in Section 5.3 of Reference 2. Deletion of this requirement is supported by implementation of new power- and flow-dependent limits for Average Planar Linear Heat Generation Rate (APLHGR) and 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.

Analyses documented in Reference 2 demonstrate that with the setpoint setdown requirement eliminated and flow- and power-dependent thermal limits implemented:

- o MCPR safety limit will not be violated as a result of any AOOs;
- o All fuel thermal-mechanical design bases will remain within the licensing limits described in the GE generic fuel licensing report GESTAR-II; and
- o 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.

Change 2 - 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.

Change 5 - 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.

Change 10 - 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 "≤ 0.58W + 62% RTP and ≤ 120% RTP" to "≤ 0.66W+ 71% RTP and ≤ 120% RTP." The Allowable Value for single loop operation is similarly changed. 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 2 and the setpoint calculations of Reference 32, 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 29 and 30), 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 2, 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.

Change 13 - A new RBM SR (SR 3.3.2.1.8) is added to verify that the upscale functions for the Low, Intermediate and High Power Range are not bypassed when in the applicable thermal power ranges. The new SR is consistent with the presentation in BWR/4 Standard Technical Specifications, Revision 2. The RBM power range break setpoint were determined by the BFN specific setpoint calculation (ED-Q2092-900118).

Change 14 - In LCO 3.3.2.1, Table 3.3.2.1-1, the RBM requirements are revised 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 2. 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 Standard Technical Specifications (NUREG-1433, Revision 2), 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 of 1.10 in Standard Technical Specifications. SL 2.1.1.2 compared to the Safety Limit of 1.07 used in Reference 2. Section 10.5 of Reference 3 documents that with RTP ≥ 90% and operating MCPR ≥ 1.40, or with RTP < 90% and operating MCPR ≥ 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 Standard TS change adds additional MCPR margin to account for increases for specific reload core analyses over the base 1.07 MCPR Safety Limit used in Reference 2 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.

Changes 15 and 16 - 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 Boiling Water Reactor Owners' Group (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.

Bases Table of Contents and Sections B 3.2.1, B 3.2.2, B 3.2.4, B 3.3.2.1, and B 3.4.1 have been revised to reflect the TS changes described above.

For the ARTS/MELLL related TS changes, Table 2 provides a comparison of the proposed wording with the recommendations of NEDC-32433P and the standard wording for the BWR/4 Standard Technical Specifications as documented in NUREG-1433, Revision 2, April 2001.

5.0 REGULATORY SAFETY ANALYSIS

The Tennessee Valley Authority (TVA) is submitting an amendment request to license DPR-33 for the Browns Ferry Nuclear Plant (BFN) Unit 1.

5.1 No Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment", as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

Power Range Neutron Monitor (PRNM) Changes:

The proposed TS changes are associated with the Nuclear Measurement Analysis and Control (NUMAC) PRNM retrofit design. The proposed changes involve modification of the Limiting Conditions for Operations (LCOs) and Surveillance Requirements for equipment designed to mitigate events which result in power increase transients. For the Average Power Range Monitor (APRM) system, the mitigating action is to block control rod withdrawal or initiate a reactor scram which terminates the power increase when setpoints are exceeded. For the Rod Block Monitor (RBM) system, the mitigating action is to block continuous control rod withdrawal prior to exceeding the Minimum Critical Power Ratio safety limit during a postulated Rod Withdrawal Error event. The worst case failure of either the APRM or the RBM systems is failure to initiate its mitigating action (failure to scram or block rod withdrawal). Failure to initiate these mitigating actions 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 and revised surveillance requirements, 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.

APRM and RBM Technical Specification (ARTS) improvements and operation in an expanded core power/flow domain, the Maximum Extended Load Line Limit (MELLL) Changes:

The proposed ARTS/MELLL changes permit expansion of the current allowable power/flow operating region and will apply a newer methodology for assuring that fuel thermal and mechanical design limits are satisfied. Operation in the MELLL region with the ARTS changes has been evaluated and there is adequate design margin for operation in the MELLL region for all events and parameters considered. 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. Therefore, design margins are not degraded by the proposed changes.

The proposed changes to the RBM system will assure that a Rod Withdrawal Error 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. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

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 equipment to perform a mitigating action does not create the possibility of a new or different kind of accident. No new external threats or release pathways are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3.0 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

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 PRNM 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: 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 have demonstrated 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 Rod Withdraw Error (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.

Based on the above, TVA concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

TVA has compared the BFN Unit 1 plant specific environmental conditions (i.e., temperature, humidity, pressure, radiation, and seismic) against the NUMAC-PRNMS qualification values and determined that the equipment is qualified to function in the Unit 1 environment. TVA has also evaluated the electromagnetic interference environment and has taken measures to reduce adverse affects. The environment at BFN and these actions ensure conformance to General Design Criterion 4 for protection against adverse environmental effects.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve:

(i) A significant hazards consideration,

(ii) A significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or

(iii) A significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 50.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. 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.

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

3. TVA letter, Pedro Salas to NRC, dated June 2, 1995, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 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."

4. TVA letter, T.E. Abney to NRC, dated March 6, 1997, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Technical Specifications (TS) Change 353R1 - 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 - Revision 1." GE Report, Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2 and 3, NEDC-32433P was included as Attachment 1.

5. TVA letter, T.E. Abney to NRC, dated April 11, 1997, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Technical Specifications (TS) Change 353S1 - 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 - Supplement 1 - Improved Standard Technical Specifications (ISTS) Format."

6. NRC letter, J.F. Williams to O.D. Kingsley, dated April 15, 1997, "Browns Ferry Nuclear Plant Units 1, 2, and 3 - Request for Additional Information Regarding Upgrade of Power Range Neutron Monitors (TAC Nos. M92503, M92504, and M92505) (TS 353)."

7. TVA letter, T.E. Abney to NRC, dated May 13, 1997, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Request for Additional Information Regarding Upgrade of Power Range Neutron Monitors (TAC. Nos. M92503, M92504, and M92505) (TS353)."

8. TVA letter, T.E. Abney to NRC, dated August 20, 1997, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Response to Request for Information - Technical Specification (TS)-353R1 and TS-353S1 - Average Power Range Monitors-Rod Block Monitor Technical Specifications (ARTS)."

9. NRC letter, J.F. Williams to O.D. Kingsley, dated September 11, 1997, "Issuance of Amendment - Browns Ferry Nuclear Plant Unit 2 (TAC No. M92504) (TS 353)."

10. NRC letter, A.W. DeAgazio to O.J. Zeringue, dated February 11, 1998, "Browns Ferry Plant Units 1, 2, and 3 - Request for Information Regarding Conversion of Custom Technical Specifications to Improved Standard Technical Specifications (TAC NOS. M96431, M96432, and M96433)."

11. TVA letter, T.E. Abney to NRC, dated March 13, 1998, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Request for Additional Information (RAI) Regarding Conversion of Custom Technical Specifications (TS) to Improved Standard Technical Specifications (TAC. Nos. M96431, M96432, and M96433) and Supplemental Change to TS-353S1."

12. NRC letter, A.W. DeAgazio to J.A. Scalice, dated September 3, 1998, "Amendment No. 213 to Facility Operating License No. DPR-68: Power Range Neutron Monitor Upgrade with Implementation of Average Power Range Monitor and Rod Block Monitor Technical Specification Improvements and Maximum Extended Load Line Limit Analyses - Technical Specification Change TS-353 (TAC No. M92505)."

13. NRC letter, A.W. DeAgazio to J.A. Scalice, dated September 17, 1998, "Browns Ferry Nuclear Plant, Unit 1 - Request for Withdrawal of Amendment Request (TAC No. M92503)."

14. TVA letter, T.E. Abney to NRC, dated October 5, 1998, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Withdrawal Of Technical Specifications (TS) Change 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."

15. TVA letter, T.E. Abney to NRC, dated September 8, 1998, "Browns Ferry Nuclear Plant (BFN) - Unit 2 - Technical Specification (TS) Change - 354 - Oscillation Power Range Monitor."

16. TVA letter, T.E. Abney to NRC, dated February 22, 1999, "Browns Ferry Nuclear Plant (BFN) - Unit 2 - Technical Specification (TS) Change - 354 - Oscillation Power Range Monitor."

17. NRC letter, L. Raghavan to J.A. Scalice, dated March 5, 1999, "Amendment Mo. 258 to Facility Operating License No. DPR-52: Oscillation Power Range Monitor Upscale Trip Function in the Average Power Range Monitor - Technical Specification Change TS-354 (TAC No. MA3556)."

18. TVA letter, T.E. Abney to NRC, dated July 28, 1999, "Browns Ferry Nuclear Plant (BFN) - Unit 3 - Technical Specification (TS) Change - 398 - Oscillation Power Range Monitor (TAC No. MA5976)."

19. NRC letter, W.O. Long to J.A. Scalice, dated September 27, 1999, "Browns Ferry Nuclear Plant, Unit 3 - Issuance of Amendment Regarding Oscillation Power Range Monitor (TAC No. MA5976)."

20. General Electric NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

21. TVA letter, A.S. Bhatnagar to NRC, dated July 25, 2001, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change - 415 - Deletion of 120-day Required Action for Restoration of Oscillation Power Range Monitor (OPRM) Function - Emergency TS Change Request for Unit 2."

22. NRC letter, K.N. Jabbour to J.A. Scalice, dated July 26, 2001, "Browns Ferry Nuclear Plant Unit 2 - Issuance of Emergency Amendment to Delete the 120-day Required Action for Restoring the Oscillation Power Range Monitor Function (TAC No. MB2470)."

