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

October 5, 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 10 REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NOS. MC3812, MC3743, AND MC3744)

By letters dated June 28, 2004 and June 25, 2004 (ADAMS Accession Nos. ML041840109 and ML041840301, respectively), TVA submitted license applications to the NRC for the EPU of BFN Unit 1 and BFN Units 2 and 3, respectively. On September 27, 2006, the NRC staff issued the Round 10 RAI regarding the EPU license amendment requests. Enclosure 1 to this submittal provides TVA's responses to the Round 10 RAI questions.

In addition to the Round 10 RAI responses, included in Enclosure 1 is a supplement to TVA's response to question ACVB-62 that was provided in response to the Round 9 RAI by TVA letter of September 15, 2006.

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In a previous EPU submittal, by letter dated May 15, 2006 (ML061450390), TVA provided the Supplemental Reload Licensing Report (SRLR) for the Cycle 7 operation of BFN Unit 1. The analyses summarized in that SRLR are based on a Cycle 7 operating plan that assumed EPU operation (i.e., 120% of original licensed thermal power (OLTP)). However, the licensing of BFN Unit 1 for EPU operation is now planned to occur in two steps as outlined in TVA's letter of September 22, 2006 (ML062680459). Because the initial operation of Unit 1 Cycle 7 will be at 105% of OLTP, the core analyses are being reperformed consistent with an interim 105% OLTP level of 3458 MWt. To support operation of BFN Unit 1 at 105% OLTP, TVA and its fuel supplier are reevaluating the core design for Cycle 7 and performing revised analyses, which will result in a revised SRLR. TVA expects to provide the revised SRLR to the NRC by January 31, 2007.

Note that Enclosure 1 contains information that General Electric Company (GE) 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), such information should be withheld from public disclosure. Enclosure 2 is a redacted version of Enclosure 1 with the GE proprietary material removed and is suitable for public disclosure. Enclosures 1 and 2 contain an affidavit from GE supporting this request for withholding from public disclosure.

Enclosures 3 and 4 provide information requested in Round 10 RAI question APLA-27/29 for BFN Units 2 and 3, respectively.

To facilitate NRC's review of the proposed TS-418 TS changes, TVA has remarked in Enclosure 5, the current Unit 2 and Unit 3 TS pages to reflect those changes necessary for full EPU operation. These changes reflect recently issued TS amendments and a revision to a page (i.e., Unit 2 TS page 3.7-17) that was incorrectly marked in TVA's submittal of September 1, 2006 (ML062500197).

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).

Two new regulatory commitments are made in this submittal. Enclosure 6 describes these commitments. If you have any questions regarding this letter, please contact me at (256)729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 5th day of October 2006.
Sincerely,

William D. Crouch

William D. Crouch

Manager of Licensing
and Industry Affairs

Enclosures:

1. Response to Round 10 RAI Questions (Proprietary Information Version)
2. Response to Round 10 RAI Questions (Non-Proprietary Version)
3. Reply to RAI APLA-27/29 for Unit 2
4. Reply to RAI APLA-27/29 for Unit 3
5. EPU TS Changes Remarked Using Current TS Pages
6. Regulatory Commitments

Enclosures
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ENCLOSURE 2
TENNESSEE VALLEY AUTHORITY
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 10 REQUEST FOR ADDITIONAL INFORMATION (RAI)
(TAC NOS. MC3812, MC3743, AND MC3744)
RESPONSE TO ROUND 10 RAI QUESTIONS

(NON-PROPRIETARY VERSION)

This enclosure is a redacted version of the response to NRC's September 27, 2006, Round 10 RAI questions in Enclosure 1 with the proprietary material removed. This enclosure contains an affidavit from General Electric Company supporting the request for withholding the proprietary information contained in Enclosure 1 from public disclosure.

AFFIDAVIT

I, George B. Stramback, state as follows:

(1) I am Manager, Regulatory Services, General Electric Company ("GE") and 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-06-349 Larry King (GE) to J. Valente (TVA), *GE Responses to NRC Requests for Additional Information – SBWB – 50/75,* dated September 29, 2006. The proprietary information in Enclosure 1, *GE Responses to NRC RAI – SBWB-50/75*, 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, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, 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.

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 29th day of September 2006.

George B. Stramback
General Electric Company

NRC RAI EEMB-115/85

The Nuclear Regulatory Commission (NRC) staff requested a discussion of any weld reinforcement following fatigue cracking of drain channel in the BFN steam dryers. In its response to this request (identified as EEMB-C.1 on page E1-106 of the July 26 submittal), the Tennessee Valley Authority (TVA, the licensee) states it has periodically inspected the Units 2 and 3 repaired drain channel welds subsequent to 105 percent original licensed thermal power (OLTP) operation.

a. Identify the inspection technique used for Units 2 and 3 and explain whether that technique was qualified to detect fatigue cracks.

b. Specify whether these periodic inspections will be performed subsequent to 105 percent OLTP operation for Unit 1. If so, identify the inspection technique to be used and explain whether that technique is qualified to detect fatigue cracks.

TVA Response to EEMB-115/85

The inspection technique originally used to visually examine the Unit 2 and Unit 3 steam dryer drain channel welds was VT-3. When GE Service Information Letter (SIL) No. 474, "Steam Dryer Drain Channel Cracking," was issued, a VT-3 inspection was considered adequate to detect fatigue cracking of this component. Using this technique, fatigue cracks were identified and subsequently repaired on the steam dryers for Units 2 and 3. Limited VT-1 inspections were made on both Unit 2 and Unit 3 steam dryers during refueling outages that occurred in the Spring of 2005 and the Spring of 2004, respectively, and no fatigue cracking was identified.

Prior to EPU operation, the drain channels of all three BFN units are required to be inspected to meet VT-1 requirements per the guidelines of BWRVIP-139 and GE SIL No. 644, Revision 1. VT-1 is the current examination method recommended for this component and is considered adequate to detect fatigue cracking.

The examination method used for the Unit 1 steam dryer drain channel inspections was VT-1. Following implementation of EPU, periodic inspection of the BFN steam dryers will be conducted in accordance with the guidance of SIL No. 644, Revision 1, and BWRVIP-139, which include the use of the VT-1 inspection technique.

NRC RAI EEMB-116/86

The NRC staff requested a discussion of the post-modification inspection procedures for the BFN steam dryer modifications. In its response to this request (identified as EEMB-C.19 on pages E1-125 and 126 of the July 26 submittal), TVA stated that the post-modification inspection will be conducted employing visual inspection (VT-2). Discuss the adequacy of this inspection method, and the ability to conduct a more detailed inspection of the BFN Unit 1 steam dryer.

TVA Response to EEMB-116/86

The response to RAI question EEMB-C.19 contained a typographical error. The correct reference to the visual inspection required is VT-1. VT-1 is currently recommended by the GE SIL 644, Revision 1, and BWRVIP-139 for these dryer examinations of concern, and is required by the TVA BFN Reactor Pressure Vessel Internals Inspection procedure.

NRC RAI EEMB-117/87

Section 9.9 of Rev. 2 of the steam dryer stress report states that TVA plans to use pressure transducers mounted in holes in the main steam lines (MSLs) to measure fluctuating pressures as input to the acoustic circuit model (ACM). Provide a schematic of the proposed installation, which shows clearly the location of the pressure transducer with respect to the inner surface of the MSL walls. Since pressure transducers exposed to steam flow will measure acoustic pressure and turbulence traveling through the MSLs and over the pressure transducer, quantify any bias error or uncertainty that might be introduced to the dryer leads computed with the Bounding Pressure ACM by the presence of turbulence-induced pressures in the ACM inputs.

TVA Response to EEMB-117/87

As discussed with the NRC staff on September 28, 2006, TVA will install strain gages on the exterior of the main steam lines instead of the dynamic pressure transducers.

NRC RAI EEMB-118

In the July 26, 2006 response, the licensee indicated that Unit 1 is currently performing restart modifications and that the final stress analysis results, which reflect the as-built configuration, are not available for most of the reactor coolant pressure boundary and balance-of-plant systems. Provide the schedule for completion of the piping system evaluation for Unit 1. Upon completion, provide the evaluation summary for piping systems and their supports including main steam, feedwater, recirculation, residual heat removal, and torus-attached piping systems. The information should include the calculated maximum stresses and fatigue usage factors, as necessary, for piping systems and their supports similar to those provided for the Units 2 and 3 extended power uprate (EPU) evaluation.

