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Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

TVA-WBN-TS-00-06

10CFR 50.90

Richard T. Purcell
Site Vice President, Watts Bar Nuclear Plant

June 7, 2000

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of) Docket No. 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 00-06 - INCREASE UNIT 1 REACTOR POWER TO 3459 MWt

In accordance with the provisions of 10 CFR 50.90, TVA is submitting a request for an amendment to WBN's license NPF-90 to change the Operating License and Technical Specifications for Unit 1. The proposed license amendment would increase the full core thermal power rating by 1.4% from 3411 MWt to 3459 MWt, based on planned installation of the improved Caldon, Incorporated (Caldon) Leading Edge Flow Meter, LEFM✓™ (LEFM) feedwater flow measurement instrumentation during the upcoming Unit 1 Cycle 3 refueling outage. Enclosure 1 to this letter provides the description and evaluation of the proposed change, a detailed safety analysis, and TVA's determination that the proposed change does not involve a significant hazards consideration and is exempt from environmental review.

As addressed in Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the (LEFM✓™) System," the LEFM will enable determination of core power level with improved measurement uncertainties, thereby allowing a power uprate. The Staff's review and approval of ER-80P is documented in NRC's Safety Evaluation for Texas Utilities' (TU) Comanche Peak Unit 2, dated March 8, 1999. In support of the TVA license amendment request, Enclosure 2 provides proprietary supplemental information for the Caldon Topical Report; Enclosure 3 provides a non-proprietary version. In addition, WBN's NSSS Vendor, Westinghouse Electric Company, has performed a power calorimetric measurement uncertainty calculation for use of the LEFM. Proprietary and non-proprietary summaries of this information are provided in Enclosures 4 and 5, respectively.

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TVA's amendment request is consistent with the power uprate license amendment granted to TU for Comanche Peak Unit 2, dated September 30, 1999. In order to facilitate NRC's review of the enclosed WBN application, TVA has addressed NRC Staff questions raised in the licensing process for the Comanche Peak Unit 2 power uprate. Non-proprietary and proprietary versions of this information are provided in Enclosures 6 and 7 respectively, and incorporated into the license amendment request where practical. TVA's implementation of the proposed license amendment is based upon the revised requirements of 10 CFR 50, Appendix K, "Emergency Core Cooling System Evaluation Models, ECCS," as recently approved by the NRC Commission and issued on June 1, 2000 (65 FR 34913), with an effective date of July 31, 2000.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The WBN Plant Operations Review Committee and the WBN Nuclear Safety Review Board have reviewed this proposed change and have determined that operation of WBN Unit 1 in accordance with the proposed change will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

Enclosure 8 contains copies of the appropriate Unit 1 TS pages marked-up to show the proposed change. Enclosure 9 forwards the revised TS pages which incorporate the proposed change.

Enclosure 2 contains information proprietary to Caldon, Inc. Accordingly, Enclosure 10 includes a Caldon Application for Withholding Proprietary Information from Public Disclosure, and an accompanying Affidavit CAW-00-02, R1 signed by Caldon, the owner of the information.

Enclosures 4 and 7 contain information proprietary to Westinghouse. Accordingly, Enclosure 11 includes Westinghouse Applications for Withholding Proprietary Information from Public Disclosure, and an accompanying Affidavits CAW-00-1398 and CAW-00-1399 signed by Westinghouse, the owner of the information. Also included are a Proprietary Information Notice and a Copyright Notice.

The above affidavits set forth the basis on which the requested information may be withheld from public disclosure by the Commission, and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.790 of the Commission's regulations. Accordingly, TVA requests that the information which is proprietary to Westinghouse and Caldon be withheld from public disclosure in accordance with 10 CFR 2.790.

Correspondence regarding the proprietary aspects of the Westinghouse information listed above, the Copyright Notice, or the supporting affidavit, should reference Westinghouse letters CAW-00-1398 and CAW-00-1399 and be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355. Correspondence regarding the proprietary aspects of the Caldon report listed above, or the supporting affidavit, should reference Caldon letter CAW-00-02, R1 and be addressed to Calvin R. Hastings, President and CEO, Caldon Incorporated, 1070 Banksville Avenue, Pittsburgh, Pa., 15216.

Enclosure 12 lists the commitments made in this submittal.

TVA requests that approval be provided prior to completion of the Unit 1 Cycle 3 refueling outage scheduled to begin September 10, 2000, and that the revised TS be made effective prior to increasing reactor power above 3411 MWt. To facilitate this review TVA and its contractors are available to meet with NRR as often as necessary to resolve staff review questions.

If you have any questions about this change, please contact P. L. Pace at (423) 365-1824.

Sincerely,

W.R. Lagergren

for R. T. Purcell

Enclosures
cc: See page 3

Subscribed and sworn to before me
on this 7th day of June , 2000.

Judy C. Lancaster
Notary Public
My Commission Expires 2-28-2001

cc (Enclosure):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1 - DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-00-06
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. BACKGROUND AND REASON FOR THE PROPOSED CHANGE

Watts Bar Nuclear Plant (WBN) Unit 1 is presently licensed for a full core thermal power rating of 3411 MWt. The proposed license amendment would increase the core power level by 1.4% to 3459 MWt. TVA has evaluated the impact of a 1.4% uprate to 3459 MWt for applicable systems, structures, components, and safety analyses.

TVA's uprate is based on eliminating unnecessary analytical margin originally required of ECCS evaluation models performed in accordance with the requirements set forth in 10 CFR 50, Appendix K (Emergency Core Cooling System Evaluation Models, ECCS) for WBN Unit 1. The currently published regulation (as revised through 60 FR 24552, May 2, 1995), mandates consideration of an assumed reactor operating power level of 102% of the licensed power level for ECCS evaluation models of light water power reactors. However, the NRC has recently approved a change to the requirements of 10 CFR 50, Appendix K, whereby licensees are provided with the option of maintaining the current 2% power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin. For the latter case, the proposed alternative reduced margin must be demonstrated to account for uncertainties due to power level instrumentation error. Based on the proposed use of the improved Caldon LEFM instrumentation to determine core power level with a power measurement uncertainty of less than 0.6%, TVA proposes to reduce the licensed power uncertainty required by 10 CFR 50, Appendix K, for modest increases of up to 1.4% in the licensed power level using current NRC approved methodologies.

The basis for the amendment request is that the Caldon instrumentation provides a more accurate indication of feedwater flow (and correspondingly reactor thermal power) than assumed during the development of Appendix K requirements. Complete technical support for this conclusion is discussed in detail in the Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," as approved in NRC's Safety Evaluation for TU Electric, dated March 8, 1999, and supplemented by Caldon Engineering Report 160P, provided in Enclosure 2 of this letter. The improved thermal power measurement accuracy obviates the need for the full 2% power margin assumed in Appendix K, thereby increasing the thermal power available for electrical generation.

Along with the proposal to increase the reactor thermal power to 3459 MWt, TVA also proposes continued use of the topical reports identified in Watts Bar Unit 1 Technical Specification 5.9.5b. These reports describe the NRC approved analytical methodologies used to determine the core operating limits for Watts Bar Unit 1, including the small break and large break loss of coolant accidents analyses. In some of these topical reports, reference is made to the use of a 2% uncertainty applied to the reactor power, consistent with 10 CFR 50, Appendix K. As discussed below, TVA's TS change request includes proposed clarifying information to TS 5.9.5b that these topical reports be approved for use consistent with this license amendment request. Further, TVA considers that the change in the power uncertainty does not constitute a significant change as defined in 10 CFR50.46 and Appendix K.

II. DESCRIPTION OF THE PROPOSED CHANGE

The proposed license amendment would revise the Watts Bar Nuclear Plant (WBN) Unit 1 operating license and TS to increase the core power level by 1.4% to 3459 MWt. The power uprate is based on the use of the Caldon, Inc., LEFM for determination of main feedwater flow and the associated determination of reactor power through the performance of a daily calorimetric, currently required by WBN Unit 1 TS. Specifically, as illustrated by the markup of the current WBN Unit 1 operating license and TS in Enclosure 8, the following changes are proposed:

(1) The Operating License for Watts Bar Unit 1 (NPF-90), Section 2.C(1), identifies the maximum core thermal power level for which WBN Unit 1 is authorized to operate as 3411 MWt. TVA proposes changing the maximum core power level to 3459 MWt.

(2) The definition of RATED THERMAL POWER in the Technical Specifications is changed to read:

"RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt."

(3) The Reactor Core Safety Limits of TS Section 2.1 are revised as specified in a revised Figure 2.1.1-1 (Reactor Core Safety Limits).

(4) LCO 3.7.1 (Main Steam Safety Valves) and Table 3.7.1-1, OPERABLE Main Steam Safety Valves versus Maximum Allowed Power, are revised to reflect the maximum allowed power for operation with inoperable MSSVs. With one inoperable MSSV per loop, the power reduction is revised from 59% RTP to 58% RTP. With multiple inoperable safety valves per loop, the power reduction and associated reduction in high flux reactor trip setpoints is 41% RTP (two inoperable MSSVs) and 25% RTP (three inoperable MSSVs). In addition, the TS Bases for LCO 3.7.1 is revised to reflect that the MSSV analysis which was performed at 102% RTP (at the current power level of 3411 MWt) equates to an analysis performed at 100.6% RTP for the proposed RTP of 3459 MWt.

(5) TS Bases for LCO 3.7.6, Condensate Storage Tank (CST) is revised to reflect that the CST analysis which was performed at 102% RTP (at the current power level of 3411 MWt) equates to an analysis performed at 100.6% RTP for the proposed RTP of 3459 MWt.

(6) Section 5.9.5(b), Analytical Methods for Core Operating Limits Report (COLR) is revised from:

"b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents"

to read:

"b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.

The approved analytical methods are specifically those described in the following documents."

TVA notes that the above proposed paragraph for TS 5.9.5b is consistent with the TU Electric license amendment for the Comanche Peak Unit 2 uprate. As discussed in response to Question Number 2 of TXX Letter No. 99203 (Refer to Enclosure 6), inoperability of the LEFM will not require an immediate reduction to 3411 MWt. The specific actions required by operations personnel will be included within a revised Technical Requirement in the WBN Technical Requirements Manual (TRM). In addition, TVA notes that the CPSES Unit 2 license amendment included an additional commitment in this TS Section (Section 5.6.5b of the CPSES Unit 2 Tech Specs) that reads as follows:

"Future revisions of approved analytical methods listed in this technical specification that currently assume 102 percent of rated power shall include the condition given above allowing use of 101 percent of rated power in safety analysis methodology when the LEFM is used for feedwater flow measurement."

TVA considers it would be more appropriate to retain a similar licensing basis commitment in the WBN Final Safety Analysis Report for only the WBN-specific analyses. Accordingly, prior to implementation of the proposed license amendment, TVA will issue an approved changed to the WBN FSAR that will stipulate that future revisions of the WBN-specific topical reports listed in TS Section 5.9.5b that currently assume 102 percent of rated power shall include the condition given above allowing use of 100.6 percent of rated thermal power in safety analysis methodology when the LEFM is used for feedwater flow measurement.

(7) Section 5.9.5(b), Analytical Methods for Core Operating Limits Report (COLR) is revised to add a reference to the following Caldon Topical Reports for LEFM as follows:

"6. Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™," Revision 0, May 2000; as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No.   ."

III. SAFETY ANALYSIS

0.1 APPROACH

The Watts Bar Power Uprate Project has been completed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," dated 1983. Since its submittal to the NRC, the methodology has been successfully used as the basis for power uprate projects on over twenty pressurized water reactor units, including Diablo Canyon Units 1 and 2, Turkey Point Units 3 and 4, and Comanche Peak Unit 2.

The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents and nuclear fuel as well as interfaces between the NSSS and Balance of Plant (BOP) systems. Inherent in this methodology are key points that promote correctness, consistency and licensability. The key points include the use of well-defined analysis input assumptions/parameter values, use of currently approved analytical techniques and use of currently applicable licensing criteria and standards.

The evaluations and analyses described herein have been completed consistent with this methodology. Section 1 provides an overview of the LEFM and its application at WBN. Section 2 discusses the NSSS thermal and hydraulic parameters, which are modified as a result of implementing the power uprate and serve as the basis for all of the evaluations and analysis. Section 3 concludes that no design transient modifications are required to accommodate the revised design conditions in Section 2. Sections 4 and 5 present the systems and component evaluations performed for the revised design conditions. Section 6 discusses the results of the accident analyses and evaluations for the SGTR, SLB Mass and Energy, LOCA Mass and Energy, LOCA and non-LOCA/Transient analyses. Section 7 summarizes the effects of the uprate on the balance of plant (secondary) systems based upon a preliminary heat balance evaluation. Section 8 provides an analysis of the effects of the power uprate on the WBN Unit 1 electrical power systems and Section 9 provides a summary of the radiological evaluation.

0.2 GENERAL LICENSING METHODOLOGY FOR PLANT ANALYSES USING PLANT POWER LEVEL

The reactor and/or NSSS thermal power are used as inputs to most plant safety, component and system analyses. These analyses generally model the core and/or NSSS thermal power in one of four ways.

First, some analyses apply a 2% increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the 1.4% uprate conditions. The power calorimetric uncertainty calculation described in Section 6.7 indicates that with the LEFM installed, the power measurement uncertainty, based on a 95% probability at a 95% confidence interval, is less than 0.6%. Thus, these analyses only need to reflect a 0.6% power measurement uncertainty. Accordingly, the existing 2% uncertainty can be allocated such that 1.4% is allocated to provide sufficient margin to address the 1.4% uprate to 3459 MWt and 0.6% is retained in the analysis to still account for the measurement uncertainty. In addition, for these types of analyses, it is shown that they still employ other conservative assumptions not affected by the 1.4% uprated power. Taken together, the use of the calculated 95/95 power measurement uncertainty and retention of conservative assumptions indicate that the margin of safety for these analyses would not be reduced.

Second, some analyses employ a nominal power level. These analyses have either been evaluated or re-performed for the 1.4% increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.4% uprate conditions.

Third, some of the analyses already employ a 4.5% increase in core power to provide analysis margin to offset plant changes. These analyses would normally be performed at the nominal power level; however, due to prudent planning purposes, they have been performed at the higher power level. For these analyses, this available margin has been used to offset the 1.4% uprate. Consequently, the analyses have not been re-performed and continue to retain sufficient analysis margin.

Fourth, some of the analyses are performed at 0% power conditions or do not actually model the core power level. Consequently, these analyses have not been re-performed since they are unaffected by the core power level.

