Westinghouse Electric Corporation **Energy Systems**

Nuclear Services Division

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March 17, 1997
CAW-97-1088

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Samuel L. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-14738, Rev. 0, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Tennessee Valley Authority Watts Bar Unit 1," (Proprietary)

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-97-1088 signed by the owner of the proprietary information, Westinghouse Electric Corporation. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by the Tennessee Valley Authority.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-97-1088, and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: Kevin Bohrer/NRC (12H5)

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

CAW-97-1088

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 17 day
of March, 1997

Rose Marie Payne
Notary Public

Notarial Seal
Rose Marie Payne, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Nov. 4, 2000
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Division, of the Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Unit.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Energy Systems Business Unit in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Tennessee Valley Authority Watts Bar Unit 1," WCAP-14738, Rev. 0 (Proprietary), for Watts Bar Unit 1, being transmitted by the Tennessee Valley Authority letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel L. Collins. The proprietary information as submitted for use by the Tennessee Valley Authority for Watts Bar Unit 1 is expected to be applicable in other licensee submittals in response to related technical specification changes.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the methods for determining instrumentation uncertainties.

(b) Provide the specific design information related to the parameters that are considered for each safety function.

(c) Assist the customer obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculation, evaluation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing testing and analytical methods and performing tests.

Further the deponent sayeth not.

ENCLOSURE 1

PROPOSED LICENSE AMENDMENT

PART 1 - EMERGENCY CORE COOLING SYSTEMS

I. Description of Proposed License Amendment

The proposed amendment would revise the Technical Specifications (TSs) to address Cycle 2 core design changes. The Cycle 2 core design for Watts Bar will include a longer fuel cycle and more highly enriched fuel (from 3.1 percent to 3.7 percent). To accommodate Cycle 2 and future core designs, the Refueling Water Storage Tank (RWST) and accumulator boron concentrations will be increased to provide enough boron in the sump to meet the Large Break Loss of Coolant Accident (LBLOCA) requirement for sump boron concentration. This requirement is that during a LBLOCA, the core will remain subcritical from boron provided by the Emergency Core Cooling System (ECCS), which takes suction from the RWST and containment sump.

The increase in RWST and accumulator boron concentrations will require changes to the plant TSs. These changes are summarized as follows, and the TS mark-ups are provided in Enclosure 3.

Accumulator Boron Concentration (Technical Specification Section 3.5.1)

The boron concentration range has been changed from 1900 to 2100 ppm to a range of 2400 to 2700 ppm to reflect the boron concentration increase.

RWST Boron Concentration (Technical Specification Section 3.5.4)

The boron concentration range has been changed from 2000 to 2100 ppm to a range of 2500 to 2700 ppm to reflect the boron concentration increase.

BASES - Applicable Safety Analyses (Technical Specification Section B 3.5.4)

The minimum RWST boron concentration has been changed from 2000 ppm to 2500 ppm to reflect the minimum value used in the post-Loss of Coolant Accident (LOCA) sump analysis for core subcriticality. Also, the maximum RWST boron concentration has been changed from 2100 ppm to 2700 ppm to reflect the maximum value used in the hot leg switchover time calculation.

II. Basis for Proposed License Amendment

The following is an evaluation of the impact of these changes and provides basis for these changes.

A. LOCA Related Analyses

1. LBLOCA

The large break LOCA analysis does not explicitly model the boron concentration level of the accumulators or RWST. During the LBLOCA, the core becomes subcritical due to voids generated by the rapid system depressurization associated with the large break transient. Though not modeled in the analysis, any additional boron injected due to the increase in the concentration levels would increase the margin by which the core is maintained in a subcritical condition. The calculated Peak Clad Temperature (PCT) is not a function of the boron concentration level in the core. Thus, an increase in the accumulator and RWST boron concentrations would have no adverse effect on the large break LOCA analysis results.

2. SBLOCA

The small break LOCA analysis does not take credit for the boron present in the RWST and the accumulators. Though not modeled in the analysis, any additional boron injected due to the increase in the concentration levels would increase the margin by which the core is maintained in a subcritical condition. The calculated PCT is not a function of the boron concentration level in the core. Thus, an increase in the accumulator and RWST boron concentrations would have no adverse effect on the small break LOCA analysis results.

3. Blowdown Reactor Vessel and Loop Forces

The LOCA blowdown hydraulic loads occur within the first few seconds of the LOCA transient and thus are not a function of the boron concentration level in the accumulators or RWST. Thus, an increase in the boron concentration levels in the accumulators and RWST would have no effect on the LOCA hydraulic forces calculation.

4. Post LOCA Long Term Core Cooling Requirements

The licensing basis commitment is that the reactor will remain shutdown by borated ECCS water residing in the sump following a LOCA. Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water will result in the reactor core remaining subcritical assuming all control rods are out. Minimum boron concentrations are assumed in the calculation for each borated water source. For this calculation, the minimum RWST boron concentration is 2500 ppm and the minimum accumulator concentration is 2400 ppm.

Westinghouse has performed the calculation which verifies that the sump solution will contain enough boron to maintain the reactor in a shutdown condition following a LOCA. The calculation demonstrates that the required boron concentration is 1741 ppm which is well below the calculated sump concentration of 1855 ppm. Thus, the sump contains enough boron to maintain the reactor in a shutdown condition.

5. <u>Hot Leg Switchover Time to Prevent Boron Precipitation</u>

The hot leg recirculation switchover time is determined for inclusion in emergency procedures to preclude boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level and on the Reactor Coolant System (RCS), RWST, accumulator and other (i.e., ice melt) water volumes and boron concentrations. In the event of a cold leg break during which the ECCS is aligned to the RCS cold legs, boron concentration in the core region increases due to boil-off of water. To preclude boron precipitation, the ECCS is realigned to the RCS hot legs at the hot leg switchover time.

The increase in the maximum RWST and accumulator boron concentrations results in a reduction in the hot leg switchover time because initial mixed boron concentrations are higher, and the precipitation concentration is reached sooner. The current hot leg switchover value of 12 hours will be changed to 9 hours.

B. <u>Non-LOCA Transient Related Analysis</u>

The following non-LOCA accidents model the RWST and/or the accumulator boron concentrations.

1. <u>Steamline Break at Full Power</u>

During this event, the ECCS is actuated via the low steam pressure signal and provides borated water to the core from the RWST. In addition, if sufficiently low RCS pressures are reached during the accident, the accumulators are capable of providing additional boron. The safety analysis assumes the minimum allowable boron concentrations in each of these sources, thus minimizing the amount of boron delivered to the core. Increasing the core boron concentration would reduce the magnitude of the power increase resulting from a loss of shutdown margin. Therefore, an increase in the RWST and accumulator boron concentrations would provide a benefit to the analysis.

2. Loss of Normal Feedwater

During this event, the ECCS is actuated via the low steam pressure signal and provides borated water to the core from the RWST. Minimum boron concentrations are modeled in the analysis. However, since the cooldown following reactor trip as a result of continued blowdown of the faulted Steam Generator (SG) is not sufficient to cause recriticality for Watts Bar, the analysis is not sensitive to changes in boron concentrations. If a post-trip loss of shutdown margin were to occur, an increased boron concentration would reduce the magnitude of the return-to-power and would therefore benefit the analysis results. Finally, it should be noted that during this event, RCS pressures remain considerably above that which would lead to actuation of the accumulators.

3. Inadvertent Operation of ECCS

Actuation of the ECCS is postulated as the initiating event for this analysis, and maximum RWST boron concentration is assumed. During this accident, RCS pressures remain considerably above that which would lead to actuation of the accumulators.

Two cases are analyzed, one to investigate minimum Departure from Nucleate Boiling Ratio (DNBR) and one to investigate pressurizer filling. For the minimum DNBR case, the limiting DNBR occurs at event initiation, and increasing the RWST boron concentration will not alter this analysis trend. Therefore, the increased concentration will not adversely impact the minimum DNBR.

Reactor trip is assumed to occur at event initiation for the case analyzed for pressurizer filling. The core boron concentration does not impact the post-trip decay heat generation and therefore the pressurizer filling results are insensitive to the amount of boron delivered to the RCS. Therefore, the increased boron concentration in the RWST will not adversely impact the analysis results.

C. Streamline Break Mass and Energy (M&E) Releases

The steamline break M&E analyses are performed for the containment integrity evaluation, compartment pressurization analysis and equipment qualification. These analyses assume the minimum allowable boron concentrations for the RWST and accumulators to minimize the amount of boron delivered to the core. An increase in these concentrations would increase the shutdown margin during the event. Thus, the increased boron concentrations would be a benefit in the analysis.

D. Steam Generator Tube Rupture (SGTR)

During the SGTR event, a low pressurizer pressure signal actuates the safety injection system which delivers flow from the RWST to the RCS. The borated water from the RWST helps to maintain the reactor in a shutdown condition after the tube rupture has occurred. The increase in the RWST concentration is a benefit in the analysis since it will lead to a higher boration rate and ultimately increase the overall RCS boron concentration. The overall increase in boron concentration will enhance the shutdown margin in the analysis. The accumulators are not modeled in the event since the RCS pressure remains above the accumulator injection pressure.

E. Containment Mass and Energy Releases

The LOCA temperature and pressure response analyses which are performed for containment integrity, compartment evaluation, and equipment qualification do not model the RWST and accumulator boron concentrations. Thus, the changes in concentration do not affect these analyses.

F. NSSS Systems and Components

1. Mechanical Components and Systems

The impact of an increase in the boron concentration range in the RWST and accumulators was assessed with respect to the mechanical and fluid system components. This increase in concentration will cause a decrease in the pH of the liquid and therefore require a review regarding the integrity of the RWST and accumulator materials, as well as other RCS component materials. This evaluation demonstrates that the integrity and operability of potentially affected equipment and systems will be maintained.

The RWST provides borated water to the refueling canal, charging pumps, safety injection pumps, containment spray pumps, and residual heat removal pumps. The accumulators supply water to the RCS during certain accident conditions. The immediate effect of raising the boric acid concentration in the RWST to 2700 ppm will be a decrease in the pH of the liquid. To assess the magnitude of this decrease, pH values of boric acid solutions containing 2000, 2300 and 2700 ppm at 40°F, 77°F, and 125°F were computed. These values are listed in the table below. The lowest and highest temperatures chosen, 40°F and 125°F, represent the range the RWST is expected to experience while 77°F is the temperature which the RWST liquid is expected to exhibit most of the time.

Table
pH of Boric Acid Solutions

Boron (ppm)	pH at 40 °F	pH at 77 °F	pH at 125 °F
2000	4.57	4.57	4.59
2300	4.49	4.49	4.52
2700	4.39	4.39	4.43

An inspection of the above table confirms that the pH of the RWST and accumulator liquids decreases very slightly when the boron concentration is increased from 2000 ppm to 2700 ppm. Specifically, at an average temperature of 77°F, the pH decreases by as little as 0.18 units. This minimal pH decrease is not expected to cause new concerns regarding the integrity of the RWST material or any other stainless steel surfaces that may come in contact with the RWST and accumulator liquids in the above temperature range.

In addition, structural carbon steel surfaces in containment during either the injection or recirculation phase following a postulated LOCA are protected by paint against corrosion. Wherever there are unprotected carbon steel surfaces, some corrosion is expected to take place in the moist air of the containment. Should the unprotected surfaces receive a spray of RWST liquid containing 2700 ppm boron during the injection phase following a LOCA, the slightly lower pH of the above liquids still is in the pH range where corrosion rates are nearly independent of pH. It is the opinion of Westinghouse that the slight pH decrease of the RWST and accumulator liquids resulting from the proposed increase in boron concentration to 2700 ppm will not cause any new corrosion concerns to unprotected (unpainted) carbon steel surfaces in the containment. During the recirculation phase following a LOCA, the expected pH of the containment sump is such that no significant corrosion of in-containment carbon steel surfaces is expected.

Finally, the solubility of boric acid at 44°F, 77°F, and 125°F is about 5590 ppm, 9500 ppm, and 18,530 ppm, respectively. Therefore, a boron concentration to 2700 ppm will remain in solution at the temperatures the liquids in the Watt Bar Unit 1 RWST and accumulators may experience.

2. Instrumentation and Control Systems

An increase in boron concentration can impact accident/post-accident chemistry conditions in the containment building. With respect to the environmental qualification of Class 1E equipment, such changes are only significant if the final pH of the containment sump solution differs greatly from that simulated during qualification testing. The intended objective is:

- to achieve and maintain pH above neutral (7.0) to preclude the possibility of chloride induced stress corrosion cracking, and

- to maintain a reasonable upper limit on pH (10.5 - 11.0) such that there is no significant degradation of polymer materials in the presence of strong alkali solutions.

Chloride induced stress corrosion cracking is a concern applicable to any stainless steel equipment located in the containment, but not unique to Class 1E equipment. Upper limits on pH range are established to provide adequate margin above the minimum pH (neutral 7.0) and with consideration of the likely non-metals used as vital sealing components of equipment. In practice, it is the non-metals that are selected for their endurance in the presence of the upper pH level selected by the equipment designer.

In the Westinghouse EQ program, documented as WCAP-8587, the purpose of chemistry conditions during EQ testing is to simulate a reasonable upper pH limit. The typical upper range limit value is 10.5 to 10.7 pH (varies among the specific tests performed). The intent is to affirm that chemistry, in conjunction with the extremes of pressure and temperature, does not result in a common mode failure of critical equipment/components. This is also the practice of other qualifiers of Class 1E equipment in that the choice of specific pH values simulated during testing will vary.