TVA Response to EEMB-118

Evaluations of Unit 1 piping and supports associated with restart modifications are ongoing. As noted in TVA's letter of September 22, 2006, engineering activities in support of these modifications are scheduled for completion in December 2006. Upon completion next February, an evaluation summary for piping systems (including main steam, feedwater, recirculation, residual heat removal (RHR), and torus attached piping) will be provided to the NRC staff. The information provided will include the calculated maximum stresses for piping systems similar to the information provided for the EPU application of Units 2 and 3. The design basis code of record for BFN is the USAS B31.1.0-1967 code; consequently, fatigue usage factors have not been calculated for the balance-of-plant piping systems.

NRC RAI APLA-27/29

For this request, an operator action is "important to risk" if any one of the following criteria is met: (1) Fussell Vesely (FV) importance to core damage frequency (CDF) greater than 0.005; (2) FV importance to large early release frequency (LERF) greater than 0.005; (3) risk achievement worth (RAW) importance to CDF greater than two; or (4) RAW importance to LERF greater than two. Provide the following information for operator actions modeled in the probabilistic risk assessment that are important to risk:

a. Basic event (operator action) name
b. Description
c. Where action is performed (e.g., control room, outside control room, both)
d. For the pre-EPU model;
    i. FV importance to CDF
    ii. RAW importance to CDF
    iii. FV importance to LERF
    iv. RAW importance to LERF
    v. time available to the operator from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release
    vi. Human error probability

e. For the post-EPU model:
   i. FV importance to CDF
   ii. RAW importance to CDF
   iii. FV importance to LERF
   iv. RAW importance to LERF
   v. Time available to the operator from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release
   vi. Human error probability

TVA Response to APLA-27/29

The requested information for pre-EPU and post-EPU for Units 2 and 3 is provided in Enclosures 3 and 4, respectively. The EPU values are based on the current revisions of the probabilistic risk assessments (PRAs) for Units 2 and 3. The differences between the pre-EPU PRAs and the current PRAs involve several changes including EPU operation, Unit 1 operation, and model corrections. Currently, the Unit 1 PRA is being revised to resolve comments from a recent (late September) PRA Peer Review Certification effort. Since Unit 1 did not have a pre-EPU PRA and the Unit 1 PRA was built from the Unit 2 and 3 PRAs, the information regarding Unit 2 and Unit 3 is representative of all three BFN units.

NRC RAI ACVB-62 (BFN Units 1, 2, and 3)

The August 4, 2006 response to Request for Additional Information (RAI) Risk Assessment Containment & Ventilation Branch (ACVB) 37/35 states that, for the CS pump, the operator is instructed to maintain flow less than 4000 gallons per minute (gpm) and within the NPSH limit curves. However, for determining adequate NPSH, it is assumed that the operator would reduce flow in response to the NPSH limit curves, but not less than 3125 gpm.

It appears that at a flow rate of 4000 gpm and the peak calculated suppression pool temperature, the pumps are in the acceptable region of the Emergency Operating Instruction NPSH limit curves. Therefore, explain what prompts the operator to reduce flow to 3125 gpm. If the operator can operate acceptably at 4000 gpm, address why shouldn't this more conservative flow rate be used in the NPSH analyses.

TVA Response to ACVB-62 (BFN Units 1, 2, and 3)

The following information was provided in TVA's September 15, 2006, response to the EPU Round 9 RAI:

> In the long-term LOCA event, the operator will control ECCS pump flow in accordance with the EOIs and within the NPSH limit curves based on plant symptoms. The NPSH analysis for long-term LOCA shows that at the peak suppression pool temperature, margin is available between the required and available wetwell pressures. The margin indicates that adequate NPSH would be available at 4000 GPM and the operator would not and should not be prompted by the EOIs to reduce core spray flow.
>
> In contrast to the symptomatic EOIs, the safety analysis that forms the basis for COP credit is based on worst case conditions. The objective is to demonstrate that the ECCS pumps will be able to perform their safety function given bounding events and worst case assumptions. Consistent with this objective, the pump flow rates used are the minimum acceptable flows needed for the safety function plus a conservative margin. This flow is 3125 GPM in the case of core spray. By performing the analysis at this value, the safety function is assured and COP margin is established. At flows above this value, operator action will ensure adequate NPSH is maintained as dictated by the symptoms as they exist.

The following is provided as additional clarification and supplements the previous response.

The purpose of the design basis NPSH analysis is to demonstrate the adequacy of plant design. The core spray system NPSH calculation for the long-term DBA-LOCA case shows that adequate NPSH will be available such that the operator will not be directed to reduce core spray pump flow below the flow required by the ECCS-LOCA performance analysis (10 CFR 50.46 safety analyses). The core spray pump minimum flow rate used in the ECCS-LOCA performance analysis is 3125 gpm. Therefore, the value of 3125 gpm was used in the NPSH calculation. By performing the NPSH evaluations at 3125 gpm, it is ensured that adequate NPSH is provided at the minimum flow required by the safety analyses.

NRC RAI ACVB-68/66

The staff has determined that the information in the May 24, 1976, report may not be sufficient to justify credit for a value of required net positive suction head (NPSH) less than the 3% head loss value.

a. Provide any supporting information not included in the May 24, 1976, report which supports the use of a lower value such as:
   i. accelerometer data,
   ii. time that the Residual Heat Removal (RHR) pump was in cavitation, and
   iii. the inspections performed on the pump before and after testing.
c. Describe the operational history of RHR pump 3A. Address whether pump RHR 3A experienced any abnormal operation since this testing.

TVA Response to ACVB-68/66

Except for the case of RHR pumps injecting into a broken recirculation loop, sufficient NPSH has been demonstrated to be available for all licensing basis NPSH cases and meets vendor requirements. This case is different than the remainder of the NPSH analysis in that the concern is not for pump performance, but only for pump survivability. The most limiting NPSH condition identified for the RHR pumps exists during the 5 to 10 minute approximate timeframe immediately following a postulated DBA LOCA involving a break of a recirculation discharge pipe. This event forms the basis for the short term NPSH analysis.

The NPSH results are presented in Figure ACVB-56/54-1 from the August 4, 2006, submittal (ML062220647) and indicate that available suppression pool pressure is 0.85 psi less than that required to maintain the vendor recommended NPSH. This issue dates back to 1976 when the potential for runout flow on the broken loop RHR pumps was identified in TVA letters to the NRC dated May 21, 1976, and July 21, 1976. At that time a reduced NPSH flow test was performed with the 3A RHR pump in order to demonstrate conservatism in vendor NPSH requirements.

In the RHR pump broken loop analysis, the affected pumps are not performing any function during the time period that negative NPSH margin exists. The success criteria for this analysis is no pump damage occurs that results in failure of the RHR pumps, so they can be credited later in the event (>10 minutes) while operating in the containment cooling mode in the event of a single failure (such as loss of RHR service water) which could disable the heat removal function on the other loop of RHR. The 1976 test data is referenced for EPU in response to RAI question SPSB-A-11 in TVA letters of March 23, 2006 (ML060880460 and ML060880395 for BFN

Unit 1 and BFN Units 2 and 3, respectively), to demonstrate that vendor NPSH requirements used in NPSH calculations are conservative. This is particularly true when considering the success criteria of interest in this scenario, which is no damage resulting in RHR pump failure.

The test performed by TVA in 1976 demonstrates that the vendor's NPSH curves used to calculate NPSH margin are conservative relative to the negative NPSH margin shown in the analysis. This provides reasonable assurance that the pumps will not experience severe operational problems in the broken loop transient scenario. Vendor review of the 1976 data was inconclusive with regard to determining pump behavior at reduced suction pressure and high flows; the pump vendor's (Sulzer) report E12.5.1296 R0, "NPSH Transient Study," concludes that catastrophic damage to the pump should not occur for the broken loop transient. Therefore, even if the pumps do experience some cavitation for this short period of time (~ 5 minutes), it is not expected that the pumps would be severely damaged, and thus, the pumps are expected to function later in the event in the containment cooling mode.

In response to the specific RAI questions posed:

a. By letter dated July 21, 1976, TVA provided to NRC the results of the testing performed on the RHR pump for the reduced NPSH performance tests. Included in that submittal were the pump vibration data that was taken and the durations of the tests at reduced NPSH. In August/September 1994, the 3A RHR pump impeller was replaced to address generic wear ring cracking concerns. A review of documentation associated with this replacement did not indicate any abnormal impeller wear.

b. TVA has performed a search of completed surveillances and work orders associated with the 3A RHR pump during the two year time frame following the reduced NPSH testing conducted in 1976. No anomalies in surveillance testing or pump maintenance were identified.