23. NRC letter, K.N. Jabbour to J.A. Scalice, dated September 13, 2001, "Browns Ferry Nuclear Plant Unit 3 - Issuance of Amendment to Delete the 120-day Required Action for Restoring the Oscillation Power Range Monitor Function (TAC No. MB2496) (TS-415)."

24. 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).

25. 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.

26. 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.

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

28. 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.

29. 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]

30. 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)]

31. 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) Technical Specification (ARTS) Improvements and Maximum Extended Load Line Limit (MELLL) Analyses."

32. 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.

ATTACHMENT 1
PLANT SPECIFIC INFORMATION REQUIRED
FOR POWER RANGE NEUTRON MONITOR RETROFIT

The following information in provided to address the six plant specific actions listed in Section 5 of the NRC Safety Evaluation Report (Reference 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 PRNM modification at BFN Unit 1. 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 final OPRM setpoints and margins will be confirmed as part of a separate OPRM Technical Specification (TS) change submittal to be provided to the NRC for review prior to activation of the OPRM trip function in the PRNM system (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 contains two groups of technical changes. One group of changes is related to the proposed PRNMS modification; these changes are labeled "**[PRNM]**" in Enclosure 1. 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. The TS changes are consistent with the standard BWR/4 Improved Standard Technical Specification format (NUREG-1433, Revision 2).

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

Table 2 provides an itemized summary of the ARTS/MELLL-related changes and identifies the corresponding portions of NEDC 32433P, April 1995. The NRC Safety Evaluation Report did not specifically request a comparison of the ARTS/MELLL TS changes. However, TVA is providing this comparison to assist NRC in their review. Also included as part of this review, is a comparison with NUREG-1433, Revision 2, Standard Technical Specifications - General Electric Plants BWR/4.

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.

TVA maintains a series of controlled drawings which provide the environmental parameters for various plant areas. For the control rooms, this information is documented on drawing 47W225-3 for temperature, humidity, pressure, and radiation. The BFN control room environmental conditions are bounded by the PRNM qualification parameters. 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 are qualified for continuous operation in a humidity range of 10% to 90%, as documented in Section 4.4.2.2.1.3 of NEDC-32410P-A. 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 - As documented in section 4.4.2.2.2.3 of NEDC-32410P-A, the PRNM control room electronics is qualified for continuous operation in a pressure range of 13 to 16 psia. Normal control room pressure is maintained slightly higher than atmospheric pressure, between 0.125" and approximately 0.5" of water (gauge). Operation of the Control Bay Emergency Ventilation System, which is an infrequent activity, results in a maximum pressure of about 0.8" of water (gauge). This is within the qualified pressure range.

- Radiation - As documented in Section 4.4.2.2.3.3 of NEDC-324102-A, the PRNM control room electronics is qualified for continuous operation with a Dose Rate less than or equal to 0.001 Rads/hr and Total Integrated Dose (TID) of less than or equal to 1000 Rads. The 40 year design integrated dose has an upper limit of 350 Rads (equivalent to less than .001 Rad/hr). Thus, control room dose rates and total integrated dose are within the qualified range.

- Seismic Acceleration - Seismic acceleration spectra for BFN Class I structures are maintained by TVA's Civil Engineering department in a series of documents entitled "Master Acceleration Response Spectra (MARS) Report for Seismic Class I Structures". For the main control rooms, the seismic qualification parameters for the PRNM system are considerably in excess of the seismic spectra for the control rooms at BFN. Additionally, a site specific calculation will be performed to document the seismic qualification of the PRNM panels.

  NRC has previously reviewed the BFN seismic response spectra as documented in the Safety Evaluation Reports for the BFN Nuclear Performance Plans. (Reference: Section 2.2 of NUREG-1232, Volume 3, Supplement 2, January 23, 1991, and Section 2.2 of NUREG-1232, Volume 3, Supplement 1, October 24, 1989.)

- 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 PROPOSED TECHNICAL SPECIFICATION CHANGES
WITH NEDC-32410P-A, APPENDIX H

ENCLOSURE 1 SECTION II PROPOSED CHANGE #	NEDC-32410P-A APPENDIX H PAGE #	COMPARISON OF PROPOSED TECHNICAL SPECIFICATIONS WITH NEDC-32410P-A
4	H-3	TS-430 implements the changes as shown in the Licensing Topical Report (LTR).
6	H-7	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.
7	H-5	TS-430 implements the change shown in the LTR for SR 3.3.1.1.3. In the BFN Technical Specifications, the comparable surveillance requirement is numbered SR 3.3.1.1.11.
8	H-7	TS-430 revises 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.
9	H-7	TS-430 revises 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.
10	H-9 & 10	TS-430 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 footnotes 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.
11	H-12	TS-430 implements the change as shown in the LTR.
12	NA	SR 3.3.2.1.4 incorporates the 18 month calibration frequency specified in SR 3.3.2.1.7 of the LTR.
13	NA	SR 3.3.2.1.8 incorporated the SR and frequency specified in SR 3.3.2.1.4 of the LTR.
NA	H-14	The Bypass Time Delay does not appear in the current Unit 1 Technical Specifications. Therefore, no change is required to reflect its deletion in the LTR.
17	NA	LCO 3.10.8.1 is revised to correlate with the revisions to Table 3.3.1.1-1.

ENCLOSURE 1 SECTION II PROPOSED CHANGE #	NEDC-32410P-A APPENDIX H PAGE #	COMPARISON OF PROPOSED TECHNICAL SPECIFICATIONS WITH NEDC-32410P-A
18	NA	SR 3.10.8.1 is revised to correlate with the revisions to Table 3.3.1.1-1.
28	H-15	TS-430 revises the APRM Bases description to match the wording as presented in the LTR.
29	H-15 & 16	TS-430 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.
30	H-17 & 18	TS-430 revises the 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.
31	H-19	TS-430 revises the Bases description of APRM Function 2.c to match the wording as presented in the LTR, except that the existing name of this function is retained.
32	H-20 & 21	TS-430 implements the changes as shown in the LTR.
33	H-21	TS-430 implements the changes as shown in the LTR.
34	H-22	TS-430 implements the changes as shown in the LTR.
35	H-23	TS-430 implements the changes as shown in the LTR, except for a difference in the numbering of the reference.
36	H-24 & 25	TS-430 implements the changes as shown in the LTR, except for a difference in the numbering of the reference.
39	H-30	TS-430 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.
41	H-30	TS-430 revises the LTR Bases discussion for SR 3.3.1.1.13 to make it applicable for SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13.
42	H-27	TS-430 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.
NA	H-32	The deletion of SR 3.3.1.1.14 in the LTR has no corresponding SR in the current BFN Unit 1 Technical Specifications. Therefore, no change is required.
43	H-32	TS-430 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.

ENCLOSURE 1 SECTION II PROPOSED CHANGE #	NEDC-32410P-A APPENDIX H PAGE #	COMPARISON OF PROPOSED TECHNICAL SPECIFICATIONS WITH NEDC-32410P-A
NA	H-33	The BFN Licensing Basis, including the current Units 1, 2 and 3 Technical Specifications, do not require response time testing of Reactor Protection System. Therefore, SR 3.3.1.1.17 of the LTR has not been adopted.
44	H-34	TS-430 updates the list of references to include the reference to NEDC-32410P as shown in the LTR. Differences are due to differences in numbering of the references and TVA has incorporated the reference to the NRC approved version.
46	H-35	TS-430 implements the changes as shown in the LTR.
48	H-36	TS-430 implements the changes as shown in the LTR. Differences are due to differences in numbering of the references.
49	H-37	TS-430 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.
51	H-39	TS-430 updates the list of references to include the reference to NEDC-32410P as shown in the LTR. Differences are due to differences in numbering of the references and TVA has incorporated the reference to the NRC approved version.
53	N/A	The Bases for LCO 3.10.8 are revised to correlate with the revisions to Table 3.3.1.1-1.
54	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-430 with BWR/4 ISTS, NUREG-1433, Revision 2 and with Recommendations of NEDC-32433P

Enclosure 1 Section I Item #	BWR/4 ISTS, Revision 2 Section #	COMPARISON OF TS-430 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
2	1.1	The definition of MFLPD is indicated as "optional" in the BWR/4 ISTS. This definition is deleted by TS-430 because it is no longer required in the BFN TS due to the ARTS / MELLL change.
3	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-430, consistent with the recommendation of NEDC-32433P, Section 11, Item (1).
5	SR 3.3.1.1.2	BWR/4 ISTS indicates that the deleted phrase is optional. TS-430 makes this change consistent with deletion of LCO 3.2.4, as described above.
10	Table 3.3.1.1-1	TS-430 revises the APRM flow-biased scram setpoint consistent with the recommendations of NEDC-32422P, Section 11, Item (7).
13	SR 3.3.2.1.8	TS-430 adds a new surveillance requirement to support the change to power-dependent RBM setpoints. This SR has the same requirements as SR 3.3.2.1.4 in the BWR/4 ISTS. However, the setpoints reflect the Browns Ferry specific setpoint calculation. This change is consistent with NEDC-32422P, Section 11, Items (8) and (9).
14	Table 3.3.2.1-1	TS-430 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 of 1.10 in ISTS Safety Limit 2.1.1.2 versus that used in NEDC-32422P (1.07).
15	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 a discussion.

