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

Brian O'Grady
Vice President, Browns Ferry Nuclear Plant

September 1, 2006

TVA-BFN-TS-431
TVA-BFN-TS-418

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 Nos. 50-259 |
| Tennessee Valley Authority | ) | 50-260 |
| | ) | 50-296 |

**BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418 - EXTENDED POWER UPRATE (EPU) - RESPONSE TO ROUND 9 - REQUEST FOR ADDITIONAL INFORMATION (RAI) - SBWB RAIs (TAC NOS. MC3812, MC3743, AND MC3744)**

By letters dated June 28, 2004 (ADAMS Accession No. ML041840109) and June 25, 2004 (ML041840301), TVA submitted applications to the NRC for EPU operation of BFN Unit 1, and BFN Units 2 and 3, respectively. On September 1, 2006, the NRC staff issued the Round 9 RAIs on the EPU amendment requests. This submittal responds to the reactor systems related questions from Round 9, namely Units 2 and 3, SBWB-65 through SBWB-74, and Unit 1 SBWB-49.

Enclosure 1 to this letter provides TVA's responses to the Round 9 SBWB RAI questions related to Areva; SBWB-65 through SBWB-74.

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Enclosure 2 provides a proprietary response to SBWB-49 and contains information that General Electric (GE) Company considers to be proprietary in nature and subsequently, pursuant to 10 CFR 9.17(a)(4), 2.390(a)(4) and 2.390(d)(1), GE requests that such information be withheld from public disclosure. Enclosure 3 is a redacted version of Enclosure 2 with the proprietary material removed and is suitable for public disclosure. Enclosure 4 contains an affidavit from GE supporting this request for withholding from public disclosure.

To facilitate NRC's review of the proposed TS-418 TS changes, in Enclosure 5 TVA has remarked the original June 25, 2004, EPU TS-418 changes against the current Unit 2 and 3 TS pages.

In a matter unrelated to the Round 9 RAI, TVA was notified by NRC staff that the AREVA NP proprietary notice belonging to Enclosure 1 of the May 11, 2006, submittal (ML061360148) for the Units 2 and 3 supplemental response to NRC Round 3 RAI cannot be located and requested another copy be submitted. Enclosure 6 contains a copy of the missing AREVA NP proprietary notice.

TVA has determined that the additional information provided by this letter does not affect the no significant hazards considerations associated with the proposed TS changes. The proposed TS changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

No new regulatory commitments have been made in this submittal.

If you have any questions regarding this letter, please contact William D. Crouch at (256)729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 1st day of September, 2006.

Sincerely,

Brian O'Grady

Brian O'Grady

Enclosures:

1. Response to Round 9 Request for Additional Information - SBWB-65 through SBWB-74
2. Response to Round 9 Request for Additional Information - SBWB-49 (Proprietary Information Version)
3. Response to Round 9 Request for Additional Information - SBWB-49 (Non-Proprietary Information Version)
4. GE Affidavit For Enclosure 2
5. BFN Units 2 and 3 - TS-418 TS Changes Remarked Using Current TS Pages
6. Copy of AREVA NP Proprietary Notice Belonging to Enclosure 1 of the May 11, 2006, TVA submittal

cc: See page 4

Enclosures:
cc (Enclosures):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

RESPONSE TO ROUND 9 REQUEST FOR ADDITIONAL INFORMATION SBWB-65 THROUGH SBWB-74

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**NRC RAI SBWB-65 (Units 2 and 3)**

Provide the head flow curves used in the limiting large break loss-of-coolant accident LBLOCA analyses (battery failure case). The curves should include the head flow curve for one low pressure core spray and one low pressure coolant injection pump discharging into each recirculation line. Also, provide the limiting axial power shape used in this limiting break.

**TVA Reply to RAI SBWB-65 (Units 2 and 3)**

The limiting LOCA break characteristics are identified in Reference 9.7 of the BFN Unit 2 Reload Analysis Report (RAR) and are presented below in Table SBWB-65-1. Reference 9.7 is EMF-2950(P) Revision 1, "Browns Ferry Units 1, 2, and 3 Extended Power Uprate LOCA Break Spectrum Analysis," Framatome ANP, April 2004. The RAR, "ANP-2541, Browns Ferry Unit 2 Cycle 15 Reload Analysis," was submitted to NRC on June 12, 2006 (ADAMS Accession No. ML061670151).

| Table SBWB-65-1<br>Limiting LOCA<br>Break Characteristics | |
|---|---|
| Location | recirculation discharge pipe |
| Type/size | split/0.5 $ft^2$ |
| Limiting Single failure | battery failure |
| Axial power shape | mid-peaked |
| Initial state | 102% EPU/105% rated flow |

For the battery failure case for the recirculation piping

discharge side break, one core spray loop (two core spray pumps) would be available. The head flow curve for the low pressure core spray system is provided below in Table SBWB-65-2.

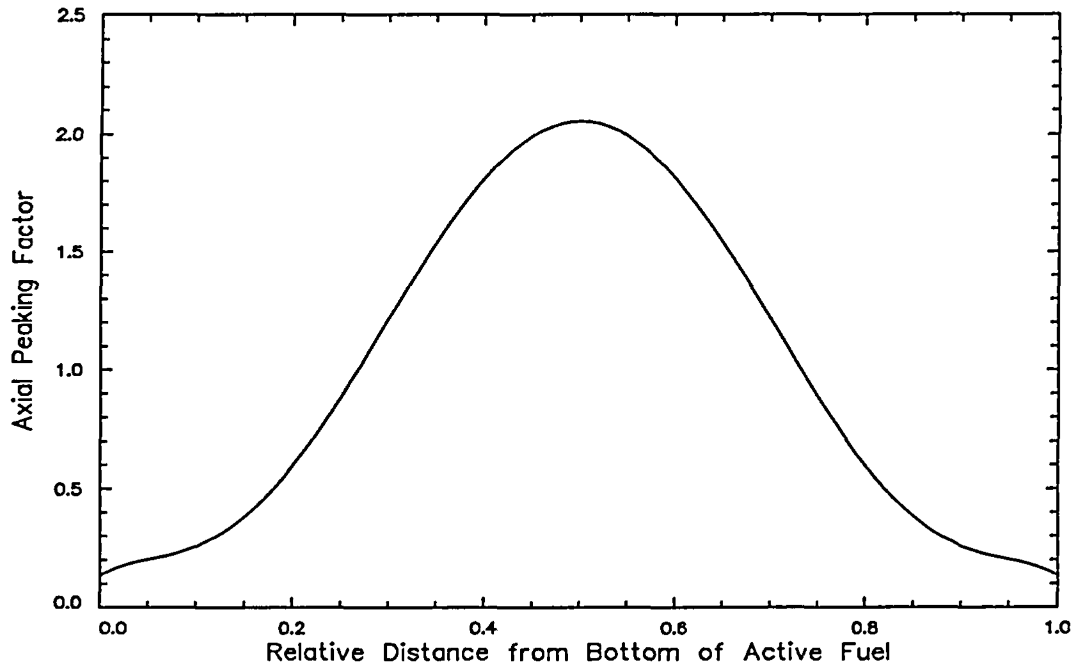
| Table SBWB-65-2<br>Core Spray Flow Rate Versus Pressure | |
|---|---|
| Vessel to Drywell ΔP (psid) | Flow Rate (gallons per minute)<br>2 Pumps Into 1 Sparger |
| 0 | 6,935 |
| 105 | 5,435 |
| 289 | 0 |

For the battery failure case for recirculation suction side break, two low pressure coolant injection (LPCI) system pumps into one LPCI loop would be available. The head flow curve for LPCI is provided below in Table SBWB-65-3.

| Table SBWB-65-3<br>LPCI Flow Rate Versus Pressure | |
|---|---|
| Vessel to Drywell ΔP (psid) | Flow Rate (gallons per minute)<br>2 Pumps Into 1 LPCI Loop |
| 0 | 17,240 |
| 20 | 16,540 |
| 319.5 | 0 |

The limiting axial power shape in the limiting LOCA analysis is shown below in Figure SBWB-65-1.

**Figure SBWB-65-1**
**Limiting Axial Power Shape (LOCA)**




<u>NRC RAI SBWB-66 (Units 2 and 3)</u>

In the Reload Analysis Report (RAR), submitted June 12, 2006, different minimum critical power ratio (MCPR) values are given for different operating conditions. However, the operating MCPR for normal operation (base case) with all the equipment available is not given.

a. Provide the operating limit MCPR with all equipment in operation.

b. Address which transient is the most limiting transient out of the five transients given on page 5-1 in determining the operating MCPR.

c. Provide a table indicating the limiting transient for pressurization and non-pressurization transients.