1 LEADING EDGE FLOW METER (LEFM)

The LEFM is an ultrasonic flow meter consisting of an electronic cabinet in the Auxiliary Instrument Room and a measurement section (spool piece) located in the 32-inch main feedwater header line. Each measurement section holds eight ultrasonic transducer assemblies, secured in their own transducer housing which forms the pressure boundary. Each transducer may be removed at full power conditions without disturbing the pressure boundary. The LEFM uses acoustic energy pulses to determine the final feedwater mass flow rate. Transducers that transmit and receive the pulses are mounted in the LEFM spool piece at an angle of 45 degrees to the flow. The sound will travel faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The LEFM uses these transient times and time differences between pulses to determine the fluid velocity and temperature. The system uses a single digital system controlled by software to employ the ultrasonic transit time method to measure 4 line integral velocities at precise locations with respect to the pipe center line. The system numerically integrates the four velocities measured according to the method described in Caldon's Topical Report ER-80P. Although its use for calorimetric input is not nuclear safety-related, the system's software has been developed and will be maintained under a Verification and Validation (V&V) program. The V&V program has been applied to all system software and hardware, and includes a detailed code review. The mass flow rate is displayed on the local display panel and transmitted to the plant process computer for use in the calorimetric measurement. The feedwater mass flow rate is calculated and is used by the plant computer to determine the reactor thermal output based on an energy balance of the secondary system.

The LEFM is an improved system for use in determining and monitoring feedwater flow in nuclear power plants. The LEFM provides on-line verification of the accuracy of the feedwater flow and temperature measurements upon which NSSS thermal power determinations are based. In addition, the LEFM provides a significant improvement in accuracy and an increase in reliability of flow and temperature measurements.

These accuracies are valid while the instruments are performing as designed. The on-line verification features of the LEFM provide the ability to assure that performance is consistent with the design basis.

The LEFM system measures the transit times of pulses of ultrasonic energy traveling along chordal acoustic paths through the flowing fluid. This technology provides significantly higher accuracy and reliability than flow instruments which use differential pressure measurements and temperature instruments which use conventional thermocouple or resistance thermometers.

The LEFM provides measurements of feedwater mass flow and temperature yielding a total uncertainty of ±0.6% of reactor thermal power, substantially more accurate than the typical ±2% RTP obtained with the conventional venturi-based instrumentation, or the ±1.4% RTP uncertainty obtainable with precision, venturi-based instrumentation.

The LEFM indications of feedwater mass flow and feedwater temperature will be directly substituted for the venturi-based mass flow indication and the RTD temperature indications currently used in the plant calorimetric measurement calculation performed with the plant computer. The plant computer will then calculate enthalpy and thermal power as it does now. The venturi-based feedwater flow measurement will continue to be used for feedwater control and other functions that it currently fulfills. Further, the venturi-based indication may be adjusted periodically on the basis of the LEFM indication, so that it serves as a backup calorimetric power determination in the event that the LEFM system is not available.

2 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) PARAMETERS

2.1 INTRODUCTION AND BACKGROUND

The NSSS parameters are the fundamental parameters which are used as input in all the NSSS analyses. They provide the Reactor Coolant System (RCS) and secondary system conditions (temperatures, pressures, flow) that are used as the basis for the design transient, system, component and accident evaluations.

The parameters are established using conservative assumptions in order to provide bounding conditions to be used in the NSSS analyses. For example, the RCS flow assumed is the RCS design flow which is a conservatively low flow that accounts for flow measurement uncertainty.

2.2 INPUT PARAMETERS AND ASSUMPTIONS

The NSSS thermal power for the uprating analysis was set at 3475 MWt (3459 MWt core). This is approximately 1.4% higher than the current NSSS power rating of 3427 MWt (3411 MWt core). The feedwater temperature was adjusted from 440 to 441.8°F to accommodate the power increase. All other input parameters remained the same as those used for the current licensing basis.

2.3 DISCUSSION OF PARAMETER CASES

Table 2-1 provides the NSSS parameter cases which were generated and used as the basis for the uprating project. The current design parameters are also shown for comparison purposes.

A review of Table 2-1 reveals that the 1.4% uprate leads to a 0.4°F change in T_{cold} and T_{hot}, which is considered very small. Also, the changes in the secondary side conditions are relatively small since:

- T_{steam} decreased by 1.4°F
- P_{steam} decreased by 11 psi
- Q_{steam} (flow) increased by 1.9%
- T_{FW} increased by 1.8°F

Table 2-1: NSSS Performance Parameters

Parameter	Current (10% SGTP/2% RTDF) Including 16 MWt [1]Net Heat Input	1.4% Uprate at 10% SGTP
NSSS Power, MWt	3427	3475
Reactor Power, MWt	3411	3459
RCS Thermal Design Flowrate, gpm/loop	93,100	93,100
RCS Minimum Measured Flow (gpm)	379,100	379,100
RCS Best Estimate Flow (gpm/loop)	99,200	99,300
RCS Pressure, psia	2250	2250
Core Bypass Flow, %	9.0	9.0
RCS Temperatures, °F		
Core Outlet	623.9	624.4
Vessel Outlet, T_{hot}	618.7	619.1
Core Average	592.8	592.8
Vessel Average	588.2	588.2
Vessel/Core Inlet, T_{cold}	557.7	557.3
Steam Generator Outlet	557.5	557.0
Steam Generator		
Steam Temperature, °F	539.4	538.0
Steam Pressure, psia	958	947
Steam Flow, 10^6 lb/hr total	15.08	15.36
Feed Temperature, °F	440	441.8
Tube Plugging %	10	10

1 The revised RCP heat addition values were evaluated per the WBN 10CFR50.59 program as part of an upgrade performed in support of WBN Unit 1 Cycle 2. It was determined that no impact to nuclear safety existed and that no unreviewed safety questions were created by more accurately accounting for the net heat input from the RCPs. The heat input estimate was revised from a nominal 14 MWt to 16 MWt.

3 DESIGN TRANSIENTS

3.1 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) DESIGN TRANSIENTS

The revised design conditions in Table 2-1 and the NSSS design transients applicable to the uprated conditions serve as primary inputs to the evaluation and analysis of the NSSS systems and components. Current primary and secondary design transients were reviewed in order to determine their continued applicability for the revised design conditions.

Primary Side Transients

The review of the primary side design conditions listed in Table 2-1 indicate that the full power temperature values of vessel outlet and vessel inlet (hot leg and cold leg) vary by less than or equal to 0.4°F from the previously applicable design values. Also, the vessel average temperature was not changed. Given the conservative assumptions used to develop the current design transients (e.g., initial conditions, unavailability of control systems during certain transients), a 0.4°F change in primary side full power temperatures is considered insignificant during all transient conditions. Therefore, the revised conditions have a negligible impact on the primary side design transients, and the previously applicable NSSS design transients for the primary side continue to apply, without modification, at the revised design conditions.

Secondary Side Transients

With regard to secondary side design parameters, the revised design conditions in Table 2-1 indicate that the plant may operate with slightly lower full power values for steam temperature and steam pressure and slightly higher values for feedwater temperature. Lower nominal steam temperatures (e.g., from 539.4 to 538.0°F) and pressures (e.g., from 958 to 947 psia) result in relatively small changes in initial conditions from those reflected in the current NSSS design transients. Similarly, a 1.8°F increase in feedwater temperature (i.e., 440 to 441.8°F) is insignificant in comparison to the analyzed feedwater temperature transients.

The small variations in these parameters were either shown to be enveloped by the existing transient curves or encompassed by the conservative assumptions used to develop the design transients. Thus, it was determined that the existing secondary side transients remained valid for the revised design conditions.

3.2 AUXILIARY EQUIPMENT DESIGN TRANSIENTS

The review of the NSSS auxiliary equipment design transients was based on a comparison between the revised operating conditions in Table 2-1 and the parameters which make up the current auxiliary equipment design transients. A review of the current auxiliary equipment transients determined that the only transients potentially impacted by the revised conditions are those temperature transients impacted by full load NSSS operating temperatures, namely T_{HOT} and T_{COLD}. These transients are currently based on an assumed full load NSSS worst case T_{HOT} of 630°F and worst case T_{COLD} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

A comparison of the limiting operating values for T_{HOT} and T_{COLD} of 619.1°F and 557.3°F, respectively, with the existing values indicates that they are still well within the design. Thus, the 1.4% uprate does not require any changes to these transients.

4 NSSS SYSTEMS

This chapter presents the results of the evaluations and analyses performed in the NSSS systems area to support the revised design conditions in Table 2-1. The systems addressed in this chapter include Fluid Systems and NSSS/BOP interface Systems. The results and conclusions of each analysis are presented within each subsection.

4.1 NSSS FLUID SYSTEMS

4.1.1 REACTOR COOLANT SYSTEM (RCS)

The RCS consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump (RCP), which circulates the water through the loops and reactor vessel, and a steam generator (SG), where heat is transferred to the main steam system (MSS). In addition, the RCS contains a pressurizer which controls the RCS pressure through electrical heaters, water sprays, power operated relief valves (PORVs) and spring loaded safety/relief valves. The steam discharged from the PORVs and safety/relief valves flows through interconnecting piping to the pressurizer relief tank (PRT).

Various assessments were performed to help demonstrate that the RCS design basis functions could still be met at the revised design conditions.

It was demonstrated that the minimum required pressurizer spray flow of 900 gpm can be achieved for the 1.4% uprate conditions defined in Table 2-1. Also, the maximum expected T_{HOT} at the revised design conditions is 619.1°F. This temperature is well below the RCS loop design temperature of 650°F.

With respect to the PRT discharge analysis, the nominal full load pressurizer steam volume is essentially unaffected by the uprate since the RCS average temperature of 588.2°F has not changed. Thus, the existing discharge analysis is essentially unaffected.

4.1.2 Chemical and Volume Control System (CVCS)

The CVCS provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of water letdown from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the shell side of the regenerative heat exchanger and then through a letdown orifice. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled, low pressure water leaves the reactor containment and enters the auxiliary building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank (VCT).

In the assessment of CVCS operation at revised RCS operating temperatures, the maximum expected RCS T_{COLD} must be less than or equal to the applicable CVCS design temperature and less than or equal to the heat exchanger design inlet operating temperature. The former criterion supports the functional operability of the system and its components. The latter criterion confirms that the heat exchanger design operating conditions remain bounding.

With regards to the CVCS thermal performance, the T_{COLD} of 557.3°F is still lower than the design system inlet temperature of 560°F. Also, it is much lower than the shell side design temperature of 650°F for the regenerative heat exchanger. The excess letdown path is used to process excess effluents associated with fluid expansion during plant heatup and thus, is unaffected by the revised T_{COLD} at full power conditions. If operated during power conditions, the excess letdown heat exchanger outlet flow is throttled to maintain the desired outlet temperature and flow. Therefore, operation of the CVCS is unaffected by the temperature change.

4.1.3 Safety Injection System (SIS)

The SIS is an Engineered Safeguards System used to mitigate the effects of postulated design basis events. The basic functions of this system include providing short- and long-term core cooling, and maintaining core shutdown reactivity margin. The SIS is made-up of three subsystems. The passive portion of the system is the four accumulator vessels which are connected to each of the RCS cold leg pipes. Each accumulator contains borated water under pressure (nitrogen cover gas). The borated water automatically injects into the RCS when the pressure within the RCS drops below the operating pressure of each of the accumulators.

The 'active' part of the SIS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two safety injection (SI) pumps and two residual heat removal (RHR) pumps take suction from the Refueling Water Storage Tank (RWST) and deliver borated water to four cold leg connections via the accumulator discharge lines. In addition, two centrifugal charging pumps take suction from the RWST on SI actuation and provide flow to the RCS via separate SI connections on each cold leg. This arrangement of SI pumps can provide safety injection flow at any RCS pressure up to the set pressure of the pressurizer safety valves.

The revised design conditions have no direct effect on the overall performance capability of the SIS. These systems will continue to deliver flow at the design basis RCS and containment pressures since there are no changes in the RCS operating pressure.

4.1.4 Residual Heat Removal System

The RHRS is designed to remove sensible and decay heat from the core and reduces the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHRS is used to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations.

The RHRS consists of two RHR heat exchangers, two RHR pumps and associated piping, valves and instrumentation. During system operation, coolant flows from one hot leg of the RCS to the RHR pumps, through the tube side of the residual heat exchangers and back to theRCS cold legs. The RHR heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell.

A single train cooldown analysis and a normal cooldown analysis were performed to address the uprated reactor power (3459 MWt). A single train cooldown is defined as cooling the RCS from 350°F at four hours after plant shutdown to 200°F by employing one RHR Pump, one RHR heat exchanger and one train of component cooling. The overall single train cooldown should be achieved within 36 hours after plant shutdown based on system design requirements. An analysis was performed to demonstrate that the cooldown can still be accomplished within 36 hours at the 1.4% uprate conditions.

In addition, a normal cooldown is defined as cooldown assuming all equipment available. The system design basis is that the normal cooldown can be achieved within 20 hours following plant shutdown. Normal cooldown is defined as cooling the RCS from 350°F at four hours after plant shutdown to 140°F using two trains of cooling equipment (2 trains of RHR, component cooling and essential raw cooling water). At the 1.4% uprate conditions, the cooldown can still be achieved within the 20 hour basis.

4.1.5 Cold Overpressure Mitigation System (COMS)

COMS is designed to protect the RCS from overpressure events when the RCS temperature is below 350°F. Changes to full power operating parameters, such as NSSS power, do not impact COMS. Thus, the existing COMS analysis is unaffected.

4.1.6 Control System Response

Condition I transients are evaluated to confirm that the plant can appropriately respond to these transients without generating a reactor trip or ESFAS actuation. The transients of concern include:

- 10% step load increase
- 10% step load decrease
- 50% load rejection
- 5%/minute ramp load increase

The analysis methodology for these transients employs a 2% power calorimetric uncertainty to increase the power level to 102%. Consistent with the discussion provided in Section 0.2, the improved thermal power measurement accuracy obviates the need for the full 2% power measurement margin assumed in the analysis.

Furthermore, the power measurement margin is but one of many conservative assumptions used in the analysis. For example, the analysis assumes a minimum available steam dump capacity of 40% of full load. As noted in Section 4.2.2, the actual steam dump capacity will be at or above 42.4% of full load for the most limiting 1.4% power uprated conditions. Also, the analysis conservatively assumes that the plant operates with the more limiting beginning of life (BOL) fuel reactivity conditions - which provide the more severe reactivity response, and hence transient conditions. Taken together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the above noted transients can be accommodated without resulting in a reactor trip or ESFAS actuation.

4.1.7 Spent Fuel Pool Cooling

Spent fuel pool cooling calculations were reviewed and found to have adequate margin for the expected decay heat load increase that will be proportional to the power increase.