A calculation of the post LOCA sump pH with the higher boron concentrations indicates that the minimum long term sump pH will be reduced from 8.1 to 8.0. The 8.0 pH value will not result in an adverse impact to the qualification Class 1E equipment or its components. There is no impact to the qualification of Class 1E equipment in the Westinghouse or TVA scope of supply. It is concluded that the boron concentration increase in the RWST and Accumulators will not result in an unreviewed safety question with respect to the qualification of Class 1E equipment.

3. Emergency Operating Procedures (EOPs)

TVA will revise the Emergency Operating Procedures to reflect the new hot leg switchover time defined previously in the above "Hot Leg Switchover Time to Prevent Boron Precipitation" section of this submittal.

4. Radiological Dose and Hydrogen Production

The increase in RWST and accumulator born concentrations and subsequent slight decrease in containment sump and spray pH does not impact the LOCA dose evaluation since pH being a measure of acidity has no direct bearing on radionuclide concentration that will exist in the sump and core fluid. Therefore, as pH is not a function of radionuclide concentration, the proposed change in RWST and accumulator boron concentration will not affect the LOCA radiological dose calculations and the present analysis remains bounding. The slight decrease in sump, core and spray fluid pH has been evaluated to not significantly impact the corrosion rate (and subsequent generation of Hydrogen) of Aluminum and Zinc inside containment so that the present analysis remains bounding. In addition, the decreased sump, core and spray fluid pH will not affect the amount of hydrogen generated from the radiolytic decomposition of the sump and core solution.

III. Environmental Consideration

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

PROPOSED LICENSE AMENDMENT

PART 2 - SAFETY LIMITS, INSTRUMENTATION, AND REACTOR COOLANT SYSTEM

I. Description of Proposed License Amendment

Watts Bar has experienced hot leg temperature fluctuations, including random spikes, which decrease the operating margin to both the OTΔT and OPΔT reactor trip setpoints. These fluctuations result from a phenomenon know as upper plenum anomaly, which has been observed at other Westinghouse plants. Although it has been shown not to be a safety concern, it can impact normal operation and has caused the plant, in some instances, to experience OT alarms during steady-state operation. To offset the effects of these temperature fluctuations and reduce the incidence of the resulting alarms, the OTΔT and OPΔT setpoints have been enhanced to increase the operating margin associated with these trip functions. The increased margin decreases the OTΔT and OPΔT sensitivity to the temperature fluctuations. The revised setpoints are identified in Table 1.

The OTΔT and OPΔT trip functions are modeled in various safety analyses including the non-LOCA (transient) analyses, steamline break mass and energy releases, and steam generator tube rupture event. Also, the NSSS is designed to respond to Condition I design transients without incurring an OTΔT or OPΔT trip. These safety analyses and Condition I transients are addressed in this safety evaluation.

In addition, Watts Bar has decided to reduce the plant thermal design flow from 97,500 gpm per loop (total of 390,000 gpm) to 93,100 gpm per loop (total of 372,400 gpm) to accommodate 10% steam generator tube plugging and a 2% reduction in thermal design flow (RTDF). For 10% steam generator tube plugging, the thermal design flow is reduced from 97,500 to 95,000 gpm. For an additional 2% RTDF, the thermal design flow is reduced from 95,000 to 93,100 gpm. The total RTDF from 97,500 gpm to 93,100 gpm is 4.51%. The NSSS performance parameters have been modified to reflect plant operation at the 10% steam generator tube plugging and an additional 2% RTDF. A comparison of the modified parameters to the existing parameters is provided in Table 2.

The reduction in TDF to 93,100 gpm affects the Technical Specifications and is addressed in this safety evaluation. The level of steam generator tube plugging is not reported in the Technical Specifications. Therefore, tube plugging is not directly addressed in this safety evaluation. However, some of the safety analyses discussed in this safety evaluation have already been performed for the 10% tube plugging and TDF of 93,100 gpm and will bound plant operation at 0% tube plugging and a TDF of 93,100 gpm. These safety analyses include the steamline break mass and energy releases, steam generator tube rupture, containment mass and energy releases, NSSS components and systems, NSSS/BOP Interface Systems, Control Systems Evaluation, and the Revised Thermal Design Procedure (RTDP) and Setpoint Study. The discussion of these analyses includes provisions for the TDF of 93,100 gpm and the 10% tube plugging, even though the provisions for the tube plugging are not needed to support the Technical Specification changes.

Westinghouse has recommended that ΔT_o and T_{AVG} be normalized quarterly for consistency with the safety analysis. Implementation of this recommendation includes a tolerance of 0.6°F for the normalization of ΔT_o and a 1°F tolerance for the normalization of T_{AVG} (identified as T′ and T″ in the Technical Specifications). These tolerances will ensure that the OTΔT and OPΔT parameters remain conservative with respect to the analysis assumptions. The use of tolerances will help WBN to determine whether the indicated ΔT and T_{AVG} may be left as is, or must be rescaled. These tolerances have been incorporated as biases into the uncertainty analysis for the affected protection system functions. These functions include the OTΔT, OPΔT and vessel ΔT equivalent to power (used in the SG low-low water level reactor trip and ESFAS functions). As a result of implementing these biases into the protection system functions (and the changes to the OTΔT/OPΔT setpoints and reduced TDF), the Allowable Values in the Technical Specifications for the OTΔT, OPΔT and vessel ΔT equivalent to power functions have been modified.

The tolerances are not directly modeled as an input to the safety analysis. They are indirectly modeled in the safety analysis because they can potentially affect the trip setpoints for the OTΔT/OPΔT and vessel ΔT equivalent to power functions. However, it has been determined that the use of tolerances does not require changes to the trip setpoints, and thus, the tolerances are not specifically discussed in the safety analyses in this safety evaluation. The incorporation of the tolerances into the setpoint uncertainty analysis is further discussed in RTDP and Setpoint Study Section.

This safety evaluation has been prepared to allow for plant operation during Cycle 2 with the revised OTΔT and OPΔT setpoints, the thermal design flow of 93,100 gpm and the tolerances for ΔT_o, T′ and T″. To obtain sufficient DNB margin for the OTΔT and OPΔT setpoints, reduced TDF and Cycle 2 design features, it was necessary to implement the RTDP. The RTDP program changes the uncertainty treatment for core power, T_{AVG}, pressurizer pressure and RCS flow. These uncertainties have been incorporated, where applicable, into the safety analyses addressed in this Safety Evaluation. A brief discussion of the RTDP program is provided in the RTDP and Setpoint Study Section.

The following Technical Specifications and Bases will be changed to incorporate the OTΔT/OPΔT margin enhancement, thermal design flow of 93,100 gpm and tolerances for ΔT_o, T′ and T″. The applicable Technical Specification mark-ups are provided in Enclosure 3.

Reactor Core Safety Limits (Figure 2.1.1-1)

The use of the RTDP to improve DNB margin leads to a modification of the Reactor Core Safety Limits.

Reactor Trip System Instrumentation (Table 3.3.1-1, page 4) and ESFAS Instrumentation (Table 3.3.2-1, page 4)

The Allowable Values for the Vessel ΔT equivalent to Power input to Steam Generator Water Level Low-Low have been changed to reflect the addition of a 0.6°F tolerance to the measurement of ΔT_o. This tolerance has been included as a bias in the uncertainty analysis to facilitate the quarterly measurement of ΔT_o.

Reactor Trip System Instrumentation (Table 3.3.1-1, pages 7 and 8)

The revised reactor core safety limits lines allow for changes in the OTΔT/OPΔT reactor trip setpoints to improve operating margin. The allowable values for these functions have changed as a result of including tolerance for ΔT_o, T′ and T″ in the uncertainty analysis. Several setpoint gains and time constants have been modified to enhance plant operation.

RCS Pressure, Temperature and Flow DNB Limits (Section 3.4.1)

The RCS average temperature limit has been revised to account for the change in uncertainty from implementing RTDP. The total RCS flow has been modified to account for the reduced thermal design flow from 97,500 gpm to 93,100 gpm. The total flow value in the Technical Specification includes an allowance for instrument uncertainty.

Bases - Reactor Core Safety Limits (Section B 2.1.1)

The "Safety Limits" section has been modified to reflect the use of the RTDP methodology.

Bases - Nuclear Enthalpy Rise Hot Channel Factor (Section B 3.2.2)

The "Applicable Safety Analyses" section has been modified to reflect the use of the RTDP methodology.

Bases - Reactor Trip System Functions OTΔT, OPΔT, and Steam Generator Water Level Low-Low (Vessel ΔT Equivalent to Power) (Section B 3.3.1)

The "Applicable Safety Analyses, LCO, and Applicability" sections for the OTΔT, OPΔT, and Steam Generator Water Level Low-Low (Vessel ΔT) trip functions have been expanded to address the quarterly examination of ΔT_0, T′, and T″. The tolerances are used to determine if these parameters should be reset during the examination.

Bases - Reactor Trip System Functions Reactor Coolant Flow - Low (Single Loop and Two Loops) (Section B 3.3.1)

These "Applicable Safety Analyses, LCO, and Applicability" sections have been modified to indicate that the Reactor Coolant Flow - Low Trip Setpoint and Allowable Value are specified in percent of thermal design flow adjusted for uncertainties and not nominal flow.

Bases - Engineered Safety Feature Actuation System (ESFAS) Instrumentation (Section B 3.3.2)

In the "Applicable Safety Analyses, LCO and Applicability section for the Steam Generator Water Level Low-Low (Vessel ΔT), a reference to Bases section B 3.3.1 has been added for a discussion of the required MODES and normalization of the vessel ΔT input to the trip time delay function.

Bases - RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) (Section B 3.4.1)

In the "Applicable Safety Analyses" section, the RCS average temperature limit, RCS average temperature analytical limit and pressurizer pressure analytical limit have been revised to account for the change in uncertainty from implementing RTDP. In the "LCO" section, the flow measurement uncertainty allowance using control board indication has also been revised as a result of implementing RTDP.

II. Basis for Proposed License Amendment

The following is an evaluation of the impact of these changes and provides basis for these changes.

A. LOCA Related Analyses

1. Large Break LOCA (LBLOCA) (FSAR Section 15.4)

The current FSAR LBLOCA analysis was performed using the NRC approved 1981 ECCS Evaluation Model with BASH (WCAP-10266-P-A, Revision 2). The analysis input parameters do not need to be modified for the reduced TDF and changes in the OTΔT/OPΔT setpoints. The current analysis for Cycle 1 uses a TDF of 93,100 gpm. Also, the LBLOCA analysis does not model the OTΔT and OPΔT trip functions.

Watts Bar will perform a BASH analysis to support the Cycle 2 design parameters as a part of the core re-load process. The Cycle 2 design parameters are not expected to require changes to the Technical Specifications.

2. Small Break LOCA (SBLOCA) (FSAR Section 15.3)

A revised SBLOCA analysis was performed to accommodate the TDF of 93,100 gpm and the Cycle 2 design parameters. The analysis does not model the OTΔT/OPΔT functions. The revised analysis used the NRC approved NOTRUMP code and included a new NRC approved condensation model (COSI). The results of the analysis demonstrate that the limiting break is still a 4-inch break and the PCT is 1127°F, which is significantly lower than the 10CFR50.46 limit of 2200°F.

3. Blowdown Reactor Vessel and Loop Forces

The LOCA hydraulic forces are used in the structural qualification of various RCS components. Table 2 indicates that the T_{cold} has been reduced from 559.1 to 557.7°F as a result of the reduced TDF of 93,100 gpm and the 10% steam generator tube plugging. The reduction in T_{cold} would lead to an increase in the LOCA hydraulic forces. However, it has been determined that the increase in forces is insignificant and is bounded by margins that exist in the current analysis. Therefore, the LOCA hydraulic forces remain within the existing licensing basis limits.

4. Post-LOCA Long Term Core Cooling Requirements

The licensing commitment is that the reactor will remain shutdown by borated ECCS water residing in the sump following a LOCA. Since credit for the control rods is not taken for LBLOCA, the borated ECCS water will result in the reactor core remaining subcritical assuming all control rods are 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 reduced thermal design flow of 93,100 gpm does not change the water volumes and associated boric acid concentrations.

5. Hot Leg Switchover Time to Prevent Boron Precipitation

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 boil off of nearly 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. The reduced TDF does not change the RCS mass, and thus does not impact the hot leg switchover time.

B. Non-LOCA Related Analyses

All of the non-LOCA analyses discussed in this section (Section 3.2) have been performed with the Cycle 2 design features ($F_Q = 2.5$, $F_{\Delta H} = 1.60$), the reduced TDF of 93,100 gpm and the revised OTΔT/OPΔT setpoints. Each of these items can adversely affect the minimum DNB ratio, thus reducing safety analysis margin. Therefore, in order to help offset these items, a different treatment of operating condition uncertainties is utilized. The different treatment is the Revised Thermal Design Procedure which is discussed in WCAP-14738 and summarized as follows, as it relates to the non-LOCA analyses.

Investigations of DNBR in the applicable non-LOCA safety analyses, as currently documented in the FSAR Chapter 15, account for uncertainties in power, flow, temperature and pressure individually. In addition, uncertainties in the peaking factors are also individually accounted for explicitly. Therefore, the uncertainty in each of these critical parameters is applied in the conservative direction for each analysis in order to yield the most limiting results. This treatment of uncertainties is referred to as the Standard Thermal Design Procedure (STDP).

However, the uncertainties in the various critical parameters mentioned above are independent of each other and the effect of these uncertainties can be statistically combined. The applicable DNBR-related analyses can then be performed assuming initial conditions consistent with the nominal plant design conditions and then the minimum DNBR results can be compared to a minimum DNBR limit which includes an accounting for the statistically combined effects of the various uncertainties. This method of analysis is referred to as RTDP and provides a significant benefit in DNBR margin when compared to STDP. Therefore, the RTDP methodology is being incorporated into the non-LOCA analyses for the applicable DNB events in order to obtain the margin necessary to incorporate the desired changes.