TVA Reply to RAI SBWB-66 (Units 2 and 3)

Response to part a

The operating limit MCPR for all equipment in service is provided in Tables 5.5 through 5.10 of the RAR under the headings "Base Case Operation." Since BFN Technical Specifications (TS) require 12 of the 13 Main Steam Safety/Relief Valves (MSRVs) to be operable, the base case pressurization events all assume one MSRV is inoperable.

Response to part b

The limiting pressurization transient varies depending on operating domain and equipment out-of-service (EOOS) conditions.

The limiting pressurization transient for BFN is typically either Load Rejection without Bypass (LRNB), Turbine Trip without Bypass (TTNB), or Feedwater Controller Failure (FWCF). The LRNB and TTNB are very similar events - primarily differentiated by small differences in turbine valve closure timing. For Unit 2 Cycle 15, the rated power MCPR operating limit for base case operation was set by FWCF with LRNB and TTNB showing similar results. This is illustrated in the ΔCPR results provided in RAR Sections 5.1.1 through 5.1.3.

The most limiting Control Rod Withdrawal Error (CRWE) MCPR shown in RAR Table 5.12 is equivalent to the MCPR for the base case FWCF with Nominal Scram Speed at rated power at the Near End of Cycle exposure. At other operating domains and EOOS conditions, the CRWE is typically non-limiting. As shown in RAR Sections 5.1.1 through 5.1.3, the Loss of Feedwater Heating (LFWH) event is non-limiting.

Response to part c

Table SBWB-66-1 provides the location in the RAR of the limiting pressurization and non-pressurization results. As discussed above in the response to part b, the limiting pressurization transient (FWCF) is shown in the tables presented in RAR Sections 5.1.1 through 5.1.3.

In AREVA NP methodology, the fuel loading error (misloaded and misoriented bundle) analyses are defined as infrequent events as described in Section 5.6 of the RAR. Consequently, the only traditional non-pressurization events for which CPR impacts are

evaluated are LFWH and CRWE. The CRWE event is the more limiting of these and is shown in RAR Table 5.12.

| **Table SBWB-66-1** | |
|---|---|
| Event Classification | Limiting Rated Power Results |
| Pressurization Transients (LRNB, TTNB, FWCF) | RAR Sections 5.1.1 to 5.1.3 |
| Non-pressurization Transient | RAR Table 5.12 |

**<u>NRC RAI SBWB-67 (Units 2 and 3)</u>**

In the RAR for Unit 2, the SLMCPR assumed for two loop operation is 1.08, but in the proposed Technical Specification (TS) 2.1.1.2, the safety limit MCPR (SLMCPR) is specified as 1.07. Address which is correct. For Unit 3, the proposed TS SLMCPR is 1.08. Address why the proposed SLMCPR values are different.

**<u>TVA Reply to SBWB-67 (Units 2 and 3)</u>**

The SLMCPR values for Unit 2 Cycle 15 operation, which will be the first Unit 2 EPU operating cycle, were submitted to NRC on June 12, 2006, in Section 7.1.1 of the RAR. The SLMCPR values for Unit 2 Cycle 15 are 1.08 for two recirculation loop operation and 1.10 for single recirculation loop operation. The SLMCPR values for the current Unit 2 operating cycle (Cycle 14) are also 1.08/1.10 as shown in TS 2.1.1.2 (page 2.0-1) of current Unit 2 TS. Thus, the SLMCPR values in the RAR for Unit 2 Cycle 15 are the same as those in current Unit 2 TS and no SLMCPR TS changes are needed for the first Unit 2 EPU cycle.

In the original June 25, 2004, TS-418 submittal, TS page 2.0-1 was marked up to show a change to the scaling factor in TS 2.1.1.1 for the thermal power safety limit. The 1.07 SLMCPR value referenced in the RAI was on the mark-up TS page, but was not shown as a change and is not the current Unit 2 TS SLMCPR value. The current Unit 3 TS 2.1.1.2 SLMCPR values (1.09/1.11) also differ from the 1.08/1.10 shown on Unit 3 page 2.0-1 of the original TS-418 submittal.

Several other TS pages in the original June 25, 2004, Units 2 and 3 TS-418 EPU submittal have been subsequently superseded by

other TS changes. To facilitate NRC's review of the Units 2 and 3 EPU license applications, TVA has remarked the TS-418 proposed TS changes from the original June 25, 2004, submittal onto current TS pages in Enclosure 5. For completeness, an entire copy of the proposed Unit 2 and 3 TS changes is included in Enclosure 5. TS pages which have been superseded can be distinguished by the hand mark-ups. If the TS page has not changed, a copy of the original TS-418 pages is provided and is distinguished by the absence of hand mark-ups.

As NRC is aware, the SLMCPR values for a given operating cycle are determined as part of the routine core design and reload analysis process. If an SLMCPR TS change is required as a result of a cycle-specific reload analysis, a TS change request is submitted to NRC, typically 3 to 5 months prior to the refueling outage. The timing of the TS submittal results from the fact that the SLMCPR for a specific cycle cannot be determined prior to the completion of the core design.

The Unit 3 SLMCPR for its initial EPU cycle (Cycle 14 - Spring 2008) will be determined as part of the normal core design and reload analysis process. A TS change will be submitted to NRC, if required, several months prior to the Spring 2008 refueling outage per standard protocol. The Unit 3 Cycle 14 core will consist entirely of AREVA NP Atrium-10 fuel.

**NRC RAI SBWB-68 (Units 2 and 3)**

As stated in the Executive summary of Enclosure 5 of the June 25, 2004, submittal EMF-2982(P), or the Framatome Uprate Safety Analysis Report (FUSAR), the FUSAR provides results for the fuel-related analyses for a reference core of ATRIUM-10 fuel. Therefore, for fuel related issues concerning Units 2 and 3, the NRC staff has focused the review on the FUSAR rather than Enclosure 4 of the June 25, 2004, submittal Power Uprate Safety Analysis Report (PUSAR) which contained the fuel related analyses for a reference core of GE-14 fuel. For many of the RAI responses for fuel-related issues, TVA refers to the PUSAR rather than the FUSAR. For example, in response to SRXB-A.2, TVA stated that:

> ...the conclusions of the PUSAR, NEDC-33047, are applicable and bounding for both Units 2 and 3.

Also, in response to SRXB-A.22, TVA stated that:

> The scenario and sequence of events remain valid for

the fuel-related EPU analyses and are consistent with the event descriptions presented in the UFSAR.

Confirm that similar conclusions can be made for the FUSAR.

Additionally, in response to SRXB-A.22, TVA stated that:

In most cases, the PUSAR analysis remains applicable for ATRIUM-10 fuel.

Identify the areas where the PUSAR analysis is not applicable for ATRIUM-10.

**TVA Reply to RAI SBWB-68 (Units 2 and 3)**

The Units 2 and 3 EPU FUSAR (AREVA NP) should be considered an addendum to the General Electric (GE) supplied PUSAR. Section 1.1 of the FUSAR states:

"…This report summarizes the impact that operation with ATRIUM-10 fuel at EPU conditions has on the Reference 1 [PUSAR] evaluations for Browns Ferry Units 2 and 3. The ATRIUM-10 EPU analyses follow the NRC-approved generic format and content described in References 2 [ELTR1] and 3 [ELTR2]."

Specifically, AREVA NP evaluated each of the PUSAR sections to determine if there was any fuel-related impact. The approach taken is discussed in Section 1.2 of the FUSAR, but it may be summarized that only fuel-related analyses of the PUSAR were explicitly analyzed with the ATRIUM-10 reference core. The FUSAR retains the section numbering corresponding to the PUSAR to simplify comparison between the two reports. The FUSAR explicitly states if the PUSAR evaluation is not dependent upon fuel design. For the other analyses, the FUSAR provides an evaluation of the impact ATRIUM-10 has on the PUSAR analyses.

In the answer to question SRXB-A.2 (Unit 2 and 3), TVA answered that the GE analyses performed for the PUSAR were based upon bounding parameters for all three BFN units. The major differences between units that were identified in the response to SRXB-A.2 included steamline pressure drop (due to Main Steam Isolation Valve differences) and Emergency Core Cooling System (ECCS) leakage (due to differences in repairs). The lowest steamline pressure drop was assumed to provide the most severe response for pressurization transient abnormal operating occurrences. Similarly, the highest combined ECCS leakage was assumed to conservatively maximize the calculated peak clad temperatures for LOCA. The same approach was taken in the AREVA

NP analyses whose results are provided in the FUSAR. Therefore, the results in the FUSAR are applicable and bounding for both units 2 and 3.