<u>NRC RAI ACVB-69/67</u>

Describe the peak short-term loss of coolant accident (LOCA) suppression pool temperature at 105% power. Provide the service water temperature assumed in this analysis.

TVA Response to ACVB-69/67

The peak short-term (< 10 minutes) LOCA suppression pool temperature utilized for 105% power is 149.7°F. Service water temperature is not an input into the short-term (<10 minutes) analysis because the transition to RHR suppression pool cooling mode does not occur until 10 minutes into the event.

NRC RAI ACVB-70/68

Verify that at 105% power, for the short-term LOCA, the available NPSH is always greater than the required NPSH at the peak RHR pump flow (11,500 gpm) without reliance on the testing reported in the May 24, 1976 report.

TVA Response to ACVB-70/68

At 105% OLTP, the available suppression pool pressure for the RHR pumps that inject into the broken recirculation system discharge piping in the short term analysis would improve relative to 120% OLTP, but the lower power level alone would not result in a positive NPSH margin for the short-term case. This is determined by accounting for the difference in vapor pressure between peak suppression pool temperature short-term at 105% OLTP (see ACVB-69/67) and the same value at 120% OLTP (149.7°F versus 155.4°F). However, conservatism in the overall NPSH analysis is sufficient to offset the negative NPSH margin even at 120% OLTP and provides a high degree of confidence that the RHR pumps will be available after the initial broken loop operation. For example:

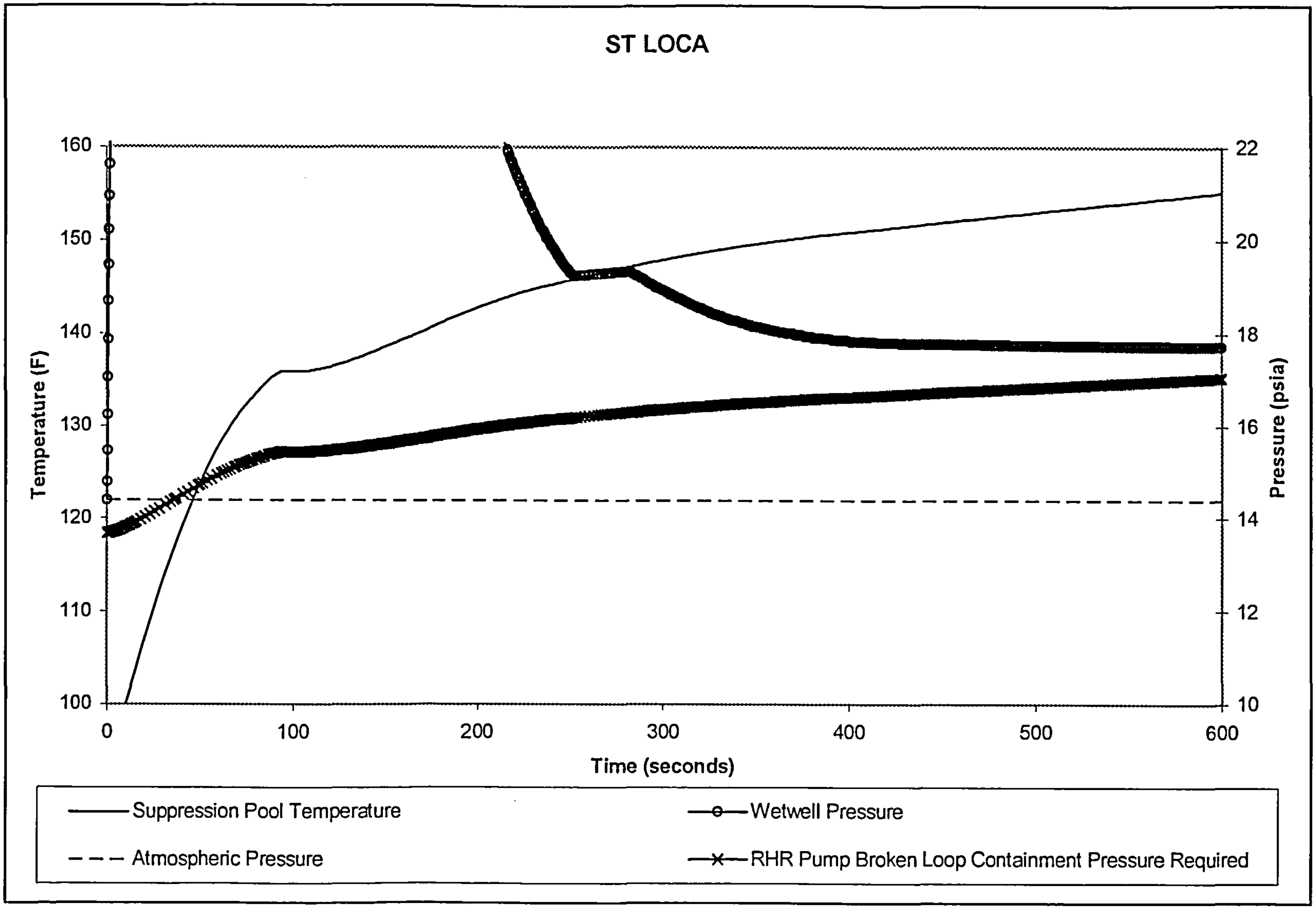
- 100% mixing of the broken loop flow with the drywell atmosphere is assumed, which results in under-prediction of drywell and suppression pool pressure during the short term analysis. This is conservative relative to the likely break geometries and limited distribution of cooler water throughout the drywell.

- Initial drywell relative humidity is assumed to be 100%, which limits the amount of non-condensable gas initially present in the drywell airspace. However, due to the design of the drywell and drywell cooling system, relative humidity is not expected to exceed 50%.

- NPSH required limits in the calculations are based on 11,500 gpm, whereas the calculated maximum flow for the broken loop of RHR is 11000 gpm. This results in a more restrictive NPSH required value.

An NPSH sensitivity analysis at 120% OLTP was performed assuming 50% relative humidity and 11000 gpm which shows that more realistic, but still conservative, values for these assumptions results in a positive NPSH margin for the broken loop case. Results are shown in Figure ACVB-70/68-1.

Figure ACVB-70/68-1
NPSH Requirements for DBA-LOCA - Short Term



NRC RAI SBWB-50/75

Provide the sequence of events tables for the limiting Appendix K Large Break LOCA and the limiting Appendix K Small Break LOCA (0.06 ft^2) discharge break with a battery failure and only 5 automatic depressurization system (ADS) valves actuated. The staff also requests the licensee to provide the low pressure coolant system and low pressure coolant injection head versus flow curve, limiting axial power shape, and ADS relief valve set pressure and relief capacity used in the analysis

TVA Response to SBWB-50 (BFN Unit 1)

The information requested regarding the low pressure core spray (LPCS) and low pressure coolant injection (LPCI) head versus flow curves was transmitted to NRC on September 1, 2006, in the response to SBWB-49 from the TVA EPU Round 9 RAI reply (ML062500197). These flow curves are independent of the break size, break location, and fuel type. The axial power shapes in the hot and average bundle for GE14 fuel with the plant operating at rated EPU power and flow were also provided in the referenced SBWB-49 response. These shapes are used for both the limiting large and small pipe breaks for GE14 fuel.

The sequence of events (SOEs) for the limiting Appendix K large break LOCA is shown in Table SBWB-50-1. The limiting large break LOCA is a DBA recirculation suction line break with a battery failure. The sequence of events for the limiting Appendix K small break LOCA (0.06 ft^2), which is a recirculation discharge line break with a battery failure and six automatic depressurization system (ADS) valves actuated, is shown in Table SBWB-50-2. The LOCA analyses with five ADS valves actuated were performed as a sensitivity study, however, these five ADS analyses are not licensing basis calculations.

The key parameters for the ADS are shown in Table SBWB-50-3. The key parameters for the relief valves are shown in Table SBWB-50-4.