1. <u>Initial Power Conditions Assumed in the Safety Analyses (FSAR Section 15.1.2)</u>

   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 uncertainties in the initial operating conditions have been recalculated as part of this program, and the new uncertainties are shown below:

a.	Core power	±2 percent allowance for calorimetric error
b.	Average RCS temperature	±6°F allowance for deadband and measurement error
c.	Pressurizer pressure	+70 / -50 psi allowance for steady state fluctuations and measurement error

   The new RCS flows used in the analysis are a TDF of 372,400 gpm and a minimum measured flow of 379,100 gpm.

   These changes represent an increase in the pressure uncertainty (from ±46 psi) and a decrease in the temperature uncertainty (from ±6.5°F) compared to the values currently documented in the FSAR. The bases for these uncertainty allocations are explained in the RTDP report contained in Enclosure 5. The FSAR changes will be submitted accordingly. The uncertainty in core power remains unchanged.

   The effect of the uncertainties have 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.

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

Based on the revised peaking factors and RCS flowrate, a revised set of core thermal limits was prepared using RTDP methods to identify the combinations of power, T-in (into the core), and pressurizer pressure which delineate the region beyond which either the DNB design basis is not met or vessel exit boiling would occur.

These limits provide the basis for a new set of Overtemperature and Overpower ΔT reactor protection system functions. The OTΔT and OPΔT setpoints identified in Enclosure 3, adjusted for uncertainties, have been used in the OTΔT and OPΔT protection functions.

An illustration of the new protection functions, without dynamic compensation, is shown in Figure 1. Note also that the anticipatory nature of these functions has been reduced for this program by moving the trip line farther away from the normal operating point. The reduction in the anticipatory nature of these functions provides additional margin between the nominal operating conditions and the trip function conditions, thus allowing greater operational flexibility. However, the reduction in the anticipatory nature increases the time required to achieve a reactor trip signal under accident situations.

The revised OTΔT/OPΔT setpoints have been incorporated into the applicable accident analyses presented in the following sections. Since the results of these analyses are shown to be acceptable, the modified trip functions are acceptable.

3. 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 Rod Cluster Control Assembly (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 reduced TDF has an adverse effect on the minimum DNBR, which cannot be offset by utilization of RTDP since this procedure is not applicable for analyses initiated from zero power conditions. Neither the OTΔT nor the OPΔT reactor trip functions are modeled in this analysis.

This accident was reanalyzed for the new conditions and it was shown that the DNB design basis was satisfied. In addition, the limiting maximum fuel centerline temperature was calculated to be 2234°F, which is considerably less than that which would lead to fuel centerline melting (4800°F). The results of this analysis are therefore acceptable and the conclusions documented in the FSAR remain valid.

4. 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 reduced TDF has an adverse effect on the minimum DNBR. Since the primary protection functions are the Power Range Neutron Flux - High and OTΔT functions, the changes made in the OTΔT function may delay the time required to achieve reactor trip.

This accident has been reanalyzed for the new conditions and OTΔT protection function. The RTDP methods were used to help provide additional DNB margin, and it was shown that the DNB design basis was satisfied for all cases. The results of this analysis were therefore found to be acceptable and the conclusions documented in the FSAR remain valid.

5. Rod Cluster Control Assembly Misalignment (FSAR Section 15.2.3)

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

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

- Statically misaligned RCCA

The dropped RCCA(s) transient is analyzed using the methodology described in WCAP-10297/10298 and is investigated to demonstrate that the DNB design basis remains met. The reduced TDF has an adverse effect on the minimum DNBR while the RTDP methodology provides a DNBR margin benefit. Since cases that yield a reactor trip on OTΔT do not experience the high power levels resulting from control rod withdrawal overshoot, cases that result in reactor trip following the dropped RCCA(s) are not limiting. The consequences of this accident have been analyzed for the new conditions and the minimum DNBR remains above the limit value. Therefore, the conclusions in the FSAR remain valid.

The analysis of a dropped RCCA bank results in a very rapid negative reactivity insertion that quickly generates a reactor trip via the Power Range Negative Flux Rate trip function, thus terminating the transient. The high power levels resulting from control rod withdrawal overshoot do not occur for this case due to the early reactor trip. The DNBR limit is still not challenged for the reduced TDF and the conclusions in the FSAR remain valid.

Like the dropped RCCA(s)/bank accidents above, the statically misaligned RCCA analysis is performed to verify that the DNB design basis is met. The reduced TDF has an adverse effect on the minimum DNBR while the RTDP methodology provides a DNBR margin benefit. Evaluation of this situation for the new conditions indicates that the minimum DNBR remains above the limit value. Therefore, the conclusions in the FSAR remain valid.

6. <u>Uncontrolled Boron Dilution (FSAR Section 15.2.4)</u>

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 reduced TDF does not directly impact this analysis. However, changes made to the OTΔT setpoint can potentially impact the results of the case analyzed during full power operation with manual rod control. The dilution leads to a slow reactivity insertion essentially equivalent to an Uncontrolled RCCA Bank Withdrawal at

Power. At a conservative insertion rate of 0.6 pcm/sec, there are more than 15 minutes available for operator action from the time of the trip alarm (OTΔT) to a loss of shutdown margin. Therefore, the results for this case are acceptable and the conclusions in the FSAR remain valid. No other cases are impacted by the scope of this evaluation.

7. 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) (RCP(s)) which may occur from an undervoltage condition in the electrical supply to the RCP(s) 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 reduced TDF has an adverse effect on the minimum DNBR, this accident has been reanalyzed. The OTΔT and OPΔT protection functions are not modeled. The RTDP methods were used to help provide additional DNB margin, and it was shown 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.

8. Startup of an Inactive Reactor Coolant Loop (FSAR Section 15.2.6)

This transient is caused by starting an idle Reactor Coolant Pump without bringing the inactive loop hot leg temperature close to the core inlet temperature. This causes both an increase in core coolant flow and injection of cold water into the core which could result in a rapid core power increase due primarily to moderator reactivity feedback. However, Watts Bar Technical Specification 3.4.4 requires that all four loops must be in operation during both Modes 1 and 2. Initiation of this accident from a lower mode of operation, when RCS temperatures would be uniform, would not challenge the DNBR limit. The changes addressed in this evaluation would not adversely affect the consequences of such an event. Therefore, the conclusions documented in the FSAR remain valid.

9. 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 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. The reduced TDF has an adverse effect on the minimum DNBR and the relaxed OTΔT setpoint delays reactor trip. This accident has been reanalyzed for the new conditions and OTΔT protection function. The RTDP methods were used to help provide additional DNB margin, and it was shown that the DNB design basis was satisfied. The results of this case were found to be acceptable and the conclusions documented in the FSAR remain valid.

The case without pressure control is analyzed to confirm that the primary and secondary pressures remain below 110% of design. This case was reanalyzed with the increased positive RCS pressure uncertainty and the reduced TDF. Reactor trip occurs on High Pressurizer Pressure, and the maximum RCS and Main Steam System pressures were 2652 and 1281 psia, respectively. Since each of these pressures is below 110% of the respective design pressures, the results are acceptable and the conclusions documented in the FSAR remain valid.

10. Loss of Normal Feedwater (FSAR Section 15.2.8)

This event is analyzed, both with and without available offsite power, to demonstrate that the auxiliary feedwater system is of sufficient capacity to remove core decay heat, stored energy and RCS pump heat following the loss of main feedwater to the steam generators. This is verified by demonstrating that the thermal expansion in the RCS coolant inventory is sufficiently limited such that at no point does the pressurizer become water solid. The limiting results occur for operation at full power. However, this event is also analyzed at part-power conditions for both partial and complete Loss of Normal

Feedwater transients in order to validate the Trip Time Delay (WCAP-13462, Revision 1) system delay times.

This event is not analyzed to investigate DNBR or fuel temperature response, and the OTΔT and OPΔT protection functions are not modeled. The only changes addressed in this evaluation which impact this analysis are the reduced TDF and the new uncertainties in initial conditions. The analysis results are not sensitive to the RCS flowrate, so the flow reduction will not adversely impact the analysis results. A comparison of the new initial condition uncertainties discussed previously shows that the new uncertainty in T-avg is bounded by the current value and that the uncertainty in core power is unchanged. Although the pressurizer pressure uncertainty is no longer bounded by the current analysis, the analysis results are not significantly sensitive to the initial pressure. RCS overpressurization is not a concern for this accident, and pressurizer sprays and PORVs are modeled in order to maximize pressurizer insurge. The current analysis results shown in the FSAR demonstrate that considerable margin exists to the pressurizer fill criterion, and a slight variation in the initial pressure will not change the conclusions of the analysis. Therefore, the conclusions documented in the FSAR remain valid.

11. 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.

The reduced TDF has an adverse effect on the minimum DNBR. In addition, the relaxation in the ΔT trip functions can delay the time required to reach a reactor trip signal. This accident has been reanalyzed for the new conditions. The RTDP methods were used for cases evaluated at full power, and it was shown that the DNB design basis was satisfied for each case. For the cases analyzed at zero power conditions, when RTDP methods are not applicable, it was demonstrated that the maximum reactivity insertion rate was less limiting than that assumed for the Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition event (FSAR Section 15.2.1). The results are therefore bounded by the results of the RCCA withdrawal transient. The results of this analysis are therefore acceptable for these conditions, and the conclusions in the FSAR remain valid.

12. 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. The reduced TDF has an adverse effect on the minimum DNBR, while incorporation of RTDP methods provides a DNBR benefit. Since the primary protection function for this event is OTΔT, the changes made in the OTΔT function may delay the time required to achieve reactor trip. Also, the analysis is not sensitive to the changes in the initial design steam flow, temperature and pressure identified in Table 2. However, the transient response to this accident is relatively mild such that the OTΔT setpoint is typically not reached for this event and the reactor stabilizes at a new equilibrium condition without generating a reactor trip.

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.

13. 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. The reduced TDF has an adverse effect on the minimum DNBR. Since the primary protection function for this event is OTΔT, the changes made in the OTΔT function may delay the time required to achieve reactor trip.

This accident has been reanalyzed for the new conditions and OTΔT protection function. The RTDP methods were used to help provide additional DNB margin, and it was shown that the DNB design basis was satisfied. The results of this analysis were therefore found to be acceptable and the conclusions documented in the FSAR remain valid.

14. <u>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)</u>

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 minimum DNBR limit is not exceeded as a result of any recriticality. The results of the Major Rupture cases bounds that of the Accidental Depressurization cases.

The event was reanalyzed for the new conditions and included the flow contribution from one centrifugal charging pump (CCP) and one safety injection pump (SIP) (i.e., one high head and one intermediate head pump). The intermediate head pump provides a considerable amount of borated safety injection flow but is conservatively ignored in the analysis currently presented in the FSAR. However, due to the low RCS pressures following a steamline break or accidental depressurization, flow from the SIP can also be credited. A comparison of the previously analyzed CCP-only data with CCP plus SIP flow data, indicates that below approximately 1430 psia, CCP plus SIP data provides a benefit.

For the major steamline rupture cases, the RCS pressure drops very rapidly such that the RCS pressure at the time ECCS flow begins delivery is considerably lower than 1430 psia. The resulting increase in flow delivery due to inclusion of the SIP contribution, therefore, has a beneficial impact on the core response.

For the accidental depressurization case, the RCS pressure also drops below 1430 psia, but within 5 seconds of the time which safety injection flow begins. The resulting increase in flow delivery due to inclusion of the SIP contribution leads to faster purging of the potentially unborated water initially in the injection lines, such that the borated flow is available earlier and with a greater flowrate than is currently analyzed. Note that the water is assumed to be unborated but could potentially be at prevailing RWST concentration. Therefore, the increased flowrate for this event also has a beneficial impact on the core response.

The reanalysis of this event yields a relatively small positive reactivity excursion and therefore only a very small return-to-power. Analysis of this response indicates that the DNB design basis continues to be met. Therefore, the conclusions in the FSAR remain valid.

15. <u>Inadvertent Operation of Emergency Core Cooling System (FSAR Section 15.2.14)</u>

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.

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. The reduced TDF has an adverse effect on the minimum DNBR, but the analysis trends would be unchanged and the minimum DNBR would continue to correspond to the RCS conditions during normal operation. The minimum DNBR is still above the DNBR limit and the conclusions documented in the FSAR for this case remain valid.

The case analyzed for pressurizer filling does not model OTΔT/OPΔT trip functions; therefore, these items do not impact the analysis. In addition, the analysis results are not sensitive to variations in the RCS flow, so the reduced TDF will not impact the results of this analysis. However, the uncertainties in initial conditions have been redetermined and the new pressure uncertainty does not bound the current value, and as a non-DNB-related analysis, uncertainties in initial conditions are included in this investigation. Initial pressure is conservatively minimized in the analysis in order to reduce the pressure at which the pressurizer sprays would be actuated, thus minimizing the pressure against which the ECCS system must deliver. As mentioned in Section 3.2.1, the new pressure uncertainty is 4 psi greater in the negative direction than the current uncertainty. This difference is very small, and sufficient margin exists in the analysis such that this minor difference would not change the conclusions of this analysis. Therefore, the conclusions documented in the FSAR for this case remain valid.

16. <u>Single Rod Cluster Control Assembly Withdrawal at Full Power (FSAR Section 15.3.6)</u>

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. The relaxation in the OTΔT setpoint and the reduced TDF adversely impact the DNB response while incorporation of RTDP methods provide DNB margin.

Evaluation of this transient for the revised conditions indicates 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 remain valid.