4.2 NSSS/BOP INTERFACE SYSTEMS

The following Balance-of-Plant (BOP) fluid systems were reviewed for compliance with applicable Nuclear Steam Supply Systems (NSSS)/BOP interface guidelines at the revised design conditions in Table 2-1. It was determined that these guidelines were still met with the 1.4% uprate

4.2.1 Main Steam System

The following summarizes the evaluation of the major components of the main steam system (MSS) relative to the revised design conditions. The major components of the MSS are the steam generator main steam safety valves (MSSVs), the SG atmospheric relief valves (ARVs), and the main steam isolation valves (MSIVs).

Steam Generator Main Steam Safety Valves

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the MSS design pressure (the maximum pressure allowed by the ASME B&PV Code). Watts Bar has twenty MSSVs with a total capacity of 16.64 x 10^6 lb/hr (0% SGTP condition), which provides about 108.1 percent of the maximum calculated steam flow of 15.39 x 10^6 lb/hr for the revised design conditions. Therefore, based on the revised design conditions, the capacity of the installed MSSVs meets the sizing criterion.

Steam Generator Atmospheric Relief Valves (ARVs)

The primary function of the ARVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the ARVs in conjunction with the Auxiliary Feedwater System (AFWS) permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the Residual Heat Removal System (RHRS) can be placed in service. During cooldown, the ARVs are either automatically or manually controlled. Each ARV P&I controller automatically compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a tube rupture event in conjunction with loss of offsite power, the ARVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

For the revised design conditions, each steam generator ARV is required to have a capacity at least equal to 64,000 lbs/hr at 100 psia inlet pressure. At these conditions, this capacity permits a plant cooldown to RHRS operating conditions (at a cooldown rate of 50°F/hr) assuming a minimum of 2 hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFWS. This is based on one train of auxiliary feedwater (AFW) operating and flow going through two SGs. Since the design capacity of the installed ARVs meets the sizing criteria, the valves are adequately sized for the 1.4% uprate conditions.

Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the original design requirements specified that the MSIVs must be capable of closure within 6 seconds of receipt of a closure signal against steam break flow conditions in either the forward or reverse direction.

Rapid closure of the MSIVs following postulated steam line breaks causes a differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for differential pressure increase and thrust loads are controlled by the steam line break area, throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load operating pressure are not impacted by the revised design conditions, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change. Consequently, the revised design conditions have no significant impact on the interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low power conditions where the revised design conditions have no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, the revised design conditions have no significant impact on the interface requirements for the MSIV bypass valves.

4.2.2 Steam Dump System

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant rated electrical load without a reactor trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the Reactor Control System, which accommodates 10% of the load reduction. A steam dump capacity of 40 percent of rated steam flow at full load steam pressure also prevents MSSV lifting following a reactor trip from full power. Watts Bar has twelve condenser steam dump valves. The total capacity for all twelve valves of 42.4% at the revised design conditions exceeds the Westinghouse sizing criterion of 40 percent of rated steam flow.

4.2.3 Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The revised design conditions will impact both feedwater volumetric flow and system pressure drop. The major components of the C&FS are the feedwater isolation valves, the feedwater regulator valves, and the Condensate and Feedwater System pumps.

Feedwater Isolation Valves/Feedwater Regulator Valves

The feedwater isolation valves (FIVs) are located outside containment and downstream of the feedwater regulator valves (FRVs). The valves function in conjunction with the primary isolation signals to the FRVs and backup trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive reactor coolant system cooldown. To accomplish this function, the FRVs and the backup FIVs must be capable of fast closure, that is within 6.5 seconds following receipt of any feedwater isolation signal.

The quick-closure requirements imposed on the FRVs and the backup FIVs causes dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam break from no load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions (i.e. maximum possible feedwater flow during a steam line break from no load conditions) are not impacted by the revised design conditions since the no load conditions are not affected by the 1.4% uprate, the design loads and associated stresses resulting from rapid closure of these valves will not change.

Condensate and Feedwater System (C&FS) Pumps

The C&FS available head in conjunction with the FRV characteristics must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the FRVs at rated flow (100 percent power) should be approximately equal to the dynamic losses from the feed pump discharge through the steam generator. In addition, adequate margin should be available in the FRVs at full load conditions to permit a C&FS delivery of 96 percent of rated flow with a 100 psi pressure increase above the full load pressure with the FRVs fully open. At the revised design conditions, the operation of the FRVs in conjunction with the feedwater pump speed control program should continue to meet these requirements.

To provide effective control of flow during normal operation, the FRVs are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0-1600 psig). Additionally, rapid closure of the FRVs is required in 6.5 seconds after receipt of a trip close signal in order to mitigate certain transients and accidents. These requirements are still applicable at the revised design conditions.

4.2.4 Auxiliary Feedwater System (AFWS)

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink of the steam generators. The system provides feedwater to the SGs during normal unit startup, hot standby, and cooldown operations and also functions as an Engineered Safeguards System. In the latter function, the AFWS is directly relied upon to prevent core damage and system overpressurization in the event of transients and accidents such as a loss of normal feedwater or a secondary system pipe break.

Auxiliary Feedwater Storage Requirements

The AFWS pumps are normally aligned to take suction from the condensate storage tank (CST). To fulfill the Engineered Safety Features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to CST inventory requirements is the loss-of-offsite power (LOOP) transient. In the event of a LOOP, sufficient CST useable inventory must be available to bring the unit from full power to hot standby conditions, maintain the plant at hot standby for 2 hours, and then cooldown the RCS to the residual heat removal system cut-in temperature (350°F) in 5 hours. In light of these design bases requirements, the analysis-of-record concluded that the tank should be designed to accommodate a minimum useable inventory of 200,000 gallons. The minimum CST useable inventory of 200,000 gallons is based on reactor trip from 102 percent of rated core power (3411 MWt). Consistent with the discussion provided in Section 0.2, the improved thermal power measurement accuracy obviates the need for the full 2% power measurement margin assumed in the analysis.

4.2.5 Steam Generator Blowdown System

The Steam Generator Blowdown System is used in conjunction with the Chemical Feed and Sampling Systems to control the chemical composition of the steam generator shell water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator water.

The Watts Bar Steam Generator Blowdown System is designed to handle a blowdown rate in the range of 5 gpm to 65.5 gpm per steam generator. The actual blowdown flows required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. These required flow rates are not expected to be significantly impacted by the 1.4% power uprate, since neither the addition of dissolved solids nor the rate of addition of particulates into the steam generators will be significantly impacted by power uprate.

Since the proposed NSSS operating parameters permits a variation in full load steam pressure to 947 psia, the inlet pressure to the steam generator blowdown and sampling system can also vary accordingly. The Steam Generator Blowdown control valve has been reviewed for capacity of mass flow versus the available head and is found to be adequate for the designed blowdown flowrate at the reduced full load steam generator pressure.

5 NSSS COMPONENTS

5.1 REACTOR VESSEL

The reactor vessel (RV) was evaluated at the revised design conditions in two areas: the structural acceptability of the vessel, and the reactor vessel integrity in terms of the impact due to neutron fluence.

5.1.1 Reactor Vessel Structural Evaluation

This evaluation assesses the effects that the 1.4% uprating conditions have on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress reports and addenda. As noted in Section 3.1 of this report, the design transients did not require modification. However, the normal vessel outlet temperature increases from 618.7°F to 619.1°F - thereby increasing the T_{hot} variation in the outlet nozzles during normal plant loading and plant unloading. Therefore, the normal plant loading and plant unloading are considered to be more severe transients at the outlet nozzles.

The evaluation considers a worst case set of operating parameters from the current design basis parameters and the 1.4% uprate parameters, but only the 1.4% uprate vessel outlet temperature presents a more severe condition than the conditions considered in the design basis analysis. The vessel inlet temperature for the 1.4% uprate (557.3°F) resulted in a reduced temperature variation during normal plant loading (557°F to 557.3°F versus

557 °F to 557.7°F) and plant unloading for those regions of the reactor vessel assumed to be in contact with vessel inlet water during normal reactor operation. Thus, only the outlet nozzles required evaluation for the worst case operating parameter effects. All of the remaining regions (including the upper head, main closure, inlet nozzles, vessel shell and bottom head) are analyzed to be in contact with the vessel inlet water. Therefore, the current reactor vessel stress reports remain applicable for the limiting locations in these regions.

The reactor vessel outlet nozzles were evaluated for the effects of the increased T_{hot} variations during the normal plant loading and plant unloading transients. The evaluation results show that the maximum ranges of primary plus secondary stress intensity and maximum cumulative fatigue usage factors reported for the outlet nozzles are negligibly affected by the 0.4°F increase in the vessel outlet temperature. Therefore, all of the maximum ranges of stress intensity remain within their applicable acceptance criteria, and the maximum cumulative fatigue usage factors remain below 1.0 for the 1.4% uprate conditions.

5.1.2 Reactor Vessel Integrity - Neutron Irradiation

The revised design conditions in Table 2-1 can affect the analyses generally in two ways. One way is that changes in T_{COLD} may affect the value used in the various analysis methods. The second way is that a 1.4% increase in core power can increase the neutron fluences experienced by the vessel.

The current analyses assume that the T_{COLD} is maintained between 530 and 590°F. The T_{COLD} of 557.3°F for the 1.4% uprate (see Table 2-1), is between 530° and 590°F. Thus, the temperature assumption for the analysis is not affected.

With regard to the neutron flux that impinges on the reactor pressure vessel (RPV), the existing fast (E > 1.0 MeV) neutron fluence projections bound the corresponding projections for the 1.4% uprate conditions. The existing fast neutron RPV exposures were originally generated after the first operating cycle was completed. As a result, these projections were based on exposure rates which conservatively assume that "out-in" loading patterns are continually being utilized. In reality, however, low-leakage loading patterns have been used for Cycles 2 and 3 at Watts Bar. An evaluation was performed to demonstrate that transforming from 'out-in' to low-leakage loading patterns substantially outweighs the adverse impact of increasing the reactor core power by 1.4%. Consequently, the existing design-basis RPV exposure projections remain bounding. Thus, the fluence values used in the existing design bound those for the 1.4% uprating.

Surveillance Capsule Withdrawal Schedule

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel in order to effectively monitor the condition of the reactor vessel materials under actual operating conditions. Since the revised fluence projections do not exceed the fluence projections used in development of the current withdrawal schedules, then the current withdrawal schedules remain valid.

Heat-up and Cooldown Pressure - Temperature Limit Curves

A review of the current applicability dates of the heatup and cooldown curves for the pressure and temperature limits was performed. This review was accomplished by comparing the fluence projections used in the current calculation of the adjusted reference temperature (ART) for all the beltline materials in the reactor vessels to the fluence based on the revised design conditions.

Since the revised fluence projections do not exceed the fluence projections used in developing the ART values, then the heatup and cooldown curves remain valid and conservative for the given EFPY of the curves.

Pressurized Thermal Shock (PTS)

The RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates and beltline circumferential weld seams for end-of-life plant operation. The current RT_{PTS} values do not exceed the screening criteria of the PTS Rule. Since the fluence projections at the revised design conditions do not exceed the fluences used in developing the RT_{PTS} values, the existing RT_{PTS} values remain valid and conservative.

Emergency Response Guideline (ERG) Limits

The current peak inside surface RT_{NDT} values at EOL was calculated to be 253°F. Since the revised fluence projections after the power uprating will not exceed the fluence projections used in development of the current peak inside surface RT_{NDT} values at EOL, the RT_{NDT} values will remain in the ERG Pressure Temperature Limit Category I after uprating. Therefore, there is no need to change the current Emergency Response Guidelines for the 1.4% uprate conditions.

Upper Shelf Energy (USE)

An evaluation was performed to assess the impact of the revised conditions on the USE values for all reactor vessel beltline materials in the reactor vessel. Since the neutron fluences used in the vessel design have not been increased as a result of the revised design conditions, it is not necessary to decrease the USE at EOL conditions. Thus, the existing values remain applicable.

5.2 REACTOR INTERNALS

The reactor internals support and orient the fuel and control rod assemblies, absorb control rod assembly dynamic loads and transmit these and other loads to the reactor vessel. The internals also direct flows through the fuel assemblies, provide adequate cooling to various internals structures and support in-core instrumentation. The changes in the RCS temperatures, reported in Table 2-1, produce changes in the boundary conditions experienced by the reactor internals components. Also, increases in core power may increase nuclear heating rates in the lower core plate, upper core plate and baffle-barrel region. Several analyses have been performed to demonstrate that the reactor internals can perform their intended design functions at the revised design conditions.

5.2.1 Thermal and Hydraulic Evaluations

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. The principal core bypass flows are the baffle barrel region, vessel head cooling spray nozzles, vessel outlet nozzle gap, baffle plate cavity gap and thimble tubes. An analysis was performed to demonstrate that the core bypass flow remains less than the current design value, and thus is acceptable.

RCCA Drop Time Analyses

The Technical Specifications require that the RCCA drop time be less than or equal to 2.7 seconds. The revised design conditions, in particular the reduced T_{COLD}, can increase the drop time due to the increased fluid density. An evaluation was performed to confirm that the current value of 2.7 seconds could be met at the revised design conditions.

Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large diameter Belleville type spring of rectangular cross section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and upper internals flange, and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation demonstrated that the spring would maintain a net clamping force and that the reactor internals assembly would remain seated and stable at the revised design conditions.

Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high velocity jet of water. This jet is created by higher pressure water being forced through gaps between the baffle plates which surround the core. In order to minimize the propensity for flow induced vibration, the crossflow emanating from baffle joint gaps must be limited to a specific momentum flux, V^2h; that is, the product of the gap width, h, and the square of the baffle joint jet velocity, V^2. This momentum flux varies from point to point along the baffle plate due to changes in pressure differential across the plate and the local gap width variations. In addition, the modal response of the vibrating fuel rod must be considered. That is, a large value of local momentum flux impinging near a grid is much less effective in causing vibration than the same V^2h impinging near the mid span of a fuel rod. It was determined that operation at the revised design conditions would have a negligible affect on the momentum flux.

5.2.2 Mechanical Evaluations

The revised design conditions do not affect the current design bases for seismic and LOCA loads. Thus, it was not necessary to re-evaluate the structural affects from seismic OBE and SSE loads, and the LOCA hydraulic and dynamic loads. With regards to flow and pump induced vibration, the current analysis uses a mechanical design flow which did not change for the revised design conditions. The revised design conditions will slightly alter the T_{COLD} and T_{HOT} fluid densities which will slightly change the forces induced by flow. However, these changes are insignificant when compared to the current design temperature ranges. Thus, the revised design conditions do not affect the mechanical loads.

5.2.3 Structural Evaluations

Evaluations are performed to demonstrate that the structural integrity of the reactor components is not adversely affected by the change in RCS conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth which must be addressed in the design and analysis of various components. The core support structures affected by the revised design conditions (Table 2-1) are discussed in the following sections.