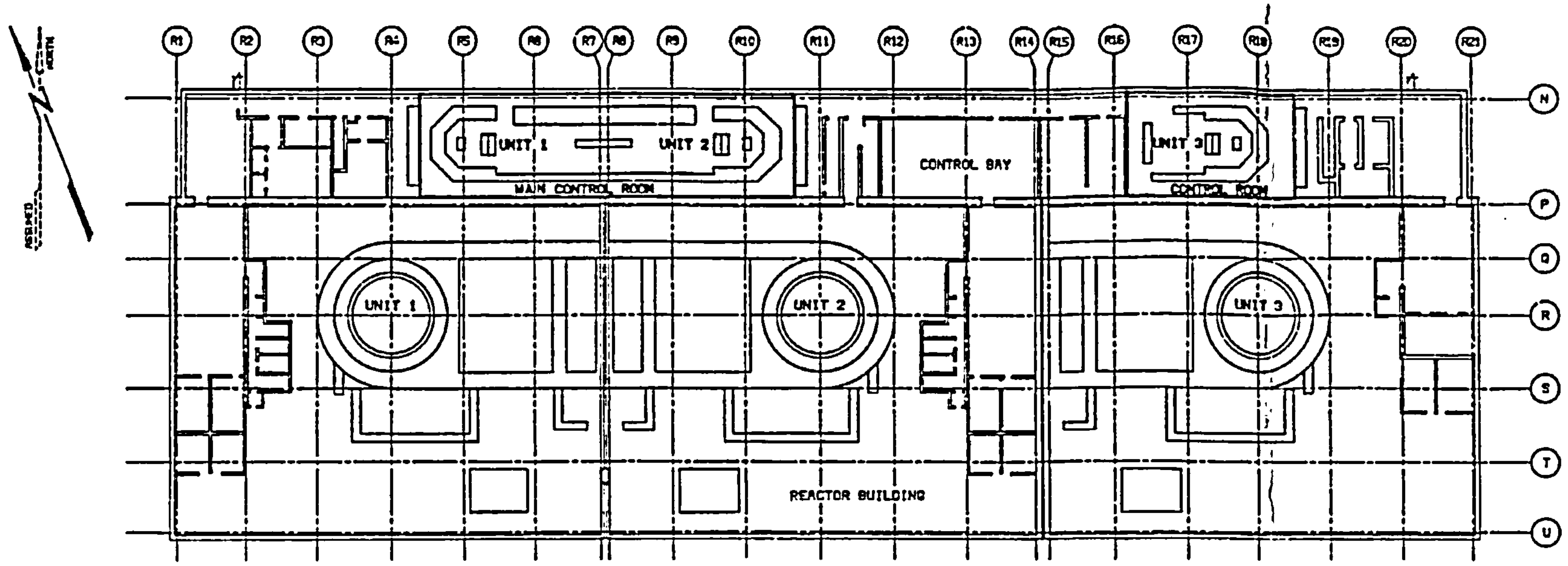
Enclosure 1 Section I Item #	BWR/4 ISTS, Revision 2 Section #	COMPARISON OF TS-430 WITH BWR/4 ISTS, NUREG-1433, REVISION 1 AND WITH RECOMMENDATIONS OF NEDC-32433P, MELLL AND ARTS IMPROVEMENT PROGRAM ANALYSIS FOR BROWNS FERRY UNITS 1, 2, AND 3
16	N/A	The Power / Flow stability map is expanded to add the 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.
20	B 3.2.1	TS-430 revises the Bases discussion of APLHGR limits to reflect the change to power and flow dependent limits. The revised wording is consistent with the discussion in Rev. 2 of the BWR/4 ISTS. This change is also consistent with NEDC-32433P, Section 11, Items (4), (5), and (10).
22	B 3.2.1	TS-430 revises the Bases discussion of APLHGR LCO to reflect the change to power and flow dependent limits. The revised wording is similar to the discussion in Rev. 2 of the BWR/4 ISTS.
23	B 3.2.1	TS-430 revises the references in the APLHGR Bases section to include the new ARTS/MELLL analysis and include additional references identified in BWR/4 ISTS.
25	B 3.2.2	TS-430 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 with the exception of specific references. This change is consistent with NEDC-32422P, Section 11, Items (2), (3) and (10).
26	B 3.2.2	TS-430 revises the references in the MCPR Bases section to include the new ARTS/MELLL analysis and include additional references to the Browns Ferry specific analysis and identified in BWR/4 ISTS.
27	B 3.2.4	TS-430 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). The bases discussion is optional in the BWR/4 ISTS.
32	B 3.3.1.1	TS-430 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). The bases discussion is optional in the BWR/4 ISTS.

Enclosure 1 Section I Item #	BWR/4 ISTS, Revision 2 Section #	COMPARISON OF TS-430 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
46	B 3.3.2.1	TS-430 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.
50	B 3.3.2.1	TS-430 revises the Bases discussion of RBM SR 3.3.2.1.8 to reflect implementation of ARTS improvements. The revised wording is similar to the discussion in Rev. 1 of the BWR/4 ISTS. The revised wording is the same, except for frequency, as the Units 2 and 3 Bases. The wording is not the same as the discussion in Rev. 2 of the BWR/4 ISTS. TVA has elected to keep the Units 2 and 3 wording in this Bases section for unit fidelity. This change is consistent with NEDC-32433P, Section 11, Item (10).
52	N/A	In the Bases discussion for the power / flow stability map, the core flow value is changed from 45% to 50% to correspond 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.

ATTACHMENT 2

DRAWING 47W225-3

CONTROL BAY ENVIRONMENTAL DATA



PLAN — EL 617.0

BUILDING LOCATION	OPERATIONAL CONDITION (NOTE A)	TEMPERATURE (°F)	RELATIVE HUMIDITY (%) (NOTE G)	PRESSURE (PSIA) (NOTE B & H)	TOTAL 40 YEAR INTEGRATED DOSE (RADS) (NOTE C)	INTEGRATED ACCIDENT DOSE (RADS) (NOTES A & C)	AREA TYPE (NOTE J)	FLOODING
CONTROL ROOMS EL 617.0	1	(AVG)78 (MAX)78 (MIN)78	50 80 40	ATM(+) ATM(+) ATM(+)	~~NA~~ 5x10² NA	NA NA NA	A	(NOTE K)
	2	(MAX)104 (MIN)60 } (NOTE D)	80 10	ATM(+) ATM(+)	NA NA	<1x10² NA		
	5	NA	NA	FIGURE 1 (47W225-1)	NA	NA		

GENERAL NOTE:

FOR OUTSIDE ENVIRONMENTAL CONDITIONS, DEFINITIONS OF MILD AND HARSH ENVIRONMENTS, AND GENERAL INFORMATION, SEE THE BFN DESIGN CRITERIA FOR ENVIRONMENTAL DESIGN (REFERENCE 2).

X'D OUT NOTES DO NOT APPLY TO THIS SHEET.

NOTES:

A. OPERATIONAL CONDITION DEFINITIONS
   1. NORMAL
   2. ABNORMAL (THIS ALSO INCLUDES ACCIDENT RADIATION LEVELS FOR LOCA CONDITIONS.)
   3. LOCA
   5. TORNADO

B. ATM INDICATES A PRESSURE EQUAL TO ATMOSPHERIC PRESSURE WILL BE PRESENT. NORMAL ATMOSPHERIC PRESSURE AT BFN IS 14.4 PSIA. ATM(+) INDICATES A PRESSURE SLIGHTLY GREATER, ATM(-) INDICATES A PRESSURE SLIGHTLY BELOW.

C. ALL 40 YEAR INTEGRATED RADIATION DOSES SHOWN ARE UPPER LIMITS FOR THE SUMMATION OF THE GAMMA AND BETA CONTRIBUTIONS UNLESS OTHERWISE INDICATED. TOTAL RADIATION DOSE CAN BE OBTAINED BY ADDING THE 40 YEAR INTEGRATED AND ACCIDENT DOSES. A LOCA IS THE ONLY DBE AFFECTING THESE AREAS.

D. THESE ABNORMAL TEMPERATURES COULD OCCUR AS A RESULT OF OUTSIDE TEMPERATURE EXCURSIONS, TEMPORARY, GREATER THAN DESIGN HEAT LOADS, OR DEGRADED ENVIRONMENTAL CONTROL SYSTEM OPERATION. THIS CONDITION COULD EXIST FOR UP TO 8 HOURS PER EXCURSION AND WILL OCCUR FOR LESS THAN 1% OF THE PLANT LIFE.

X. THE SGTS WILL BE OPERATED 10 HOURS PER MONTH CONTINUOUSLY FOR TESTING AND MAINTENANCE. FOR 4 MONTHS EACH YEAR. ROOM AMBIENT TEMPERATURE COULD REACH 110°F DURING THIS TESTING (40 HOURS TOTAL). FOLLOWING A LOCA AND SIMULTANEOUS LOSS OF ELECTRICAL POWER TO THE VENTILATION FANS, SPACE TEMPERATURE COULD REACH 140°F. THIS CONDITION WILL EXIST FOR 24 HOURS AND SHOULD BE CONSIDERED TO OCCUR ONCE DURING THE PLANT LIFE.

X. THESE SPACES WILL REACH 120°F ONCE EACH MONTH FOR 1 HOUR PLUS ONCE EVERY 18 MONTHS FOR 24 HOURS (CONTINUOUS) DURING PERIODIC TESTING OF THE DIESELS.

G. THE MAXIMUM AND MINIMUM ABNORMAL RELATIVE HUMIDITIES GIVEN WILL OCCUR <1% OF THE TIME WITH A DURATION OF 8 HOURS FOR EACH EXCURSION. THE MAXIMUM TEMPERATURE ASSOCIATED WITH 90% RH IS 80°F.

H. THE PROBABILITY OF A TORNADO IS 1.71 X 10^{-3} OR ONCE EVERY 600 YEARS. THE FIGURE LISTED DEFINES THE PRESSURE VS TIME RELATIONSHIP DURING A DESIGN BASIS TORNADO FOR THE SUBJECT ROOM(S).