Table SBWB-50-1
Sequence of Events for GE14 EPU Appendix K Limiting Large Break
(Recirculation Suction Line DBA Break with Battery Failure)

Event	Time (sec)
[[	
	]]

Table SBWB-50-2
Sequence of Events for GE14 EPU Appendix K Limiting Small Break
(Recirculation Discharge Line 0.06 ft^2 Break with Battery Failure - 6 ADS Valves)

Event	Time (sec)
[[	
	]]

Table SBWB-50-3
Key Automatic Depressurization System Parameters

Variable	Units	Analysis Value
Total number of valves available	-	6
Total number of valves assumed available in analysis	-	6
Minimum flow capacity per valve at vessel pressure	lbm/hr psig	800000 1125
Initiating signal to start ADS blowdown timer		
ECCS ready permissive (at least 1 LPCI or 2 core spray pumps are running)[1] and		
Low-low-low water level (L1)	in. AVZ[2]	372.5
and		
Low water level (L3)	in. AVZ[2]	518
and either		
High drywell pressure	psig	2.6
or		
High drywell pressure bypass timer elapsed[3]	sec	360
Automatic timer delay time from initiating signa completed to initiation of valve opening.	sec	120

[1] For small recirculation line breaks, the ECCS ready permissive occurs 21 seconds after L1 is reached. This time delay includes a 2 sec. signal processing delay.

[2] Above vessel zero.

[3] Bypass timer starts on low-low-low water level (L1) signal.

Table SBWB-50-4
Key Relief Valve Parameters

Valve Group	A	B	C
Number of valves in group	4	4	5
Opening setpoint, psig	1135.0	1145.0	1155.0
Closing setpoint, psig	1100.5	1110.2	1119.9
SRV capacity at 103% of 1090 psid, lbm/hr	870,000	870,000	870,000

<u>TVA Response to SBWB-75</u> (BFN Units 2 and 3)

The AREVA NP limiting licensing basis LOCA for ATRIUM-10 fuel is a 0.5 ft^2 recirculation line discharge split break with a battery failure (SF-BATT) at 102% EPU power and 105% rated core flow as specified in TVA's September 1, 2006 (ML062500197), reply to the EPU Round 9 RAI question SBWB-65. For this analysis, the available Emergency Core Cooling System is a single LPCS loop with 6 ADS valves operable. The licensing basis LOCA analysis was performed with 6 ADS valves and the limiting Peak Clad Temperature (PCT) case is based on this analysis. Therefore, no 5 ADS valve LOCA cases are being submitted. The injection head versus flow curves for LPCS and LPCI as well as the axial power shape for the limiting LOCA case were also provided in the TVA reply to NRC RAI SBWB-65. The ADS valve characteristics used in break spectrum analysis are attached in Table SBWB-75-1. The ADS Safety Relief Valve pressure setpoints are not used in the LOCA analyses since the ADS valves are actuated based by reactor water level signals after timer delays.

Table SBWB-75-2 provides the sequence of events (SOEs) for the limiting large break LOCA, which is a 1.0 discharge coefficient recirculation line suction guillotine (DEG) break with battery failure. The SOE for a small break LOCA (0.05 ft^2 recirculation line discharge split break with battery failure) is shown in Table SBWB-75-3. This is the closest break size to the requested 0.06 ft^2 break that was analyzed in the AREVA NP break spectrum. The SOE for the limiting PCT LOCA analysis (0.5 ft^2 recirculation line discharge split break with battery failure) is provided in Table SBWB-75-4.

Table SBWB-75-1
Automatic Depressurization System (ADS) Parameters

Parameter	Value
Number of valves installed	6
Number of valves available	6
Minimum flow capacity of available valves	4.8 Mlbm/hr at 1125 psig
ADS Initiating Signals and Setpoints	
Water level[1]	L1 (372.5 in)
LPCS ready permissive[2]	L1 + 40 sec (max)
ADS Time Delays	
Delay time (from ADS timer permissive to time valves are open)	120 sec

[1] Relative to vessel zero.

[2] ADS timer initiation occurs after level trip L1 is met and LPCS pumps reach the ADS ready permissive. Credit is conservatively not taken for the RHR pump ready permissive that would occur 8 seconds earlier.

Table SBWB-75-2
Event Times for Large Break LOCA 1.0 DEG Recirculation Line Suction SF-BATT Mid-Peaked Axial 102% EPU 105% Flow

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.5
Low-low liquid level, L2 (448 in)	5.7
Low-low-low liquid level, L1 (372.5 in)	7.3
Jet pump uncovers	8.4
Recirculation suction uncovers	11.3
Lower plenum flashes	13.9
LPCS valve pressure permissive	38.0
LPCI valve pressure permissive	38.0
LPCI high-pressure cut off	39.0
LPCS valve starts to open	40.0
LPCS permissive for ADS timer	40.0
LPCI valve starts to open	40.0
LPCS high-pressure cutoff	41.2
LPCS pump at rated speed	43.0
LPCS flow starts	43.0
LPCI pump at rated speed	44.0
LPCI flow starts	44.0
RDIV pressure permissive	49.4
RDIV starts to close	51.4
Rated LPCS flow	70.5
Blowdown ends	70.5
LPCS valve fully open	73.0
LPCI valve fully open	80.0
RDIV fully closed	87.4
Bypass reflood	94.5
Core reflood	100.8
PCT	100.8
ADS valves open	160.0

Table SBWB-75-3
Event Times for Small Break LOCA 0.05 ft^2 Split Recirculation Line Discharge SF-BATT Mid-Peaked Axial 102% EPU 105% Flow

Event	Time(sec)
Initiate break	0.0
Initiate scram	0.5
Low-low liquid level, L2 (448 in)	132.9
Low-low-low liquid level, L1 (372.5 in)	215.1
LPCS permissive for ADS timer	244.1
LPCS pump at rated speed	247.1
Jet pump uncovers	329.2
ADS valves open	364.1
Lower plenum flashes	366.9
LPCS valve pressure permissive	517.2
LPCS valve starts to open	519.2
LPCS high-pressure cutoff	536.7
LPCS flow starts	536.7
Recirculation suction uncovers	541.0
LPCS valve fully open	552.2
RDIV pressure permissive	594.3
RDIV starts to close	596.3
PCT	610.0
RDIV fully closed	632.3
Bypass reflood	674.5
Rated LPCS flow	695.2
Blowdown ends	695.2
Core reflood	695.2

Table SBWB-75-4
Event Times for Limiting Break LOCA 0.5 ft^2 Split Recirculation Line Discharge SF-BATT Mid-Peaked Axial 102% EPU 105% Flow

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.5
Low-low liquid level, L2 (448 in)	16.4
Low-low-low liquid level, L1 (372.5 in)	26.7
Jet pump uncovers	35.5
LPCS permissive for ADS timer	55.7
Recirculation suction uncovers	57.2
LPCS pump at rated speed	58.7
Lower plenum flashes	71.3
ADS valves open	175.7
LPCS valve pressure permissive	193.1
LPCS valve starts to open	195.1
LPCS high-pressure cutoff	201.4
LPCS flow starts	201.4
RDIV pressure permissive	222.0
RDIV starts to close	224.0
LPCS valve fully open	228.1
RDIV fully closed	260.0
Rated LPCS flow	277.2
Blowdown ends	277.2
Core reflood	358.3
PCT	358.3
Bypass reflood	421.4

ENCLOSURE 3
TENNESSEE VALLEY AUTHORITY
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 10 REQUEST FOR ADDITIONAL INFORMATION (RAI)
(TAC NOS. MC3812, MC3743, AND MC3744)

TVA REPLY TO RAI APLA-27/29 FOR UNIT 2

This enclosure provides TVA's Reply to APLA-27/29 for Unit 2.

The following footnote applies to the attached tables:

(1) The Riskman quantification method produced a number of FV negative values and RAW values less than 1.0. The underlying cause for these valves can be found in the frequency truncation of scenarios. These values are presented in the attached tables as 0 (FV) and 1 (RAW).