The primary inputs to the evaluations are the revised design conditions in Table 2-1 and the gamma heating rates. The gamma heating rates were modified, as required, to account for the associated 1.4% increase in core power.

Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel into which baffle plates are installed, supported by bolting interconnecting former plates to the baffle and core barrel. The baffle to former bolts restrain the motion of the baffle plates that surround the core. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, seismic loads, as well as secondary loads consisting of preload, and thermal loads resulting from RCS temperatures and gamma heating rates. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. In addition to providing structural restraint, the baffles also channel and direct coolant flow such that a coolable core geometry can be maintained.

The thermal stresses in the core barrel shell in the core active region are primarily due to temperature gradients through the thickness of the core barrel shell. These temperature gradients are caused by the fluid temperatures between the inside and outside surfaces and the contribution of gamma heating.

It was determined that the existing structural analysis was still bounding for the revised design conditions since the gamma heating rates and thermal assumptions used in the present analysis bound those for the 1.4% uprate conditions. Thus, these components are structurally adequate for the revised design conditions.

Lower Core Plate Structural Analysis

The lower core plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid flow through the plate. The fluid flow is provided to each fuel assembly and the baffle-barrel region flow exits through the holes. The plate is bolted at the periphery to a ring welded to the inside diameter of the core barrel. The center span of the plate is supported by the lower support columns which are attached at the lower end to the lower support plate.

Temperature differences between components of the lower support assembly induce thermal stresses in the lower core plate. In addition, due to the lower core plate's proximity to the core and thermal expansion of fuel rods at power, the heat generation rates in the lower core plate due to gamma heating cause a significant temperature increase in this component. Thermal expansion of the lower core plate is restricted by the lower support columns, lower support plate and core barrel. These restraining items are exposed to the inlet temperature and have heat generation rates much lower than those found in the lower core plate.

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plate is not adversely affected by the revised design conditions. These evaluations conclude that the fatigue usage is still less than 1.0 and the plate is structurally adequate at the revised design conditions.

Upper Core Plate Structural Analysis

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, thus serving as the transitioning member for the control rods in entry and retraction from the fuel assemblies. It also controls coolant flow in its exit from the fuel assemblies and serves as a boundary between the core and the exit plenum. The upper core plate is restrained from vertical movement by the upper support columns which are attached to the upper support plate assembly. The lateral movement is restrained by four equally spaced core plate alignment pins.

The normal and upset stresses in the upper core plate are mainly due to hydraulic, seismic, and thermal loads. The total thermal stresses are due to secondary membrane stress and surface skin stress. Evaluations were performed to determine the impact that the revised design conditions had on the structural integrity of the upper core plate. As a result of the evaluation, it was concluded that the fatigue usage is still less than 1.0 and the plate is structurally adequate for the revised design conditions.

5.3 CONTROL ROD DRIVE MECHANISMS (CRDMS)

The pressure boundary portions of the CRDMs are exposed to the vessel/core inlet fluid. The conditions in Table 2-1 indicate that the maximum decrease in vessel/core inlet temperature was from 557.7°F to 557.3°F. An analysis was performed to determine the impact that the revised design conditions had on the stress and fatigue usage of the CRDM components. The results indicate that the stress and fatigue usage are still within applicable limits.

5.4 REACTOR COOLANT LOOP PIPING AND SUPPORTS

The revised design conditions were reviewed for impact on the existing design basis analysis for the reactor coolant loop piping, primary equipment nozzles, primary equipment supports and the pressurizer surge line piping. The temperature changes associated with the revised conditions can cause potential changes in the loads in the components to be reconciled. Also, the changes in temperatures can potentially impact the applicable fatigue usage values since these temperature are used as initial conditions in some of the design transients.

The evaluation of the reactor coolant loop piping, the primary equipment nozzles, the primary equipment supports and the pressurizer surge line piping indicated that the potential increase in loads was bounded by the loads in the existing analyses. Also, the stresses and fatigue usage values for the reactor coolant loop piping, the primary equipment nozzles and the pressurizer surge line piping are unaffected due to the conservative nature of the analyses (e.g. - conservative grouping of more severe transients). Thus, there were no changes to the existing loads, critical locations, stresses and fatigue usage factors.

In addition, the current leak-before-break (LBB) evaluation was performed for the primary loop piping and pressurizer surge lines to provide technical justification for eliminating large primary loop pipe rupture as the structural design basis. In order to demonstrate the elimination of RCS primary loop and surge line pipe breaks, the following objectives were achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack which yields a detectable leak rate.

- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability.
- Demonstrate margin on applied load.
- Demonstrate that fatigue crack growth is negligible.

An evaluation confirmed that the revised design conditions had a negligible effect on these LBB conclusions.

5.5 REACTOR COOLANT PUMPS (RCPs)

5.5.1 STRUCTURAL ANALYSIS

The pressure boundary portions of the RCPs are exposed to the steam generator outlet fluid. Table 2-1 indicates that the steam generator outlet temperature decreased from 557.5°F to 557.0°F. An analysis was performed to determine the impact that the revised design conditions had on the stress and fatigue usage of the RCP components. The results indicate that the stress and fatigue usage are still within the applicable limits.

5.5.2 RCP Motor Analysis

The RCP motors were evaluated based on the revised design conditions for continuous operation at the hot loop rating and operation at the revised cold loop rating, as well as the design basis starting conditions and loads on thrust bearings. The evaluation confirmed that the revised design conditions have a negligible impact on these design bases, thus, the RCPs will continue to operate at their applicable hot and cold loop operating ratings. Also, the RCPs will be able to accelerate at the design basis starting conditions, and the thrust bearings will not exceed their load ratings.

5.6 STEAM GENERATORS

5.6.1 STRUCTURAL INTEGRITY

The bases for the existing structural and fatigue analyses of the steam generators are contained in the applicable D3 Steam Generator Stress Reports. An evaluation was performed to demonstrate that the ASME limits are still maintained at the revised design conditions. This evaluation considered the most critical components with regards to stress and fatigue usage. The primary input for the evaluation was the revised design conditions in Table 2-1. As noted in Section 3.1, it was not necessary to modify the secondary side design transients for the revised design conditions.

The evaluation considered these inputs by developing scaling factors needed to calculate the increased stress and fatigue usage. The results indicate that all applicable fatigue usage values are still less than the allowable limit of 1.0. Thus, the evaluation demonstrates that the steam generators meet the requirements of the ASME Code at the revised design conditions.

5.6.2 Thermal-Hydraulic Performance

The following evaluations and analyses were performed to assess the impact that the revised design conditions had on the thermal-hydraulic performance of the steam generators.

Circulation Ratio/Bundle Liquid Flow

The circulation ratio is a measure of tube bundle liquid flow in relation to steam flow and is primarily a function of steam flow. The bundle liquid flow minimizes the accumulation of contaminants on the tube sheet and in the bundle. At the revised design conditions, there is a slight reduction in the bundle flow which has a minimal affect on its function. Thus, the bundle flows are still adequate.

Hydro-dynamic Stability - Damping Factor

The hydrodynamic stability of a steam generator is characterized by a damping factor. A negative value of this parameter indicates a stable unit indicating that small perturbations of steam pressure or circulation ratio will diminish rather than grow in amplitude. An evaluation confirmed that the damping factor would still remain highly negative at the revised design conditions, and thus, the steam generators are expected to remain hydro-dynamically stable.

Secondary Side Pressure Losses

Evaluations were performed to demonstrate that the revised design conditions have a negligible affect on the steam line pressure losses. Thus, there is a negligible impact on the feedwater system operation.

5.6.3 Moisture Carryover

Evaluations were performed to predict the amount of moisture carryover from the steam generators at the revised design conditions. Increases in steam flow, and reductions in steam temperature and pressure can increase the moisture carryover. The evaluations concluded that the moisture carryover would still be acceptable at the revised design conditions.

5.6.4 U-Bend Fatigue Evaluation

An evaluation was performed to determine the impact that the revised design conditions had on the U-bend fatigue evaluation performed for the steam generators. The evaluation first assessed the steam flow and pressure, corresponding to the revised design conditions, for which the affected tubes would require plugging. It was determined that at the revised design conditions, no additional tubes would be required for plugging.

5.6.5 Evaluation of Steam Generator Tube Degradation Mechanisms

The revised design conditions (See Table 2-1) will have a negligible impact on the existing and potential tube degradation mechanisms. Table 2-1 indicates that the design T_{hot} is expected to increase by 0.4°F for the 1.4% uprate and is considered to be the most sensitive operating parameter with respect to corrosion. The primary system pressure of 2250 psia is unchanged, and the steam pressure is reduced and is considered to have a secondary effect on corrosion. These changes are expected to have an insignificant effect on the tube corrosion mechanisms since they are relatively minor and are comparable to the range of uncertainties used in assessing corrosion.

With regard to pre-heater wear, the 1.4% uprate conditions result in a slight increase in flow through the main feedwater nozzle which can impact the rate of wear. This slight increase in flow is not expected to result in a significant increase in the wear rate, and the resultant flow is within the pre-heater design flow. For anti-vibration bar (AVB) wear, the slightly increased steam flow and reduced steam pressure can impact the flow induced vibration and wear. The revised design conditions will have a negligible impact on the projected AVB wear rate. Thus, the 1.4% uprate does not significantly impact the present AVB wear.

5.6.6 Tube Plugging and Repair Criteria

A preliminary assessment has been performed to confirm that the existing 40% through wall plugging criteria will remain adequate for the 1.4% uprate conditions. The present design calculation is based on the original plant design conditions. The 1.4% uprate conditions result in a reduced steam pressure which is more limiting. The present calculation is judged to have sufficient margin to accommodate the design reduction in steam pressure and still adequately account for continued tube degradation and eddy current uncertainties such that the 40% through wall plugging limit will remain valid. A subsequent calculation will be performed to further substantiate this judgment.

5.6.7 Evaluation of ARC for F* and ODSCC at TSP Intersections

TVA recently submitted to the NRC a request to implement the F* alternate repair criteria (ARC) for the roll expanded portion of the tube within the tubesheet. An additional request was made to implement a voltage based repair criteria for tubes affected by ODSCC at tube support plate intersections. These requests are included in TVA Letters to NRC: dated April 10, 2000, "Watts Bar Nuclear Plant - Unit 1 Tech. Spec. Change No. WBN TS-99-013-Alternate Steam Generator Tubesheet Region Plugging Criterion (F*)" and "Watts Bar Nuclear Plant - Unit 1 Tech. Spec. Change No. WBN TS-99-014-Steam Generator Alternate Repair Criteria for Axial Outside Diameter Stress Corrosion Cracking (ODSCC)".

A review of the supporting calculations for these transmittals confirmed that the F* length and voltage based repair criteria remain applicable for the 1.4% uprate conditions. The F* criterion is based on a primary to secondary differential pressure of 1400 psi. Table 2-1 indicates that the 1.4% uprate conditions result in a 1303 psi (2250-947 psi) differential. This pressure differential is bounded by the 1400 psi and does not result in a change to the F* length, and the present submittal remains valid for the 1.4% uprate conditions.

The ODSCC ARC was developed to replace the application of the generic 40% depth plugging criterion for tube cracking at elevations corresponding to tube support plate intersections. The loading conditions compared to applicable criteria are only operative during faulted conditions, since the tube degradation is confined to the tube/tube support plate intersection crevice during normal operation. Hence, a slight increase in the normal operation primary to secondary differential pressure is inconsequential in relation to the ARC. The structural and leakage criteria do apply during the application of faulted loading conditions, however, these are unaffected by the 1.4% uprate. Thus, the 1.4% uprate should have no affect on the application of the ARC.

5.7 PRESSURIZER

The limiting locations from a structural standpoint on the pressurizer are the surge nozzle, the spray nozzle, and the upper shell at the point of spray impingement. The limiting operating condition of the pressurizer occurs when the RCS pressure is high and the RCS hot leg temperature (T_{HOT}) and cold leg temperature (T_{COLD}) are low. The pressurizer structural evaluation was performed by comparing the key inputs in the current pressurizer stress report with the revised design conditions in Table 2-1. The results indicate that the design conditions used in the current analysis are still bounding for the revised design conditions. Thus, there is no impact on the current analysis.

5.8 NSSS AUXILIARY EQUIPMENT

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves and tanks. It was determined that the existing design values used in the fatigue analysis for these components envelop those reported in Table 2-1. Also, as noted in Section 3.2, the current auxiliary equipment design transients remain applicable for the 1.4% conditions. Thus, the components will continue to meet their current design criteria since the fatigue usage values for each component will still be less than the allowable limit of 1.0.

6.0 NSSS ACCIDENT ANALYSES

6.1 STEAM GENERATOR TUBE RUPTURE (SGTR) EVALUATION - (FSAR SECTION 15.4)

The SGTR analysis includes an analysis to demonstrate margin to steam generator overfill and a thermal and hydraulic analysis to ensure that the offsite radiation doses resulting from the event are less than the allowable values specified in Standard Review Plan 15.6.3 and 10 CFR 100. Radiological consequences of an SGTR are discussed in Section 9.

Margin to Overfill Evaluation

The margin to overfill analysis is performed to demonstrate that the ruptured steam generator does not overfill before the primary-to-secondary break flow is terminated. The purpose of this analysis is to ensure that the steam generator does not overfill and that liquid does not enter the steam generator exit nozzle and affect the main steam line and associated piping supports; and there is no liquid discharge through the MSSVs or PORVs.

The licensing basis analysis for the SGTR margin to overfill event is initiated from 100% power. A 2% power measurement uncertainty is applied in the actual calculation to increase the initial power level to 102%. Consistent with the discussion provided in Section 0.2, the improved thermal power measurement accuracy obviates the need for the full 2% power measurement margin assumed in the analysis.

Furthermore, the power measurement margin is but one of many conservative assumptions used in the analysis. For example, the analysis uses an initial steam generator mass which is higher than the actual initial mass, thereby decreasing the initial margin to overfill. Also, the analysis applies 120% of the ANS 1971 decay heat curve to maximize the decay heat, thereby maximizing the primary to secondary leakage and decreasing the margin to overfill. Taken together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the margin of safety would not be reduced.

6.2 STEAM LINE BREAK EVALUATION

Several steam line break (SLB) and feedline break (FLB) analyses are performed to support the plant design. A SLB analysis is conducted for a break inside containment to determine the containment pressure and temperature response. Another SLB analysis is conducted for a break outside containment to assess the environmental qualification of safety-related equipment. Short-term SLB and FLB analyses are performed to assess the pressurization of compartments and subcompartments located inside and outside containment. Radiological consequences are discussed in Section 9.