X. THE MAXIMUM POSSIBLE FLOOD LEVEL OF 572.5 COULD CAUSE FLOODING AT THE INTAKE PUMPING STATION.

J. AREAS LISTED IN THE TABLE ARE DIVIDED INTO TWO CATEGORIES AND ARE DEFINED AS FOLLOWS:
   A. SPACES THAT ARE MAINTAINED AT OR NEAR 85°F BY REDUNDANT ENVIRONMENTAL CONTROL SYSTEMS SERVED BY ONSITE EMERGENCY ELECTRICAL POWER.
   B. SPACES NOT MAINTAINED BY REDUNDANT ENVIRONMENTAL CONTROL SYSTEMS SERVED BY ONSITE EMERGENCY ELECTRICAL POWER.

K. THE MAXIMUM POSSIBLE FLOOD LEVEL OF 572.5 WILL NOT AFFECT THESE AREAS.

X. THESE AREAS ARE BELOW THE MAXIMUM POSSIBLE FLOOD LEVEL; THEREFORE, FLOODING IS LIKELY TO OCCUR. SAFETY RELATED EQUIPMENT IN THESE ROOMS SHOULD BE LOCATED ABOVE EL 572.5 OR OTHERWISE PROTECTED.

M. THE LOCA TEMPERATURE DETERMINED BY MD-Q0031-880240 IS BOUNDED BY THE ABNORMAL MAXIMUM TEMPERATURE.

N. DELETED

REFERENCE:

1. SUMMARY OF MILD ENVIRONMENTAL CONDITIONS FOR BROWNS FERRY NUCLEAR PLANT (MD-Q0999-910030).
2. BROWNS FERRY NUCLEAR PLANT GENERAL DESIGN CRITERIA FOR ENVIRONMENTAL DESIGN (BFN-50-7151).
3. BROWNS FERRY NUCLEAR PLANT HVAC ADEQUACY ANALYSIS MD-Q0031-880240.

REV NO	CHANGE REF	DATE	DFTR	CHKR	DSGN	RVWR	APPD	APPD	APPD	ISSD
4	DCN S33300A	10-27-95	KML	KSP	N/A	ROC	KSP	TFJ	ROC	HB/V
	DELETED NOTE N AND REVISED DOSE RATES									
3	CAT 4	10-14-94	LMQ	JK	BLS	BCR	BCR	N/A	HEC	HLB
	REVISED NOTE N AND REFERENCE NO.1									
2	CAT 4	9-8-93	RLA	WAL	CRS	JEM	CP	DAF	JEM	JEM
	REVISED PER CALCULATION ND-Q0999-910030 REVISION 6 AND MD-Q0031-880240 REVISION 6 AND ISSUED FOR USE									
1	DCN S15576A	2-16-93	AKW	APL	BSR	DWP	JDM	VFB	PER	JEM
	REVISED TO INCLUDE LOCA TEMPERATURES FROM MD-Q0031-880240 R4; DELETED TORNADO PRESSURE BOUNDARY CURVE; ADDED DRAWING REFERENCE FOR TORNADO PRESSURE BOUNDARY CURVE FROM CALCULATION MD-Q0999-920128									

SCALE: NTS EXCEPT AS NOTED

CONTROL BAY
UNITS 1-3

ENVIRONMENTAL DATA
ENVIRONMENT - MILD
EL 617.0

BROWNS FERRY NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY
DIVISION OF ENGINEERING DESIGN

SUBMITTED | RECOMMENDED | APPROVED

INSPECTED AND APPROVED FOR ISSUE

NO ISSUE FOR SCN-P3138

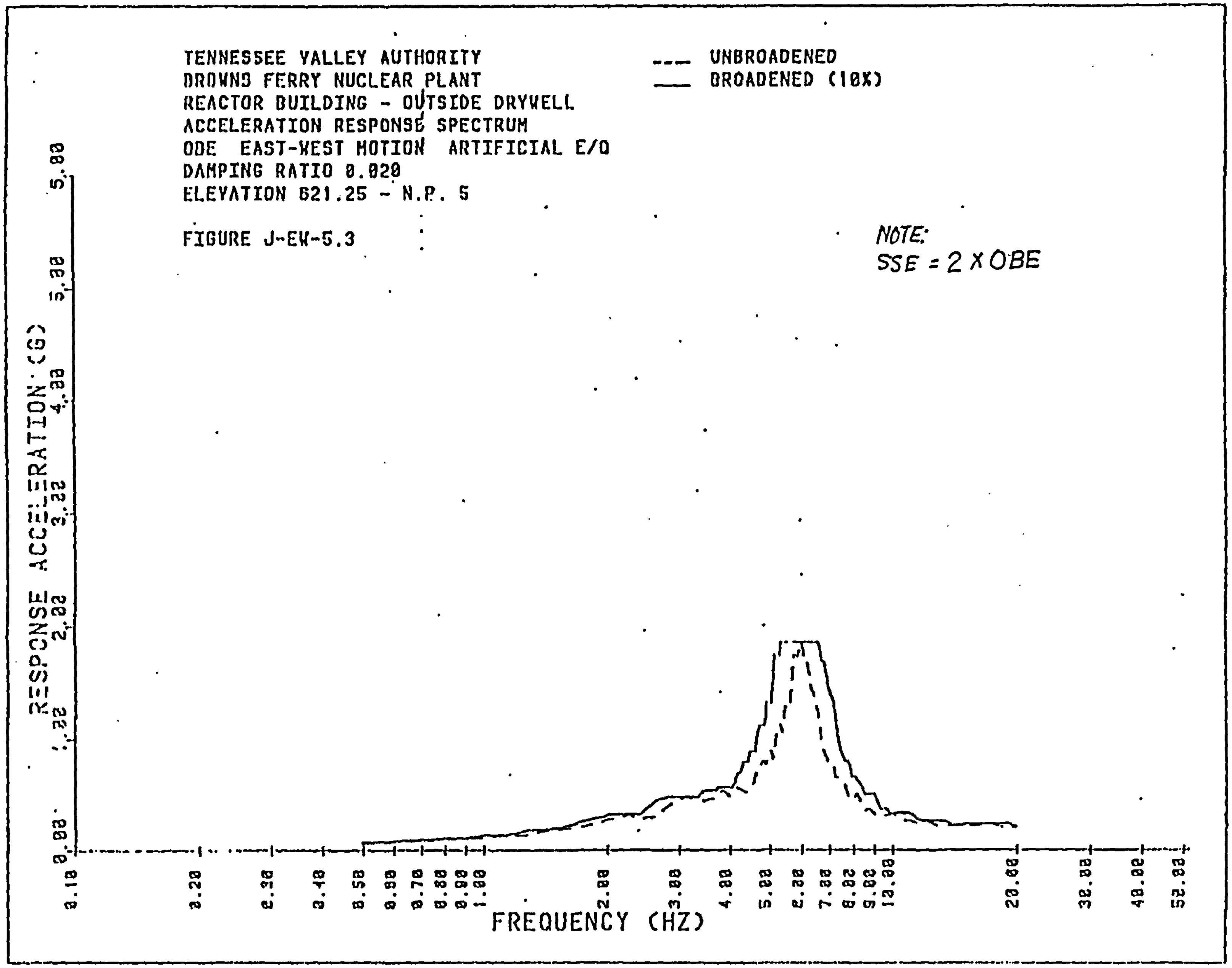
KNOXVILLE 2-15-84 | 67 | N | 47W225-3 | R4

RECORD DRAWING AS CONSTRUCTED

ATTACHMENT 3

EXCERPTS FROM MASTER ACCELERATION RESPONSE SPECTRA

REPORT FOR SEISMIC CLASS I STRUCTURES


TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
REACTOR BUILDING - OUTSIDE DRYWELL
ACCELERATION RESPONSE SPECTRUM
OBE EAST-WEST MOTION ARTIFICIAL E/Q
DAMPING RATIO 0.020
ELEVATION 621.25 - N.P. 5
FIGURE J-EW-5.3
UNBROADENED
BROADENED (10X)
NOTE:
SSE = 2 X OBE
RESPONSE ACCELERATION (G)
FREQUENCY (HZ)
0.00
1.00
2.00
3.00
4.00
5.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
2.00
3.00
4.00
5.00
6.00
7.00
8.00
9.00
10.00
20.00
30.00
40.00
50.00