As noted above, the PUSAR was reviewed to determine which evaluations were potentially fuel-dependent. As stated in the RAI question, TVA responded to SRXB-A.22 stating that in most cases the PUSAR analysis remains applicable for ATRIUM-10 fuel and if not, they are addressed in the FUSAR. The basis of this statement is that a large number of the PUSAR evaluations are not dependent upon fuel type and a significant number of the fuel type dependent analyses are insensitive to the change in fuel design. Of the remaining fuel dependent analyses, specific calculations were performed to show that acceptable results were still obtained at EPU conditions. No unacceptable results were obtained in either the PUSAR or FUSAR.

**NRC RAI SBWB-69 (Units 2 and 3)**

For Units 2 and 3, RAI SRXB-A.9 indicates that the peak calculated pressure for the reactor overpresssure analysis is 1204 pounds per square inch gage (psig). Address whether the response is applicable only for Unit 2 and 3, or does it also apply to Unit 1. Confirm that the proposed Units 2 and 3 TS Surveillance Requirement (SR) 3.1.7.6 standby liquid control system pump discharge test pressure of 1275 psig is satisfactory considering the operating margin for the pump discharge relief set pressure. Address why the change in pressure proposed for Unit 1 (1275 psig to 1325 psig in SR 3.1.7.6) is not applicable for Units 2 and 3.

**TVA Reply to RAI SBWB-69 (Units 2 and 3)**

The subject Units 2 and 3 RAI SRXB-A.9 response references Section 6.5 and Table 9.4 of the GE PUSAR; the conclusions of which are based on the GE Anticipated Transient Without Scram (ATWS) analysis. Since the PUSAR constitutes the base analysis for Unit 1, it follows that the RAI SRXB-A.9 response is equally applicable to all three units.

On Units 2 and 3, the TS SR criteria for the standby liquid control (SLC) system pump discharge test pressure was previously raised from 1275 psig to 1325 psig as part of the first power uprate license amendment. This change was approved by NRC for Units 2 and 3 in the Safety Evaluation Report dated September 8, 1998 (ML020100022). Therefore, no further change in the SLC pump discharge test pressure SR criteria was requested in TS-418, but was required for Unit 1 as proposed in

TS-431. If TS-431 is approved, all three BFN units will have the same SLC pump discharge test pressure SR criteria of 1325 psig.

The response to Units 2 and 3 RAI SRXB-A.10 discusses the adequacy of the operating margin for the SLC pump discharge relief set pressure.

**NRC RAI SBWB-70 (Units 2 and 3)**

In RAR Section 5.6, Fuel Loading Error, the acceptance criteria is given, provide the associated analyses for uprated conditions.

**TVA Reply to RAI SBWB-70 (Units 2 and 3)**

Sections 5.6.1 (Mislocated Fuel Assembly) and 5.6.2 (Misoriented Bundle) of the RAR state that the analysis was performed for each event. The result given in the RAR is that neither the transient Linear Heat Generation Rate limit nor 0.1% dryout limits weas exceeded. Therefore, the dose acceptance criteria are met since the fuel failure thresholds are not exceeded.

**NRC RAI SBWB-71 (Units 2 and 3)**

In RAR Section 5.6.1, identify the topical report and the evaluation model used for the Mislocated Fuel Assembly event.

**TVA Reply to SBWB-71 (Units 2 and 3)**

The Mislocated Fuel Assembly event analysis methodology is defined in topical report XN-NF-80-19(P)(A), Vol. 1, Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983. The acceptance criterion for the event is defined in XN-NF-80-19(P)(A), Vol. 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986. The evaluation model used is CASMO4/MICROBURN-B2 as defined in topical report EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO4/MICROBURN-B2," Siemens Power Corporation, October 1999 (RAR Reference 8.5).

**NRC RAI SBWB-72 (Units 2 and 3)**

In RAR Section 5.6.2, identify the topical report and the

evaluation model used for the Misoriented Fuel Bundle.

**TVA Reply to SBWB-72 (Units 2 and 3)**

The Misoriented Fuel Bundle event analysis methodology is defined in topical report XN-NF-80-19(P)(A), Vol. 1, Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983. The acceptance criterion for the event is defined in XN-NF-80-19(P)(A), Vol. 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986. The evaluation model used is CASMO4/MICROBURN-B2 as defined in topical report EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO4/MICROBURN-B2," Siemens Power Corporation, October 1999 (RAR Reference 8.5).

**NRC RAI SBWB-73 (Units 2 and 3)**

Table 9.2 of the FUSAR does not include the following events: Loss of Auxiliary Power, Main Condenser Vacuum, Recirculation Flow Controller Failure, Trip of one pump, Trip of two pumps, Recirculation flow controller failure, or Start-up of idle pump. Confirm whether these events were analyzed and documented.

**TVA Reply to RAI SBWB-73 (Units 2 and 3)**

The events for which results are presented in Table 9.2 of the FUSAR are identified in Table E-1 of ELTR-1, GE Licensing topical report NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate." Table 9-2 of the PUSAR presents results for the same events. Analyses for the events identified in this RAI were performed for BFN Units 2 and 3 at EPU conditions and have been documented and reviewed per AREVA NP quality assurance procedures. Results for these events are presented below in Table SBWB-73-1.

| Table SBWB-73-1 | | | | |
|---|---|---|---|---|
| Event | Peak Neutron Flux (% of rated EPU) | Peak Heat Flux (% of rated EPU) | ΔCPR | MCPR Operating Limit |
| Loss of Condenser vacuum | 292 | 117 | 0.25 | 1.33 |
| Loss of Auxiliary Transformers | 100 | 100 | 0.17 | 1.25 |
| Loss of Auxiliary Power Grid | 286 | 116 | 0.25 | 1.33 |
| Recirculation Flow Control Failure - Decreasing Flow | 100 | 100 | 0.04 | 1.12 |
| Trip of one recirculation pump | 100 | 100 | 0.04 | 1.12 |
| Trip of two recirculation pumps | 100 | 101 | 0.13 | 1.21 |
| Startup of idle recirculation pump* | 83 | 82 | 0.29 | 1.37 |

* The startup of an idle recirculation pump event was initiated from 66% power/52% flow. The power dependent MCPR limits increase as power decreases so the results of this event are non-limiting.

**NRC RAI SBWB-74 (Units 2 and 3)**

Address why the anticipated transient without scram analysis was not done in the RAR.

**TVA Reply to RAI SBWB-74 (Units 2 and 3)**

The ATWS reactivity margin with SLCS is provided in Section 4.2.2 of the Unit 2 Cycle 15 RAR. The contents of the the RAR are defined by NRC-approved documents. This value is not in the RAR since it is not specified in Appendix A of topical report XN-NF-80-19, Volume 4, Revision 1. However, the ATWS overpressure analysis is performed by AREVA NP as part of their reload process and the result for Unit 2 Cycle 15 is 1486 psig in the vessel lower plenum. This meets the acceptance criteria of ≤1500 psig (ASME service Level C limit of 120% of the design pressure).

# ENCLOSURE 3

## TENNESSEE VALLEY AUTHORITY
## BROWNS FERRY NUCLEAR PLANT (BFN)
## UNIT 1

## RESPONSE TO ROUND 9 REQUEST FOR ADDITIONAL INFORMATION - SBWB-49 (NON-PROPRIETARY INFORMATION VERSION)

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This enclosure provides TVA's response to RAI SBWB-49 from NRC's September 1, 2006, Round 9 Request for Additional Information for BFN Unit 1.

**NRC RAI SBWB-49**

Provide the head flow curves used in the limiting large break loss-of-coolant accident analyses (battery failure case). The curves should include the head flow curve for one low pressure core spray and one low pressure coolant injection pump discharging into each recirculation line. Also, provide the limiting axial power shape used in this limiting break.

**TVA Reply to SBWB-49**

The information requested is provided in the attached figures. The information provided is not dependent on the break size and location, but the axial power shape for the hot bundle depends on the fuel type and the power/flow conditions. Figure SBWB-49-1 shows the axial power shapes in the hot and average bundle as a function of the distance above the bottom of the core. The hot bundle shape is based on GE14 fuel with the plant operating at rated Extended Power Uprate power and flow.

Figure SBWB-49-2 shows the flow into the shroud of one core spray loop (two core spray pumps) as a function of the differential pressure between the vessel and drywell.

Figure SBWB-49-3 shows the Low Pressure Coolant Injection (LPCI) flow curves for one LPCI pump and two LPCI pumps injecting into one LPCI loop. The LPCI flow is the flow into the jet pumps as a function of the differential pressure between the vessel and drywell. The curve for two LPCI pumps into one LPCI loop is also included because it represents the available LPCI systems for recirculation suction pipe breaks when the battery failure is the single failure.