SLB Mass and Energy Release Inside and Outside Containment

SLB mass and energy releases inside containment are used as input boundary conditions to a containment analysis performed to determine the transient pressure and temperature response. SLB mass and energy releases outside containment are used as input boundary conditions to an environmental qualification of safety-related equipment and instrumentation.

The licensing basis analyses for these events is initiated from 100% power and applies a 2% power measurement uncertainty to increase the initial power level to 102%. Consistent with the discussion provided in Section 0.2, the improved thermal power measurement accuracy obviates the need for the full 2% power margin assumed in the analysis.

In addition, the present analysis of record uses an initial feedwater enthalpy based on a feedwater temperature of 444°F which bounds the design value of 441.8°F identified in Table 2-1. Thus, the analysis is unaffected by the small increase in feedwater temperature from 440 to 441.8°F.

Furthermore, the power measurement margin is but one of many conservative assumptions used in the analysis to maximize the quantity of released mass and energy. For example, the analysis assumes the use of reactivity feedback characteristics, including minimum shutdown margin, to promote a conservatively higher "return to power." Also, the initial and trip values for the steam generator water mass, as well as the main feedwater and auxiliary feedwater flows, are selected to conservatively maximize secondary side energy available for heat transfer during an inside containment release, and conservatively increase the propensity for superheating during an outside containment release. Taken together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the margin of safety would not be reduced.

Feedline Break (FLB) in the Main Steam Valve Vault (MSVV)

The analysis for a double ended FLB in the Main Steam Valve Vaults (MSVV) has been reviewed to determine if there are any impacts from a 1.4% uprate. The results of FLB analysis are used to establish the maximum flooding level in the MSVVs. A break location upstream of the FIVs within the MSVV has previously been found to produce the highest flood level. An initial feedwater temperature of 425°F was conservatively assumed to maximize the fluid density and thus the mass release from the break. Since the final feedwater temperature will increase 1.8°F with a 1.4% uprate, this assumption becomes even more conservative. Flow and pressure conditions within the feedwater system piping were initialized at 100% power. The 1.4% uprate is accompanied by a lower main steam pressure (minimum 11 psid, 958 psia - 947 psia). The feedwater pump discharge pressure is set by the main feed pump turbine (MFPT) speed program. This program maintains a differential pressure between the main steam and feedwater headers by adjusting MFPT speed (i.e., by throttling inlet steam to the MFPT). This calculated differential pressure will only increase slightly (~2 psi) due to the 1.4% uprate increase in steam flow. The overall effect of a 1.4% uprate would result in a lower feedwater header pressure. Thus the initial pressure condition at the break location would be decreased and the mass release would be bounded by the current analysis of record.

Short Term SLB and FLB Mass and Energy Releases

Short-term SLB mass and energy releases are used as input to a compartment or subcompartment pressurization analysis, inside or outside containment. The analytical method for the short-term SLB mass and energy releases is a hand calculation in which the steam system inventory is released outside a control volume over a short duration. The critical analysis parameters are defined at no-load conditions to maximize the short-term release (e.g., higher initial steam generator pressure). These no-load conditions are unaffected by the 1.4% uprate and thus the existing releases remain bounding.

Short-term FLB mass and energy releases are used as input to a compartment or sub-compartment pressurization analysis. The analytical method for the short-term FLB mass and energy releases is a calculation in which the main feedwater system inventory is released outside a control volume over a short duration. The only thermal-hydraulic related inputs to the mass and energy release calculation are the no-load steam generator pressure (~1100 psia) and the saturation pressure of the main feedwater at full power (temperature = 440°F). Table 2-1 indicates that the 1.4% uprate resulted in a 1.8°F increase in feedwater temperature from 440 to 441.8°F. The no-load steam generator pressure was not affected. The 1.8°F increase in feedwater temperature would result in a slight increase in the mass and energy which was bounded by releases used in the applicable sub-compartment analysis.

6.3 LOCA MASS AND ENERGY RELEASES

Long-Term LOCA/Containment Integrity Analysis

This analysis demonstrates the ability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA. The analysis of record presently assumes an NSSS thermal power of 3579 MWt which is about 4.4% greater than current NSSS power of 3427 MWt). In addition, the analysis applies an extra 2% power measurement uncertainly to the 3579 MWt value to account for power measurement uncertainty. Consistent with the discussion provided in Section 0.1, the improved thermal power measurement accuracy obviates the need for the full 2% power margin assumed in the analysis.

The power measurement margin is but one of many conservative assumptions used in the analysis. For example, as noted above, a key assumption is the assumed initial power level of 3579 MWt, which is about 4.4% higher than the present thermal power. Taken together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the margin of safety would not be reduced.

Short-Term LOCA Mass and Energy Release Analysis

Several evaluations are performed to support the loop subcompartment, reactor cavity and pressurizer enclosure analysis. The analysis inputs that may potentially change with the uprate are the initial RCS fluid temperatures. Since this event lasts for approximately 3 seconds, the single effect of power is not significant.

The short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses. This correlation was used to conservatively evaluate the impact of the changes in the RCS inlet and outlet temperatures from the 1.4% uprate relative to those used in the current analysis of record. The use of the lower temperatures (See Table 2-1) maximizes the critical mass flux in the Zaloudek correlation.

Loop Subcompartment Analysis

The loop subcompartment analysis is performed to ensure that the walls of the loop subcompartments, including the lower crane wall, upper crane wall, operating deck, and the containment shell, can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) which accompanies a LOCA. Also, this analysis verifies the adequacy of the ice condenser performance.

A vessel outlet temperature of 617.3°F and a vessel/core inlet temperature of 557.7°F, both conservatively bounded low for short-term considerations, were evaluated for the 10% SGTP program performed in 1997. (See TVA Letter to NRC dated March 27, 1997, "Watts Bar Nuclear Plant (WBN) - Proposed License Amendment - Cycle 2 Core Reload Changes - Technical Specification Change No. 96-013" and NRC's related September 11, 1997 issuance of WBN Unit 1 Tech Spec Amendment No. 7) It was concluded that the short-term releases, for the changes identified in the 10% SGTP submittal, could increase by approximately 20% when compared to the original plant design conditions. However, the basis for this 20% increase was a previous evaluation for the turbine volumetric flow test which evaluated a more limiting set of initial RCS conditions that included a vessel outlet temperature of 607.3°F and a vessel/core inlet temperature of 549.1°F. These values bound the 1.4% uprate program values of 619.1°F and 557.3°F, respectively.

Furthermore, based upon a comparison of the RCS temperature conditions evaluated for the 10% SGTP Program and the 1.4% Uprate program values, it is judged that 15% of the 20% increase can be associated with the 1.4% Uprate Program, the remaining 5% could be considered excess margin. Thus, for the 1.4% uprate program, the releases are only expected to increase by 15% when compared to the original plant design conditions.

The Transient Mass Distribution (TMD) program described in Section 6.2.1.3.4 of the Watts Bar FSAR was used for the current licensing basis subcompartment analysis. There are margins in the current subcompartment calculations that would offset the predicted 15% increase in mass and energy releases. For example, splitting the break flow into the TMD elements on both sides of the break, as opposed to assuming all flow goes to one element, would offset the increase. Thus, the current licensing basis mass and energy releases remain bounding.

Reactor Cavity Analysis

The reactor cavity analysis is performed to ensure that the walls in the immediate proximity of the reactor vessel can maintain their structural integrity during the short pressure pulse which accompanies a LOCA within the reactor cavity region. Loadings on the reactor vessel are also determined.

The 127 sq. in. reactor vessel inlet break is presently used in the existing sub-compartment calculation. As discussed above, it was estimated that the peak releases would conservatively increase by approximately 15% for the 1.4% uprate program. However, based upon results of the structural analysis of the reactor coolant system, a better estimate of the break size is 45 sq. in. The reduced rates from this reduced break size more than offset the predicted 15% increase. For example, the releases are approximately proportional to the break size, and as such, the releases would be reduced by a factor of (127/45 = 2.8). This excess margin more than offsets the predicted increase. Since the current mass and energy releases remain bounding, the current reactor cavity pressure analysis remains bounding. The margin of safety would not be reduced.

Pressurizer Enclosure Analysis

The pressurizer enclosure analysis is performed to ensure that the walls in the immediate proximity of the pressurizer enclosure can maintain their structural integrity. Loadings acting across the pressurizer are also determined.

The current licensing basis pipe break is a severance in the spray line. Comparing the pipe size assumed in the current analysis versus the as-built piping, the margin in the releases just due to the currently assumed break size is greater than 25%. The break sizes used in the current analysis are 0.1963 ft^2 for the cold leg spray nozzle and 0.08727 ft^2 for the pressurizer spray nozzle. The as-built break sizes are 0.1469 ft^2 for the cold leg spray nozzle and 0.06447 ft^2 for the pressurizer spray nozzle. The difference in break sizes leads to greater than 25% margin in the mass and energy releases. This more than offsets the predicted 15% increase in mass and energy releases. Therefore, the current mass and energy releases remain bounding, and the current pressurizer enclosure pressure analysis remains bounding. The margin of safety would not be reduced.

Maximum Reverse Pressure Differential Analysis

Following a LOCA, the pressure and temperature in the lower compartment of containment increases which forces the air in the lower compartment into the upper compartment and increases the pressure in the upper compartment. As the temperature in the lower compartment decreases with time, the pressure in the lower compartment also decreases. Eventually the pressure in the lower compartment becomes less than the pressure in the upper compartment, which creates a reverse differential pressure across the operating deck. This analysis is used to predict this reverse differential pressure and to ensure the structural adequacy of the operating deck.

The analysis of record is a generic and conservative analysis discussed in FSAR Section 6.2.1.3.11. The dead-ended compartments adjacent to the lower compartment are assumed to be swept of air during the initial blowdown. This is a very conservative assumption, since this will maximize the air forced into the upper ice bed and upper compartment thus raising the compression pressure for the operating deck. In addition, it will minimize the noncondensables in the lower compartment.

The mass and energy releases utilized serve only as a vehicle to initiate the event and to purge the lower and the dead-ended compartment air. Any increases in releases during the post-blowdown period would result in the lower compartment pressure remaining at a higher value, and thus would reduce the reverse differential pressure. The mass and energy releases are extracted from a model used to maximize the LOCA PCT and not from a model used to maximize the peak containment pressure. It is judged that the RCS temperature changes and the resulting effects would not affect the results of the maximum reverse pressure differential calculation.

The purpose of this analysis is to show that significant margin exists in the design. The existing peak calculated differential pressure of 0.65 psi is significantly lower than the structural design and load carrying capability of the operating deck. Thus the 1.4% uprating will have a minimal impact, if any, on the analysis and there is significant analysis margin available. The current analysis of record remains bounding, and the margin of safety would not be reduced.

6.4 LOCA RELATED ANALYSES

Best Estimate LBLOCA Analysis (BELBLOCA)

The current licensing basis BELBLOCA analysis employs a nominal core power of 3411 MWt. A study was performed for the 1.4% increase in core power to 3459 MWt.

The study was performed by use of the MONTEC computer code. The PCT impact due to the 1.4% uprate is a penalty of 12 °F which was expected since an increase in power level is expected to increase the PCT; and due to the magnitude of the power increase, the PCT impact was expected to be small. The resultant PCT value of 1773 °F is still well below the 2200 °F limit.

Small Break LOCA (SBLOCA)

A SBLOCA analysis is performed to demonstrate the peak clad temperature (PCT) during the SBLOCA is less than the 10 CFR 50.46 limit of 2200°F. As discussed in Section I of this report, the WBN licensing basis analysis methodology employs a 2% calorimetric uncertainty for reactor power level in accordance with the original requirements of 10 CFR 50, Appendix K (ECCS Evaluation Models). However, with the NRC's recent approval of the change to the requirements of Appendix K, TVA is proposing a relaxation of the 2% power uncertainty margin based on the planned use of the improved Caldon LEFM instrumentation to determine core power level with a power measurement uncertainty of less than 0.6%. Thus, consistent with Section III.0.2 of this report, the current WBN SBLOCA analysis of record does not require re-performance for the 1.4% power uprate. Instead, the existing 2% uncertainty margin for the ECCS analysis may be reduced such that 1.4% is allocated to provide sufficient margin to address the 1.4% uprate to 3459 MWt and 0.6% is retained in the analysis to still account for the power measurement uncertainty.

Blowdown Reactor Vessel and Loop Forces

The purpose of a LOCA hydraulic forces analysis is to generate the hydraulic forcing functions and hydraulic loads that occur on Reactor Coolant System (RCS) components as a result of a postulated loss-of-coolant accident (LOCA). These forcing functions and loads are considered in the structural design of the NSSS components. The hydraulic forcing functions and loads that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. An evaluation was performed to demonstrate that the current LOCA hydraulic forcing functions remain applicable for the 1.4% uprate conditions in Table 2-1. In general, LOCA hydraulic forces increase with an increase in RCS coolant density and consequently, LOCA hydraulic forces increase with lower RCS temperatures. Additionally, Westinghouse has historically determined that cold leg breaks yield more limiting results. Therefore, the reduction in cold leg temperature was evaluated with respect to the impact on hydraulic forces. The temperature change between the current analysis and the value reported in Table 2-1 was estimated to increase the LOCA forces (due to the density increase) by less than 2% of their current values. The frequency of the LOCA forces pressure transient remained essentially unchanged and the amplitude of the LOCA decompression wave remained smaller when compared to the current values.

The 2% increase in forces was offset by a more representative characterization of the loop at the break location. This approach results in about a 17% decrease in loop force at the break location. Thus, the current LOCA forces remain bounding.

Post-LOCA Long-Term Core Cooling (LTCC)

The Westinghouse licensing position for satisfying the requirements of 10CFR50.46, Paragraph (b), Item (5), "Long-term cooling," concludes that the reactor will remain shut down by borated ECCS water residing in the RCS/sump following a LOCA. Since credit for the control rods is not taken for large break LOCA, the borated ECCS water provided by the RWST and accumulators must have a concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady-state conditions at the initiation of a LOCA and considers sources of both borated and unborated fluid in the post-LOCA containment sump. The other sources of water considered in the calculation of the sump boron concentration are the RCS, ECCS/RHR piping, the boron injection tank (BIT) and piping and ice condenser inventory. The water volumes and associated boric acid concentrations are not directly affected by the 1.4% power uprate. The Cycle specific core re-load licensing process provides confirmation that these volumes and concentrations are adequate. Thus, there is no impact on the LTCC analysis.

Hot Leg Switchover

For a cold leg break post-LOCA, ECCS injection into the cold leg will circulate around the top of the full downcomer and out the broken cold leg. Flow stagnation in the core and the boiling off of near pure water will increase the boron concentration of the remaining water. As the boron concentration increases, the boron will eventually precipitate and potentially inhibit core cooling. Thus, at a designated time after a LOCA, the ECCS configuration is switched to hot leg injection to flush the core with water and keep the boron concentration below the precipitation point.