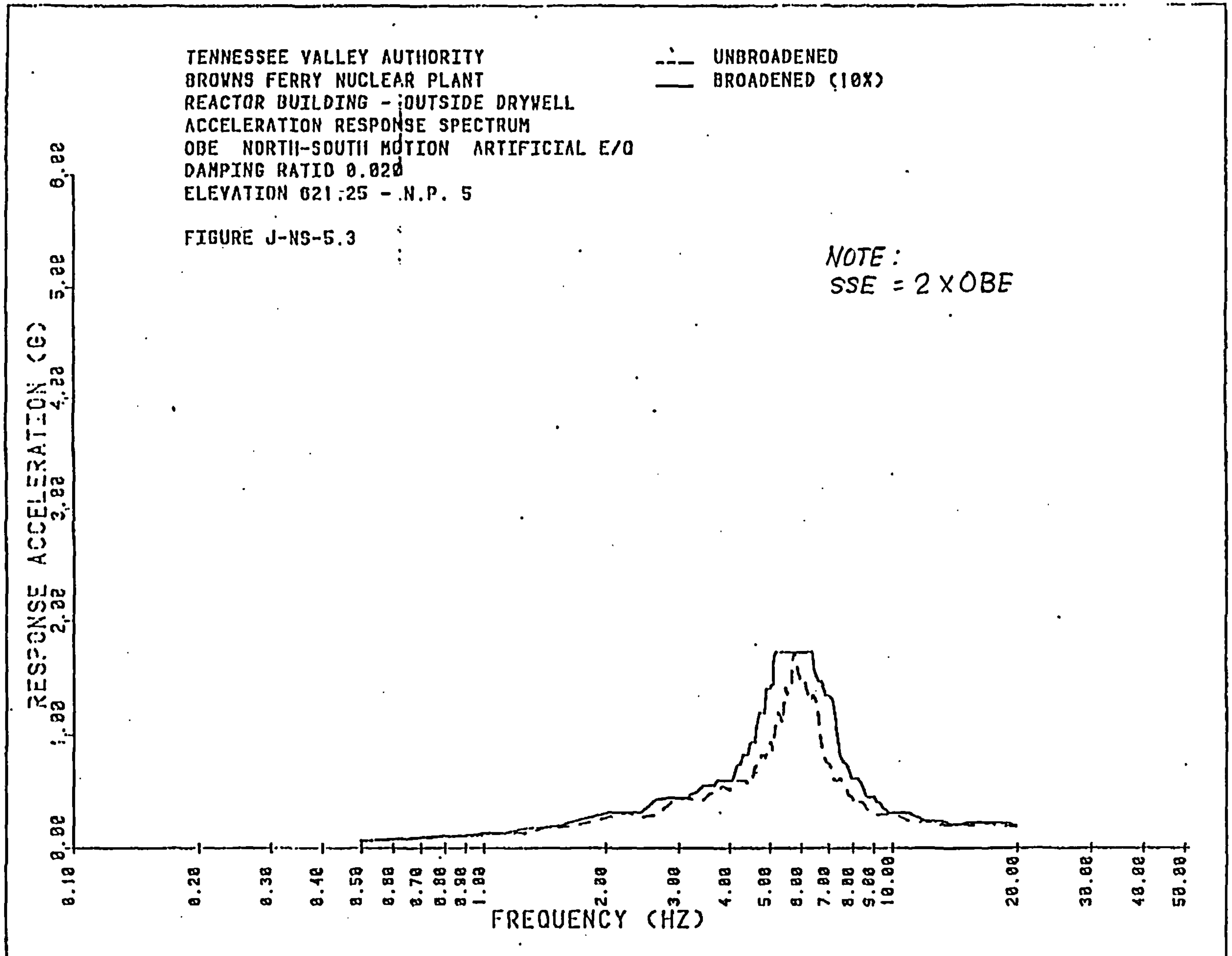
DIGITIZED DATA FOR
BROADENED SPECTRUM

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
REACTOR BUILDING - OUTSIDE DRYWELL
ACCELERATION RESPONSE SPECTRUM
→OBE EAST-WEST MOTION ARTIFICIAL E/Q
DAMPING RATIO 0.020
ELEVATION 621.25 - N.P. 5

NOTE:
SSE = 2 x OBE

FIGURE J-EW-5.3

FREQUENCY	ACCELERATION	FREQUENCY	ACCELERATION	FREQUENCY	ACCELERATION
.5000	.0714	4.8680	1.1099	9.4099	.3644
.8000	.1079	4.9451	1.3181	9.6491	.3644
1.1000	.1352	5.0553	1.3181	9.9000	.3303
1.4000	.1887	5.1000	1.4654	9.9028	.3298
1.7000	.2442	5.1724	1.7410	11.0000	.3298
2.0000	.3089	5.2803	1.7410	11.1000	.3222
2.0455	.3209	5.3571	1.8582	11.4000	.2946
2.4115	.3209	6.5476	1.8582	11.7000	.2734
2.7000	.4529	6.6000	1.8232	12.0000	.2616
2.8125	.4716	6.7032	1.6854	12.2440	.2579
3.3603	.4716	6.7901	1.6854	13.4146	.2579
3.4615	.5291	6.9000	1.5517	13.5000	.2538
3.6962	.5291	7.2000	1.2626	13.8000	.2383
3.7500	.5570	7.5000	.8624	13.9828	.2261
4.0469	.5570	7.6360	.7902	14.7679	.2261
4.2000	.6832	7.7465	.7902	15.0000	.2355
4.2056	.6871	7.8000	.7717	18.3333	.2355
4.2518	.6871	7.9664	.6497	18.4259	.2347
4.3689	.7896	8.0882	.6497	19.1304	.2347
4.4439	.7896	8.1000	.6459	19.2000	.2340
4.5000	.8752	8.4000	.5620	19.5000	.2240
4.5455	.8839	8.5497	.4936	19.8000	.2144
4.6257	.8839	9.0164	.4936	19.9557	.2127
4.7368	1.1099	9.3000	.3887	20.0000	.2127



TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
REACTOR BUILDING - OUTSIDE DRYWELL
ACCELERATION RESPONSE SPECTRUM
OBE NORTH-SOUTH MOTION ARTIFICIAL E/Q
DAMPING RATIO 0.020
ELEVATION 621.25 - N.P. 5
FIGURE J-NS-5.3
UNBROADENED
BROADENED (10X)
NOTE:
SSE = 2 X OBE
RESPONSE ACCELERATION (G)
FREQUENCY (HZ)
0.00
1.00
2.00
3.00
4.00
5.00
6.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
2.00
3.00
4.00
5.00
6.00
7.00
8.00
9.00
10.00
20.00
30.00
40.00
50.00

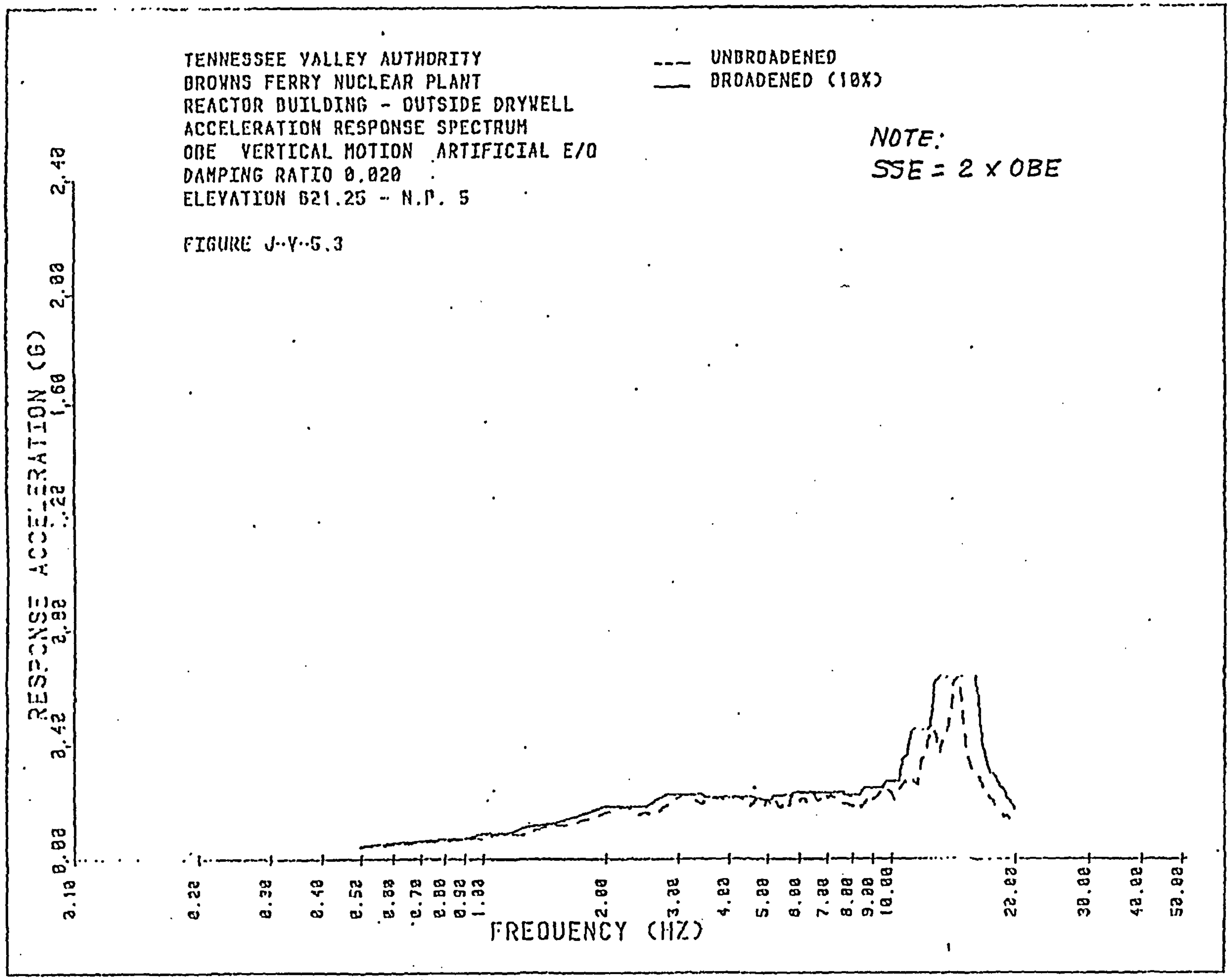
DIGITIZED DATA FOR
BROADENED SPECTRUM

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
REACTOR BUILDING - OUTSIDE DRYWELL
ACCELERATION RESPONSE SPECTRUM
--> OBE NORTH-SOUTH MOTION ARTIFICIAL E/Q
DAMPING RATIO 0.020
ELEVATION 621.25 - N.P. 5