[[

{3}]]

Figure SBWB-49-1. Axial Power Shapes in the Hot and Average Bundles

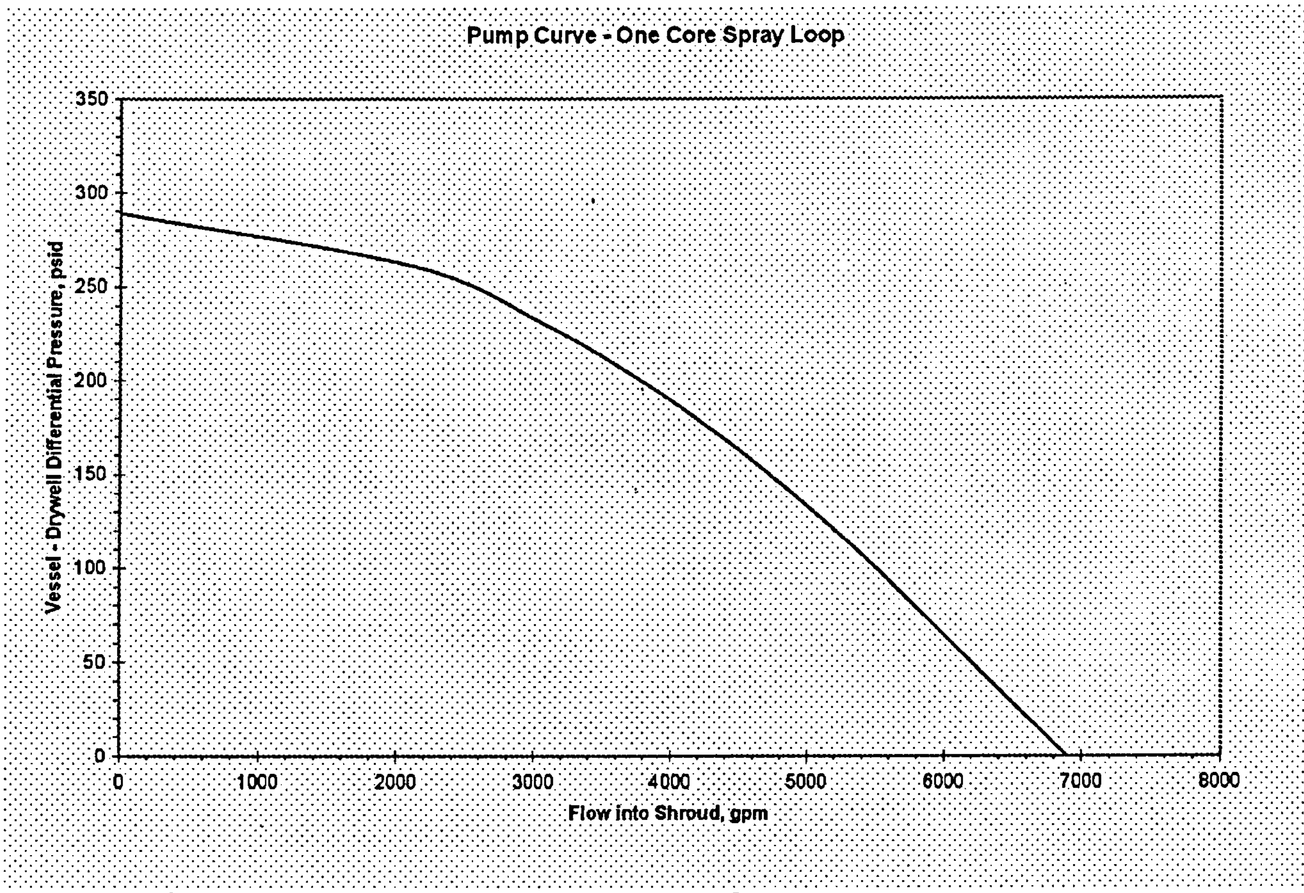


Figure SBWB-49-2. Pump Curve for One Core Spray Loop

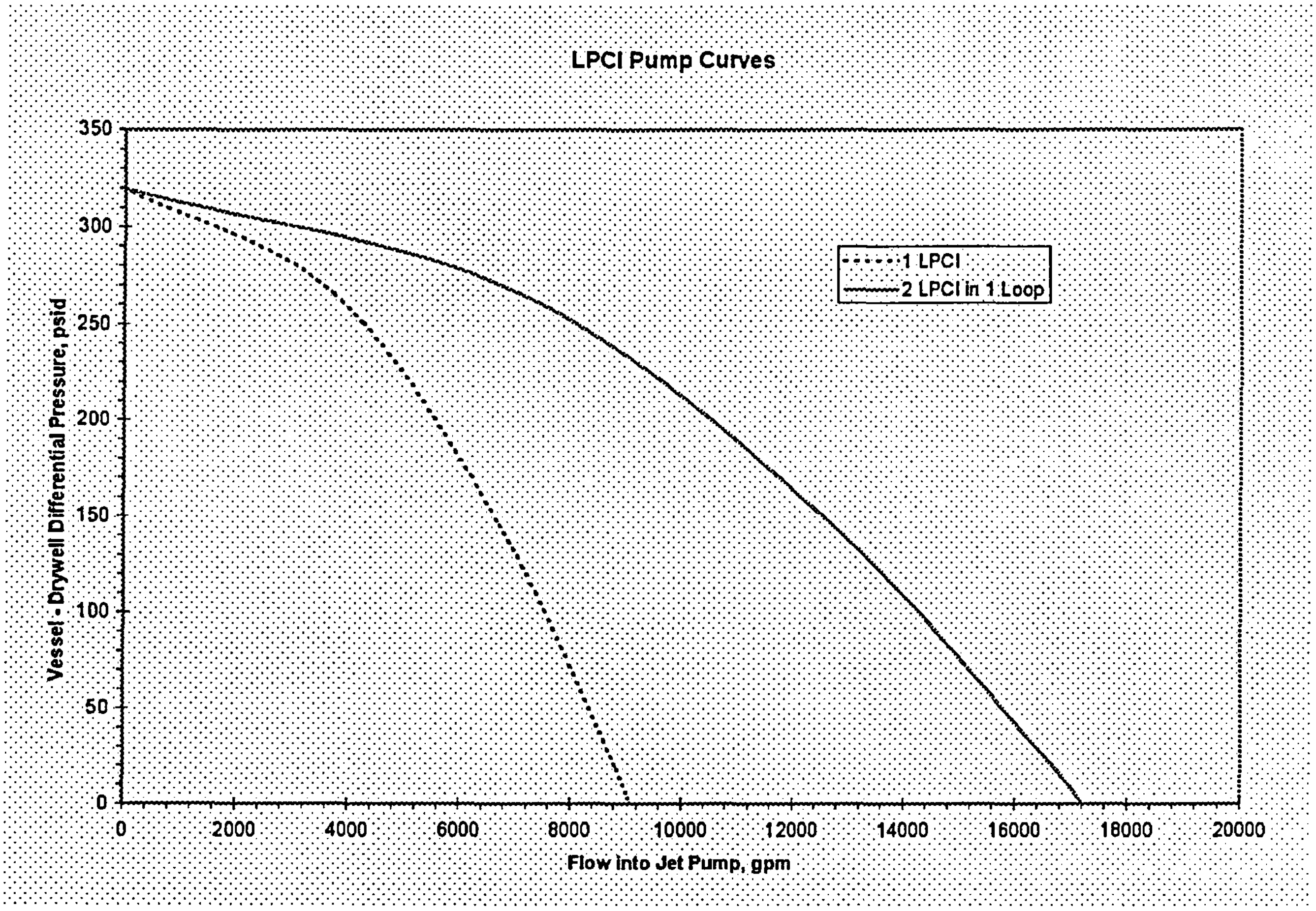


Figure SBWB-49-3. LPCI Pump Curves

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

GE AFFIDAVIT FOR ENCLOSURE 2

# AFFIDAVIT

I, Louis M. Quintana, state as follows:

(1) I am Manager, Licensing, General Electric Company ("GE"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 to GE letter GE-ER1-AEP-06-346, Larry King (GE) to J. Valente (TVA), *GE Response to NRC Request for Additional Information – SBWB-49,* dated August 30, 2006. The proprietary information is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission</u>, 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority (or his delegate) for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions from evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability for the power uprate of a GE BWR, utilizing analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of and applied to perform evaluations of the transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the underlying evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 30th day of August 2006.

Louis M. Quintana

Louis M. Quintana
Manager, Licensing

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

TS-418 TS CHANGES REMARKED USING CURRENT TS PAGES

sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

3952

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 295, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 253 to Facility Operating License DPR-52, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 253. For SRs that existed prior to Amendment 253, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 253.