As with SBLOCA, the licensing basis analysis methodology for Hot Leg Switchover employs a 2% calorimetric uncertainty for reactor power level in accordance with the original requirements of 10 CFR 50, Appendix K. Therefore, consistent with Section 0.2 of this report and in the same manner as SBLOCA, the current WBN HL Switchover analysis does not require re-performance for the 1.4% power uprate. Instead, the existing 2% uncertainty margin for the ECCS analysis may be reduced such that 1.4% is allocated to provide sufficient margin to address the 1.4% uprate to 3459 MWt and 0.6% is retained in the analysis to still account for the power measurement uncertainty.

6.5 NON-LOCA/TRANSIENT ANALYSES

6.5.1 Non-LOCA/Transient Analyses Performed With Statistical Methods

Initial Power Conditions Assumed in the Safety Analyses (FSAR Section 15.1.2)

The uncertainties in initial operating conditions (i.e., power, flow, temperature and pressure) are not explicitly included in the transient assessment part of the DNB-related analyses which use the RTDP methodology. However, these uncertainties are accounted for in the calculation of the core design evaluation of the DNBR safety analysis limit. Also, the uncertainties are applied to the applicable accident analyses which are not analyzed to investigate the minimum DNBR response.

The only uncertainty modified as a result of using the LEFM is the power measurement uncertainty which now is ± 0.6% as noted in Section 6.7 of this report. All of the other uncertainties (i.e. average RCS temperature, pressurizer pressure and RCS flow) did not need to be modified.

The effect of the revised power measurement uncertainty has been accounted for in the analysis/evaluation of the various non-LOCA accidents discussed below. For the analyses which utilize the RTDP method for the calculation of the minimum DNBR, these uncertainties are accounted for in the minimum DNBR safety analysis limit rather than being accounted for explicitly in the analyses.

Trip Points and Time Delays to Trip Assumed in Accident Analyses (FSAR Section 15.1.3)

Based on the 1.4% increased core power, a revised set of core thermal limits was prepared using the RTDP method. It was not necessary to change the DNB design basis since existing analysis margin was used to offset the reduction in margin from the increased core power.

Using the revised set of core thermal limits, it was determined that the OTΔT and OPΔT setpoints did not need to be modified to accommodate the increased core power.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (FSAR Section 15.2.1)

This accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of one or more RCCA banks, resulting in a rapid power excursion. This transient is promptly terminated by a reactor trip on the Power Range High Neutron Flux-low setpoint. Due to the inherent thermal lag in the fuel pellet, heat transfer to the RCS is relatively slow, and the goal of the analysis is to demonstrate that the minimum DNBR is above the limit value.

The analysis is performed at hot zero power conditions. However, since the existing neutron high flux setpoint (credited in the analysis) as a percentage of full power stays the same and the full power assumed in the analysis increases, the overall neutron flux trip value increases in terms of the absolute power. Thus, an evaluation was performed to confirm that this small change in the power level at trip has a negligible impact on the analysis results. It was found that the power increase is very rapid during the transient, such that the small power increase would have a negligible effect on the trip time. Thus, the DNB design basis and limiting maximum fuel centerline temperature remain unaffected and the conclusions documented in the FSAR remain valid.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.2.2)

This event is defined as an inadvertent addition of reactivity to the core caused by the withdrawal of RCCA banks when the core is above the no-load condition. The event is analyzed at 10%, 60% and 100% of rated thermal power assuming beginning-of-life and end-of-life reactivity conditions and a spectrum of reactivity insertion rates. Unless terminated by manual or automatic action, the power mismatch between the reactor core power generation and the steam generator heat extraction results in a coolant temperature increase that could potentially lead to a departure from nucleate boiling. Therefore, in order to prevent damage to the fuel clad, the reactor protection system is designed to terminate the transient before the DNB limit is violated.

The present DNB limiting case is the 60% power event. An evaluation was performed to confirm that this is still the limiting case. In addition, an evaluation was performed at this limiting case to confirm that the impact of the 1.4% increased core power had a negligible effect on the analysis results. This was primarily performed to confirm that the existing high neutron flux safety analysis limit of 118% was still adequate for the increased core power. It was found that the power increase is very rapid during the transient, such that the small power increase would have a negligible effect on the trip time. Thus, the results of this analysis were found to be acceptable and the conclusions documented in the FSAR remain valid.

Rod Cluster Control Assembly Misalignment (FSAR Section 15.2.3)

The Rod Cluster Control Assembly (RCCA) Misalignment analysis includes the following events:

- One or more dropped RCCAs within the same group
- A dropped RCCA bank
- Statically misaligned RCCA

These transients are investigated to demonstrate that the DNB design basis remains met. An evaluation was performed to confirm that the existing statepoints used in the current analysis are still bounding for the 1.4% uprate conditions. Also, it was confirmed that there is sufficient DNB margin to accommodate the 1.4% power increase. Thus, the results of this analysis were found to be acceptable, and the conclusions documented in the FSAR remain valid.

Partial and Complete Loss of Forced Reactor Coolant Flow (FSAR Sections 15.2.5 and 15.3.4)

The partial/complete loss of forced reactor coolant flow events may result from mechanical or electrical failure(s) in the

Reactor Coolant Pump(s) (RCPs) which may occur from an undervoltage condition in the electrical supply to the RCPs or from a reduction in motor supply frequency to the RCPs due to a frequency disturbance on the power grid. These analyses demonstrate that the minimum DNBR remains above the limit value. The limiting results are obtained at full power conditions and occur very quickly following initiation of the event.

Since the 1.4% increase in core power has an adverse effect on the minimum DNBR, this accident has been reanalyzed. The analysis results show that the DNB design basis was satisfied for each case. The results were therefore found to be acceptable, and the conclusions documented in the FSAR remain valid.

Loss of External Electrical Load and/or Turbine Trip (FSAR Section 15.2.7)

This event is defined as a complete loss of steam load from full power without a direct reactor trip, or a turbine trip with or without a direct reactor trip and is analyzed to demonstrate 1) that primary and secondary pressures remain below 110% of design and 2) that the minimum DNBR remains above the safety analysis limit value. The impact that the 1.4% increased core power has on the primary and secondary pressures is discussed below in Section 6.5.2.

The Loss of Load/Turbine Trip analysis includes cases both with and without automatic pressure control. Although cases have historically been analyzed with both minimum and maximum reactivity feedback conditions, this accident, as a heatup event, is limiting at minimum feedback conditions. Maximum feedback cases are bounded by the minimum feedback cases and therefore do not need to be separately addressed. The case with pressure control is analyzed to investigate the RCS heatup effect on the DNBR response.

An evaluation was performed to confirm that the existing results used in the current analysis are still bounding for the 1.4% uprate conditions. Thus, the results of this analysis were found to be acceptable, and the conclusions documented in the FSAR remain valid.

Excessive Heat Removal Due to Feedwater System Malfunctions (FSAR Section 15.2.10)

Reductions in the feedwater temperature or additions of large amounts of feedwater to the steam generators result in excessive heat removal from the plant primary coolant system. Analyses are performed under both full power and no-load conditions to demonstrate that the DNB design basis is met. Both single loop and multiple loop malfunctions are considered, as well as operation with both manual and automatic rod control.
An evaluation was performed to confirm that the existing results used in the current analysis are still bounding for the 1.4% uprate conditions. Thus, the results of this analysis were found to be acceptable, and the conclusions documented in the FSAR remain valid.

Accidental Depressurization of the Reactor Coolant System (FSAR Section 15.2.12)

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve and is analyzed under full-power conditions to determine the minimum DNBR. An evaluation was performed to confirm that the existing results in the current analysis are still bounding for the 1.4% uprate conditions. The results of this analysis were found to be acceptable, and the conclusions documented in the FSAR remain valid.

Inadvertent Operation of Emergency Core Cooling System (FSAR Section 15.2.14)

This analysis assumes that the safety injection system is inadvertently actuated. Two separate cases are considered for this event. A case that assumes no reactor trip as a result of ECCS actuation is investigated to verify that the DNBR safety limits are not violated. Reactor trip is eventually provided by the Low Pressurizer Pressure function; neither the OTΔT function nor the OPΔT function is credited.

A case is also analyzed to investigate the potential for pressurizer filling due to continued ECCS injection and reactor coolant expansion resulting from residual heat generation. This case assumes a reactor trip coincident with event initiation and is addressed below in Section 6.5.2.

The case analyzed for minimum DNBR demonstrates that the most limiting condition occurs at event initiation and that the DNB ratio increases from that point. Thus, the event is generally not DNB limiting. An evaluation was performed to confirm that the existing results in the current analysis are still bounding for the 1.4% uprate conditions. The results of this analysis were found to be acceptable, and the conclusions documented in the FSAR remain valid.

Single Reactor Coolant Pump Locked Rotor (FSAR Section 15.4.4)

A single Reactor Coolant Pump (RCP) locked rotor event is based on the sudden seizure of a RCP impeller or failure of the RCP shaft. The analysis includes a RCS pressure and fuel rod temperature transient evaluation. The impact on the RCS pressure is discussed below in Section 6.5.2. A reactor trip via the Low RCS Flow protection function terminates this event very quickly.

Since the 1.4% increase in core power has an adverse effect on the minimum DNBR, this accident has been reanalyzed. The analysis results show that the DNB design basis was satisfied for each case, and the number of rods that undergo DNB is less than the current limit. The results were therefore found to be acceptable, and the conclusions documented in the FSAR remain valid.

Steamline Break at Power with Coincidental Rod Withdrawal

Although not a part of the Watts Bar FSAR, TVA requested that Westinghouse perform a special steamline break core response analysis for Watts Bar with the assumption of coincidental Rod Cluster Control Assembly withdrawal due to exposure of the turbine impulse transmitters or the excore detector equipment to an adverse environment. This event is simulated by modeling a steamline rupture occurring at full power conditions with a coincident withdrawal of RCCA Bank D. The cases were analyzed assuming a range of steamline break sizes to determine the limiting case with respect to minimum DNBR.

This accident has been reanalyzed for the 1.4% uprate conditions. Additional cases were analyzed to confirm that the maximum linear heat generation rate was acceptable. Also, evaluations were performed to confirm that the DNB design basis was met. Since the DNB design basis was met and the maximum heat generation rate at the hot spot did not exceed that which would cause fuel melt, the results of this analysis were found to be acceptable.

Excessive Load Increase Incident (FSAR Section 15.2.11)

This transient is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. Cases are evaluated at beginning-of-life and end-of-life conditions with and without rod control to demonstrate that the DNB design basis is met.
Also, the analysis is not sensitive to the changes in the initial design steam flow, temperature and pressure identified in Table 2-1.
A comparison of the plant conditions assuming conservatively bounding deviations in core power, average coolant temperature, and RCS pressure to the conditions corresponding to those required to exceed the core thermal limits indicates that the minimum DNBR remains above the limit value for all cases. Therefore, the conclusions documented in the FSAR remain valid.

Single Rod Cluster Control Assembly Withdrawal at Full Power (FSAR Section 15.3.6)

In terms of overall system transient response, this event is similar to that presented in the Uncontrolled RCCA Bank Withdrawal at Power event, except that local power peaking in the area of the withdrawn RCCA results in a lower minimum DNBR. The analysis credits a reactor trip on OTΔT and shows that less than 5% of the fuel rods would be expected to experience a DNBR less than the limit value.

This event is addressed as a part of the Cycle 4 core re-load process to confirm that less than 5% of the fuel rods are expected to experience a DNBR less than the limit value. Therefore, the conclusions of the FSAR will remain valid.

6.5.2 Non-LOCA/Transient Analyses Employing a 2% Calorimetric Uncertainty

The following Non-LOCA/Transient analyses are currently analyzed with an explicit 2% power measurement uncertainty to increase the initial power level to 102%. This explicit 2% power uncertainty bounds the 1.4% power uprate since the power uncertainty has been reduced to 0.6%.

- Startup of an Inactive Reactor Coolant Loop (FSAR Section 15.2.6)
- Loss of External Electrical Load and/or Turbine Trip - overpressure analysis (FSAR Section 15.2.7)
- Loss of Normal Feedwater (FSAR Section 15.2.8)
- Inadvertent Operation of Emergency Core Cooling System - Overfill Analysis (FSAR Section 15.2.14)
- Major Rupture of a Main Feedwater Pipe - 102 % power case (FSAR Section 15.4.2.2)
- Single Reactor Coolant Pump Locked Rotor - overpressure, maximum clad temperature, and maximum zirconium-water reaction analysis (FSAR Section 15.4.4)
- Rupture of a Control Rod Drive Mechanism Housing - 102% Power Cases (FSAR Section 15.4.6)

Consistent with the discussion provided in Section 0.2, the improved thermal power measurement accuracy obviates the need for the full 2% power uncertainty assumed in the analysis. Furthermore, the power measurement margin is but one of many conservative assumptions used in these analyses. For example, the analyses, where applicable, apply uncertainties in the conservative direction to RCS average temperature, RCS pressure, pressurizer level, and steam generator level. Thermal design flow is assumed rather than minimum measured flow. The engineered safeguards feature equipment is also conservatively modeled by minimizing or maximizing flowrates, temperatures, and response times, as appropriate. Taken together, the improved thermal power measurement uncertainty and conservative assumptions provide substantial conservatism such that the margin of safety would not be reduced.

Also, the Rupture of a Control Rod Drive Mechanism Housing (FSAR Section 15.4.6) event models the neutron flux high trip setpoint, which has not been changed for the 1.4 % uprate conditions. Thus, it was necessary to confirm that the event acceptance criteria would continue to be met. The event is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the Reactor Coolant System would eject the control rod and drive shaft to the fully withdrawn position. The transient responses for the hypothetical RCCA ejection event are analyzed at beginning and end of life for both full and zero power operation in order to bound the entire fuel cycle and expected operating conditions. The analyses are performed to show that the fuel and clad limits are not exceeded. Since this study is not performed to evaluate the minimum DNBR, the RTDP method is not utilized (the limiting fuel rod is conservatively assumed to undergo DNB very early in the transient, thus maximizing fuel temperature response).

An evaluation was performed to confirm that the existing high neutron flux safety analysis limits of 35% and 118% are still adequate for the 1.4% increased power. It was found that the power increase is very rapid during the transient, such that the small power increase would have a negligible effect on the trip time. Thus, the results indicate that the fuel pellet enthalpies remain below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel and the maximum amount of fuel melted at the hot spot is less than 10%. Also, the average clad temperature at the hot spot remains below 3000°F and the zirconium-water reaction is less than 16%. Therefore, the results of this analysis are acceptable and the conclusions in the FSAR remain valid.