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OAD1 Basic Event: HOAD1	Inhibit ADS actuation, given ATWS with an unisolated RPV	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	3.33E-03	1.05E-02
RAW, CDF	3.27E+00	3.05E+00
FV, LERF	1.61E-02	4.31E-02
RAW, LERF	1.19E+01	9.43E+00
Human Error Prob. (Monte Carlo Mean)	1.47-03	5.09E-03
Time Available*	Time to -122" dependent on suppression pool heatup, but approx. 10 minutes. Four min. provided by timer after reaching -122" for 14 min.	Time to -122" dependent on suppression pool heatup, but approx. 8.5 minutes. Four min. provided by timer after reaching -122" reducing time available to 12.5 min
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OAD2 Basic Event: HOAD2	Inhibit ADS actuation, given ATWS with an isolated RPV	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	1.96E-04	1.44E-03
RAW, CDF	1.13E+00	1.15E+00
FV, LERF	9.66E-04	5.96E-03
RAW, LERF	1.65E+00	1.62E+00
Human Error Prob. (Monte Carlo Mean)	1.48E-03	9.49E-03
Time Available*	Level drops to -122" within 2 min. without injection, Cont. Press. > 2.45 psig when RPV is isolated. Must inhibit prior to 95 sec-timeout.	Level drops to -122" within 105 seconds without injection. No change to timeout length following -122", so time available reduces by 15 seconds.
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

<table>
<tr><th colspan="3">Unit 2 Human Failure Events</th></tr>
<tr><th>Operator Action Name</th><th colspan="2">Description</th></tr>
<tr><td>Split Fraction: OF1
Basic Event: HOF1</td><td colspan="2">Control one Feedwater Pump and hotwell level, given auto control was successful</td></tr>
<tr><td></td><td>Pre-EPU (Model C2051602)</td><td>Post-EPU (Model U2060706)</td></tr>
<tr><td>FV, CDF</td><td>1.62E-03</td><td>4.16E-03</td></tr>
<tr><td>RAW, CDF</td><td>5.37E+00</td><td>2.62E+00</td></tr>
<tr><td>FV, LERF</td><td>3.00E-04</td><td>8.00E-04</td></tr>
<tr><td>RAW, LERF</td><td>1.81E+00</td><td>1.31E+00</td></tr>
<tr><td>Human Error Prob. (Monte Carlo Mean)</td><td>3.70E-04</td><td>2.56E-03</td></tr>
<tr><td>Time Available*</td><td>Monitor during cooldown (up to 24 hours). Respond to alarm within 5 min to avoid automatic trip</td><td>Action is required after water has been injected in the RPV</td></tr>
<tr><td colspan="3">Where is this operator action performed?
Control Room</td></tr>
<tr><td colspan="3">*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.</td></tr>
</table>

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHC1 Basic Event: HOHC1	Control RPV level and pressure with HPCI and/or RCIC during first 6 hours	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	0.00E+00(1)	1.09E-01
RAW, CDF	1.00E+00(1)	1.10E+02
FV, LERF	6.71E-05	2.13E-02
RAW, LERF	1.06E+00	2.25E+01
Human Error Prob. (Monte Carlo Mean)	1.06E-03	9.92E-04
Time Available*	Continuous requirement - react within 5 min of high level alarm to prevent automatic HPCI trip at +55"	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHC2 Basic Event: HOHC2	Control RPV level and press with HPCI during first 6 hours, given RCIC failed or insufficient	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	0.00E+00(1)	4.49E-03
RAW, CDF	1.00E+00(1)	6.21E+00
FV, LERF	0.00E+00(1)	6.63E-04
RAW, LERF	1.00E+00(1)	1.77E+00
Human Error Prob. (Monte Carlo Mean)	9.18E-04	8.62E-04
Time Available*	Continuous requirement; react within 5 min of high level alarm to prevent automatic HPCI trip at +55"	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHC3 Basic Event: HOHC3	Control RPV level and press during first 6 hours, given HPCI failed.	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	0.00E+00(1)	5.67E-03
RAW, CDF	1.00E+00(1)	8.56E+00
FV, LERF	0.00E+00(1)	8.99E-04
RAW, LERF	1.00E+00(1)	2.20E+00
Human Error Prob. (Monte Carlo Mean)	7.36E-04	7.49E-04
Time Available*	Continuous requirement; after recovery of RPV level react within 5 min after alarm to prevent automatic HPCI trip at +55"	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHL2 Basic Event: HOHL2	Recover and control RPV level and pressure with HPCI and/or RCIC up to 24 hours, given short term control failed	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	0.00E+00[(1)]	1.18E-01
RAW, CDF	1.00E+00	2.71E+01
FV, LERF	0.00E+00[(1)]	2.29E-02
RAW, LERF	1.00E+00	6.05E+00
Human Error Prob. (Monte Carlo Mean)	4.49E-03	4.51E-03
Time Available*	Continuous requirement. React to alarm within 15 min of indication to prevent automatic trip at +55"	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OLA1 Basic Event: HOLA1	Control LPCI to maintain RPV level at TAF, given ATWS	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	8.23E-03	6.41E-03
RAW, CDF	1.10E+00	1.08E+00
FV, LERF	3.95E-02	2.63E-02
RAW, LERF	1.47E+00	1.31E+00
Human Error Prob. (Monte Carlo Mean)	7.75E-02	7.84E-02
Time Available*	Continuous requirement for close control until sub-criticality and refill	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OLP1 Basic Event: HOLP1	Control RPV level using LPCI mode of RHR or the Core Spray System	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	7.01E-03	2.06E-03
RAW, CDF	3.64E+02	1.10E+02
FV, LERF	1.92E-02	2.35E-03
RAW, LERF	9.96E+02	1.26E+02
Human Error Prob. (Monte Carlo Mean)	1.93E-05	1.89E-05
Time Available*	Initiate after cooldown. Over 2 hours to core uncovery from normal RPV level with no injection	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OLP2 Basic Event: HOLP2	Open the hardened wetwell vent, partial AC power available	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	1.71E-01	1.32E-01
RAW, CDF	2.03E+00	1.80E+00
FV, LERF	0.00E+00[1]	2.01E-02
RAW, LERF	1.00E+00[1]	1.12E+00
Human Error Prob. (Monte Carlo Mean)	1.43E-01	1.42E-01
Time Available*	Hours for suppression pool heat up	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OP1 Basic Event: None (Screening Value)	Operator depressurizes RPV (Level 2)	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	0.00E+00(1)	0.00E+00(1)
RAW, CDF	1.00E+00(1)	1.00E+00(1)
FV, LERF	1.99E-02	8.11E-03
RAW, LERF	1.02E+00	1.01E+00
Human Error Prob. (Monte Carlo Mean)	4.55E-01	4.55E-01
Time Available*	Not time sensitive	Not time sensitive
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OP3 Basic Event: None (Screening Value)	Operator depressurizes RPV (Level 2)	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	0.00E+00(1)	0.00E+00(1)
RAW, CDF	1.00E+00(1)	1.00E+00(1)
FV, LERF	2.07E-02	8.04E-02
RAW, LERF	1.33E+00	2.28E+00
Human Error Prob. (Monte Carlo Mean)	5.90E-02	5.90E-02
Time Available*	Not time sensitive	Not time sensitive
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

<table>
<tr><td colspan="3">Unit 2 Human Failure Events</td></tr>
<tr><td>Operator Action Name</td><td colspan="2">Description</td></tr>
<tr><td>Split Fraction: ORVD1
Basic Event: HORVD1</td><td colspan="2">Emergency depressurize given failure of HPCI and RCIC</td></tr>
<tr><td></td><td>Pre-EPU (Model C2051602)
(note the corresponding split fraction in the pre EPU model is ORVD2)</td><td>Post-EPU (Model U2060706)</td></tr>
<tr><td>FV, CDF</td><td>1.02E-01</td><td>3.11E-01</td></tr>
<tr><td>RAW, CDF</td><td>2.21E+02</td><td>1.60E+03</td></tr>
<tr><td>FV, LERF</td><td>2.20E-02</td><td>6.37E-02</td></tr>
<tr><td>RAW, LERF</td><td>4.82E+01</td><td>3.29E+02</td></tr>
<tr><td>Human Error Prob. (Monte Carlo Mean)</td><td>4.66E-4</td><td>1.95E-04</td></tr>
<tr><td>Time Available*</td><td>30 minutes to recognize need to emergency depressurize. 3 to 5 minutes to -190" once -162" (= TAF) reached</td><td>Ten min available based on MAAP CASE01, loss of all injection into vessel</td></tr>
<tr><td colspan="3">Where is this operator action performed?
Control Room</td></tr>
<tr><td colspan="3">*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.</td></tr>
</table>