(3) The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's

1.1 Definitions (continued)

| | |
|---|---|
| PHYSICS TESTS | PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:<br>a. Described in Section 13.10, Refueling Test Program; of the FSAR;<br>b. Authorized under the provisions of 10 CFR 50.59; or<br>c. Otherwise approved by the Nuclear Regulatory Commission. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt. 3952 |
| SHUTDOWN MARGIN (SDM) | SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:<br>a. The reactor is xenon free;<br>b. The moderator temperature is 68°F; and<br>c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM. |

(continued)

# 2.0 SAFETY LIMITS (SLs)

## 2.1 SLs

### 2.1.1 Reactor Core SLs

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

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THERMAL POWER shall be ≤ ~~25~~% RTP.

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

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single loop operation.

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

### 2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

## 2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

SURVEILLANCE REQUIREMENTS (continued)

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| | Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1. | Once within 8 hours after discovery that SPB concentration is > 9.2% by weight<br>AND<br>12 hours thereafter |
| 203<br>SR 3.1.7.5 | Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is ≥ ~~186~~ pounds. | 31 days |
| SR 3.1.7.6 | Verify the SLC conditions satisfy the following equation:<br>$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$<br>where,<br>C = sodium pentaborate solution concentration (weight percent)<br>Q = pump flow rate (gpm)<br>E = Boron-10 enrichment (atom percent Boron-10) | 31 days<br>AND<br>Once within 24 hours after water or boron is added to the solution |
| SR 3.1.7.7 | Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig. | 24 months |

(continued)

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25~~ 23% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Any APLHGR not within limits. | A.1 | Restore APLHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

same AS originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.1.1 | Verify all APLHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ ~~25~~ 23% RTP<br><br>AND<br><br>24 hours thereafter |

same as
originally
submitted

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25~~ 23% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Any MCPR not within limits. | A.1 | Restore MCPR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < ~~25~~23% RTP | 4 hours |

same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.2.1 | Verify all MCPRs are greater than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ ~~25~~ 23% RTP<br><br>AND<br><br>24 hours thereafter |
| SR 3.2.2.2 | Determine the MCPR limits. | Once within 72 hours after each completion of SR 3.1.4.1<br><br>AND<br><br>Once within 72 hours after each completion of SR 3.1.4.2 |

same as originally submitted

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25~~23% RTP.

ACTIONS

| CONDITION | | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|---|
| A. | Any LHGR not within limits. | A.1 | Restore LHGR(s) to within limits. | 2 hours |
| B. | Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

Same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.3.1 | Verify all LHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ ~~25~~23% RTP<br><br>AND<br><br>24 hours thereafter |

same as originally submitted

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. -------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. --------------------------------- One or more Functions with one or more required channels inoperable in both trip systems. | B.1 | Place channel in one trip system in trip. | 6 hours |
| | OR | | |
| | B.2 | Place one trip system in trip. | 6 hours |
| C. One or more Functions with RPS trip capability not maintained. | C.1 | Restore RPS trip capability. | 1 hour |
| D. Required Action and associated Completion Time of Condition A, B, or C not met. | D.1 | Enter the Condition referenced in Table 3.3.1.1-1 for the channel. | Immediately |
| E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | E.1 | Reduce THERMAL POWER to < ~~30~~26% 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 |

(continued)

same as originally submitted

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------------------------<br>Not required to be performed until 12 hours after THERMAL POWER ≥ ~~25~~23% RTP.<br>---------------------------------------------------------<br><br>Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ ~~25~~23% RTP. | 7 days |
| SR 3.3.1.1.3 | -------------------------NOTE------------------------<br>Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.<br>---------------------------------------------------------<br><br>Perform CHANNEL FUNCTIONAL TEST. | 7 days |

(continued)

Same as originally submitted

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. | 24 months |
| SR 3.3.1.1.13 | -------------------------NOTE------------------------<br>Neutron detectors are excluded.<br>-------------------------------------------------------<br><br>Perform CHANNEL CALIBRATION. | <br><br><br><br>24 months |
| SR 3.3.1.1.14 | Perform LOGIC SYSTEM FUNCTIONAL TEST. | 24 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~~26% RTP. | 24 months |
| SR 3.3.1.1.16 | -------------------------NOTE------------------------<br>For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.<br>-------------------------------------------------------<br><br>Perform CHANNEL FUNCTIONAL TEST. | <br><br><br><br><br><br>184 days |
| SR 3.3.1.1.17 | Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 25% and recirculation drive flow is < 60% of rated recirculation drive flow. | 24 months |



same as originally submitted

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<br>SR 3.3.1.1.3<br>SR 3.3.1.1.5<br>SR 3.3.1.1.6<br>SR 3.3.1.1.9<br>SR 3.3.1.1.14 | ≤ 120/125 divisions of full scale |
| | 5(a) | 3 | H | SR 3.3.1.1.1<br>SR 3.3.1.1.4<br>SR 3.3.1.1.9<br>SR 3.3.1.1.14 | ≤ 120/125 divisions of full scale |
| b. Inop | 2 | 3 | G | SR 3.3.1.1.3<br>SR 3.3.1.1.14 | NA |
| | 5(a) | 3 | H | SR 3.3.1.1.4<br>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<br>SR 3.3.1.1.6<br>SR 3.3.1.1.7<br>SR 3.3.1.1.13<br>SR 3.3.1.1.16 | ≤ ~~15~~13% RTP |
| b. Flow Biased Simulated Thermal Power - High | 1 | 3(b) | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.7<br>SR 3.3.1.1.13<br>SR 3.3.1.1.16 | ~~≤ 0.66 W + 66% RTP and ≤ 120% RTP~~(c) |
| c. Neutron Flux - High | 1 | 3(b) | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.7<br>SR 3.3.1.1.13<br>SR 3.3.1.1.16 | ≤ 120% RTP |

(continued)

≤ 0.55 W
+ 65.5% RTP and
≤ 120% RTP

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

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

(c) [~~.66~~55 W + ~~66~~65.5% - ~~.66~~55Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating.

same as originally submitted

Table 3.3.1.1-1 (page 3 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 |
|---|---|---|---|---|---|
| 7. Scram Discharge Volume Water Level - High (continued) | | | | | |
| b. Float Switch | 1,2 | 2 | G | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 46 gallons |
| | 5(a) | 2 | H | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 46 gallons |
| 8. Turbine Stop Valve - Closure | ≥ ~~30~~26% RTP | 4 | E | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14<br>SR 3.3.1.1.15 | ≤ 10% closed |
| 9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low | ≥ ~~30~~26% RTP | 2 | E | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14<br>SR 3.3.1.1.15 | ≥ 550 psig |
| 10. Reactor Mode Switch - Shutdown Position | 1,2 | 1 | G | SR 3.3.1.1.12<br>SR 3.3.1.1.14 | NA |
| | 5(a) | 1 | H | SR 3.3.1.1.12<br>SR 3.3.1.1.14 | NA |
| 11. Manual Scram | 1,2 | 1 | G | SR 3.3.1.1.8<br>SR 3.3.1.1.14 | NA |
| | 5(a) | 1 | H | SR 3.3.1.1.8<br>SR 3.3.1.1.14 | NA |
| 12. RPS Channel Test Switches | 1,2 | 2 | G | SR 3.3.1.1.4 | NA |
| | 5(a) | 2 | H | SR 3.3.1.1.4 | NA |
| 13. Deleted | | | | | |

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

same as originally submitted

## 3.3 INSTRUMENTATION

### 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels of feedwater and main turbine high water level trip instrumentation per trip system shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ ~~25~~23% RTP.

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more feedwater and main turbine high water level trip channels inoperable, in one trip system. | A.1 Place channel(s) in trip. | 7 days |
| B. One or more feedwater and main turbine high water level trip channels inoperable in each trip system. | B.1 Restore feedwater and main turbine high water level trip capability. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

same as originally submitted

## 3.3 INSTRUMENTATION

### 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

   1. Turbine Stop Valve (TSV) - Closure; and

   2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

<u>OR</u>

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ 30% RTP.

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ACTIONS

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

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more channels inoperable. | A.1 | Restore channel to OPERABLE status. | 72 hours |
| | OR | | |
| | A.2 | ------------NOTE------------<br>Not applicable if inoperable channel is the result of an inoperable breaker.<br>--------------------------------<br>Place channel in trip. | 72 hours |
| B. One or more Functions with EOC-RPT trip capability not maintained.<br>AND<br>MCPR and LHGR limit for inoperable EOC-RPT not made applicable. | B.1 | Restore EOC-RPT trip capability. | 2 hours |
| | OR | | |
| | B.2 | Apply the MCPR and LHGR limit for inoperable EOC-RPT as specified in the COLR. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 | Reduce THERMAL POWER to < 30% RTP. | 4 hours |

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SURVEILLANCE REQUIREMENTS

------------------------------------------------NOTE------------------------------------------------

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 EOC-RPT trip capability.