17. Major Rupture of a Main Feedwater Pipe (FSAR Section 15.4.2.2)

This event is analyzed, both with and without available offsite power, to demonstrate that the auxiliary feedwater system has sufficient capacity to remove core decay heat, stored energy, and RCS pump heat following the rupture of a main feedwater line that results in an inventory blowdown from one steam generator and a loss of main feedwater to all loops. This is demonstrated by showing that no hot leg boiling occurs prior to the time at which the secondary-side heat removal capacity exceeds the RCS heat generation rate. The limiting results occur for operation at full power. However, this event is also analyzed at part-power conditions in order to validate Trip Time Delay (WCAP-13462, Rev. 1) system delay times.

This event is not analyzed to investigate DNBR or fuel temperature responses, and the OTΔT and OPΔT protection functions are not credited. The only changes addressed in this evaluation which impact this analysis are the reduced TDF and the new uncertainties in initial conditions. The hot leg saturation margin, which is used as an event acceptance criterion, is not sensitive to variations in RCS flowrate and this change does not impact the analysis results. A comparison of the new initial condition uncertainties discussed above shows that the uncertainty in T-avg is bounded by the current value and that the uncertainty in core power is unchanged. Although the pressurizer pressure uncertainty is no longer bounded by the current analysis, sensitivity studies performed for this accident have demonstrated that the analysis results are not sensitive to the initial pressure. Therefore, the conclusions documented in the FSAR remain valid.

18. 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. A reactor trip via the Low RCS Flow protection function terminates this event very quickly. The OTΔT and OPΔT protection functions are not modeled.

The reduced TDF has an adverse effect on the fuel thermal transient. 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 the fuel temperature response).

This accident has been reanalyzed for the new conditions and yielded the following results:

Maximum RCS Pressure: 2640 psia
Maximum Clad Temperature: 1825°F
Maximum $Zr-H_2O$ reaction: 0.3%

The maximum RCS pressure is less than that which would cause stresses to exceed the faulted condition stress limits. In addition, the PCT is considerably less than 2700°F to meet the requirement of maintaining a coolable core geometry for a locked rotor/shaft break analysis for a zirconium-water reaction less that 16 percent. Therefore, the fuel and RCS integrity is maintained, and the conclusions in the FSAR remain valid.

A separate case was analyzed to determine the percentage of fuel rods that are predicted to undergo DNB as a consequence of this accident. The increased peaking factors and decreased flow adversely impact this analysis. However, the RTDP methods are used to help offset these penalties. The analysis confirms that the number of rods that undergo DNB is less than 13% and is therefore bounded by the percentage calculated for the current analysis. Thus, the conclusions in the FSAR remain valid.

19. <u>Rupture of a Control Rod Drive Mechanism Housing (FSAR Section 15.4.6)</u>

The rod ejection 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 to show that the fuel and clad limits are not exceeded.

The reduced TDF has an adverse effect on the fuel thermal transient. 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).

This accident has been reanalyzed for the new conditions and yielded the following results at the hot spot:

Case	Maximum Clad Average Temperature (°F)	Maximum $Zr\text{-}H_2O$ Reaction (%)	Maximum Fuel Stored Energy (cal/gm)	Maximum Fuel Melt (%)
Zero Power BOL	2334	1.3	132	0
Zero Power EOL	2708	3.1	150	0
Full Power BOL	2217	0.8	173	3.5
Full Power EOL	2136	0.7	166	2.8

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.

20. <u>Steamline Break at Power with Coincidental Rod Withdrawal</u>

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.

The reduced TDF has an adverse effect on the minimum DNBR and on the maximum linear heat generation rate (expressed in terms of kw/ft of fuel rod). Since the primary protection functions are the Low Steam Pressure ESF function and the OPΔT function, the changes made in the OPΔT function can also delay the time required to achieve reactor trip.

This accident has been reanalyzed for the new conditions and OPΔT protection function. The RTDP methods were used to help provide additional DNB margin, and it was shown that the DNB design basis was satisfied. Additional cases were analyzed to confirm that the maximum linear heat generation rate was acceptable without the RTDP methodology since this criteria is not DNB-related. 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.

C. Steamline Break Mass and Energy Releases

The safety evaluations presented in this section have been performed for 10% SGTP and a TDF of 93,100 gpm and will bound plant operation at 0% SGTP and a TDF of 93,100 gpm. Thus, the discussion presented in this section includes both the 10% tube plugging and TDF of 93,100 gpm, even though the provisions for 10% tube plugging are not needed to support the proposed Technical Specification changes.

1. SLB Mass and Energy Releases Inside Containment

An evaluation of the SLB mass and energy releases indicates that the existing licensing basis remains more limiting. 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. A limiting analysis for the mass and energy releases following the SLB is a function of the amount of heat transferred from the primary side of the RCS to the secondary side and out the postulated pipe rupture. Therefore, the quantity of heat transfer in the steam generators is maximized for the mass and energy release analysis.

Conditions which define maximum primary-to-secondary heat transfer include 0% steam generator tube plugging (SGTP) and a TDF of 97,500 gpm for forced convection heat transfer. The assumptions of 10% SGTP and a TDF of 93,100 gpm are in the opposite direction for maximizing the heat transfer. Thus, each represents a benefit to the analysis conditions. The existing assumptions for 0% SGTP and a TDF of 93,100 gpm remain conservative.

The reactor trip protection functions are provided by a secondary-side SI signal (low steam pressure) or a containment pressure SI signal, both producing a reactor trip signal. Thus, the OTΔT/OPΔT protection functions are not needed (and not credited) in the analysis of the SLB mass and energy releases inside containment.

2. <u>SLB Mass and Energy Releases Outside Containment</u>

SLB mass and energy releases outside containment are used as input boundary conditions to an environmental qualification of safety-related equipment and instrumentation. A limiting analysis for the mass and energy releases following the SLB is a function of the amount of heat transferred from the primary side of the RCS to the secondary side and out the postulated pipe rupture. Therefore, the quantity of heat transfer in the steam generators is maximized for the mass and energy release analysis.

Conditions which define maximum primary-to-secondary heat transfer include 0% SGTP and a TDF of 97,500 gpm for forced convection heat transfer. The assumptions of 10% SGTP and a TDF of 93,100 gpm are in the opposite direction for maximizing the heat transfer. Thus, each represents a benefit to the analysis conditions. The existing assumptions for 0% SGTP and TDF of 97,500 gpm remain conservative.

The reactor trip protection functions are provided by a secondary-side SI signal (low steam pressure), an OPΔT signal, or a steam generator low-low water level signal. Conservative assumptions are in the direction of minimizing the steam generator inventory to produce an earlier onset of superheated steam resulting from tube bundle uncovery. Therefore, changes to the OPΔT setpoints (which delay reactor trip) could have an adverse impact on the SLB mass and energy releases outside containment. A revised calculation of the SLB mass and energy releases outside containment has been performed and documented for Watts Bar.

The analysis of the impacts on equipment qualification outside containment has been completed. Revised temperature profiles have been generated and are being incorporated into the Equipment Qualification (EQ) Program.

3. <u>Short-Term SLB Mass and Energy Releases</u>

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 only analysis input to the calculation that could be affected by either 10% SGTP or TDF of 93,100 gpm is the steam generator steam mass at hot-zero-power conditions. The OTΔT and OPΔT trip functions are not modeled due to the short duration of the transient. The 10% SGTP and TDF of 93,100 gpm do not change the steam mass at zero power, since there is no change in the calculated no-load temperature and pressure. Based on this evaluation, it is concluded that these changes do not affect the licensing basis analysis for the short term SLB mass and energy releases outside and inside containment.

4. Short-Term FLB Mass and Energy Releases

Short-term FLB mass and energy releases are used as input to a compartment or subcompartment pressurization analysis. The analytical method for the short-term FLB mass and energy releases is a hand 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). Since neither of these inputs has changed with respect to the analysis of record, the results of the short-term feedline break mass and energy releases remain valid. The 10% SGTP, TDF of 93,100 gpm, and OTΔT/OPΔT trips are not modeled since they do not affect these initial conditions for the main feedwater system inventory.

5. Radiological Steam Releases for Dose Calculations

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. The only analysis inputs to the calculation that could be affected by either 10% SGTP or TDF of 93,100 gpm are the secondary system boundary condition values. Specifically, these are the steam pressure and steam temperature, and the full-power and zero-power steam generator steam and water masses. The steam pressure and temperature are higher for the 0% SGTP and TDF of 97,500 gpm condition than for the 10% SGTP and TDF of 93,100 gpm condition and thus are more limiting. Thus, the existing releases are bounding.

D. Steam Generator Tube Rupture (SGTR) (FSAR Section 15.4.3)

The safety evaluations presented in this section have been performed for 10% SGTP and a TDF of 93,100 gpm and will bound plant operation at 0% tube plugging and a TDF of 93,100 gpm. Thus, the discussion presented in this section includes both the 10% tube plugging and TDF of 93,100 gpm, even though the provisions for 10% tube plugging are not needed to support the proposed Technical Specification changes.

The licensing basis SGTR analysis for Watts Bar is presented in WCAP-13575, Revision 1. 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 10CFR100.

1. Margin to Overfill Analysis

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 performing 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 TDF of 93,100 gpm, 10% tube plugging and revised OTΔT/OPΔT setpoints have been incorporated into the margin to overfill analysis. The results demonstrate that the margin to overfill is still adequate and has been enhanced from 119 to 222 cubic feet.

2. Offsite Dose Calculation

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. Currently, Westinghouse calculates the applicable steam releases and break flows for use in the dose analysis.

The steam release calculation has been modified to include the TDF of 93,100 gpm, 10% SGTP, and the revised OTΔT/OPΔT setpoints. The revised steam releases and break flows have been evaluated as to their impact on the SGTR dose analysis. The results of the evaluation indicate that the beta, gamma and inhalation (thyroid) doses for the SGTR event decrease slightly from those currently shown in Table 15.5-19 of the FSAR, e.g., 0.0604, 0.3560, and 15.7606 rem from 0.6341, 0.3732, and 16.10, respectively of the Exclusion Area Boundary (EAB), and 0.1403, 0.08269, and 3.661 rem from 0.1473, 0.867, and 3.74 at the Low Population Zone. However, in all cases the resulting dose rates at the EAB and low population zones are still less than 10 percent of the 10 CFR 100 limits, and the control room operator doses from this event are still well below the GDC 19 reference values.

E. Containment Mass and Energy Releases

The safety evaluations presented in this section have been performed for 10% SGTP and TDF of 93,100 gpm and will bound plant operation at 0% tube plugging and a TDF of 93,100 gpm. Thus, the discussion presented in this section includes both the 10% SGTP and TDF of 93,100 gpm, even though the provisions for 10% SGTP are not needed to support the proposed Technical Specification changes.

1. 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 containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure. Analysis results are also used to support environment qualification.

The 10% SGTP and TDF of 93,100 gpm change the RCS temperatures that exist before the LOCA occurs. Higher initial temperatures are more limiting since the initial energy content is maximized, which causes a faster ice melt and higher peak pressure. The analysis uses the maximum composite RCS T_{HOT} and T_{COLD} from the 0% SGTP/TDF of 97,500 gpm and the 10% SGTP/TDF of 93,100 gpm conditions. (See Table 2 for calculated values.) It was determined that by using the maximum composite temperature, the initial blowdown energy release during a LOCA increased by 2.16×10^6 Btu. The increase in the energy release can be offset by taking credit for a more recent decay heat standard. The current model utilizes an older ANSI decay heat standard. Using the more recent 1979 decay heat model and the actual core power has shown a benefit of 23.69×10^6 Btu. This benefit more than offsets the previously noted increase.

2. Short-Term LOCA Mass and Energy Release Analysis

a. Introduction

Several evaluations are performed to support the loop subcompartment, reactor cavity, and pressurizer enclosure analysis. The 10% SGTP and TDF of 93,100 gpm change the RCS temperatures that exist before the LOCA occurs. Lower initial RCS temperatures are more limiting since it is expected that the lower temperatures would increase the initial mass flow into the compartment.

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 deviations in the RCS inlet and outlet temperatures from the 10% SGTP and TDF of 93,100 gpm relative to those used in the current analysis of record.

The use of the lower temperatures maximizes the critical mass flux in the Zaloudek correlation. The analysis uses the minimum composite RCS T_{HOT} and T_{COLD} that are calculated for the 0% SGTP/TDF of 97,500 gpm and the 10% SGTP/TDF of 93,100 gpm conditions. (See Table 2 for calculated values.)

b. <u>Loop Subcompartment Analysis</u>

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 helps to verify the adequacy of the ice condenser performance.

Based upon the evaluation which considered the lowest composite initial RCS temperature conditions discussed above, the short-term releases used for the subcompartment analyses could increase by approximately 20%.

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 20% 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. Additionally, and more importantly, analysis was conducted on a similar ice condenser design that showed the temperature reduction was actually a benefit instead of a penalty. Thus, the current licensing basis mass and energy releases remain bounding.

c. 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 currently forms the licensing basis for this subcompartment. It was estimated, as previously noted, that the peak releases would conservatively increase by approximately 20%. However, based upon results of the structural analysis of the reactor coolant system, a better estimate of the break size is 45 sq. in. (see WCAP-8889). The reduced rates from this reduced break size more than offset the predicted 20% increase. For example, it is expected that 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 more than offsets the 20% increase. Since the current mass and energy releases remain bounding, the current reactor cavity pressure analysis remains bounding.

d. 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 licensing basis 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 the greater than 25% margin in the mass and energy releases. This more than offsets the predicted 20% increase in mass and energy releases. Therefore, the current mass and energy releases remain bounding, and the current pressurizer enclosure pressure analysis remains bounding.