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: ORVD2 Basic Event: HORVD2	Emergency depressurize by manually opening MSRVs, given manual control of HPCI and RCIC failed	
	Pre-EPU (Model C2051602) (this is ORVD3 in the pre EPU model)	Post-EPU (Model U2060706)
FV, CDF	0.00E+00(1)	1.18E-01
RAW, CDF	1.00E+00	1.73E+00
FV, LERF	0.00E+00(1)	2.29E-02
RAW, LERF	1.00E+00	1.14E+00
Human Error Prob. (Monte Carlo Mean)	6.32E-03	1.40E-01
Time Available*	30 minutes to recognize need to emergency depressurize. 3 to 5 minutes to -190" once -162" (= TAF) reached	Ten min available based on MAAP CASE01, loss of all injection into vessel
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSL1 Basic Event: HOSL1	Activate SLC unisolated RPV	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	1.20E-02	3.79E-02
RAW, CDF	3.72E+00	3.32E+00
FV, LERF	5.77E-02	1.55E-01
RAW, LERF	1.40E+01	1.05E+01
Human Error Prob. (Monte Carlo Mean)	4.40E-03	1.61E-02
Time Available*	3 to 5 min available to avoid level/ power control requirement. (HCR used 240 sec.)	Lowered T_w in HCR model from 180 to 158 seconds t to reflect 7/8 available time due to assumed 120%/105% power ratio
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSL2 Basic Event: HOSL2	Activate SLC isolated RPV	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	5.83E-03	1.99E-02
RAW, CDF	1.28E+00	1.24E+00
FV, LERF	2.80E-02	8.14E-02
RAW, LERF	2.36E+00	2.00E+00
Human Error Prob. (Monte Carlo Mean)	2.02E-02	7.54E-02
Time Available*	at 50% power SP reaches 110 F in about 2 min and 170 F in about 7 min. Used 180 second as time available	EPU lowered time available to 158 seconds
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSP1 Basic Event: HOSP1	Align RHR for suppression pool (SP) cooling	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	1.18E-01	6.94E-02
RAW, CDF	1.56E+03	9.48E+02
FV, LERF	3.44E-02	3.69E-02
RAW, LERF	4.55E+02	5.04E+02
Human Error Prob. (Monte Carlo Mean)	7.57E-05	7.33E-05
Time Available*	Not time sensitive - about 90 min before SP temperature exceeds 140 F	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSP2 Basic Event: HOSP2	Align RHR for SP cooling, given ATWS	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	3.57E-03	2.62E-03
RAW, CDF	1.58E+00	1.43E+00
FV, LERF	1.74E-02	1.10E-02
RAW, LERF	3.82E+00	2.81E+00
Human Error Prob. (Monte Carlo Mean)	6.15E-03	6.01E-03
Time Available*	Approximately 9 min until HCTL if unit at 50% power	Time available due to integrated ATWS heat generation for baseline was very conservative and judged to not be worsened by EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSP3 Basic Event: HOSP3	Align RHR for SP cooling, given one path unavailable	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	6.48E-04	4.78E-04
RAW, CDF	1.05E+01	8.09E+00
FV, LERF	0.00E+00(1)	1.06E-04
RAW, LERF	1.00E+00(1)	2.57E+00
Human Error Prob. (Monte Carlo Mean)	6.82E-05	6.74E-05
Time Available*	Not time sensitive - much more than 1 hour before SP temperature exceeds 140 F	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSV1 Basic Event: HOSV1	Defeat MSIV closure logic, given ATWS with turbine trip	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	1.89E-02	1.28E-02
RAW, CDF	1.19E+00	1.13E+00
FV, LERF	9.11E-02	5.28E-02
RAW, LERF	1.91E+00	1.55E+00
Human Error Prob. (Monte Carlo Mean)	9.11E-02	8.70E-02
Time Available*	Accomplish in first 10 min of transient, after reaching BIIT; circa 7 minutes before SP reaches 110°F, forcing lowering of level	Time available due to integrated ATWS heat generation for baseline was very conservative and judged to not be worsened by EPU
Where is this operator action performed? Back panels within the Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 2 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSW1 Basic Event: HOSW1	Transfer Mode Switch to REFUEL/ SHUT DOWN in response to scram	
	Pre-EPU (Model C2051602)	Post-EPU (Model U2060706)
FV, CDF	2.40E-03	1.23E-03
RAW, CDF	4.22E+00	2.68E+00
FV, LERF	7.25E-03	0.00E+00(1)
RAW, LERF	1.07E+01	1.00E+00(1)
Human Error Prob. (Monte Carlo Mean)	7.44E-04	7.31E-04
Time Available*	Not time significant for typical pressure reduction rates	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

ENCLOSURE 4
TENNESSEE VALLEY AUTHORITY
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 10 REQUEST FOR ADDITIONAL INFORMATION (RAI)
(TAC NOS. MC3812, MC3743, AND MC3744)

TVA REPLY TO RAI APLA-27/29 FOR UNIT 3

This enclosure provides TVA's Reply to APLA-27/29 for Unit 3.

The following footnote applies to the attached tables:

[1] The Riskman quantification method produced a number of FV negative values and RAW values less than 1.0. The underlying cause for these valves can be found in the frequency truncation of scenarios. These values are presented in the attached tables as 0 (FV) and 1 (RAW).