----------------------------------------------------------------------------------------------------

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.4.1.1 | Perform CHANNEL FUNCTIONAL TEST. | 92 days |
| SR 3.3.4.1.2 | Verify TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ ~~30~~26% RTP. | 24 months |
| SR 3.3.4.1.3 | Perform CHANNEL CALIBRATION. The Allowable Values shall be:<br><br>TSV - Closure: ≤ 10% closed; and<br><br>TCV Fast Closure, Trip Oil Pressure - Low: ≥ 550 psig. | 24 months |
| SR 3.3.4.1.4 | Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation. | 24 months |

same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.2.1 | -------------------------NOTES---------------------<br>1. Not required to be performed until 4 hours after associated recirculation loop is in operation.<br>2. Not required to be performed until 24 hours after > ~~25~~23% RTP.<br>-----------------------------------------------------------<br><br>Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop:<br><br>a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.<br>b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.<br>c. Each jet pump flow differs by ≤ 10% from established patterns. | 24 hours |

same as
originally
submitted

## 3.7 PLANT SYSTEMS

### 3.7.1 Residual Heat Removal Service Water (RHRSW) System ~~and Ultimate Heat Sink (UHS)~~

LCO 3.7.1

-------------------------------------NOTE-------------------------------------
The number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for ≥ 24 hours.
------------------------------------------------------------------------------

Four RHRSW subsystems ~~and UHS~~ shall be OPERABLE with the number of OPERABLE pumps as listed below:

1. 1 unit fueled - four OPERABLE RHRSW pumps.
2. 2 units fueled - six OPERABLE RHRSW pumps.
3. 3 units fueled - eight OPERABLE RHRSW pumps.

APPLICABILITY: MODES 1, 2, and 3.

same as
originally
submitted

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One required RHRSW pump inoperable. | A.1 | -------------NOTES-----------<br>1. Only applicable for the 2 units fueled condition.<br>2. Only four RHRSW pumps powered from a separate 4 kV shutdown board are required to be OPERABLE if the other fueled unit has been in MODE 4 or 5 for ≥ 24 hours.<br>---------------------------------<br><br>Verify five RHRSW pumps powered from separate 4 kV shutdown boards are OPERABLE. | Immediately |
| | OR | | |
| | A.2 | Restore required RHRSW pump to OPERABLE status. | 30 days |

(continued)

same as originally submitted

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. One RHRSW subsystem inoperable. | B.1 | -------------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling - Hot Shutdown," for RHR shutdown cooling made inoperable by the RHRSW system. -------------------------------- | |
| | | Restore RHRSW subsystem to OPERABLE status. | 30 days |
| C. Two required RHRSW pumps inoperable. | C.1 | Restore one inoperable RHRSW pump to OPERABLE status. | 7 days |
| D. Two RHRSW subsystems inoperable.. | D.1 | ------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.7, for RHR shutdown cooling made inoperable by the RHRSW System. -------------------------------- | |
| | | Restore one RHRSW subsystem to OPERABLE status. | 7 days |

(continued)

same as originally submitted

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| E. Three or more required RHRSW pumps inoperable. | E.1 | Restore one RHRSW pump to OPERABLE status. | 8 hours |
| F. Three or more RHRSW subsystems inoperable. | F.1 | -------------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by the RHRSW System. --------------------------------- | |
| | | Restore one RHRSW subsystem to OPERABLE status. | 8 hours |
| G. Required Action and associated Completion Time not met. | G.1 | Be in MODE 3. | 12 hours |
| ~~OR~~ | AND | | |
| ~~UHS inoperable~~ | G.2 | Be in MODE 4. | 36 hours |

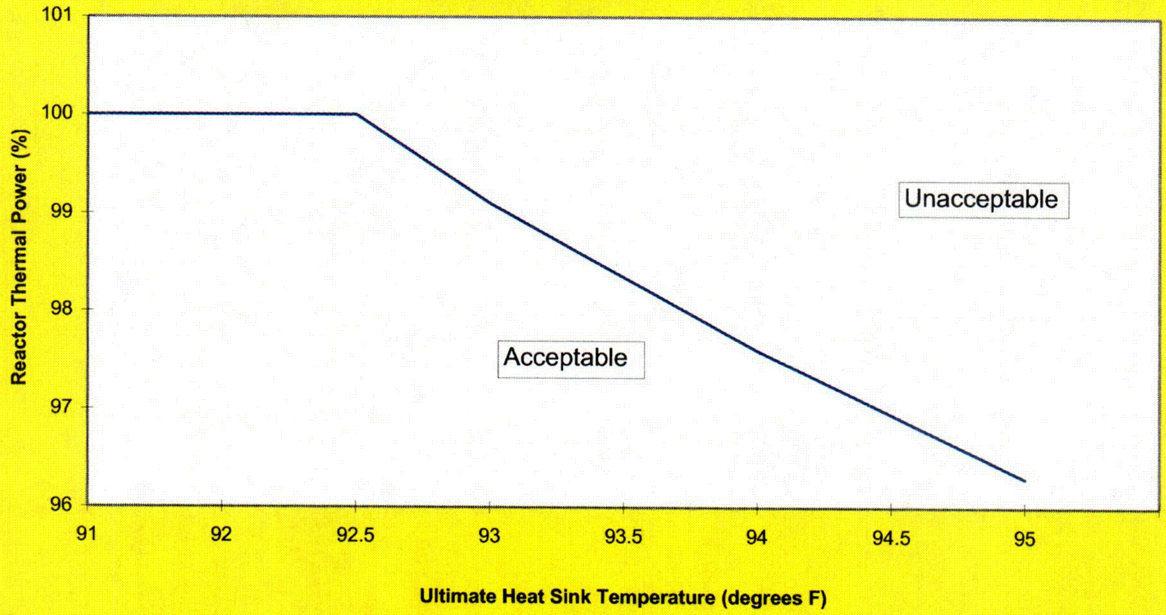
Same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.1.1 | Verify each RHRSW manual and power operated valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position. | 31 days |
| ~~SR 3.7.1.2~~ | ~~Verify the average water temperature of UHS is within the limits specified in Figure 3.7.1-1.~~ | ~~24 hours UHS temperature ≤ 91°F~~<br><br>~~AND~~<br><br>~~1 hour UHS temperature >91°F~~ |

Same as originally submitted

Figure 3.7.1-1
Reactor Thermal Power Versus Ultimate Heat Sink Temperature Limit




Delete Figure.

same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.2.1 | ~~-------------------------NOTE-------------------------~~<br>~~Refer to SR 3.7.1.2 for additional UHS requirements.~~<br>~~------------------------------------------------------~~<br><br>Verify the average water temperature of UHS is $\leq$ 95°F. | 24 hours |
| SR 3.7.2.2 | -------------------------NOTE-------------------------<br>Isolation of flow to individual components does not render EECW System inoperable.<br>------------------------------------------------------<br><br>Verify each EECW system manual and power operated valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position. | 31 days |
| SR 3.7.2.3 | Verify each required EECW pump actuates on an actual or simulated initiation signal. | 24 months |

Same as
originally
submitted

## 3.7 PLANT SYSTEMS

### 3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

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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. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < 25% RTP. | 4 hours |

## 5.5 Programs and Manuals

---

5.5.12 <u>Primary Containment Leakage Rate Testing Program</u> (continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 50.6 psig. The maximum allowable primary containment leakage rate, $L_a$, shall be 2% of primary containment air weight per day at $P_a$. 48.5

Leakage Rate acceptance criteria are:

a. The primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following the testing performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A test; and

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) Air lock door seals leakage rate is $\leq 0.02\ L_a$ when the overall air lock is pressurized to $\geq 2.5$ psig for at least 15 minutes.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

---

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ~~3458~~ megawatts thermal.

3952

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 253, except for Amendment No. 248, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 212 to Facility Operating License DPR-68, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 212. For SRs that existed prior to Amendment 212, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 212.

1.1 Definitions (continued)

| | |
|---|---|
| PHYSICS TESTS | PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:<br>a. Described in Section 13.10, Refueling Test Program; of the FSAR;<br>b. Authorized under the provisions of 10 CFR 50.59; or<br>c. Otherwise approved by the Nuclear Regulatory Commission. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt. 3952 |
| SHUTDOWN MARGIN (SDM) | SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:<br>a. The reactor is xenon free;<br>b. The moderator temperature is 68°F; and<br>c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM. |

(continued)

# 2.0 SAFETY LIMITS (SLs)

## 2.1 SLs

### 2.1.1 <u>Reactor Core SLs</u>

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

23

THERMAL POWER shall be ≤ 25% RTP.