NOTE:
SSE = 2 X OBE

FIGURE J-NS-5.3

FREQUENCY	ACCELERATION	FREQUENCY	ACCELERATION	FREQUENCY	ACCELERATION
.5000	.0715	4.6172	.9343	11.0000	.3139
.8000	.1076	4.7368	1.1884	11.1000	.3037
1.0000	.1348	4.8722	1.1884	11.4000	.2813
1.1256	.1348	4.9451	1.4002	11.7000	.2633
1.2000	.1561	5.0376	1.4002	11.9508	.2471
1.5000	.1950	5.1000	1.4307	12.2222	.2471
1.8000	.2668	5.2326	1.7186	12.3000	.2422
2.0455	.3183	6.3953	1.7186	12.3229	.2407
2.4304	.3183	6.5987	1.4605	13.0952	.2407
2.7000	.4337	6.7073	1.4605	13.2000	.2397
2.8125	.4446	6.8750	1.3418	13.5000	.2329
3.1554	.4446	7.0513	1.3418	13.8000	.2171
3.3000	.4873	7.2000	1.2476	13.8956	.2122
3.4615	.5493	7.5000	.8085	14.6219	.2122
3.6625	.5493	7.8000	.7189	14.7000	.2143
3.7500	.5945	7.9527	.6166	15.0000	.2235
4.0604	.5945	8.2090	.6166	18.3333	.2235
4.2000	.7294	8.4000	.5705	18.5019	.2213
4.2056	.7333	8.7000	.4580	19.1304	.2213
4.2540	.7333	8.7113	.4562	19.2000	.2194
4.3269	.8213	9.0164	.4562	19.5000	.2064
4.4376	.8213	9.3000	.3980	19.7425	.1982
4.5000	.9222	9.6000	.3412	20.0000	.1982
4.5455	.9343	9.8321	.3139		


TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
REACTOR BUILDING - OUTSIDE DRYWELL
ACCELERATION RESPONSE SPECTRUM
OBE VERTICAL MOTION ARTIFICIAL E/Q
DAMPING RATIO 0.020
ELEVATION 621.25 - N.P. 5
FIGURE J-V-5.3
--- UNBROADENED
— BROADENED (10%)
NOTE:
SSE = 2 x OBE
RESPONSE ACCELERATION (G)
FREQUENCY (HZ)

DIGITIZED DATA FOR
BROADENED SPECTRUM

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
REACTOR BUILDING - OUTSIDE DRYWELL
ACCELERATION RESPONSE SPECTRUM
→ OBE VERTICAL MOTION ARTIFICIAL E/Q
DAMPING RATIO 0.020
ELEVATION 621.25 - N.P. 5

NOTE:
SSE = 2X OBE

FIGURE. J-V-5.3

FREQUENCY	ACCELERATION	FREQUENCY	ACCELERATION	FREQUENCY	ACCELERATION
.5000	.0471	5.6940	.2213	11.5385	.4494
.8000	.0700	5.7000	.2223	12.3742	.4494
1.0000	.0873	5.8442	.2294	12.6000	.5040
1.1458	.0873	6.2491	.2294	12.9000	.6143
1.2000	.0971	6.2500	.2295	13.2000	.6310
1.5000	.1235	7.6389	.2295	13.2353	.6330
1.8000	.1614	7.6813	.2219	16.1765	.6330
2.0455	.1819	7.8571	.2219	16.2000	.6236
2.5000	.1819	8.0357	.2234	16.5000	.5038
2.7000	.2094	8.4082	.2234	16.8000	.3856
2.8125	.2251	8.6538	.2459	17.1000	.3406
3.4375	.2251	9.5269	.2459	17.4000	.3061
3.5409	.2146	9.6000	.2533	17.7000	.2938
3.7397	.2146	9.7826	.2685	18.0000	.2848
3.7500	.2157	10.2088	.2685	18.3000	.2643
4.5833	.2157	10.2273	.2701	18.6000	.2474
4.8000	.2085	10.5038	.2701	18.9000	.2333
4.8368	.2059	10.8000	.3501	19.2000	.2096
5.1525	.2059	11.1000	.3958	19.5000	.2000
5.2326	.2213	11.4000	.4444	19.8000	.1830

ENCLOSURE 2

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

TECHNICAL SPECIFICATION CHANGE (TS-430) -
POWER RANGE NEUTRON MONITOR UPGRADE WITH IMPLEMENTATION
OF AVERAGE POWER RANGE MONITOR AND ROD BLOCK MONITOR
TECHNICAL SPECIFICATION IMPROVEMENTS AND MAXIMUM
EXTENDED LOAD LINE LIMIT ANALYSES

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

I. AFFECTED PAGE LIST

i	B i	B 3.3-13	B 3.3-59
1.1-5	B 3.2-1	B 3.3-14	B 3.3-60
3.2-7	B 3.2-2	B 3.3-15	B 3.3-67
3.2-8	B 3.2-5	B 3.3-16	B 3.3-68
3.3-1	B 3.2-7	B 3.3-30	B 3.3-70
3.3-3	B 3.2-10	B 3.3-31	B 3.3-71
3.3-4	B 3.2-15	B 3.3-32	B 3.4-5
3.3-5	B 3.2-16	B 3.3-36	B 3.4-10
3.3-6	B 3.2-17	B 3.3-37	B 3.10-43
3.3-7	B 3.2-18	B 3.3-39	B 3.10-47
3.3-18	B 3.2-19	B 3.3-40	
3.3-19	B 3.2-20	B 3.3-41	
3.3-20	B 3.2-21	B 3.3-42	
3.4-3	B 3.2-22	B 3.3-44	
3.4-4	B 3.3-9	B 3.3-46	
3.10-22	B 3.3-10	B 3.3-57	
3.10-24	B 3.3-12	B 3.3-58	

II. MARKED PAGES

See attached.

TABLE OF CONTENTS

CHANGE 1

Deleted: 3.2.4 . . . Average Power Range Monitor (APRM) ¶ . . . . Gain and Setpoints . 3.2-7

(continued)

CHANGE 2

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.

Deleted: MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

Deleted: 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.

(continued)

**CHANGE 3
(Page 1 of 2)**

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

CHANGE 3
(Page 2 of 2)

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.

CHANGE 4

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
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

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.

CHANGE 5

	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

Deleted: plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints"

(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 MWD/T average core exposure
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)

CHANGE 6

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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	------NOTE------ Neutron detectors are excluded. ------ 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

Change 7

Deleted: Adjust the channel to conform to a calibrated flow signal.

Deleted: 18 months

Change 8

Change 9

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

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

(continued)

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

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

(c) [0.66 W + 71% - 0.66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Change 10
(Page 1 of 2)

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SR 3.3.1.1.14
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Change 10
(Page 2 of 2)

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)

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1

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2

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F

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SR 3.3.1.1.7¶
SR 3.3.1.1.8¶
SR 3.3.1.1.14¶

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≥ 3% RTP

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(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.

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.

Change 11

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	184 days
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

Deleted: 92

Change 12

Deleted: 92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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
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 ≥ 27% and ≤ 62% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 62% and ≤ 82% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 82% RTP.	18 months

Change 13

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

Change 14

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)~~
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

Deleted: b

Deleted: (a),(b)

Deleted: c

Deleted: (a),(b)

Deleted: ≥ 3% RTP

(a) THERMAL POWER ≥ 27% and ≤ 62% RTP and MCPR less than the value specified in the COLR.

(b) THERMAL POWER > 62% and ≤ 82% RTP and MCPR less than the value specified in the COLR.

Deleted: ≥ 90% RTP and MCPR < 1.44

Deleted: ≥ 29% and < 90% 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 > 82% and < 90% RTP and MCPR less than the value specified in the COLR.

(g) THERMAL POWER ≥ 90% RTP and MCPR less than the value specified in the COLR.

(h) THERMAL POWER ≥ 27% and < 90% RTP and MCPR less than the value specified in the COLR.

(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 AND Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated

Change 15

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Change 16

REPLACE
WITH
ATTACHED
FIGURE

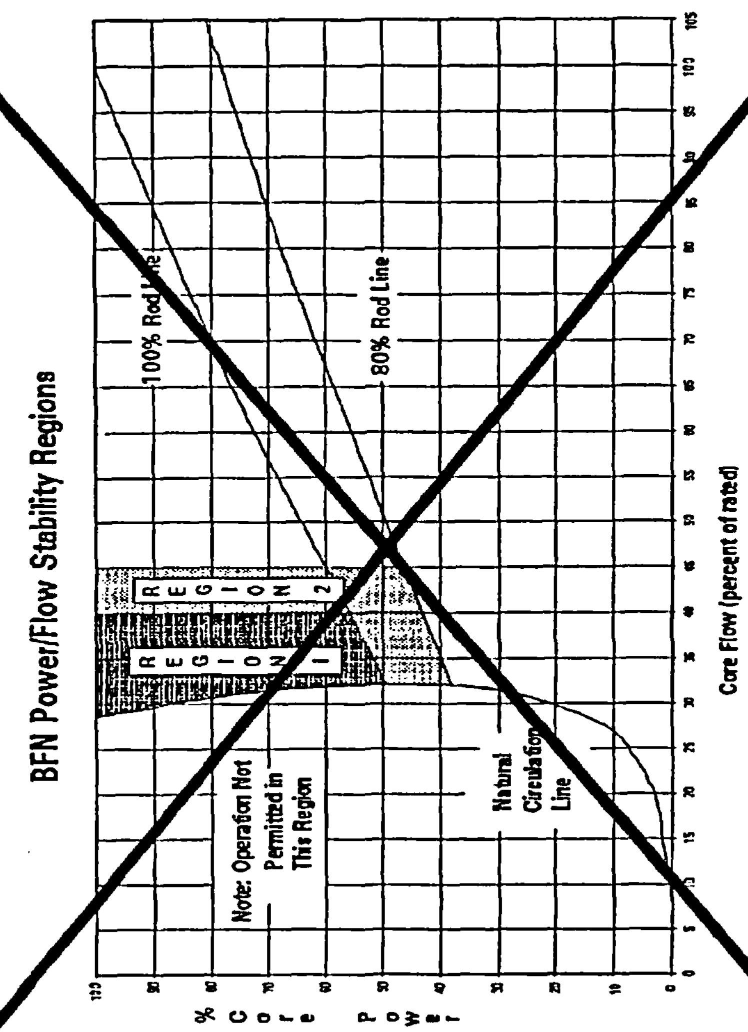

Figure 3.4.1-1
THERMAL POWER VERSUS CORE FLOW STABILITY REGIONS

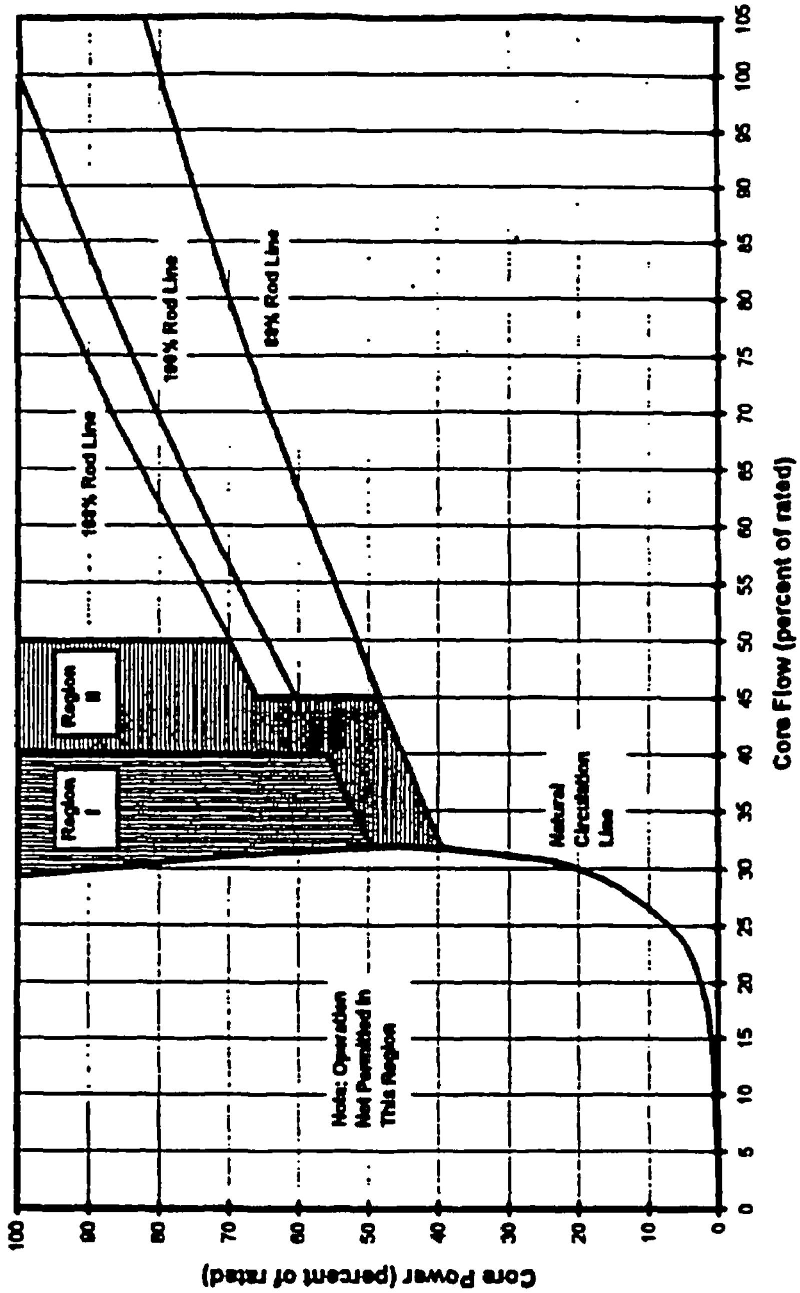

Figure 3.4.1-1

THERMAL POWER VERSUS
CORE FLOW STABILITY
REGIONS

3.10 SPECIAL OPERATIONS

Change 17

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.

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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

Change 19

Deleted: B 3.2.4 . . . Average Power Range Monitor (APRM)¶ . . . . Gain and Setpoints . B 3.2-15

(continued)

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.

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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.

Deleted: and

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.

Deleted: APLHGR limits are developed as a function of exposure and fuel bundle type.

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 (Ref. 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 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.

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(Page 2 Of 2)

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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(Page 1 of 3)

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

Deleted: 7

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 limit.** With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure

Change 22

(continued)

dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 10). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR.

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(continued)

BASES (continued)

REFERENCES

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

Change 23

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Change 21
(Page 3 of 3)

BASES (continued)

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(Page 1 of 2)

APPLICABLE SAFETY ANALYSES

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

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Deleted: 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 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

(continued)

between 25% RTP and the previously mentioned bypass power level.

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

Change 25 (Page 2 of 2)

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.**

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

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

REFERENCES

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

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9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

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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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(Page 1 of 8)

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.

(continued)

BASES

BACKGROUND (continued)

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, 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,

Change 27
(Page 2 of 8)

(continued)

BASES

BACKGROUND (continued)

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.

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.

(continued)

BASES

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(Page 4 of 8)

APPLICABLE SAFETY ANALYSES (continued)

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. [illegible]).

(continued)

BASES (continued)

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

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.

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.

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(Page 5 of 8)

(continued)

BASES (continued)

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.

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(Page 6 of 8)

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 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.

(continued)

BASES

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ACTIONS
(continued)

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.

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.

(continued)

BASES (continued)

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 (continued)	Average Power Range Monitor

Change 28

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Average Power Range Monitor

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

Change 29
(Page 1 of 2)

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.

Deleted: 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.

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.

(continued)

BASES

Change 29
(Page 2 of 2)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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.

Deleted: 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.

(continued)

BASES

Change 30

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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.

Deleted: 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 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 total drive flow signal representative of total core flow. 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.

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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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(Page 1 of 2)

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.

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

(continued)

BASES

Change 31
(Page 2 of 2)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	<u>2.c. Average Power Range Monitor Fixed Neutron Flux - High</u> *(continued)* 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.

Deleted: 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.

(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.

Change 32

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.

(continued)

BASES

Change 33

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.d. Average Power Range Monitor - Inop

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Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions. This Function (Inop) provides assurance that the 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 unbypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

Deleted: 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 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.

Deleted: 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.

There is no Allowable Value for this Function.

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

(continued)

Change 34

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.

(continued)

Change 35

ACTIONS (continued)

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.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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.

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(Page 1 of 3)

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.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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.

Change 36
(Page 2 of 3)

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

(continued)

Change 36
(Page 3 of 3)

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. For the typical Function with one-out-of-two taken twice logic and the IRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve - Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).

Deleted: and APRM

(continued)

BASES

Change 37

SURVEILLANCE REQUIREMENTS (continued)

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 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.

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.

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.

(continued)

BASES

Change 38

SURVEILLANCE REQUIREMENTS (continued)

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.

Deleted: and APRM

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.)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 and SR 3.3.1.1.6 *(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.

SR 3.3.1.1.7

Change 39
(Page 1 of 2)

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 MWD/T average core exposure Frequency is based on operating experience with LPRM sensitivity changes.

Deleted: and

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.

(continued)

Change 39
(Page 2 of 2)

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SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16 (continued)

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.

Change 40

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

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 includes physical inspection and actuation of the switches.

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(continued)

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(Page 2 of 2)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

A 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 MWD/T 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 IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 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.

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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

Change 42

SURVEILLANCE REQUIREMENTS *(continued)*

SR 3.3.1.1.11

(Deleted)

Deleted: 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.

Change 43

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 (continued)

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.**

Change 44

BASES

BACKGROUND (continued)	During refueling, shutdown, and low power operations, the primary indication of neutron flux levels is provided by the SRMs or special movable detectors connected to the normal SRM circuits. The SRMs provide monitoring of reactivity changes during fuel or control rod movement and give the control room operator early indication of subcritical multiplication that could be indicative of an approach to criticality.
APPLICABLE SAFETY ANALYSES	Prevention and mitigation of prompt reactivity excursions during refueling and low power operation is provided by LCO 3.9.1, "Refueling Equipment Interlocks;" LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"; IRM Neutron Flux - High and Average Power Range Monitor (APRM) Neutron Flux - High, (Setdown) Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation." The SRMs have no safety function and are not assumed to function during any FSAR design basis accident or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.
LCO	During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, subcritical multiplication and reactor criticality, and neutron flux level and reactor period until the flux level is sufficient to maintain the IRMs on Range 3 or above. All but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core.

Change 45

(continued)

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.

Change 46
(Page 1 of 2)

Deleted: A signal from one average power range

(continued)

BASES

BACKGROUND (continued)

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).

Deleted: monitor (APRM) channel assigned to each Reactor Protection

Deleted: System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system.

Change 46
(Page 2 of 2)

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 on 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

(continued)

BASES (continued)

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.**

Change 47
(Page 1 of 2)

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 for RBM OPERABILITY.