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OAD1 Basic Event: HOAD1	Inhibit ADS actuation, given ATWS with an unisolated RPV	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	2.19E-03	5.74E-03
RAW, CDF	2.49E+00	2.15E+00
FV, LERF	1.47E-02	3.87E-02
RAW, LERF	1.10E+01	8.78E+00
Human Error Prob. (Monte Carlo Mean)	1.47E-03	4.95E-03
Time Available*	Time to -122" dependent on suppression pool heatup, but approx. 10 minutes. Four min. provided by timer after reaching -122" for 14 min.	Time to -122" dependent on suppression pool heatup, but approx. 8.5 minutes. Four min. provided by timer after reaching -122" reducing time available to 12.5 min.
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OAD2 Basic Event: HOAD2	Inhibit ADS actuation, given ATWS with an isolated RPV	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	1.28E-04	7.66E-04
RAW, CDF	1.09E+00	1.08E+00
FV, LERF	8.79E-04	5.22E-03
RAW, LERF	1.59E+00	1.56E+00
Human Error Prob. (Monte Carlo Mean)	1.48E-03	9.19E-03
Time Available*	Level drops to -122" within 2 min. without injection, Cont. Press. > 2.45 psig when RPV is isolated. Must inhibit prior to 95 sec-timeout.	Level drops to -122" within 105 seconds without injection. No change to timeout length following -122", so time available reduces by 15 seconds.
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OF1 Basic Event: HOF1	Control one Feedwater Pump and hotwell level, given auto control was successful	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	1.05E-03	2.44E-03
RAW, CDF	3.83E+00	1.88E+00
FV, LERF	2.75E-04	7.66E-04
RAW, LERF	1.74E+00	1.28E+00
Human Error Prob. (Monte Carlo Mean)	3.70E-04	2.75 E-03
Time Available*	Monitor during cooldown (up to 24 hours). Respond to alarm within 5 min to avoid automatic trip	Action is required after water has been injected in the RPV
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHC1 Basic Event: HOHC1	Control RPV level and pressure with HPCI and/or RCIC during first 6 hours	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	0.00E+00(1)	6.17E-02
RAW, CDF	1.00E+00(1)	6.23E+01
FV, LERF	1.17E-03	2.17E-02
RAW, LERF	2.10E+00	2.26E+01
Human Error Prob. (Monte Carlo Mean)	1.06E-03	1.01E-03
Time Available*	Continuous requirement - react within 5 min of high level alarm to prevent automatic HPCI trip at +55"	*Same as pre-EPU*
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHC2 Basic Event: HOHC2	Control RPV level and press with HPCI during first 6 hours, given RCIC failed or insufficient	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	0.00E+00(1)	2.69E-03
RAW, CDF	1.00E+00(1)	3.83E+00
FV, LERF	1.01E-05	8.13E-04
RAW, LERF	1.01E+00	1.86E+00
Human Error Prob. (Monte Carlo Mean)	9.18E-04	9.50E-04
Time Available*	Continuous requirement; react within 5 min of high level alarm to prevent automatic HPCI trip at +55"	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHC3 Basic Event: HOHC3	Control RPV level and press during first 6 hours, given HPCI failed	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	0.00E+00(1)	3.20E-03
RAW, CDF	1.00E+00(1)	5.15E+00
FV, LERF	0.00E+00(1)	9.42E-04
RAW, LERF	1.00E+00(1)	2.22E+00
Human Error Prob. (Monte Carlo Mean)	7.36E-04	7.69E-04
Time Available*	Continuous requirement; after recovery of RPV level react within 5 min after alarm to prevent automatic HPCI trip at +55"	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OHL2 Basic Event: HOHL2	Recover and control RPV level and pressure with HPCI and/or RCIC up to 24 hours, given short term control failed.	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	0.00E+00(1)	6.76E-02
RAW, CDF	1.00E+00	1.62E+01
FV, LERF	0.00E+00(1)	2.28E-02
RAW, LERF	1.00E+00	6.13E+00
Human Error Prob. (Monte Carlo Mean)	4.49E-03	4.42E-03
Time Available*	Continuous requirement. React to alarm within 15 min of indication to prevent automatic trip at +55"	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OLA1 Basic Event: HOLA1	Control LPCI to maintain RPV level at TAF, given ATWS	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	5.40E-03	3.66E-03
RAW, CDF	1.06E+00	1.04E+00
FV, LERF	3.60E-02	2.46E-02
RAW, LERF	1.43E+00	1.29E+00
Human Error Prob. (Monte Carlo Mean)	7.75E-02	7.89E-02
Time Available*	Continuous requirement for close control until sub-criticality and refill	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OLP1 Basic Event: HOLP1	Control RPV level using LPCI mode of RHR or the Core Spray System	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	7.80E-03	2.25E-03
RAW, CDF	4.05E+02	1.25E+02
FV, LERF	2.19E-02	3.39E-03
RAW, LERF	1.14E+03	1.88E+02
Human Error Prob. (Monte Carlo Mean)	1.93E-05	1.81E-05
Time Available*	Initiate after cooldown. Over 2 hours to core uncovery from normal RPV level with no injection	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OLP2 Basic Event: HOLP2	Open the hardened wetwell vent, partial AC power available	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	1.52E-01	9.16E-02
RAW, CDF	1.91E+00	1.55E+00
FV, LERF	7.50E-03	2.06E-02
RAW, LERF	1.05E+00	1.12E+00
Human Error Prob. (Monte Carlo Mean)	1.43E-01	1.44E-01
Time Available*	Hours for suppression pool heat up	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OP1 Basic Event: None (Screening Value)	Operator depressurizes RPV (Level 2)	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	0.00E+00(1)	0.00E+00(1)
RAW, CDF	1.00E+00(1)	1.00E+00
FV, LERF	5.42E-02	6.33E-03
RAW, LERF	1.06E+00	1.01E+00
Human Error Prob. (Monte Carlo Mean)	4.55E-01	4.55E-01
Time Available*	Not time sensitive	Not time sensitive
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OP3 Basic Event: None (Screening Value)	Operator depressurizes RPV (Level 2)	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	0.00E+00(1)	0.00E+00(1)
RAW, CDF	1.00E+00(1)	1.00E+00(1)
FV, LERF	1.86E-02	7.48E-02
RAW, LERF	1.30E+00	2.19E+00
Human Error Prob. (Monte Carlo Mean)	5.90E-02	5.90E-02
Time Available*	Not time sensitive	Not time sensitive
Where is this operator action performed?		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: ORVD1 Basic Event: HORVD1	Emergency depressurize by manually opening MSRVs, given HPCI/RCIC hardware failed	
	Pre-EPU (Model C3051602) (this is ORVD2 in the pre EPU model)	Post-EPU (Model U3060706)
FV, CDF	6.71E-02	1.64E-01
RAW, CDF	1.45E+02	8.99E+02
FV, LERF	1.96E-02	5.73E-02
RAW, LERF	4.30E+01	3.15E+02
Human Error Prob. (Monte Carlo Mean)	4.66E-04	1.82E-04
Time Available*	30 minutes to recognize need to emergency depressurize. 3 to 5 minutes to -190" once -162" (= TAF) reached	Ten min available based on MAAP CASE01, loss of all injection into vessel
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: ORVD2 Basic Event: HORDV2	Emergency depressurize by manually opening MSRVs, given HPCI/RCIC failed due to operator control error	
	Pre-EPU (Model C3051602) (this is ORVD3 in the pre EPU model)	Post-EPU (Model U3060706)
FV, CDF	0.00E+00[(1)]	6.76E-02
RAW, CDF	1.00E+00	1.41E+00
FV, LERF	0.00E+00[(1)]	2.28E-02
RAW, LERF	1.00E+00[(1)]	1.14E+00
Human Error Prob. (Monte Carlo Mean)	6.32E-03	1.40E-01
Time Available*	30 minutes to recognize need to emergency depressurize. 3 to 5 minutes to -190" once -162" (= TAF) reached	Ten min available based on MAAP CASE01, loss of all injection into vessel
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSL1 Basic Event: HOSL1	Activate SLC unisolated RPV	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	7.91E-03	2.14E-02
RAW, CDF	2.79E+00	2.31E+00
FV, LERF	5.27E-02	1.43E-01
RAW, LERF	1.29E+01	9.75E+00
Human Error Prob. (Monte Carlo Mean)	4.40E-03	1.61E-02
Time Available*	3 to 5 min available to avoid level/ power control requirement. (HCR used 240 sec.)	Lowered T_w in HCR model from 180 to 158 seconds t to reflect 7/8 available time due to assumed 120%/105% power ratio
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSL2 Basic Event: HOSL2	Activate SLC isolated RPV	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	3.81E-03	1.16E-02
RAW, CDF	1.19E+00	1.14E+00
FV, LERF	2.54E-02	7.82E-02
RAW, LERF	2.23E+00	1.92E+00
Human Error Prob. (Monte Carlo Mean)	2.02E-02	7.84E-02
Time Available*	At 50% power SP reaches 110 F in about 2 min and 170 F in about 7 min. Used 180 second as time available	EPU lowered time available to 158 seconds
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSP1 Basic Event: HOSP1	Align RHR for suppression pool (SP) cooling	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	9.64E-02	5.15E-02
RAW, CDF	1.27E+03	6.78E+02
FV, LERF	4.05E-02	4.32E-02
RAW, LERF	5.35E+02	5.69E+02
Human Error Prob. (Monte Carlo Mean)	7.57E-05	7.60E-05
Time Available*	Not time sensitive - about 90 min before SP temperature exceeds 140 F	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSP2 Basic Event: HOSP2	Align RHR for SP cooling, given ATWS	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	2.29E-03	1.50E-03
RAW, CDF	1.37E+00	1.25E+00
FV, LERF	1.55E-02	1.03E-02
RAW, LERF	3.51E+00	2.70E+00
Human Error Prob. (Monte Carlo Mean)	6.15E-03	6.03E-03
Time Available*	Approximately 9 min until HCTL if unit at 50% power	Time available due to integrated ATWS heat generation for baseline was very conservative and judged to not be worsened by EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSP3 Basic Event: HOSP3	Align RHR for SP cooling, given one path unavailable	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	5.52E-04	2.35E-04
RAW, CDF	9.10E+00	4.47E+00
FV, LERF	0.00E+00(1)	1.40E-05
RAW, LERF	1.00E+00(1)	1.21E+00
Human Error Prob. (Monte Carlo Mean)	6.82E-05	6.79E-05
Time Available*	Not time sensitive - much more than 1 hour before SP temperature exceeds 140 F	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSV1 Basic Event: HOSV1	Defeat MSIV closure logic, given ATWS with turbine trip	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	1.26E-02	7.89E-03
RAW, CDF	1.13E+00	1.08E+00
FV, LERF	8.45E-02	5.35E-02
RAW, LERF	1.84E+00	1.55E+00
Human Error Prob. (Monte Carlo Mean)	9.11E-02	8.92E-02
Time Available*	Accomplish in first 10 min of transient, after reaching BIIT; circa 7 minutes before SP reaches 110°F, forcing lowering of level	Time available due to integrated ATWS heat generation for baseline was very conservative and judged to not be worsened by EPU
Where is this operator action performed? Back panels within the Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

Unit 3 Human Failure Events		
Operator Action Name	Description	
Split Fraction: OSW1 Basic Event: HOSW1	Transfer Mode Switch to REFUEL/ SHUT DOWN in response to scram	
	Pre-EPU (Model C3051602)	Post-EPU (Model U3060706)
FV, CDF	1.68E-03	8.00E-04
RAW, CDF	3.26E+00	2.10E+00
FV, LERF	6.51E-03	2.54E-04
RAW, LERF	9.74E+00	1.35E+00
Human Error Prob. (Monte Carlo Mean)	7.44E-04	7.29E-04
Time Available*	Not time significant for typical pressure reduction rates	Same as pre-EPU
Where is this operator action performed? Control Room		
*Time available is the time from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release.		

ENCLOSURE 5
TENNESSEE VALLEY AUTHORITY
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 10 REQUEST FOR ADDITIONAL INFORMATION (RAI)
(TAC NOS. MC3812, MC3743, AND MC3744)

TS-418 TS CHANGES REMARKED USING CURRENT TS PAGES

This enclosure provides a revised markup of the Unit 2 and Unit 3 TS changes for full 120% EPU operation.

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.