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

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.11 for single loop operation.

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

### 2.1.2 <u>Reactor Coolant System Pressure SL</u>

Reactor steam dome pressure shall be ≤ 1325 psig.

## 2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

SURVEILLANCE REQUIREMENTS (continued)

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| | Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1. | Once within 8 hours after discovery that SPB concentration is > 9.2% by weight<br><br>AND<br><br>12 hours thereafter |
| SR 3.1.7.5 | Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is ≥ 186 203 pounds. | 31 days |
| SR 3.1.7.6 | Verify the SLC conditions satisfy the following equation:<br><br>$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$<br><br>where,<br><br>C = sodium pentaborate solution concentration (weight percent)<br><br>Q = pump flow rate (gpm)<br><br>E = Boron-10 enrichment (atom percent Boron-10) | 31 days<br><br>AND<br><br>Once within 24 hours after water or boron is added to the solution |
| SR 3.1.7.7 | Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig. | 24 months |

(continued)

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25~~23% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any APLHGR not within limits. | A.1 Restore APLHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

Same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.1.1 | Verify all APLHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ ~~25~~23% RTP<br><br>AND<br><br>24 hours thereafter |

Same as originally submitted

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25~~23% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any MCPR not within limits. | A.1 Restore MCPR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

Same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.2.1 | Verify all MCPRs are greater than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ ~~25~~23% RTP<br><br>AND<br><br>24 hours thereafter |
| SR 3.2.2.2 | Determine the MCPR limits. | Once within 72 hours after each completion of SR 3.1.4.1<br><br>AND<br><br>Once within 72 hours after each completion of SR 3.1.4.2 |

same as originally submitted

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25~~23% RTP

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Any LHGR not within limits. | A.1 Restore LHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.3.1 | Verify all LHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ ~~25~~23% RTP<br><br>AND<br><br>24 hours thereafter |

same as originally submitted

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. -------------NOTE-------------<br>Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f.<br>---------------------------------<br>One or more Functions with one or more required channels inoperable in both trip systems. | B.1 | Place channel in one trip system in trip. | 6 hours |
| | <u>OR</u> | | |
| | B.2 | Place one trip system in trip. | 6 hours |
| C. One or more Functions with RPS trip capability not maintained. | C.1 | Restore RPS trip capability. | 1 hour |
| D. Required Action and associated Completion Time of Condition A, B, or C not met. | D.1 | Enter the Condition referenced in Table 3.3.1.1-1 for the channel. | Immediately |
| E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1. | E.1 | Reduce THERMAL POWER to < ~~30~~<u>26</u>% 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 |

(continued)

same as originally submitted

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------------------------- <br> Not required to be performed until 12 hours after THERMAL POWER ≥ ~~25~~23% RTP. <br> --------------------------------------------------------- | |
| | Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ ~~25~~23% RTP. | 7 days |
| SR 3.3.1.1.3 | -------------------------NOTE------------------------- <br> Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. <br> --------------------------------------------------------- | |
| | Perform CHANNEL FUNCTIONAL TEST. | 7 days |

(continued)

same as originally submitted

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. | 24 months |
| SR 3.3.1.1.13 | -------------------------NOTE------------------------<br>Neutron detectors are excluded.<br>---------------------------------------------------------<br>Perform CHANNEL CALIBRATION. | <br><br><br>24 months |
| SR 3.3.1.1.14 | Perform LOGIC SYSTEM FUNCTIONAL TEST. | 24 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~~26% RTP. | 24 months |
| SR 3.3.1.1.16 | -------------------------NOTE------------------------<br>For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.<br>---------------------------------------------------------<br>Perform CHANNEL FUNCTIONAL TEST. | <br><br><br>184 days |
| SR 3.3.1.1.17 | Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 25% and recirculation drive flow is < 60% of rated recirculation drive flow. | 24 months |



same as
originally
submitted

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<br>SR 3.3.1.1.3<br>SR 3.3.1.1.5<br>SR 3.3.1.1.6<br>SR 3.3.1.1.9<br>SR 3.3.1.1.14 | ≤ 120/125 divisions of full scale |
| | 5(a) | 3 | H | SR 3.3.1.1.1<br>SR 3.3.1.1.4<br>SR 3.3.1.1.9<br>SR 3.3.1.1.14 | ≤ 120/125 divisions of full scale |
| b. Inop | 2 | 3 | G | SR 3.3.1.1.3<br>SR 3.3.1.1.14 | NA |
| | 5(a) | 3 | H | SR 3.3.1.1.4<br>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<br>SR 3.3.1.1.6<br>SR 3.3.1.1.7<br>SR 3.3.1.1.13<br>SR 3.3.1.1.16 | ≤ ~~15~~13% RTP |
| b. Flow Biased Simulated Thermal Power - High | 1 | 3(b) | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.7<br>SR 3.3.1.1.13<br>SR 3.3.1.1.16 | ~~≤ 0.66 W + 66% RTP and ≤ 120% RTP~~(c) |
| c. Neutron Flux - High | 1 | 3(b) | F | SR 3.3.1.1.1<br>SR 3.3.1.1.2<br>SR 3.3.1.1.7<br>SR 3.3.1.1.13<br>SR 3.3.1.1.16 | ≤ 120% RTP |

(continued)

≤ 0.55 W
+ 65.5% RTP
and ≤ 120% RTP

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

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

(c) [~~.66~~.55 W + ~~66~~65.5% - ~~.66~~.55 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."



Same as originally submitted

Table 3.3.1.1-1 (page 3 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 |
|---|---|---|---|---|---|
| 7. Scram Discharge Volume Water Level - High | | | | | |
| b. Float Switch | 1,2 | 2 | G | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 46 gallons |
| | 5(a) | 2 | H | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 46 gallons |
| 8. Turbine Stop Valve - Closure | ≥ ~~30~~26% RTP | 4 | E | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14<br>SR 3.3.1.1.15 | ≤ 10% closed |
| 9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low | ≥ ~~30~~26% RTP | 2 | E | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14<br>SR 3.3.1.1.15 | ≥ 550 psig |
| 10. Reactor Mode Switch - Shutdown Position | 1,2 | 1 | G | SR 3.3.1.1.12<br>SR 3.3.1.1.14 | NA |
| | 5(a) | 1 | H | SR 3.3.1.1.12<br>SR 3.3.1.1.14 | NA |
| 11. Manual Scram | 1,2 | 1 | G | SR 3.3.1.1.8<br>SR 3.3.1.1.14 | NA |
| | 5(a) | 1 | H | SR 3.3.1.1.8<br>SR 3.3.1.1.14 | NA |
| 12. RPS Channel Test Switches | 1,2 | 2 | G | SR 3.3.1.1.4 | NA |
| | 5(a) | 2 | H | SR 3.3.1.1.4 | NA |
| 13. Deleted | | | | | |

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

same as originally submitted

## 3.3 INSTRUMENTATION

### 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels of feedwater and main turbine high water level trip instrumentation per trip system shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ ~~25~~23% RTP.

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more feedwater and main turbine high water level trip channels inoperable, in one trip system. | A.1 Place channel(s) in trip. | 7 days |
| B. One or more feedwater and main turbine high water level trip channels inoperable in each trip system. | B.1 Restore feedwater and main turbine high water level trip capability. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

Same as originally submitted

## 3.3 INSTRUMENTATION

### 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~30~~ 26% RTP.

same as originally submitted

ACTIONS

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

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more channels inoperable. | A.1 | Restore channel to OPERABLE status. | 72 hours |
| | <u>OR</u> | | |
| | A.2 | ------------NOTE------------<br>Not applicable if inoperable channel is the result of an inoperable breaker.<br>---------------------------------<br><br>Place channel in trip. | 72 hours |
| B. One or more Functions with EOC-RPT trip capability not maintained.<br><br><u>AND</u><br><br>MCPR and LHGR limit for inoperable EOC-RPT not made applicable. | B.1 | Restore EOC-RPT trip capability. | 2 hours |
| | <u>OR</u> | | |
| | B.2 | Apply the MCPR and LHGR limit for inoperable EOC-RPT as specified in the COLR. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 | Reduce THERMAL POWER to < ~~30~~<u>26</u>% RTP. | 4 hours |

same as originally submitted

SURVEILLANCE REQUIREMENTS

--------------------NOTE--------------------
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 EOC-RPT trip capability.
--------------------------------------------