3. <u>Maximum Reverse Pressure Differential Analysis</u>

Following a LOCA, the pressure and temperature in the lower compartment of the containment increases. This 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. As indicated in Table 2, the T_{hot} increased by about 1.4°F and the T_{cold} decreased by about 1.4°F as a result of the 10% SGTP/TDF of 93,100 gpm. Also, the T_{avg} remained unchanged by the 10% SGTP/TDF of 93,100 gpm. These changes do not affect the mass and energy releases since they would tend to offset each other in terms of increasing and decreasing the releases. Therefore, the temperature changes would not affect the results of the calculation.

Furthermore, the purpose of this analysis is to show that significant margin exists in the design. An existing peak calculated differential pressure of 0.65 psi is still applicable and is significantly lower than the structural design pressure drop capability of the operating deck. Thus, the proposed changes will have a minimal impact, if any, on the analysis and there is significant analysis margin available. The current analysis of record remains bounding.

F. NSSS Components

The safety evaluations presented in this section have been performed for 10% SGTP and a TDF of 93,100 gpm and will bound plant operation at 0% tube plugging and a TDF of 93,100 gpm. Thus, the discussion presented in this section includes both the 10% tube plugging and TDF of 93,100 gpm, even though the provisions for 10% tube plugging are not needed to support the proposed Technical Specification changes.

As mentioned in Section I, the 10% SGTP and TDF of 93,100 gpm changed the NSSS Performance Parameters. As a result, it is necessary to demonstrate that the affected NSSS Components maintain their structural integrity for the 40-year plant design life and continue to perform their design functions with the new parameters.

A review of Table 2, which contains the revised parameters, indicates that the following parameters have changed:

- T_{hot} increased by 1.4°F (617.3° to 618.7°F)
- T_{cold} decreased by 1.4°F (559.1° to 557.7°F)
- TDF decreased by 4400 gpm (97,500 to 93,100 gpm)
- Steam pressure decreased by 42 psi (1000 to 958 psi)
- Steam temperature decreased by 5.2°F (544.6° to 539.4°F)
- Steam flow decreased by 0.4% (15.14 to 15.08 x 10^6 lbs/hr)

The impact that these changes have on the affected NSSS systems and components is provided as follows.

1. NSSS Design Transient Curves

It has been determined that the 10% SGTP and TDF of 93,100 gpm do not require a change to the NSSS design transient curves. In general, the transients that are expected to occur during the 40-year plant life are used for the design of the NSSS systems and components. Primary and secondary temperature and pressure data are used to define the transient conditions. Many of these transients are initiated from 100% power and, thus, could be affected by the changes in the NSSS performance parameters.

The revised parameters were reviewed with respect to their impact on the NSSS design transient curves, and it was determined that current design transient curves are applicable without modification. The full power T_{hot} and T_{cold} remain within 2°F of the design values (T_{avg} is unchanged) and thus will have negligible impact on transient behavior. Similarly, the revised steam temperature is within 6°F of the design value, which is considered to have a negligible impact on transient behavior, considering the small changes to primary side conditions. Thus, no base transient curve revisions were made. Discussion of primary temperature and secondary parameter changes in the following sections are in reference to RCS parameters contained in Table 2.

2. Reactor Vessel

A reactor vessel analysis is performed to demonstrate that the vessel does not exceed any of the maximum ranges of stress intensity limits and fatigue usage limits in Section III of the ASME B&PV Code.

The 1.4°F increase in T_{hot} is a penalty in the analysis since a higher T_{hot} increases the temperature change and stress associated with plant loading and unloading. The reactor vessel stress analysis was performed with the new T_{hot} of 618.7°F. The results of the analysis indicate that the change in thermal stress at the outlet nozzle safe ends (due to bimetallic weld thermal expansion mismatch) would increase by less than 0.1%, and thus, the change in stress is negligible. In addition, the results indicate that the largest increase in stress within the nozzle is only 0.12 ksi which is negligible compared to the existing calculated stress, which is well below the 80.1 ksi stress limit. Thus, it was concluded that the stress intensity and the associated maximum cumulative fatigue usage factors were still within the applicable ASME Code limits.

The revised vessel inlet temperature resulted in a reduced temperature variation during normal plant loading and plant unloading for those regions of the reactor vessel assumed to be in contact with vessel inlet water during normal reactor operation. The 1.4°F decrease in T_{cold} is a benefit since a lower T_{cold} reduces the temperature change and stress during loading and unloading with the vessel parts in contact with T_{cold}. These parts include the upper head, main closure, inlet nozzles, vessel shell and bottom head. Therefore, the current reactor vessel stress reports remain applicable for the limiting locations in these parts.

3. Reactor Internals

The reactor internals support the core and direct flows within the reactor vessel. While directing the primary flow through the core, the reactor internals also establish secondary coolant flow paths for cooling the upper regions of the reactor vessel and for cooling the internals structural components. Changes in the primary coolant system flow rates and temperatures that result from changes in SGTP levels and RTDF also produce changes in the boundary conditions experienced by the reactor internal components. This then leads to changes in the temperatures experienced by these components.

The components which could be most impacted by the reduction in TDF from 97,500 gpm/loop to 93,100 gpm/loop and the vessel/core inlet temperature from 559.1° to 557.7°F are: the baffle/barrel region components (baffle plates, core barrel, baffle/former bolts), the upper/lower core plates, and the neutron panels/support system. Both structural and thermal hydraulic assessments/evaluations were performed for the reactor internal components.

An engineering assessment was performed to demonstrate that the short term and long term structural integrity of the baffle/barrel region components and the lower core plate were not adversely impacted by the change in operating conditions. The result of this structural assessment was that the reduction in TDF and the corresponding reduced heat transfer coefficients will have negligible impact on the structural integrity of the baffle/barrel region components and lower core plate. Based on engineering assessments performed for other similar type plants, it is Westinghouse's engineering judgment that the upper core plate and neutron panels/support system will not be adversely impacted by the change in operating conditions.

In addition, a thermal hydraulic analysis was performed to confirm that the design core bypass flow percentage can be maintained and that the reactor vessel head temperature will remain at the cold leg temperature. The thermal hydraulic evaluation showed that with 10 percent SGTP and 2 percent RTDF, the core bypass percentage is about 8.8 percent which remains under the current 9.0 percent design limit and the reactor vessel head average fluid temperature is maintained at T_{cold}.

Also, a Rod Control Cluster Assembly (RCCA) rod drop time evaluation was performed because with the reduced vessel/inlet temperature, the water density would increase, thereby increasing the RCCA rod drop time. The RCCA rod drop time evaluation confirmed that the current design RCCA drop time remains applicable at the new operating conditions.

In summary, the evaluation for reactor internals demonstrated that there would be no adverse impact from the proposed 10% SGTP and TDF of 93,100 gpm.

4. Reactor Coolant Piping and Supports

The revised parameters (Table 2) were reviewed for impact on the existing analysis for the following components: reactor coolant loop piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. The temperature changes associated with the 10% SGTP and TDF of 93,100 gpm necessitated reconciliation of potential load changes in the components. The changes in the temperatures and pressures were factored into the fatigue aspects of the piping evaluation. All of the thermal expansion, seismic, and LOCA analyses performed on the piping systems incorporate full power conditions. Thermal design transients related to the fatigue aspects of the analysis were also factored into the evaluation.

The revised temperatures were assessed to determine the impact on the existing analysis results for the primary reactor coolant loop piping, primary loop nozzles, and primary equipment supports. The assessment considered changes in loads generated as a result of the temperature changes and the potential impact on the components. The changes to the hot, cold and cross-over leg temperatures are negligible since they do not generate any measurable load changes. Thus, the evaluation determined that there is no impact on any existing results.

The evaluation performed for the pressurizer surge line stratification analysis focused on the fatigue analysis. The changes in T_{hot} directly affect any fatigue assessments. The 1.4°F increase in T_{hot} is a benefit since it reduces the ΔT between the pressurizer and the hot leg. Thus, the existing analysis remains bounding.

5. Leak Before Break (LBB) Analysis

The current LBB evaluation was performed for the primary loops to provide technical justification for eliminating large primary loop pipe rupture as the structural design basis. The evaluation was documented in WCAP-11985.

In order to demonstrate the elimination of RCS primary loop pipe breaks, the following objectives must be 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.

These objectives were met in WCAP-11985.

The LBB evaluations include the applied loads as the input. Both normal operating loads and the faulted loads are used as input to the evaluations.

The effect of temperature changes resulting from the revised parameters on the primary loop loads is negligible since the changes do not generate any measurable load changes. In addition, the temperature changes have a negligible impact on the material properties since these properties are relatively insensitive to small temperature changes. Since the magnitude of change in loads and material properties is negligible, the LBB margins previously calculated and documented in WCAP-11985 will remain unchanged.

6. Pressurizer

An analysis was performed to assess the impact of the revised NSSS parameters on the pressurizer components. The conditions that affect the primary plus secondary stresses, and the primary plus secondary plus peak stresses are the changes in the T_{hot}, T_{cold} and the pressurizer transients. A review of the revised temperature parameters showed that the changes in T_{hot} and T_{cold} are very small and are enveloped by the current stress analysis. The operating temperature assumed for the pressurizer is 653°F. The following is a summary of the temperature changes incurred (see Table 2 for T_{hot} and T_{cold} data):

Parameter:	Current Value:	Revised Value:	$\Delta T_{current}$:	ΔT_{SGTP}:
T_{hot}, °F	617.3	618.7	653-617.3 = 35.7	653-618.7 = 34.3
Tcold, °F	559.1	557.7	653-559.1 = 93.9	653-557.7 = 95.3

For components affected by T_{hot} (e.g., the surge nozzle), the temperature difference for the revised parameters is bounded by the current design conditions since the ΔT is reduced from 35.7 to 34.3°F. The limiting component affected by changes in T_{cold} is the spray nozzle for which the design analysis addresses a ΔT of 125°F; clearly, this bounds the new ΔT of 95.3°F.

No changes were made to the design transients (see previous section entitled NSSS Design Transient Curves) and, therefore, the transients specified in the current

Design Specification are still applicable. For this reason, it was concluded that the revised parameters will not have any impact on the pressurizer stress analysis and fatigue analysis.

7. Reactor Coolant Pump (RCP)

a. RCP Pressure Boundary

An evaluation of the pressure boundary components of the Reactor Coolant Pumps (RCP) was performed for the revised NSSS parameters in Table 2.

Since the RCPs are in the cold leg of the primary piping, the cold leg temperature is one of the parameters of interest in the fatigue evaluation. The input parameters of interest to the ASME Code pressure boundary analysis are the operating pressure and temperature and any transient changes. For the revised parameters, the pressure remains at 2250 psia, thus no pressure increase needs to be justified. It was also indicated that the system transients are applicable without modification. Therefore, only the RCS temperature changes need to be addressed.

The ASME Code pressure boundary components are described in the generic code analysis reports and the specific plant pressure boundary stress reports. The evaluation included reviewing the various generic and plant specific analyses for the RCPs. A previous evaluation had addressed the RCP shaft load stresses which show very adequate margin (in the stress analysis) to accommodate the minor (~2.3%) increase in hydraulic loads that occur due to reducing the TDF from 97,500 to 93,100 gpm.

In addition, a reduction in the current temperature from 558.8°F to 557.5°F for the revised parameters results in only a 1.3°F difference. This temperature difference (~0.25%) is judged to be insignificant and would not affect the generic stress report for the RCP.

Since there are no new transients and the original transients still bound the revised NSSS Parameters, the various Code fatigue waiver/analyses used in the generic stress reports and PBSRs remain applicable. Since the operating pressure remains the same and the operating temperature only slightly decreases, all present Code pressure boundary analyses remain conservatively applicable without further evaluation.

b. Reactor Coolant Pump Motor

The RCP motors were evaluated for the worst case loads based on the revised NSSS parameters. The RCP motors were evaluated at four conditions: continuous operation at revised hot loop rating, operation at revised cold loop rating, starting, and loads on thrust bearings.

The Equipment Specification requires the motor to drive the RCP continuously under hot loop conditions without exceeding a stator winding temperature rise of 75°C. It was determined that the worst case hot loop load under the revised operating conditions is 6685 HP. The revised load does not exceed the nameplate rating of the motor (7000 HP). The motors have been shown by test to operate within specification limits at the hot loop nameplate rating. Therefore, continuous operation at the revised load is acceptable.

With respect to operation at the revised cold loop rating, the Equipment Specification requires the motor to drive the RCP for 3,000 total hours (up to 50 hours continuously) at the specified maximum load. The evaluation determined that the worst case loop load under the revised operating conditions is 8764 HP. This represents a 0.16% increase over the nameplate cold loop rating of the motor (8750 HP). The increase in stator winding temperature due to this new load is estimated to be 0.3°C. This increase is considered to be insignificant, and therefore operation at the revised load is acceptable.

When starting, the motor is required to start across the line under cold loop conditions, with 80% starting voltage, against the reverse flow of the other three RCPs running at full speed. The limiting component for this type of starting duty is the rotor cage winding. A conservative all heat stored analysis was used to determine if the cage winding temperature exceeds the design limits which are 300°C on the bars and 50°C on the resistance rings. A new load torque curve was developed, and in reviewing it, the starting temperature rise for the rotor bars and resistance rings was calculated. The results show a bar temperature of 216.7°C and ring temperature of 36.26°C. These temperatures do not exceed the design limits. Therefore, the motor can safely accelerate the load under worst case conditions.

Finally, performance of the thrust bearings in an RCP motor can be adversely affected by excessive or inadequate loading. An evaluation of the new conditions indicate the axial down thrust is increased from 55,000 lbs. to 56,086 lbs. for hot loop operation and reduced from 75,000 lbs. to 71,828 lbs. for cold loop operation. The thrust bearing was designed for loads exceeding 101,200 lbs. Therefore, the thrust bearings are acceptable for the revised loads.

Based on the analysis of the four areas that affect motor performance, the RCP motors are considered acceptable for operation with the revised NSSS parameters.