6.5.3 Other Non-LOCA Transient Analyses

Uncontrolled Boron Dilution (FSAR Section 15.2.4)

This event is analyzed to identify the amount of time available for operator or automatic mitigation of an inadvertent boron dilution prior to complete loss of shutdown margin. This transient is considered for Watts Bar for operational Modes 1 through 2. Modes 3, 4, and 5 are addressed by operating procedures. Dilution cannot occur in Mode 6 due to administrative controls.

The critical parameters in the determination of the time available include the overall RCS active volume, the dilution flowrate, and the initial and critical boron concentrations. The 1.4% increased core power does not affect any of these input parameters or the nature or the transient.

Therefore, the conclusions in the FSAR remain valid.

Accidental Depressurization of the Main Steam System and Major Rupture of a Main Steam Line (FSAR Section 15.2.13 and 15.4.2.1)

For these events, excessive steam relief is assumed to cause an RCS cooldown that results in a positive reactivity excursion. The safety analyses are performed under zero power initial conditions and show that the DNB design basis is met. The results of the Major Rupture cases bound those of the Accidental Depressurization cases.

These events are analyzed at hot zero power conditions and it has been confirmed that the DNB design basis continues to be met for the 1.4% power increase. Therefore, the conclusions documented in the FSAR remain valid

6.6 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SETPOINTS

Westinghouse WCAP-12096, Revision 8 provides the basis for the reactor trip and engineered safety features actuation setpoints. These setpoints are used, as required, in the plant safety analyses and contained in the plant technical specifications. As discussed in this report, some of these safety analyses have been either re-performed or re-analyzed for the 1.4% uprate conditions and have been shown to meet their applicable acceptance criteria. These evaluations and analyses were performed with the current reactor trip and engineered safety feature actuation setpoints. Thus, the setpoints did not need to be revised for the 1.4% uprate program to meet the safety analysis criteria.

As discussed in Section 2 of this submittal, the 1.4% uprate modifies the RCS design conditions used in the calculation of some setpoints and associated allowable values. In particular, the full power vessel ΔT increases from its present value of 61°F to 61.8°F, which is considered in the setpoints and allowable values for the OTΔT, OPΔT, and Vessel ΔT equivalent to power reactor trip functions. Prior uprating studies have shown that the inclusion of the increased ΔT in the calculation of these setpoints, considering all effects, would actually increase the margin in these channels. Thus, the existing setpoints and allowable values for these functions are still bounding and do not require modification.

The 1.4% uprate conditions also alter the process conditions inside the steam generator which may affect the setpoints and allowable values for the steam generator narrow range water level low-low reactor trip and high-high turbine trip. Based on evaluations performed for other plant uprates, it is judged that the 1.4% uprate conditions, in particular the change in operating steam generator pressure, would have a negligible effect on the steam pressure effect and the fluid velocity effect at the narrow range water level instrument tap - which is considered in the calculation. Thus, there would be no impact on the existing setpoints and allowable values for these functions.

6.7 RTDP UNCERTAINTY CALCULATIONS

RTDP Uncertainties Excluding Power Calorimetric

Westinghouse WCAP - 14738, Revision 0 provides the basis for the RCS control system uncertainties that are used in the plant safety analyses. Most of these safety analyses have been either evaluated or re-performed for the 1.4% uprate conditions and have been shown to meet their applicable acceptance criteria. These evaluations and analyses were performed with the current uncertainties with exception to the power calorimetric measurement uncertainty - which has been modified to account for the use of the Caldon LEFM. The modified power calorimetric measurement uncertainty is discussed more fully below.

As discussed in Section 2 of this report, the 1.4% uprate increases the design full power vessel ΔT from its present value of 61°F to 61.8°F, which is considered in the uncertainty for the calorimetric RCS flow measurement. Prior uprating studies have shown that the inclusion of the increased ΔT in this calculation, considering all effects, would slightly reduce the uncertainty. Thus, the existing uncertainty for this control function is still bounding.

The 1.4% uprate conditions also alter the process conditions inside the steam generator which may affect the measurement accuracy for the steam generator narrow range water level control uncertainty. Based on prior evaluations, it is judged that the 1.4% uprate conditions have a negligible effect on the steam pressure effect and the fluid velocity effect at the narrow range water level instrument tap - which is considered in the calculations. Thus, there would be no impact on the existing uncertainty calculation for this control function.

Finally, the turbine impulse pressure increases as a direct result of the 1.4% power uprate. The turbine impulse pressure provides the percent (%) turbine power load signal to the reference T_{avg} program for reactor temperature control. The reference T_{avg} program will be re-scaled for the increase in turbine impulse pressure to provide the desired full power T_{avg} of 588.2°F at the uprated power.

Power Calorimetric Uncertainty

The power calorimetric measurement uncertainty reported in WCAP-14738, Revision 0 was re-calculated to account for the use of the LEFM uncertainties (Enclosure 4). The calculation indicates that the 95/95 power measurement uncertainty is less than 0.6%. Thus, it supports the use of a 0.6% power uncertainty in the safety analyses

7 BALANCE OF PLANT

Using the revised NSSS parameters (see Table 2-1), TVA has performed a preliminary heat balance at 101.4% reactor thermal power for the proposed uprate. The secondary side plant systems were originally designed to support the operation of the Westinghouse supplied turbine/generator. At the valves wide open or stretch condition the turbine/generator is rated at 1,269,837 kW with a steam flow of 15,900,800 lb/hr. This equates to operation at approximately 104% reactor thermal power. Therefore, no major impacts to the balance of plant is expected. Comparison of the uprate heat balance with the current 100% heat balance revealed no significant differences in pressures, temperatures, or flows for the secondary side plant systems (See Table 7-1).

The Balance of Plant systems that are being reviewed are those that are (or could be) directly affected by the power uprate. This does not include the systems that have been evaluated and discussed previously for the NSSS-BOP interface requirements (i.e., Main Steam, Feedwater, Steam Generator Blowdown, etc.).

- Extraction Steam - Each of the turbine extraction lines will realize an increase in pressure of 1.8% or less. The mass flow changes will be within ±5% of the current flows. These increases and/or changes are within the design parameters as given in the design basis documentation.

- Condensate - The condensate system does not require an increase in storage capacity of the condensate storage tank, has a modest temperature increase (≤2°F), no pressure increase, and a slight increase in flow rate. All of the preceding have been evaluated against the design bases and are judged to be bounded by existing analysis and are adequate for the 1.4% power increase.

- Heater, Drains and Vents - The heater drains have been evaluated for design pressure and temperature and have been found to be bounded for the small temperature increase expected from the power uprate. Little, if any, pressure increase will be experienced since the main steam pressure is being decreased and the associated equipment will also experience this decrease in pressure.

- Condensate Polishing - The condensate polishing system will experience a small increase in temperature which is well within the capacity of the system. An increase in the total flow of less than two percent is within the capacity of the system. The purity of the condensate is not expected to be significantly different with the power uprate and the review of the design bases documents for the condensate demineralizers indicates that the power uprate is acceptable for this system.

- Turbine/Generator Cooling - The generator is designed for ~105% of nameplate rated power. The hydrogen cooling system is also designed for this thermal loading. The incoming raw water is required to have sufficient capacity to cool this thermal load imposed on the hydrogen. Therefore, no impacts are expected for the cooling system.

- Condenser Circulating Water - The additional heat load on the main condensers will result in a slightly higher back pressure on the main turbine. This increase, however, will not be enough to restrict operation of the turbine at full load due to back pressure limitations. Therefore, the condenser circulating water system is still adequate to meet its design requirements.

- Secondary Sampling - The Secondary Sampling system is not challenged due to the lower main steam pressure and only a slight increase in the feedwater and associated feedwater makeup systems.

Based on TVA's preliminary evaluation, the Balance of Plant systems are deemed adequate for the increase in thermal loads produced by the power uprate. However, as part of the design change process for the power uprate, additional heat balance studies will be performed at higher ambient conditions to assess potential impacts on individual BOP components.

TABLE 7-1

FIELD DESCRIPTION		WATTS BAR 100.0% RTP	WATTS BAR 101.4% RTP	% Diff
STEAM GENERATOR STEAM OUTLET FLOW	#/HR	15,121,600	15,355,470	1.55%
STEAM GENERATOR STEAM OUTLET PRESS	PSIA	1000.0	980.0	-2.00%
MAIN TURBINE CONTROL VALVE INLET PRESS	PSIA	975.0	955.0	-2.05%
MAIN TURBINE CONTROL VALVE INLET TEMP	°F	541.5	539.0	-0.46%
MAIN TURBINE CONTROL VALVE INLET MOISTURE	%	0.3864	0.3823	-1.04%
MAIN TURBINE CONTROL VALVE INLET ENTH	BTU/LB	1191.3	1192.0	0.06%
HP TURBINE IMPULSE PRESS	PSIA	743.1	757.5	1.94%
HP TURBINE 1ST EXTRACTION OUTLET PRESS	PSIA	418.9	426.5	1.81%
HP TURBINE 2ND EXTRACTION OUTLET PRESS	PSIA	277.2	282.1	1.77%
HP TURBINE EXHAUST STEAM OUTLET PRESS	PSIA	172.0	174.7	1.57%
MOISTURE SEPERATOR SHELL SIDE INLET PRESS	PSIA	170.3	173.0	1.59%
MOISTURE SEPERATOR SHELL SIDE OUTLET PRESS	PSIA	165.2	167.8	1.57%
MOISTURE SEPERATOR SHELL SIDE OUTLET TEMP	°F	366.1	367.4	0.36%
1ST STAGE REHEATER SHELL SIDE OUTLET PRESS	PSIA	163.1	165.7	1.59%
1ST STAGE REHEATER SHELL SIDE OUTLET TEMP	°F	428.2	429.9	0.40%
2ND STAGE REHEATER SHELL SIDE OUTLET PRESS	PSIA	160.7	163.2	1.56%
2ND STAGE REHEATER SHELL SIDE OUTLET TEMP	°F	520.3	517.8	-0.48%
LP TURBINE REHEAT STEAM INLET PRESS	PSIA	157.4	159.9	1.59%
LP TURBINE REHEAT STEAM INLET TEMP	°F	519.7	517.2	-0.48%
LP TURBINE 4TH EXTRACTION OUTLET PRESS	PSIA	68.2	69.3	1.54%
LP TURBINE 5TH EXTRACTION OUTLET PRESS	PSIA	43.0	43.6	1.51%
LP TURBINE 6TH EXTRACTION OUTLET PRESS	PSIA	15.8	16.0	1.59%
LP TURBINE 7TH EXTRACTION OUTLET PRESS	PSIA	6.9	7.0	1.74%
MOISTURE REMOVAL STAGE 5 OUTLET PRESS	PSIA	3.1	3.2	1.61%
LP TURBINE A EXHAUST STEAM OUTLET PRESS	IN-HG	1.68	1.70	1.19%
LP TURBINE B EXHAUST STEAM OUTLET PRESS	IN-HG	2.31	2.34	1.30%
LP TURBINE C EXHAUST STEAM OUTLET PRESS	IN-HG	3.14	3.20	1.91%
MAIN CONDENSER DRAIN OUTLET TEMP	°F	116.7	117.3	0.51%
GSC TUBE SIDE OUTLET TEMP	°F	119.7	120.4	0.58%
GSC/SGBD HX BYPASS INLET FLOW	#/HR	6,494,525	6,639,690	2.24%

TABLE 7-1				
FIELD DESCRIPTION		WATTS BAR 100.0% RTP	WATTS BAR 101.4% RTP	% Diff
MFPTC TUBE SIDE INLET TEMP	°F	117.5	118.2	0.60%
MFPTC TUBE SIDE OUTLET TEMP	°F	144.0	145.1	0.76%
NO 7 FWH OUTLET TEMP	°F	168.7	169.4	0.41%
NO 6 FWH INLET TEMP	°F	169.4	170.1	0.41%
NO 6 FWH OUTLET TEMP	°F	209.2	209.9	0.33%
NO 5 FWH OUTLET TEMP	°F	263.1	263.8	0.27%
NO 4 FWH INLET TEMP	°F	263.1	263.8	0.27%
NO 4 FWH OUTLET TEMP	°F	296.9	297.9	0.34%
NO 3 FWH OUTLET TEMP	°F	363.4	364.5	0.30%
NO 2 FWH INLET TEMP	°F	363.9	365.1	0.33%
NO 2 FWH OUTLET TEMP	°F	401.4	402.8	0.35%
NO 1 FWH INLET TEMP	°F	403.2	404.6	0.35%
NO 1 FWH OUTLET TEMP	°F	439.9	441.5	0.36%
2ND STAGE REHEATER TUBE SIDE INLET FLOW	#/HR	792,344	767,432	-3.14%
2ND STAGE REHEATER TUBE SIDE INLET PRESS	PSIA	965.3	945.5	-2.05%
MOISTURE REMOVAL STAGE 1 OUTLET FLOW	#/HR	30,593	30,152	-1.44%
NO 1 EXTRACTION STEAM OUTLET FLOW	#/HR	638,995	665,643	4.17%
1ST STAGE REHEATER TUBE SIDE INLET FLOW	#/HR	571,850	586,932	2.64%
1ST STAGE REHEATER TUBE SIDE INLET PRESS	PSIA	414.8	422.2	1.78%
MOISTURE REMOVAL STAGE 2 OUTLET FLOW	#/HR	40,052	39,489	-1.41%
NO 2 EXTRACTION STEAM OUTLET FLOW	#/HR	647,624	661,979	2.22%
NO 3 EXTRACTION STEAM OUTLET FLOW	#/HR	907,378	926,234	2.08%
MOISTURE SEPERATOR DRAIN OUTLET FLOW	#/HR	1,426,208	1,430,347	0.29%
REHEAT STEAM TO MFPT LP INLET FLOW	#/HR	236,271	244,578	3.52%
NO 4 EXTRACTION STEAM OUTLET FLOW	#/HR	360,355	370,577	2.84%
NO 5 EXTRACTION STEAM OUTLET FLOW	#/HR	515,016	526,087	2.15%
MOISTURE REMOVAL STAGE 3 OUTLET FLOW	#/HR	59,794	62,749	4.94%
NO 6 EXTRACTION STEAM OUTLET FLOW	#/HR	358,220	365,304	1.98%
MOISTURE REMOVAL STAGE 4 OUTLET FLOW	#/HR	103,305	106,643	3.23%
NO 7 EXTRACTION STEAM OUTLET FLOW	#/HR	200,573	201,004	0.21%
MOISTURE REMOVAL STAGE 5 WATER OUTLET FLOW	#/HR	134,325	138,018	2.75%
MOISTURE REMOVAL STAGE 5 STEAM OUTLET FLOW	#/HR	61,593	62,622	1.67%
NO 1 FWH SATURATION PRESS	PSIA	398.0	405.1	1.78%
NO 1 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	4.2	4.3	2.38%
NO 1 FWH DRAIN COOLER APPROACH (DCA)	°F	7.7	7.8	1.30%
NO 2 FWH SATURATION PRESS	PSIA	263.3	268.0	1.79%
NO 2 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	4.2	4.3	2.38%
NO 2 FWH DRAIN COOLER APPROACH (DCA)	°F	9.3	9.4	1.08%
NO 3 FWH SATURATION PRESS	PSIA	163.4	166.0	1.59%
NO 3 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	1.9	2.0	5.26%
NO 4 FWH SATURATION PRESS	PSIA	64.8	65.8	1.54%
NO 4 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	0.9	0.9	-0.00%
NO 4 FWH DRAIN COOLER APPROACH (DCA)	°F	0.4	0.5	25.00%
NO 5 FWH SATURATION PRESS	PSIA	40.8	41.4	1.47%
NO 5 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	5.4	5.5	1.85%
NO 5 FWH DRAIN COOLER APPROACH (DCA)	°F	11.4	11.7	2.63%
NO 6 FWH SATURATION PRESS	PSIA	15.0	15.2	1.33%
NO 6 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	3.7	3.8	2.70%
NO 6 FWH DRAIN COOLER APPROACH (DCA)	°F	11.4	11.5	0.88%
NO 7 FWH SATURATION PRESS	PSIA	6.5	6.7	3.08%
NO 7 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	5.2	5.2	0.00%
NO 1 FWH DRAINS OUTLET TEMP	°F	410.9	412.4	0.37%
NO 2 FWH DRAINS OUTLET TEMP	°F	373.1	374.5	0.38%
NO 4 FWH DRAINS OUTLET TEMP	°F	263.6	264.4	0.30%
NO 5 FWH DRAINS OUTLET TEMP	°F	220.5	221.5	0.45%