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

(continued)

BASES

Change 47
(Page 2 of 2)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

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 ≥ 27% 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). **Analyses (Ref. 3) have shown that for specified initial MCPR values, the RBM is not required to be OPERABLE. These MCPR values are provided in the COLR for operations ≥ 90% RTP, and for operations ≥ 27% and < 90% RTP. For these power ranges with the initial MCPR ≥ the COLR value, no RWE event will result in exceeding the MCPR SL (Ref. 3).** Therefore, under these conditions, the RBM is also not required to be OPERABLE.

Deleted: 29

Deleted: 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 SL (Ref. 3).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.

Change 48

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).

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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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

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.

Change 49

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.

Deleted: a 92 day

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.6 (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 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.

SR 3.3.2.1.8

Change 50 (Page 1 of 2)

The RBM Setpoints are automatically varied as a function of power. Three Allowable Values are specified in 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.

Change 50
(Page 2 of 2)

BASES (continued)

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.
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.**

Change 51

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This decay ratio also helps ensure sufficient margin to an instability occurrence is maintained. The generic region has been determined to be bounded by the 80% rod line and the 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.

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(Page 1 of 2)

Deleted: 45

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

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

Change 52
(Page 2 of 2)

Deleted: 45

REFERENCES

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

BASES

APPLICABLE SAFETY ANALYSES (continued)	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 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

Change 53

(continued)

BASES (continued)

Change 54

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.

(continued)

ENCLOSURE 3

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

TECHNICAL SPECIFICATION CHANGE (TS-430) - POWER RANGE NEUTRON MONITOR UPGRADE WITH IMPLEMENTATION OF AVERAGE POWER RANGE MONITOR AND ROD BLOCK MONITOR TECHNICAL SPECIFICATION IMPROVEMENTS AND MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSES

PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)

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II. CLEAN PAGES

See attached.

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(continued)

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.

(continued)

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
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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
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
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 MWD/T average core exposure
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	-----------------------NOTE----------------------- Neutron detectors are excluded. --- 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)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

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

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

(c) [0.66 W + 71% - 0.66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

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.

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
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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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
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 ≥ 27% and ≤ 62% RTP. b. Intermediate Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 62% and ≤ 82% RTP. c. High Power Range -- Upscale Function is not bypassed when THERMAL POWER is > 82% 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 ≥ 27% and ≤ 62% RTP and MCPR less than the value specified in the COLR .

(b) THERMAL POWER > 62% and ≤ 82% RTP and MCPR less than the value specified in the COLR

(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 > 82% and < 90% RTP and MCPR less than the value specified in the COLR.

(g) THERMAL POWER ≥ 90% RTP and MCPR less than the value specified in the COLR.

(h) THERMAL POWER ≥ 27% and < 90% RTP and MCPR less than the value specified in the COLR.

(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 AND Immediately after any decrease of > 10% rated core flow while initial thermal power is > 40% of rated

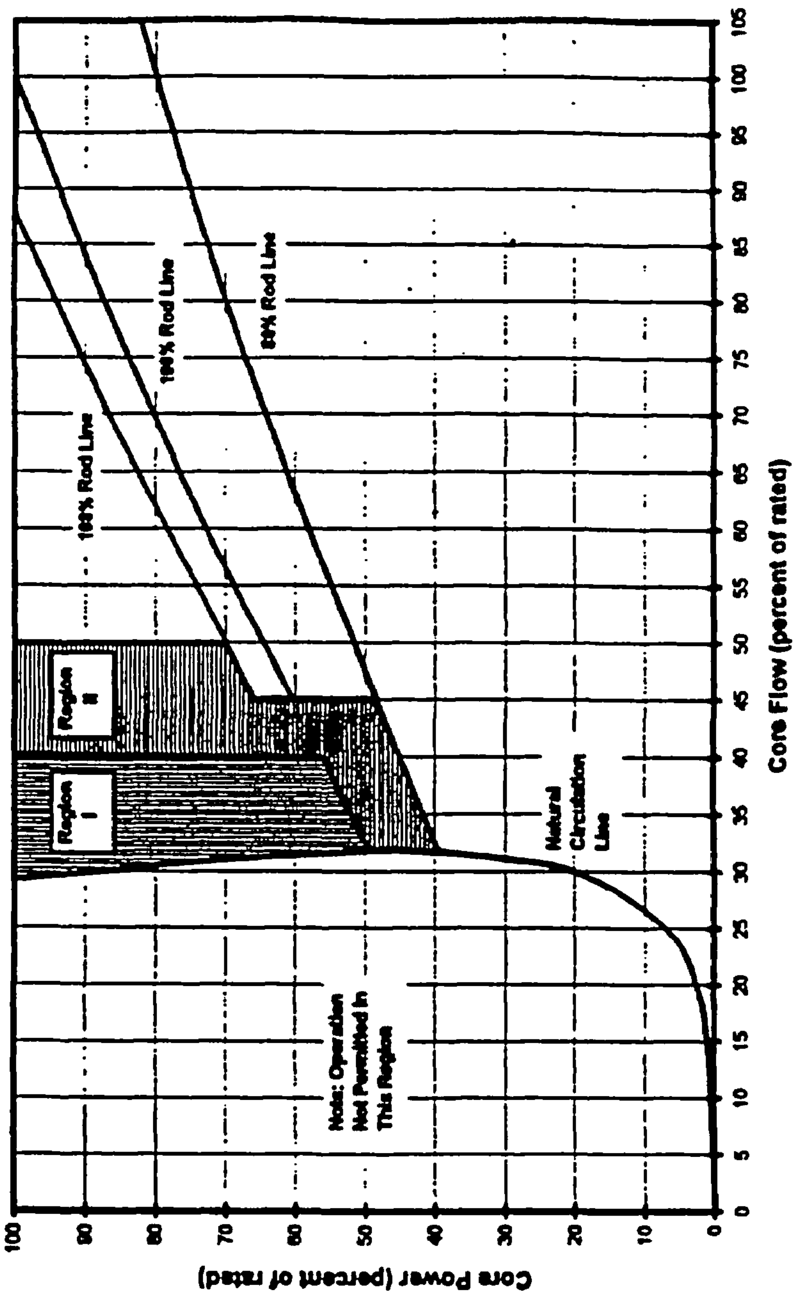

Figure 3.4.1-1

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)

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(continued)

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 (Ref. 7). Flow dependent

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 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.

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

LCO

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

(continued)

BASES (continued)

APPLICABILITY

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

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.

(continued)

BASES (continued)

REFERENCES

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

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

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 8, and 10. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.
	The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Reference 8). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Reference 6) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.
	Power dependent MCPR limits ($MCPR_p$) are determined by the one dimensional transient code (Reference 9). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.
	The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

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 nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1

If the MCPR 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 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is 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.

SR 3.2.2.2

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

(continued)

BASES (continued)

REFERENCES

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including abnormal operational transients. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3). Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for abnormal operational transients, plus an allowance for densification power spiking. The LHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 4).
LCO	The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.
APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1

If the LHGR 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is 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 slow 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 lower power levels.

REFERENCES

1. FSAR, Chapter 14.
2. FSAR, Chapter 3.
3. NUREG-0800, Standard Review Plan 4.2, Section II.A.2(g), Revision 2, July 1981.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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.

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Average Power Range Monitor

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 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	<u>2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High</u> (continued) 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. <u>2.c. Average Power Range Monitor Fixed Neutron Flux - High</u> 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

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 - Inop

Three of the four APRM channels are required to be OPERABLE for each of the APRM Functions. This Function (Inop) provides assurance that the 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 unbypassed 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.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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.

(continued)

BASES

ACTIONS (continued)

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.

(continued)

BASES

ACTIONS (continued)

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.

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.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2 (continued)

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 (or the associated trip system is in trip),

(continued)

BASES

ACTIONS

C.1 (continued)

such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve - Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).

For Function 8 (Turbine Stop Valve - Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip).

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)."

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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).

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.)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 and SR 3.3.1.1.6 (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.

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 MWD/T average core exposure 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16 (continued)

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.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

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 includes physical inspection and actuation of the switches.

A 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 MWD/T 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 IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

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.

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 (continued)

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.

BASES

BACKGROUND (continued)	During refueling, shutdown, and low power operations, the primary indication of neutron flux levels is provided by the SRMs or special movable detectors connected to the normal SRM circuits. The SRMs provide monitoring of reactivity changes during fuel or control rod movement and give the control room operator early indication of subcritical multiplication that could be indicative of an approach to criticality.
APPLICABLE SAFETY ANALYSES	Prevention and mitigation of prompt reactivity excursions during refueling and low power operation is provided by LCO 3.9.1, "Refueling Equipment Interlocks;" LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"; IRM Neutron Flux - High and Average Power Range Monitor (APRM) Neutron Flux - High, (Setdown) Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation." The SRMs have no safety function and are not assumed to function during any FSAR design basis accident or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.
LCO	During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, subcritical multiplication and reactor criticality, and neutron flux level and reactor period until the flux level is sufficient to maintain the IRMs on Range 3 or above. All but one of the channels are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core.

(continued)

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.

(continued)

BASES

BACKGROUND (continued)

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).

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 on 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.

(continued)

BASES (continued)

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.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

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 ≥ 27% 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). Analyses (Ref. 3) have shown that for specified initial MCPR values, the RBM is not required to be OPERABLE. These MCPR values are provided in the COLR for operations ≥ 90% RTP, and for operations ≥ 27% and < 90% RTP. For these power ranges with the initial MCPR ≥ the COLR value, no RWE event will result in exceeding the MCPR SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.2 and SR 3.3.2.1.3 (continued)

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.

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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 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 (continued)

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.

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)

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 (continued)

SR 3.4.1.2

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

REFERENCES

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

BASES

APPLICABLE SAFETY ANALYSES (continued)	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 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

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BASES (continued)

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.

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