8. Control Rod Drive Mechanism (CRDM)

An evaluation of the pressure boundary components of the Full-Length Control Rod Drive Mechanisms (F/L CRDM) and Part-Length Control Rod Drive Mechanisms (P/L CRDM) was performed for the revised NSSS parameters.

For the Watts Bar design, the CRDMs are exposed to the cold leg temperatures. The P/L CRDMs are not active, but the pressure boundary components remain attached to the reactor vessel head.

The input parameters of interest to the ASME Code pressure boundary analysis are the changes in operating pressure and temperature and any transient changes. For the revised conditions the pressure remains at 2250 psia, thus no pressure increase needs to be justified. It was also indicated that the system transients are applicable without modification for the revised conditions. Therefore, only the RCS operating temperature changes have been addressed. The original transients and operating temperatures are given by the respective equipment specifications.

The ASME Code pressure boundary components are described in the generic code analysis reports and the specific plant pressure boundary stress reports (PBSRs). The evaluation included reviewing the various generic and plant specific analyses for the CRDMs. The CRDMs are affected by the cold leg temperature. The present temperature of 559.1°F drops to 557.7°F for the revised conditions (see Table 2), only a 1.4°F difference. This difference is insignificant (~0.25%) and would not affect the CRDM generic reports.

9. Steam Generators (SGs)

a. SG Moisture Carryover

The as-built moisture separator packages for the steam generator Models D2, D3 (installed at Watts Bar) and D4 steam generators, in terms of their major components, are substantially the same. Each contains 12, 20" primary swirl vane separators and a double tier dryer. Full power separator loadings for all 4-loop plants with D steam generators are the same.

Early results from some of the 4-loop plants with Model D steam generators indicated that modifications were necessary to the Watts Bar Units in order to meet the 0.25% moisture carryover limit at full power. The Watts Bar separator modifications comprised the following major elements:

- Perforated plates added to dryer faces
- Removal of the mid deck plate hatches
- Supplemental upper tier drains
- Conversion from a 0.6 primary separator orifice diameter ratio to a 0.7 orifice with an orifice collar added.

The primary separator orifice ratio in the last modification refers to the orifice located at the outlet to the primary separators and its diameter as a ratio to the primary separator riser diameter. Among the 4-loop plants with Model D steam generators, only the ones with 0.7 primary separator orifices operated at full power moisture levels below 0.25%. The moisture carryover at WBN during normal full power operation has been tested at approximately 0.356 percent.

The effect of the revised NSSS parameters was evaluated for the performance of the separator packages for Watts Bar. Performance of a given moisture separator package is primarily determined by the parameters of steam flow, steam pressure, and water level. The revised parameters indicate that steam flow does not change significantly, and water level is maintained for the 10% SGTP. The only parameter affecting separator performance which shows significant change is steam pressure.

The evaluation considered another operating plant with similar equipment. The 4-loop plant with Model D separator packages most like the Watts Bar Units has the 0.7 primary separator orifices but does not have the other modifications listed above for Watts Bar. The 0.7 orifices in this plant also do not have collars designed to prevent entrainment from the mid deck, as do the orifices installed in the Watts Bar Units. Field data from this plant was used to develop an expected trend of moisture carryover versus separator parameter. This trend showed that the expected change in moisture caused by the steam pressure decrease which results from 10% SGTP and TDF of 93,100 gpm is small. Therefore, the revised NSSS parameters are expected to have minimal effect on WBN moisture carryover.

b. SG U-Bend Fatigue

An evaluation was performed on the potential for vibration of the small radius U-bends due to fluid elastic instability at the revised NSSS parameters.

The evaluation uses a one dimensional multiplier to define the effects of revised conditions in the U-bend region in relation to the reference conditions analyzed in the original evaluation. The method applies operating condition changes, principally a change in steam pressure and steam flow, toward calculation of a change in the U-bend stability ratios, called a relative stability ratio, RSR. The vibration potential and fatigue usage for each susceptible tube is then determined using these RSR values and other factors which are not a function of operating conditions.

A previous evaluation of potential U-bend vibration at current operating conditions had determined which tubes at Watts Bar Unit 1 were at potential risk to develop a North Anna type tube rupture. The earlier analysis was performed using a reference condition which bounds the current operating conditions and assumed that future operation would be at similar levels. It was determined that a single tube, R10C22 in S/G 1, required preventive action at these conditions.

The reason was that the stress ratio for the tube R10C22 (S/G 1) was 1.10, operating at the current operating conditions. This tube was recommended for plugging since the stress ratio was greater than 1.0. The stress ratios and fatigue usage are calculated assuming 40 years of operation. Tubes with stress ratios greater than 1.0 would generally not be acceptable for 40 years (or more) of operation and conversely, tubes with stress ratios less than 1.0 generally would be acceptable for 40 years of operation at the given operating conditions.

With the revised NSSS parameters, the reduction in steam pressure leads to a lower void fraction which reduces the dampening of the steam generator tubes and may increase the vibration. The evaluation for the revised NSSS parameters calculated stress ratios and total fatigue usage for the tubes which have the highest stress ratio in each of the Rows 8, 9, 10 and 11. There are no unsupported tubes in Row 12 and therefore no susceptible tubes in this row. The stress ratios and fatigue usage for the most susceptible tube in each row are seen to be well below 1.0. These tubes and all less susceptible tubes will not require preventive action. Thus, the revised NSSS parameters will cause no additional tubes (other than R10C22 in S/G 1) to become subject to significant U-bend vibration and fatigue.

c. Steam Generator Structural Evaluation

The bases of the original structural evaluation are contained in the Model D3 steam generator stress reports and used duty cycle loading events specified in the plant equipment specifications. The same bases remained applicable for the revised NSSS parameters. The effect of changes warranted by the revised thermal hydraulic conditions applicable for the 10% maximum tube plugging was incorporated. The changes, however, were not considered significant.

The primary side reactor coolant pressure remained unchanged at 2250 psia while the secondary side steam pressure reduced to 958 psia from 1000 psia at the 100% thermal power normal operating conditions. This constituted an increase of about 3% to the primary to secondary pressure differential. The secondary side temperature change from 544.6 to 539.4°F was considered minor.

The critical components considered for the revised NSSS parameters are the components subjected to primary-to-secondary differential pressure. These components are primarily, the tubesheet, the tubesheet-to-shell junctions and the tubes. All these critical components were structurally evaluated, using the original analysis as the basis of comparative assessment.

The results of the evaluation showed that the maximum stress intensities and stress intensity ranges in critical steam generator components would remain within the allowable stress limits and the fatigue usage would not exceed its corresponding limit. The results, therefore, indicated compliance with the required limits of the ASME Boiler and Pressure Vessel Code.

10. Auxiliary Equipment

An evaluation was performed to determine the effect of the revised NSSS parameters on the qualification of the auxiliary heat exchangers, tanks, pumps and valves.

First, the applicable auxiliary equipment design transients for Watts Bar were reviewed. The only transients that are potentially impacted by the revised conditions are those temperature transients that are impacted by full load NSSS operating temperatures, T_{hot} and T_{cold}. The transients used for the specification of the auxiliary system components assumed a full load NSSS T_{hot} and T_{cold} of 630°F and 560°F, respectively. These NSSS temperatures were selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

A comparison of the proposed NSSS operating temperatures for T_{hot} and T_{cold} of 618.7°F and 557.7°F, respectively, with the T_{hot} and T_{cold} values used to develop the design transients indicates that the proposed operating temperatures are less than the values assumed to develop the design transients. Therefore, the actual temperature transients are less severe than the design temperature transients.

G. NSSS Systems

The safety evaluations presented in this section have been performed for 10% SGTP and a TDF of 93,100 gpm and will bound plant operation at 0% tube plugging and a TDF of 93,100 gpm. Thus, the discussion presented in this section includes both the 10% tube plugging and TDF of 93,100 gpm, even though the provisions for 10% tube plugging are not needed to support the proposed Technical Specification changes.

As mentioned in Section I, the 10% SGTP and TDF of 93,100 gpm changed the NSSS Performance Parameters. As a result, it is necessary to demonstrate that the affected NSSS systems can still perform their intended design functions with the new parameters. The impact that these changes have on the affected systems is provided as follows.

1. Reactor Coolant System (RCS)

The revised NSSS parameters impact the fluid system performance of certain RCS related functions, not identified in Table 2. The reduction in T_{cold} from 559.1° to 557.7°F increased the density of the pressurizer spray which ultimately decreases the spray flow. It was determined that the spray flow would be 902 gpm with the lower temperature. This flow is above the design value of 900 gpm. Therefore, the pressurizer spray system will still perform its intended design function.

The reduction in TDF from 97,500 gpm to 93,100 gpm results in a reduction in the pressurizer surge line and pressurizer relief line pressure drop under analyzed conditions since the lower flow results in a lower pressure drop. Thus, the design of the pressurizer relief system for the revised conditions remains bounding.

2. Chemical and Volume Control System (CVCS)

The regenerative and letdown heat exchangers are designed to cool letdown flow from T_{cold} to 127°F. The variation in T_{cold} from 559.1° to 557.7°F is 1.4°F, which will have no significant effect on the heat exchanger structurally. The design temperature of the shell is 650°F, which is well above the T_{cold} of 557.7°F. The cooling requirements of the letdown heat exchanger under the revised conditions are bounded by the design requirements.

3. Emergency Core Cooling System (ECCS)

The ECCS flows are unaffected since there are no changes to the RCS operating pressure. The changes in T_{HOT}, T_{COLD} and TDF do not affect the ECCS flows.

4. Residual Heat Removal System (RHRS)

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 is normally placed in operation approximately four hours after reactor shutdown when the pressure and temperature of the RCS are approximately 400 psig and 350°F, respectively. Under normal operating conditions, the RHRS is designed to reduce the temperature of the reactor coolant to 140°F within 20 hours following reactor shutdown, with both trains operating. In the event of a train failure, the RHRS is designed to reduce the reactor coolant temperature to 200°F within 36 hours after reactor shutdown.

Since the initiation temperature and decay heat generation rates have not changed, the demands on the RHRS are not affected. Therefore, the RHRS is still capable of reducing the reactor coolant temperature to 140°F within the 20 hour limit for normal operating conditions, when both trains are operating. In the event of a train failure, the RHRS is still capable of reducing the reactor coolant temperature to 200°F within the 36 hour limit.

5. Spent Fuel Pool Cooling System (SFPCS)

The primary function of the SFPCS is to remove decay heat which is generated by the spent fuel pool elements stored in the pool. Decay heat generation is proportional to plant power level. Since the plant power level remains unchanged, the demands on the SFPCS are not increased. The purification function is controlled by SFPCS demineralization and filtration rates, which are not affected by the revised NSSS parameters.

H. NSSS/BOP Interface Systems Evaluations

The safety evaluations presented in this section have been performed for 10% SGTP and a TDF of 93,100 gpm and will bound plant operation at 0% tube plugging and a TDF of 93,100 gpm. Thus, the discussion presented in this section includes both the 10% tube plugging and TDF of 93,100 gpm, even though the provisions for 10% tube plugging are not needed to support the proposed Technical Specification changes.

1. Main Steam System

The impact that the revised NSSS parameters has on the design basis of several main steam system components is provided as follows.

a. Steam Generator Safety Valves

The setpoints of the steam generator safety valves are determined based on the design pressure of the steam generators (1185 psig) and the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the SGs has not changed, there is no need to revise the setpoints of the safety valves.

The steam generator safety valves must have sufficient capacity to ensure that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV code) for the worst-case loss-of-heat-sink event. Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 105 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the Main Steam System design pressure. Additionally, the capacity of any single safety valve is limited to 970,000 lb/hr at 1200 psia based on the present steam break analysis of record for a stuck-open steam generator safety valve.

Watts Bar has twenty safety valves with a total capacity of 16.65 x 10^6 lb/hr, which provides about 110.4 percent of the maximum steam flow of 15.08 x 10^6 lb/hr at the revised conditions. Therefore, the capacity of the installed MSSVs meets the sizing criterion.

b. Steam Generator Atmospheric Dump Valves

The steam generator atmospheric dump valves (ADVs) are automatically controlled by steam line pressure during plant operations. The steam generator ADVs automatically modulate open and exhaust to atmosphere whenever the steam line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steam line pressure decreases, the steam generator ADVs modulate closed and reseat at a pressure below the opening pressure. The steam generator ADV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither of these pressures changes for the proposed range of NSSS operating parameters, there is no need to change the ADV setpoint.

Also, the four steam generator ADVs are sized to have a capacity equal to about 20 percent of the steam flow used for plant design at no-load steam pressure. This capacity permits a plant cooldown to RHRS operating conditions in about 4 hours with two ADVs in service (at an assumed cooldown rate of 50°F/hr) assuming 2 hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFW system.

Based on the revised NSSS parameters, the installed ADV capacity (3.17 x 10^6 lb/hr at 1106 psia) is more than 20 percent (3.016 x 10^6 lbs/hr) of the maximum full-load steam flow (15.08 x 10^6 lbs/hr). Therefore, the ADVs are adequate for the revised conditions.

c. Main Steam Isolation Valves

The MSIVs are located outside the containment and downstream of the steam generator safety and relief valves. 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 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 significant 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 pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables are not impacted by the revised NSSS parameters, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change.

2. Steam Dump System

The NSSS Reactor Control Systems and the associated equipment are designed to provide satisfactory operation (automatic in the range of 15 to 100 percent power) without reactor trip when subjected to the following load transients:

- Loading at 5 percent of full power per minute with automatic reactor control.

- Unloading at 5 percent of full power per minute with automatic reactor control.

- Instantaneous load transients of plus or minus 10 percent of full power (not exceeding full power) with automatic reactor control.

- Load reductions of 50 percent of full power with automatic reactor control and steam dump.