TABLE 7-1				
FIELD DESCRIPTION		WATTS BAR 100.0% RTP	WATTS BAR 101.4% RTP	% Diff
NO 6 FWH DRAINS OUTLET TEMP	°F	180.8	181.7	0.50%
NO 7 HDT DRAINS OUTLET TEMP	°F	173.1	173.8	0.40%
MFPT EXHAUST STEAM OUTLET PRESS	IN-HG	9.31	9.64	3.54%
CCW INLET TEMP	°F	73.5	73.5	0.00%
CCW OUTLET TEMP	°F	107.8	108.3	0.46%
CCW INLET FLOW	#/HR	440,000	440,000	0.00%
MAIN FEED PUMP TURBINE SPEED	RPM	5,048	5,085	-0.74%
MAIN FEED PUMP TURBINE OUTPUT	KW	13224.5	13537.7	2.37%
TOTAL GENERATOR OUTPUT	MWe	1,206,610	1,220,576	1.16%
MAIN STEAM THROTTLE FLOW	#/HR	14,314,604	14,573,388	1.81%
HP TURBINE EXHAUST FLOW	#/HR	12,379,447	12,583,059	1.64%
MAIN STEAM AT LP TURBINE INLET FLOW	#/HR	9,809,590	9,981,900	1.76%
LP TURBINE A EXHAUST STAGE FLOW	#/HR	2,675,243	2,719,405	1.65%
LP TURBINE B EXHAUST STAGE FLOW	#/HR	2,675,243	2,719,405	1.65%
LP TURBINE C EXHAUST STAGE FLOW	#/HR	2,675,243	2,719,405	1.65%
CONDENSER HOTWELL DRAIN FLOW	#/HR	8,469,293	8,614,898	1.72%
NO. 1 FWH DRAIN FLOW	#/HR	1,461,932	1,463,227	0.09%
NO. 2 FWH DRAIN FLOW	#/HR	2,721,458	2,751,627	1.11%
NO. 3 HDT DRAIN FLOW	#/HR	5,055,044	5,108,208	1.05%
NO. 4 FWH DRAIN FLOW	#/HR	360,355	370,577	2.84%
NO. 5 FWH DRAIN FLOW	#/HR	875,371	896,664	2.43%
NO. 6 FWH DRAIN FLOW	#/HR	1,293,385	1,324,717	2.42%
NO. 7 HDT DRAIN FLOW	#/HR	1,597,263	1,632,364	2.20%
MFPT CONDENSER INLET FLOW	#/HR	8,644,293	8,789,898	1.68%
NO. 7 FWH TUBE INLET FLOW	#/HR	8,043,531	8,190,080	1.82%
NO. 6 FWH TUBE INLET FLOW	#/HR	9,640,794	9,822,444	1.88%
NO. 4 FWH TUBE OUTLET FLOW	#/HR	10,241,556	10,422,262	1.76%
NO. 1 FWH TUBE OUTLET FLOW	#/HR	15,296,600	15,530,470	1.53%

8 ELECTRIC POWER ANALYSIS

Electrical Distribution System

As a result of this uprate, no ac or dc auxiliary load ratings are expected to change, and the loads are not expected to experience additional demands above their ratings. Therefore, the plant auxiliary ac/dc electrical load will not change. The main generator electrical parameters remain the same, and the uprate capacity remains within the generator rating. The voltage controls and grid source impedance at the WBNP 500-kV and 161-kV grid will not be affected by this uprate; therefore, the evaluated voltages and short circuit values at different levels of station auxiliary electrical distribution system will not change as a result of this uprate.

Turbine/Generator

The electrical systems associated with the turbine auxiliary systems are not affected by the uprate.

The Unit 1 steam turbine-driven polyphase generator is a four pole machine rated at 1411 MVA, with an operating point of 1270 MWe at a 0.9 power factor. This rating is based upon 75 psig hydrogen pressure, which is supplemented with water cooling for the stator and rotor.

At the current thermal rating of Unit 1 of 3411 MWt, the Unit 1 main generator electrical output is typically 1204 Mwe. The anticipated net increase of 17 MW lies well within the nameplate rating of the generator of 1270 Mwe at 0.9 power factor. Therefore there will be no generator limitations to prevent operation at a core power of 3459 MWt.

TVA has not identified any changes to equipment protection relay settings for the generator; although some process alarm setpoints for the generator and the exciter may require adjustment.

To deliver electrical power provided by the generator to the transmission system, the unit is equipped with an isolated phase bus, three main transformers and switchyard breakers and switches. The components are rated to deliver electrical power at or in excess of the main generator nameplate rating of 1411 MVA.

Isophase Bus

The isophase bus is designed to standards ANSI/IEEE C37.20 and C37.23, IEEE Guide for Metal-Enclosed Bus and Calculating Losses in Isolated Phase Bus, with a forced cooling rating of 35,700 amps (along the main bus section) and self cooled rating of 20,600 amps each phase (at the generator and transformer terminals). These ratings are greater than the Unit 1 Main Generator rating of 33,943 stator amps (19,597 amps each phase) at 1411 MVA and are well in excess of the anticipated generator output. The Isophase Bus will support the power increase with no modifications.

Main Transformers

The Unit 1 main transformers have a total capacity of 1444 MVA, which is in excess of the main generator nameplate rating of 1411 MVA Therefore, the main bank transformers will operate within all applicable limits at the 1.4% uprated power.

Switchyard

The switchyard equipment exceeds the nameplate rating of the main generator. All 500kV switches and breakers are rated 3000 amperes, which exceeds the main generator maximum output current of approximately 1600 amperes at its nameplate rating of 1411 MVA. The switchyard will accept the additional load without the need for any hardware modifications.

500kV Grid Stability

A study was performed to determine the impact on the stability of 500kV grid for a 1.4% increase in WBN generation. This study considered a line out pre-event and a subsequent simultaneous LOCA of the WBN unit and a fault and trip of another line. The change in post-event 500kV grid voltage related to the increase in generation at WBN was less than 0.06 kV in the study cases. This is considered insignificant and not to have an impact on the stability of the grid.

161kV OFFSITE POWER SUPPLY

There is no change in the shutdown loads at WBN or the voltage requirements of these loads associated with the 1.4% upgrade. The WBN unit receives shutdown power from the 161 kV system at the Watts Bar Hydro plant. This power is supplied through two 161 kV transmission lines from the hydro switchyard to the nuclear plant common station service transformers. Since there is no inter-tie bank connecting the 500kV and 161kV grids at Watts Bar, the 161kV system is electrically remote from the 500kV system at the nuclear site. Therefore, the extra 1.4% of power generated into the 500 kV system has no significant impact on the 161kV system at WBN and the ability of the unit to safely shut down, and therefore Watts Bar will continue to be in conformance with GDC 17.

9 Radiological Consequences

LOCA and Normal Operational Effluents

A review of radiological analyses has been performed to determine the potential impact on the radiological consequences from a 1.4% reactor power level uprating. Current WBN analyses are based on 3582 MWt (i.e., 105% of 3411 MWt)for normal operation and 3565 MWt (i.e., 104.5% of 3411 MWt) for accidents. The proposed uprate is for 3459 MWt core power. Regulatory Guide 1.49 requires that analyses be performed at 102% of rated power. Since 102% of 3459 MWt (3528 MWt) is less than 3582 MWt (normal operation) and less than 3565 MWt (accident operation) the power uprate will not require WBN to reanalyze radiological calculations. Thus the 1.4% uprate condition is bounded by the current WBN analyses.

Steam Generator Tube Rupture

The offsite dose calculation is performed to demonstrate that the radioactivity released during the SGTR does not exceed the allowable values specified in Standard Review Plan 15.6.3 and 10 CFR 100. The licensing basis methodology for the offsite dose mass and energy releases is presented in WCAP-10698. The analysis of record incorporates a 2% power measurement uncertainty to increase the initial power level to 102%. Consistent with the discussion provided in Section 0.2, the improved thermal power measurement accuracy obviates the need for the full 2% power measurement margin assumed in the analysis.

The power measurement margin is but one of many conservative assumptions used in the analysis. For example, a key assumption is that a steam generator power operated relief valve "sticks" open, to maximize the outside releases. Also, the analysis employs a worse case composite break location, considering limiting characteristics of a hot and cold leg tube rupture, thereby maximizing the primary to secondary leakage and the associated outside releases. Taken together, the improved power measurement uncertainty and conservative assumptions provide substantial conservatism such that the margin of safety would not be reduced.

Main Steam Line Break

Steam releases are calculated for use in the radiological dose evaluation. The analytical method for the radiological steam releases is a hand calculation in which steam releases and feedwater flows are calculated for given time periods. A higher power level is more limiting since it increases the primary to secondary heat transfer and hence overall release rate.

The current analysis of record employs a core power level of 3565 MWt (4.5% increase from 3411 MWt) which is higher than the 1.4% uprate value of 3459 MWt. Thus, the existing analysis remains bounding for the 1.4% uprate conditions.

Waste Gas Decay Tank

Westinghouse provided TVA with WGDT radiation sources for plant operation with 18 month fuel cycles. The radiation sources were developed by assuming a 3565 MWt core power. Thus, they would bound those expected for the 1.4% uprate core power of 3459 MWt. To meet 10 CFR 100 offsite dose limits, WBN is administratively limited to significantly less WGDT inventories than that found in the original basis performed by Westinghouse. These limitations preclude any affects that the power uprate would have.

Hydrogen Sources

Westinghouse provided TVA with revised hydrogen sources from the core and sump radiolysis for 18 month fuel cycles. These sources were generated by assuming a 3582 MWt core power. Thus, they would bound those expected for the 1.4% uprate core power of 3459 MWt.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA is submitting a request for an amendment to the Watts Bar Nuclear Plant (WBN) Unit 1 Operating License and Technical Specifications. The proposed license amendment would increase the full core thermal power rating by 1.4% from 3411 Megawatts-thermal (MWt) to 3459 MWt, based on planned installation of the improved Caldon Inc., Leading Edge Flow Meter (LEFM✓™) feedwater flow measurement instrumentation during the upcoming Unit 1 Cycle 3 refueling outage. TVA's uprate is based on eliminating unnecessary analytical margin originally required of Emergency Core Cooling System Evaluation Models (ECCS) performed in accordance with the requirements set forth in 10 CFR 50, Appendix K. The NRC has recently approved a change to the Appendix K requirements whereby licensees may reduce the previously required 2% power margin between the licensed power level and the assumed power level for the ECCS evaluation, provided the reduced margin is demonstrated to account for uncertainties due to power level instrumentation error. The use of the LEFM will enable determination of core power level with improved measurement uncertainties, thereby allowing the proposed increase in licensed power level.

TVA has concluded that operation of WBN Unit 1, in accordance with the proposed change to the Technical Specifications, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

The comprehensive analytical efforts performed to support the proposed change included a review of the Nuclear Steam Supply System (NSSS) systems and components that could be affected by this change. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), loop piping and supports, reactor coolant pump, steam generator and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. The Rod Control Cluster Assembly (RCCA) drop time remains within the current limits assumed in the accident analyses. Thus, there is no increase in the consequences of the accidents which credit RCCA drop. No additional steam generator tubes will need to be plugged to preclude the potential for U-bend fatigue. In addition, a preliminary assessment indicates the existing 40% through wall plugging criteria for SG tubes will remain adequate; TVA will perform a calculation to substantiate this conclusion. Thus, subject to confirmation by calculation, there is no increase in the probability of a Steam Generator Tube Rupture (SGTR) event. The Leak Before Break analysis conclusions remain valid and thus the limiting break sizes determined in this analysis remain bounding.

All of the NSSS systems will continue to perform their intended design functions during normal and accident conditions. The pressurizer spray flow remains above its design value. Thus, the control system design analyses which credit the flow do not need to be modified for changes in this flow. The auxiliary systems and components continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/Balance of Plant (BOP) interface systems will continue to perform their intended design functions. The steam generator safety valves will provide adequate relief capacity to maintain the steam generators within design limits. The atmospheric dump valves will still relieve 20% of the maximum full load steam flow. The steam dump system will still relieve 40% of the maximum full load steam flow. The current LOCA hydraulic forcing functions are still bounding. Thus, there is no significant increase in the probability of an accident previously evaluated.

Additionally, the reduction in the power measurement uncertainty allows for certain safety analyses to continue to be used, without modification, at the 3459 MWt power level. Other safety analyses performed at a nominal power level have been either re-performed or re-evaluated at the 3459 MWt power level and continue to meet their applicable acceptance criteria. Some existing safety analyses had been previously performed at a power level greater than 3459 MWt, and thus continue to bound the 3459 MWt power level. The effects on radiation dose for the power uprate were determined to be bounded by the power levels at which current dose analyses were performed Thus, there is no significant increase in the consequences of an accident previously evaluated.

B. <u>**The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**</u>

No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

Operation at the 3459 MWt core power does not involve a significant reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the NRC or that are in compliance with all applicable regulatory review guidance and standards. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

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