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.3.4.1.1 | Perform CHANNEL FUNCTIONAL TEST. | 92 days |
| SR 3.3.4.1.2 | Verify TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ ~~30~~26% RTP. | 24 months |
| SR 3.3.4.1.3 | Perform CHANNEL CALIBRATION. The Allowable Values shall be:<br><br>TSV - Closure: ≤ 10% closed; and<br><br>TCV Fast Closure, Trip Oil Pressure - Low: ≥ 550 psig. | 24 months |
| SR 3.3.4.1.4 | Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation. | 24 months |

Same as originally submitted

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| SR 3.4.2.1 | -------------------------NOTES---------------------<br>1. Not required to be performed until 4 hours after associated recirculation loop is in operation.<br>2. Not required to be performed until 24 hours after > ~~25~~23% RTP.<br>-------------------------------------------------<br>--------<br><br>Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop:<br><br>a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns.<br>b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.<br>c. Each jet pump flow differs by ≤ 10% from established patterns. | 24 hours |

same as
originally
submitted

## 3.7 PLANT SYSTEMS

### 3.7.1 Residual Heat Removal Service Water (RHRSW) System ~~and Ultimate Heat Sink (UHS)~~

LCO 3.7.1

-------------------------------------NOTE-------------------------------------
The number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for ≥ 24 hours.
--------------------------------------------------------------------------------

Four RHRSW subsystems ~~and UHS~~ shall be OPERABLE with the number of OPERABLE pumps as listed below:

1. 1 unit fueled - four OPERABLE RHRSW pumps.
2. 2 units fueled - six OPERABLE RHRSW pumps.
3. 3 units fueled - eight OPERABLE RHRSW pumps.

APPLICABILITY: MODES 1, 2, and 3.

same as originally submitted

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One required RHRSW pump inoperable. | A.1 | -------------NOTES-----------<br>1. Only applicable for the 2 units fueled condition.<br>2. Only four RHRSW pumps powered from a separate 4 kV shutdown board are required to be OPERABLE if the other fueled unit has been in MODE 4 or 5 for $\geq$ 24 hours.<br>----------------------------------<br><br>Verify five RHRSW pumps powered from separate 4 kV shutdown boards are OPERABLE. | Immediately |
| | <u>OR</u> | | |
| | A.2 | Restore required RHRSW pump to OPERABLE status. | 30 days |

(continued)

same as
originally
submitted

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| B. One RHRSW subsystem inoperable. | B.1 | -------------NOTE------------<br>Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling - Hot Shutdown," for RHR shutdown cooling made inoperable by the RHRSW system.<br>--------------------------------<br><br>Restore RHRSW subsystem to OPERABLE status. | <br><br><br><br><br><br><br><br>30 days |
| C. Two required RHRSW pumps inoperable. | C.1 | Restore one inoperable RHRSW pump to OPERABLE status. | 7 days |
| D. Two RHRSW subsystems inoperable.. | D.1 | -------------NOTE------------<br>Enter applicable Conditions and Required Actions of LCO 3.4.7, for RHR shutdown cooling made inoperable by the RHRSW System.<br>--------------------------------<br><br>Restore one RHRSW subsystem to OPERABLE status. | <br><br><br><br><br><br>7 days |

(continued)

same as originally submitted

ACTIONS (continued)

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| E. Three or more required RHRSW pumps inoperable. | E.1 | Restore one RHRSW pump to OPERABLE status. | 8 hours |
| F. Three or more RHRSW subsystems inoperable. | F.1 | ------------NOTE------------<br>Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by the RHRSW System.<br>--------------------------------- | |
| | | Restore one RHRSW subsystem to OPERABLE status. | 8 hours |
| G. Required Action and associated Completion Time not met.<br><br>~~OR~~<br><br>~~UHS inoperable~~ | G.1 | Be in MODE 3. | 12 hours |
| | AND | | |
| | G.2 | Be in MODE 4. | 36 hours |

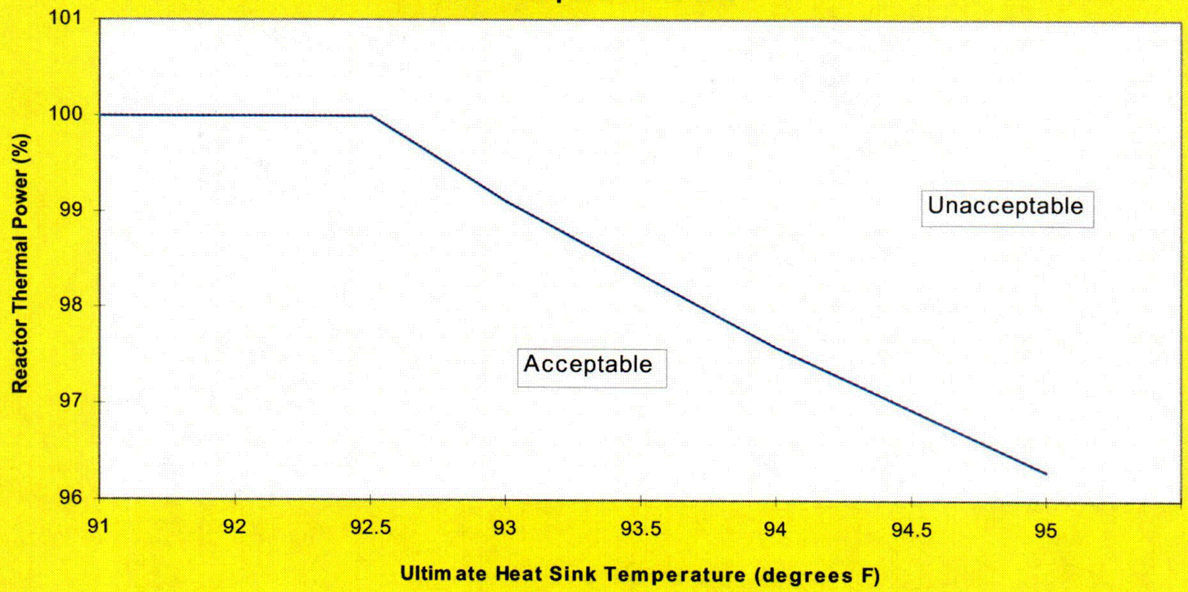
same as originally submitted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.1.1 | Verify each RHRSW manual and power operated valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position. | 31 days |
| ~~SR 3.7.1.2~~ | ~~Verify the average water temperature of UHS is ≤ 95°.~~ | ~~24 hours UHS temperature ≤ 95°F~~<br><br>~~AND~~<br><br>~~1 hour UHS temperature > 91°F~~ |

Same as originally submitted

**Figure 3.7.1-1**
**Reactor Thermal Power Versus Ultimate Heat Sink Temperature Limit**




Delete Figure

same as originally submitted

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.7.2.1 | ~~-----NOTE-----~~<br>~~Refer to SR 3.7.1.2 for additional UHS requirements.~~<br>~~----------~~<br><br>Verify the average water temperature of UHS is ≤ 95°F. | 24 hours |
| SR 3.7.2.2 | -----NOTE-----<br>Isolation of flow to individual components does not render EECW System inoperable.<br>----------<br><br>Verify each EECW system manual and power operated valve in the flow paths servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position. | 31 days |
| SR 3.7.2.3 | Verify each required EECW pump actuates on an actual or simulated initiation signal. | 24 months |

same as originally submitted

## 3.7 PLANT SYSTEMS

### 3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25~~ 23% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Requirements of the LCO not met. | A.1 | Satisfy the requirements of the LCO. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < ~~25~~23% RTP. | 4 hours |

same as originally submitted

## 5.5 Programs and Manuals

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5.5.12 <u>Primary Containment Leakage Rate Testing Program</u> (continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is ~~50.6~~ 48.5 psig. The maximum allowable primary containment leakage rate, $L_a$, shall be 2% of primary containment air weight per day at $P_a$.

Leakage Rate acceptance criteria are:

a. The primary containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following the testing performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A test; and

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) Air lock door seals leakage rate is $\leq 0.02\ L_a$ when the overall air lock is pressurized to $\geq 2.5$ psig for at least 15 minutes.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

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ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

COPY OF AREVA NP PROPREITARY NOTICE BELONGING TO ENCLOSURE 1 OF THE MAY 11, 2006, TVA SUBMITTAL

# AFFIDAVIT

COMMONWEALTH OF VIRGINIA )
) ss.
CITY OF LYNCHBURG )

1. My name is Gayle F. Elliott. I am Manager, Product Licensing in Regulatory Affairs, for AREVA NP, Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information provided to the NRC by TVA in a May 2006 letter responding to an NRC RAI regarding the Browns Ferry Nuclear Power Plant, Units 2 and 3 extended power uprate, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure.

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 4th

day of May, 2006.

Brenda C. Maddox

Brenda C. Maddox
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 7/31/07