(2) Technical Specifications 3952

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: a. Described in Section 13.10, Refueling Test Program; of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or 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: a. The reactor is xenon free; b. The moderator temperature is 68°F; and 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:

THERMAL POWER shall be ≤ ~~25~~ 23% 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 AND 12 hours thereafter
203 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: $\frac{(C)(Q)(E)}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days AND 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 AND 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 AND 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 AND 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 AND 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~~25% 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----- Not required to be performed until 12 hours after THERMAL POWER ≥ ~~25~~23% RTP. 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----- 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)

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----- Neutron detectors are excluded. ----- Perform CHANNEL CALIBRATION.	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----- 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
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 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~~13% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	~~≤ 0.66 W + 66% RTP and ≤ 120% RTP~~(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

≤ 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 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
8. Turbine Stop Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
26					
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low(d)	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 550 psig
10. Reactor Mode Switch - Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.8 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.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

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.

2C

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------------ Not applicable if inoperable channel is the result of an inoperable breaker. ------------------------------- Place channel in trip.	72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. <u>AND</u> 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% 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: TSV - Closure: ≤ 10% closed; and 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----- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > ~~25~~23% RTP. ----------------------------------	
	Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop: 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. b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns. 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---------- 1. Only applicable for the 2 units fueled condition. 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. ------------------------------	
		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---------- 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
	AND		
~~OR~~	G.2	Be in MODE 4.	36 hours
~~UHS inoperable~~			

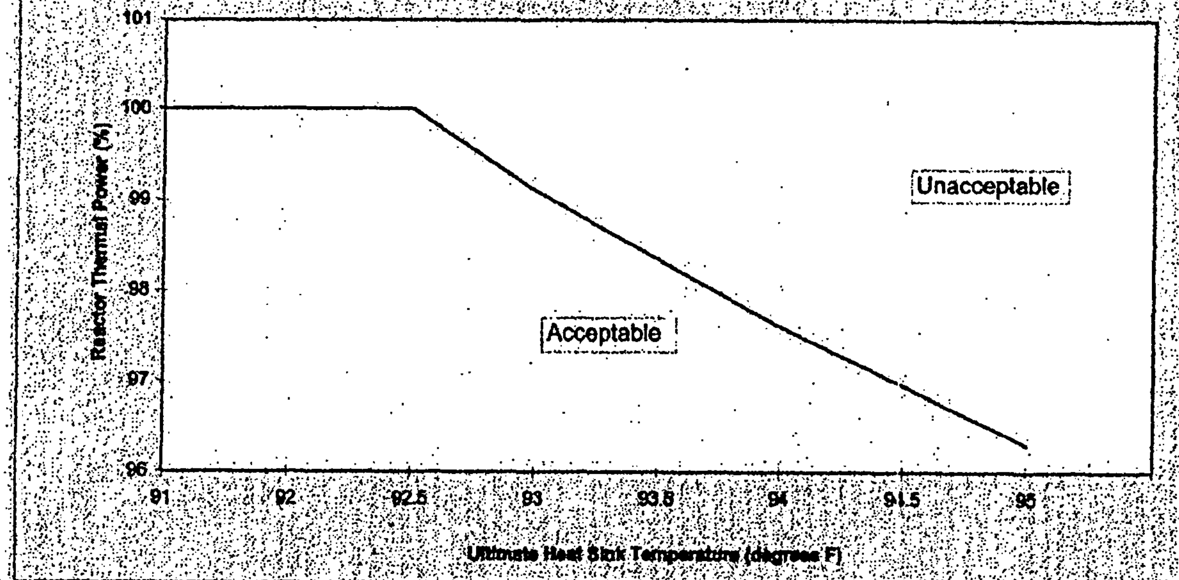
same as originally submitted

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~~ ~~AND~~ ~~1 hour UHS temperature > 91°F~~

Same as originally submitted

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



same as originally submitted

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	~~NOTE~~ ~~Refer to SR 3.7.1.2 for additional UHS requirements.~~ Verify the average water temperature of UHS is ≤ 95°F.	24 hours
SR 3.7.2.2	NOTE Isolation of flow to individual components does not render EECW System inoperable. 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

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5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (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 ≥ 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.

(2) <u>Technical Specifications</u> 3952

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: a. Described in Section 13.10, Refueling Test Program; of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or 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~~3952 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and 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.

Same as originally submitted

(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:

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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 ≥ 1.09 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 **AND** 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 pounds. 203	31 days
SR 3.1.7.6	Verify the SLC conditions satisfy the following equation: $\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days **AND** 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 AND 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 AND 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 AND 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 AND 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------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ ~~25~~23% RTP. --- 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------------------ 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)

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----- Neutron detectors are excluded. ----- Perform CHANNEL CALIBRATION.	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----- 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
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 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~~13% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	~~≤ 0.66 W + 66% RTP and ≤ 120% RTP~~(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

≤ 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 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
8. Turbine Stop Valve - Closure 26	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low(d)	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 550 psig
10. Reactor Mode Switch - Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.8 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.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

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~~ <u>20</u>% 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
	OR		
	A.2	---NOTE--- Not applicable if inoperable channel is the result of an inoperable breaker.	
		Place channel in trip.	72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. AND 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~~26% 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: TSV - Closure: ≤ 10% closed; and 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------ 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > ~~26~~23% RTP. ------ Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop: 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. b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns. 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---------- 1. Only applicable for the 2 units fueled condition. 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. ------------------------------- 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

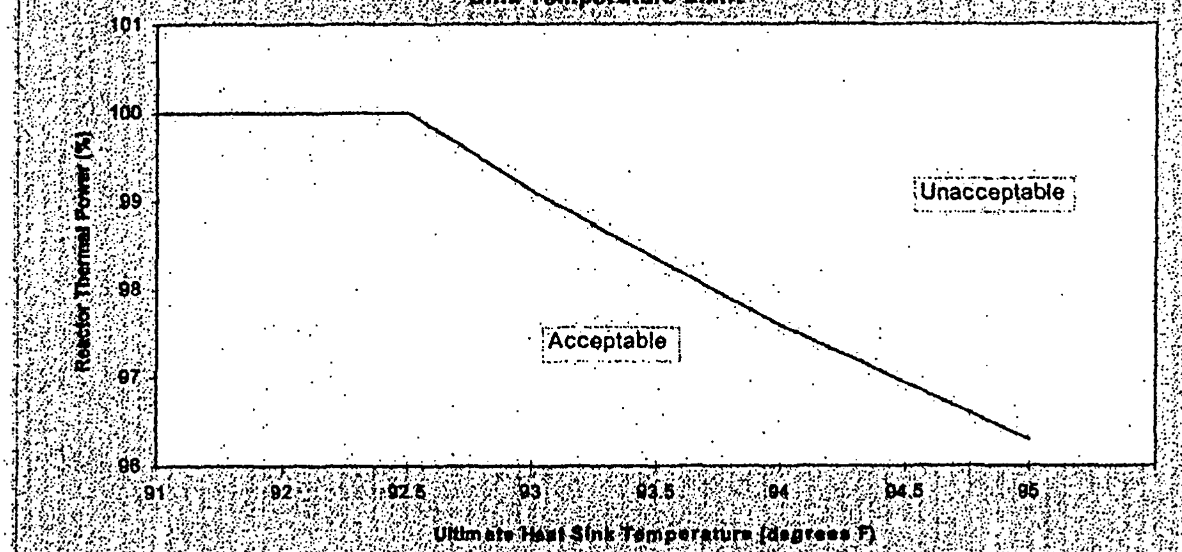
same as originally submitted

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 ≤ 95°.~~	~~24 hours UHS temperature ≤ 95°F~~ ~~AND~~ ~~1 hour UHS temperature > 91°F~~

Same as originally submitted

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



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originally
submitted

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	~~NOTE~~ ~~Refer to SR 3.7.1.2 for additional UHS requirements.~~ Verify the average water temperature of UHS is ≤ 95°F.	24 hours
SR 3.7.2.2	NOTE Isolation of flow to individual components does not render EECW System inoperable. 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

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 50.0 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 ≥ 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.

ENCLOSURE 6
TENNESSEE VALLEY AUTHORITY
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 10 REQUEST FOR ADDITIONAL INFORMATION (RAI)
(TAC NOS. MC3812, MC3743, AND MC3744)

REGULATORY COMMITMENTS

Commitments

An evaluation summary for Unit 1 piping systems (including main steam, feedwater, recirculation, residual heat removal (RHR), and torus attached piping) will be provided to the NRC staff by December 31, 2006. The information provided will include the calculated maximum stresses for piping systems similar to the information provided for the EPU application of Units 2 and 3.

Because initial operation of Unit 1 Cycle 7 will be at 105% of OLTP, the core design and associated analyses are being reperformed consistent with an interim licensed power level of 3458 MWt. TVA will provide the revised Supplemental Reload Licensing Report to the NRC by January 31, 2007.