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. (Various NSSS control systems are used to respond to the additional 10 percent load rejection.) A steam dump capacity of 40 percent of rated steam flow at full load steam pressure also prevents the steam generator safety valve lifting following a reactor trip from full power.

Watts Bar has twelve condenser steam dump valves and each valve is specified to have a flow capacity of 5.32×10^5 lbs/hr at a valve inlet pressure of 900 psia. This total capacity provides a steam dump capability of about 45.5 percent of the original maximum guaranteed steam flow (15.14×10^6 lb/hr), or 6.89×10^6 lbs/hr at a full load steam generator pressure of 1000 psia versus the sizing criterion of 40 percent of rated steam flow.

At the revised conditions, the steam dump capacity could be as low as 43.7 percent of rated steam flow due to the reduction in steam pressure. This 43.7 percent value is still above the 40 percent requirement.

To provide effective control of flow on large step load reductions or plant trip, the steam dump valves are required to go from full-closed to full-open in 7 seconds at any pressure between 50 psi less than full load pressure and steam generator design pressure. The dump valves are also required to modulate to control flow. These requirements are still applicable for the revised NSSS parameters.

3. Condensate and Feedwater System

The impact that the revised NSSS parameters has on the design basis of several condensate and feedwater system components is provided as follows.

a. Main Feedwater Isolation Valves

The main feedwater isolation valves (MFIVs) are located outside containment in the main steam valve vaults and downstream of the main feedwater regulator valves (MFRVs) located in the Turbine Building. The valves function in conjunction with the MFRVs 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 cooldowns. To accomplish this function, the MFIVs and the backup MFRVs must be capable of closure within 6.5 seconds after receipt of a closure signal under all operating and accident conditions, including a maximum flow condition with all main feedwater pumps delivering to one steam generator.

The quick-closure requirements imposed on the MFIVs and the backup MFRVs cause 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 are not impacted by the proposed change in NSSS operating parameters, the design loads and associated stresses resulting from rapid closure of these valves will not change.

b. <u>Main Feedwater Regulator Valves, Condensate and Feedwater System (C&FS) Pumps</u>

The C&FS available head in conjunction with the feedwater regulator valve characteristic 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 MFRVs valves at rated flow (100 percent power) should be approximately equal to the dynamic losses from the feed pump discharge through the steam generator (i.e., equal to the frictional resistance of feed piping, MFIV, high pressure feed water heaters, feed flow meter, and steam generator. Evaluation of the present MFRV design (considering full open Cv vs lift and linear trim) and pump speed control program at full load conditions to permit condensate and feedwater system delivery at the appropriate pressure increase above full load pressure indicates that adequate margin is available to address the new NSSS operating parameters. Therefore, with the current design MFRV and system layout the current pump speed control program can be set to accommodate the revised NSSS operating parameters.

For the revised NSSS operating parameters, the present speed control program results in a negligible change in MFRV pressure drop (about 1 psi) and a corresponding negligible change in valve lift (less than 1 percent) at 100 percent power. Therefore, operation of the MFRVs (in conjunction with the present feedwater pump speed control program) is judged to be acceptable for both steady state and transient operation.

To provide effective control of flow during normal operation, the MFRVs 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 MFRVs 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 for the revised NSSS operating parameters.

I. Control Systems Evaluation

Condition I transients are evaluated to determine that the plant can appropriately respond to the transient without a trip from the OTΔT and OPΔT protection functions. The proposed setpoint changes for OTΔT and OPΔT are a benefit since they result in an increase in the margin to OTΔT/OPΔT trip. Here, margin is defined as the difference between the setpoint and the compensated measured ΔT expressed in percent of full power ΔT. Since the margin to trip is increased, the Condition I transient acceptance criteria continue to be met.

J. RTDP Uncertainty Report and Revised Protection System Setpoint Study

1. Revised Thermal Design Procedure (RTDP)

The safety analyses discussed in this safety evaluation utilize the NRC approved RTDP methodology (WCAP-11397-P-A). The RTDP methodology is primarily used as a means to obtain additional DNB margin. This DNB margin is needed to account for the OTΔT and OPΔT setpoint changes, the reduced TDF and Cycle 2 design features. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB design limits. The statistical treatment of uncertainties is used since these parameters are independent of each other. The treatment provides margin by reducing the need to consider the parameter uncertainties in the conservative and limiting direction.

Four operating parameters uncertainties are used in the RTDP uncertainty analysis. These parameters are pressurizer pressure, primary coolant temperature (T_{avg}), reactor power and reactor coolant system flow. The uncertainty analysis for these parameters is documented in WCAP-14738, Revision 0. These uncertainties have been incorporated, as applicable, into the safety analyses discussed in this safety evaluation.

The revised uncertainties resulted in changes to the RCS average temperature and pressure values reported in the Technical Specifications. These Technical Specification changes are identified in Enclosure 3.

2. Revised Protection System Setpoint Study

The current setpoint study for protection systems is contained in WCAP-12096, Revision 6. This WCAP has been revised to account for the changes to the OTΔT and OPΔT setpoints, reduced thermal design flow of 93,100 gpm and tolerances for the measurement of ΔT_o, T′ and T″. The revisions are documented in WCAP-12096, Revision 7.

A review of WCAP-12096, Revision 7 indicates that the OTΔT, OPΔT, reactor coolant flow and vessel ΔT equivalent to power Allowable Values have been revised to reflect the changes. As a result, it was determined that the Technical Specifications for the OTΔT and OPΔT setpoints and for the vessel ΔT equivalent to power Allowable Value had to be modified. This Allowable Value is used to assess operability in the Technical Specifications and is not specifically modeled in the safety analyses. These Technical Specification changes are identified in Enclosure 3.

TABLE 1

SELECTED CHANGES TO OTΔT/OPΔT SETPOINTS
(As Reported in the Technical Specifications)

OTΔT			OPΔT		
Constant	New Value	Old Value	Constant	New Value	Old Value
Allowable Value	1.2%	(2%)	Allowable Value	1.0%	(1.8%)
K1 $\leq$	1.16	(1.0952)	K4 $\leq$	1.10	(1.091)
K2 $\geq$	.0183/°F	(.0133)	K6 $\geq$	.00162	(.00126)
K_3 =	.000900/psig	(.000647)	$\tau_4 \geq$	3 sec	(12)
$\tau_4 \geq$	3 sec	(12)			

Note: Please see Technical Specification marked up pages in Enclosure 3 for a complete representation of the changes to these setpoints.

TABLE 2

COMPARISON OF EXISTING AND MODIFIED NSSS PERFORMANCE PARAMETERS

Parameter	Current	10% SGTP/2% RTDF
NSSS Power, MWt	3425	3425
Reactor Power, MWt	3411	3411
RCS Thermal Design Flowrate, gpm/loop	97,500	93,100
RCS Pressure, psia	2250	2250
RCS Temperatures, °F		
Core Outlet	621.6	623.9
Vessel Outlet	617.3	618.7
Core Average	592.0	592.8
Vessel Average	588.2	588.2
Vessel/Core Inlet	559.1	557.7
Steam Generator Outlet	558.8	557.5
Steam Generator		
Steam Temperature, °F	544.6	539.4
Steam Pressure, psia	1000	958
Steam Flow, 10^6 lb/hr total	15.14	15.08
Feed Temperature, °F	440	440
Tube Plugging %	0	10

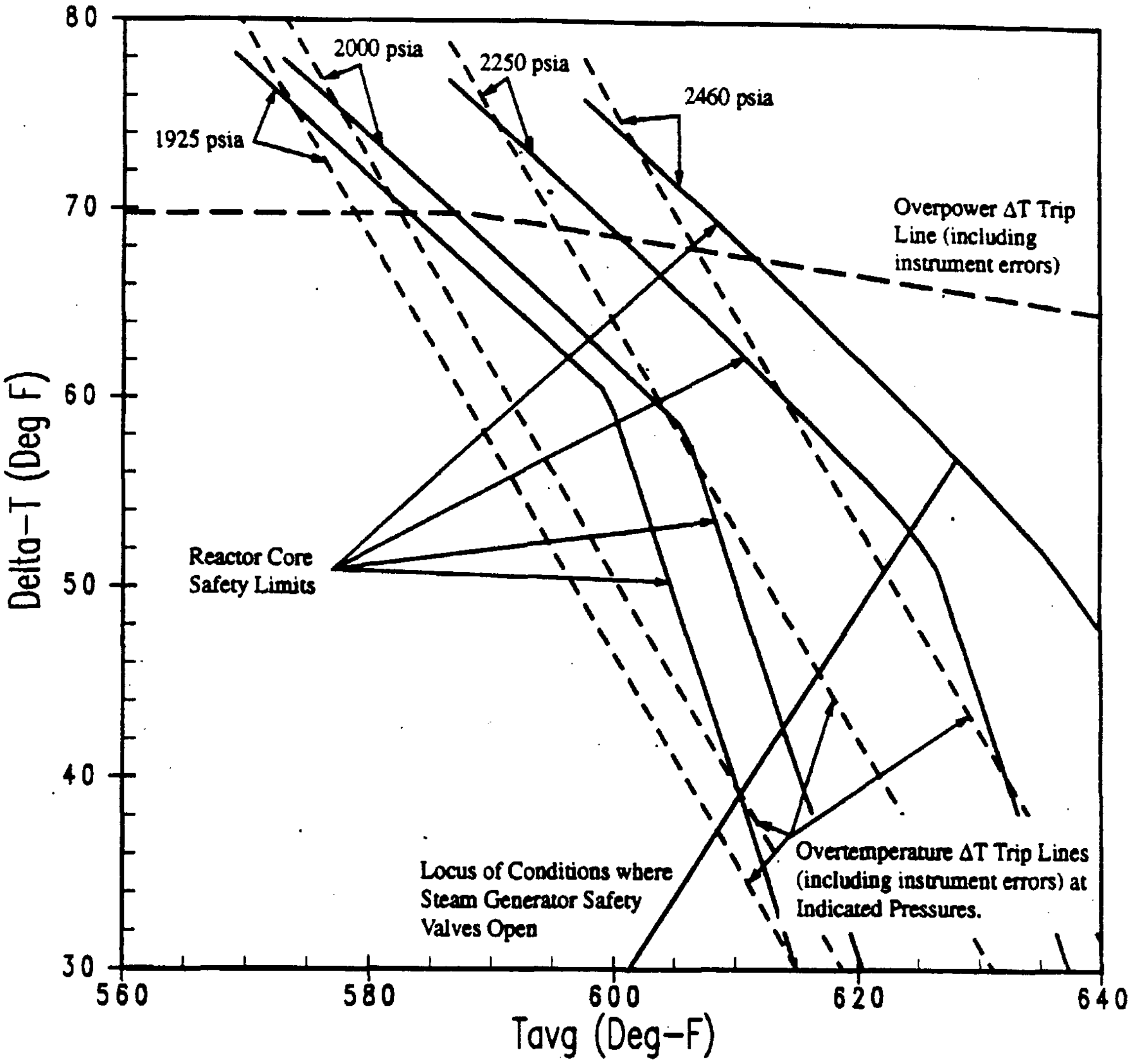

Figure 1: Illustration of OTΔT and OPΔT Protection

III. Environmental Consideration

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

ENCLOSURE 2

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

PART 1 - EMERGENCY CORE COOLING SYSTEMS

I. Description of Proposed License Amendment

The proposed amendment would revise the technical specifications (TSs) to address Cycle 2 core design changes. The Cycle 2 core design for Watts Bar will include a longer fuel cycle and more highly enriched fuel (from 3.1 percent to 3.7 percent). To accommodate this design, the RWST and accumulator boron concentrations will be increased to provide enough boron in the sump to meet the LBLOCA requirement for sump boron concentration. This requirement is that during a LBLOCA, the core will remain subcritical from boron provided by the ECCS, which takes suction from the RWST and containment sump.

The increase in RWST and accumulator boron concentrations will require changes to the plant Technical Specifications.

Accumulator Boron Concentration (Technical Specification Section 3.5.1)

The boron concentration range has been changed from 1900 to 2100 ppm to a range of 2400 to 2700 ppm to reflect the boron concentration increase.

RWST Boron Concentration (Technical Specification Section 3.5.4)

The boron concentration range has been changed from 2000 to 2100 ppm to a range of 2500 to 2700 ppm to reflect the boron concentration increase.

BASES - Applicable Safety Analyses (Technical Specification Section B 3.5.4)

The minimum RWST boron concentration has been changed from 2000 ppm to 2500 ppm to reflect the minimum value used in the post-LOCA sump analysis for core subcriticality. Also, the maximum RWST boron concentration has been changed from 2100 ppm to 2700 ppm to reflect the maximum value used in the hot leg switchover time calculation.

II. Basis for No Significant Hazards Consideration Determination

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated;

The RWST and accumulator boron concentrations do not affect any initiating event for accidents currently evaluated in the FSAR. The increased concentrations will not adversely affect the performance of any system or component which is placed in contact with the RWST or accumulator water. The integrity and operability of the stainless steel surfaces in the RWST, accumulator and affected NSSS components/systems will be maintained. The decrease in solution pH is small and will not degrade the stainless steel. Also, the integrity of the Class 1E instrumentation and control equipment will be maintained since the lower sump pH, resulting from the increased boron concentrations, is still within the applicable equipment qualification limits. These limits are set to preclude the possibility of chloride induced stress corrosion cracking and assure that there is no significant degradation of polymer materials. The design, material and construction standards of all components which are placed in contact with the RWST and accumulator water remain unaffected.

For the evaluations, the consequences of an accident previously evaluated in the FSAR will not be increased. There is no increase in the LOCA accident consequences. The changes in the concentrations increase the amount of boron in the sump during a LOCA. The increased boron in the sump is sufficient to maintain the core in a subcritical condition during a LOCA. Also, a revised hot leg switchover time has been calculated and will be implemented in the plant EOPs. Thus, there will be no boron precipitation in the core during a LOCA.

Furthermore, there is no increase in consequences of the non-LOCA events. The concentration changes are a benefit to the SLB at full power analysis due to the reduction in power during the accident. The loss of normal feedwater event is not sensitive to changes in the RWST and accumulator boron concentrations. The concentration changes do not affect the inadvertent operation of ECCS analysis since the minimum DNBR occurs at the event initiation, and the concentration changes do not affect the analysis trend.

Finally, the concentration changes are a benefit for the SLB M&E release and SGTR events since the increased boron increases the available shutdown margin for these events. In addition, the increase in RWST and accumulator boron concentrations and subsequent slight decrease in containment sump and a spray pH does not impact the LOCA dose evaluation since pH is not a function of radionuclide concentration. Therefore, the present analysis remains bounding. Also, the slight decrease in sump, core and spray fluid pH has been evaluated to not impact the corrosion rate (and subsequent generation of Hydrogen) of Aluminum and Zinc inside containment significantly that the present analysis does not remain bounding. Further, the decreased sump, core and spray fluid pH has been evaluated to not affect the amount of hydrogen generated from the radiolytic decomposition of the sump and core solution. In view of the preceding, it is concluded that the proposed change will not increase the consequences of an accident previously evaluated in the FSAR.

(2) or create the possibility of a new or different kind of accident from any accident previously evaluated;

The changes to the RWST and accumulator concentrations do not cause the initiation of any accident nor create any new credible limiting single failure. The changes do not result in a condition where the design, material, and construction standards of the RWST and accumulators and other potentially affected NSSS components, that were applicable prior to the changes, are altered. The integrity and operability of the stainless steel surfaces in the RWST, accumulator, and affected NSSS components/systems will be maintained. The decrease in solution pH is small and will not degrade the stainless steel. Also, the integrity of the Class 1E instrumentation and control equipment will be maintained during a LOCA since the lower sump pH, resulting from the increased boron concentrations, is still within the applicable equipment qualification limits. These limits are set to preclude the possibility of chloride induced stress corrosion cracking and assure that there is no significant degradation of polymer materials.

The changes do not invalidate any of the accident analyses results or conclusions. All of the safety analysis acceptance criteria continue to be met. The changes in the concentrations increase the amount of boron in the sump during a LOCA. The increased boron in the sump is sufficient to maintain the core in a subcritical condition during a LOCA. Also, a revised hot leg switchover time has been calculated and will be implemented in the plant EOPs. Thus, there will be no boron precipitation in the core during a LOCA.

Furthermore, there is no possibility of a different kind of non-LOCA event. The concentration changes are a benefit to the SLB at full power analysis due to the reduction in power increase during the accident. The loss of normal feedwater event is not sensitive to changes in the RWST and accumulator boron concentrations. The concentration changes do not affect the inadvertent operation at ECCS analysis since the minimum DNBR occurs at the event initiation, and the concentration changes do not affect the analysis trend.

Finally, the concentration changes are a benefit for the SLB M&E release and SGTR events since the increased boron increases the available shutdown margin for these events.

(3) or involve a significant reduction in a margin of safety.

The changes do not invalidate any of the non-LOCA safety analysis results or conclusions, and all of the non-LOCA safety analysis acceptance criteria continue to be met. The margin of safety associated with the licensing basis LBLOCA and SBLOCA analyses is not reduced as a result of the proposed changes. Since adequate margin to the PCT limit of 2200°F has been maintained, no degradation in the margin of safety to the design failure point (fuel melt) has been calculated. The licensing basis containment and steam line break mass and energy releases remain bounding, and the SGTR event acceptance criteria continue to be met. Furthermore, the changes do not affect the safety related performance of the RWST, accumulator or related NSSS components.

III. Summary

Based on the above, TVA has determined that operation of Watts Bar in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Therefore, operation of Watts Bar in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

PART 2 - SAFETY LIMITS, INSTRUMENTATION, AND REACTOR COOLANT SYSTEM

I. Description of Proposed License Amendment

Watts Bar has experienced hot leg temperature fluctuations, including random spikes, which decrease the operating margin to both the OTΔT and OPΔT reactor trip setpoints. These fluctuations have caused, in some cases, the plant to experience OT alarms during steady-state operation since the temperature fluctuations reduced the operating margin. To mitigate the temperature fluctuations and associated alarms, the OTΔT and OPΔT setpoints have been enhanced to increase the operating margin associated with these trip functions. The increased margin decreases the OTΔT and OPΔT sensitivity to the temperature fluctuations. The revised setpoints are identified in Table 1.

The OTΔT and OPΔT trip functions are modeled in various safety analyses including the non-LOCA (transient) analyses, steamline break mass and energy releases, and steam generator tube rupture event. Also, the NSSS is designed to respond to Condition I design transients without incurring an OTΔT or OPΔT trip. These safety analyses and Condition I transients are addressed in this safety evaluation.

In addition, Watts Bar has decided to reduce the plant thermal design flow from 97,500 gpm per loop to 93,100 gpm per loop (total of 390,000 gpm) to accommodate 10 percent steam generator tube plugging and a 2 percent reduction in thermal design flow (RTDF). For 10 percent steam generator tube plugging, the thermal design flow is reduced from 97,500 to 95,000 gpm. For an additional 2 percent RTDF, the thermal design flow is reduced from 95,000 to 93,100 gpm. The total RTDF from 97,500 gpm to 93,100 gpm is 4.51 percent. The NSSS performance parameters have been modified to reflect plant operation at the 10 percent steam generator tube plugging and an additional 2 percent RTDF. A comparison of the modified parameters to the existing parameters is provided in Table 2.

The reduction in TDF to 93,100 gpm affects the Technical Specifications and is addressed in this safety evaluation. The level of steam generator tube plugging is not reported in the Technical Specifications. Therefore, tube plugging is not directly addressed in this safety evaluation. However, some of the safety analyses discussed in this safety evaluation have already been performed for the 10 percent tube plugging and TDF of 93,100 gpm and will bound plant operation at 0 percent tube plugging and a TDF of 93,100 gpm. These safety analyses include the steamline break mass and energy releases, steam generator tube rupture, containment mass and energy releases, NSSS components and systems, NSSS/BOP Interface Systems, Control Systems Evaluation, and the Revised Thermal Design Procedure (RTDP) and Setpoint Study. The discussion of these analyses includes provisions for the TDF of 93,100 gpm and the 10 percent tube plugging, even though the provisions for the tube plugging are not needed to support the Technical Specification changes.

Also, Watts Bar has decided to implement a tolerance of .6°F for the Technical Specification Surveillance for ΔT_o and 1°F tolerance for the surveillance of T_{AVG} (identified as T′ and T″ in the Technical Specifications). The use of this tolerance will help to determine whether the indicated ΔT and T_{AVG} should be left as is, or rescaled during the surveillance. These tolerances have been incorporated as biases into the uncertainty analysis for the affected protection system functions. These functions include the OTΔT, OPΔT and vessel ΔT equivalent to power (used in the SG low-low water level trip functions). As a result of implementing these biases into the protection system functions (and the changes to the OTΔT/OPΔT setpoints and reduced TDF), the Allowable Value in the Technical Specifications for the OTΔT, OPΔT and vessel ΔT equivalent to power functions have been modified.

The tolerances are not directly modeled as an input to the safety analysis. They are indirectly modeled in the safety analysis because they can potentially affect the trip setpoints for the OTΔT/OPΔT and vessel ΔT equivalent to power functions. However, it has been determined that the use of tolerances does not require changes to the trip setpoints, and thus the tolerances are not specifically discussed in the safety analyses in this safety evaluation. See WCAP 14738 for more information regarding the incorporation of the tolerances to the setpoint uncertainty analysis.

This safety evaluation has been prepared to allow for plant operation during Cycle 2 with the revised OTΔT and OPΔT setpoints, the thermal design flow of 93,100 gpm and the tolerances for ΔT_o, T′ and T″. To obtain sufficient DNB margin for the OTΔT/OPΔT setpoint, reduced TDF and Cycle 2 design features, it was necessary to implement the RTDP. The RTDP program changes the uncertainty treatment for core power, T_{AVG}, pressurizer pressure, and RCS flow. These uncertainties have been incorporated, where applicable, into the safety analyses addressed in this Safety Evaluation. See WCAP 14738 for a discussion of the RTDP program.

The following Technical Specifications will be changed to incorporate the OTΔT/OPΔT margin enhancement, thermal design flow of 93,100 gpm and tolerances for ΔT_o, T′ and T″. The applicable Technical Specification mark-ups are provided in Enclosure 3.

Reactor Core Safety Limits (Figure 2.1.1-1)

The use of the RTDP to improve DNB margin leads to a modification of the Reactor Core Safety Limits.

Reactor Trip System Instrumentation (Table 3.3.1-1, page 4) and ESFAS Instrumentation (Table 3.3.2-1, page 4)

The Allowable Values for the Vessel ΔT Equivalent to Power input to Steam Generator Water Level Low-Low have been changed to reflect the addition of a 0.6°F tolerance to the measurement of ΔT_o. This tolerance has been included as a bias in the uncertainty analysis to facilitate the quarterly measurement of ΔT_o.

Reactor Trip System Instrumentation (Table 3.3.1-1, pages 7 and 8)

The revised reactor core safety limits lines allow for changes in the OTΔT/OPΔT reactor trip setpoints to improve operating margin. The allowable values for these functions have changed as a result of including tolerances for ΔT_o, T′ and T″ in the uncertainty analysis. Several setpoint gains and time constants have been modified to enhance plant operation.

RCS Pressure, Temperature and Flow DNB Limits (Section 3.4.1)

The RCS average temperature limit has been revised to account for the change in uncertainty from implementing RTDP. The total RCS flow has been modified to account for the reduced thermal design flow from 97,500 gpm to 93,100 gpm. The total flow value in the Technical Specification includes an allowance for instrument uncertainty.

Bases - Reactor Core Safety Limits (Section B 2.1.1)

The "Safety Limits" section has been modified to reflect the use of the RTDP methodology.

Bases - Nuclear Enthalpy Rise Hot Channel Factor (Section B 3.2.2)

The "Applicable Safety Analyses" section has been modified to reflect the use of the RTDP methodology.

Bases - Reactor Trip System Functions OTΔT, OPΔT and Steam Generator Water Level Low-Low (Vessel ΔT Equivalent to Power) (Section B 3.3.1)

The "Applicable Safety Analyses, LCO, and Applicability" sections for the OTΔT, OPΔT, and Steam Generator Water Level Low-Low (Vessel ΔT) trip functions have been expanded to address the quarterly examination of the ΔT_0, T′, and T″. The tolerances are used to determine if these parameters should be reset during the examination.

Bases - Reactor Trip System Functions - Reactor Coolant Flow - Low (Single Loop and Two Loops) (Section B 3.3.1)

The "Applicable Safety Analyses, LCO and Applicability" sections, have been modified to indicate that the Reactor Coolant Flow - Low Trip Setpoint and Allowable Value are specified in percent of thermal design flow adjusted for uncertainties and not nominal flow.

Bases - Engineered Safety Feature Actuation System (ESFAS) Instrumentation (Section B 3.3.2)

In the "Applicable Safety Analyses, LCO and Applicablity" section for the Steam Generator Water Level Low-Low (Vessel ΔT), a reference to Bases section B 3.3.1 has been added for a discussion of the required MODES and normalization of the vessel ΔT input to the Trip Time Delay function.

Bases - RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) (Section B 3.4.1)

In the "Applicable Safety Analyses" section, the RCS average temperature limit, RCS average temperature analytical limit and pressurizer pressure analytical limit have been revised to account for the change in uncertainty from implementing RTDP. In the "LCO" section, the flow measurement uncertainty allowance using control board indication has also been revised as a result of implementing RTDP.

II. Basis for No Significant Hazards Consideration Determination

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92 (c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated;

The proposed changes do not result in a condition where the design, material, and construction standards, which were applicable prior to the changes, are altered. The revised OTΔT and OPΔT setpoints do not require any hardware changes and are used for accident mitigation. Thus, the setpoint changes do not increase the probability of the accident.

All of the affected NSSS systems and components have been evaluated with the TDF of 93,100 gpm. The primary loop components (reactor vessel, reactor internals, CRDMs, loop piping and supports, reactor coolant pump, steam generator, and pressurizer) meet the applicable structural limits with the revised TDF of 93,100 gpm and will continue to perform their design functions. The RCCA drop time remains unaffected and the current design core bypass flow remains valid. No additional steam generator tubes need to be plugged to mitigate the potential for U-Bend fatigue. Also, all of the NSSS systems will still perform their intended design functions. The pressurizer spray flow remains above the design value and the pressurizer relief system remains unaffected since the TDF is lower than the current design flow and the required pressure drop is lower. The design of the auxiliary system components remains bounding for the revised TDF and the corresponding changes to the NSSS thermal hydraulic parameters. In addition, all of the NSSS/BOP interface systems will perform their intended design functions. The steam generator safety valves will provide adequate relief capacity to maintain the steam generator within applicable design limits. The ADVs will still relieve 20 percent of the maximum full load steam flow. The steam dump system will still relieve 40 percent of the maximum full load steam flow.

All of the applicable acceptance criteria for the accidents described in the FSAR continue to be met. The LBLOCA analysis currently uses a TDF of 93,100 gpm. Thus, no adjustments are required for the LBLOCA input parameters to accommodate the TDF of 93,100 gpm. The SBLOCA has been performed with the TDF of 93,100 gpm, and the corresponding PCT is well below the 2200°F limit. The post LOCA boron concentration and the hot leg switchover time are unaffected. The revised thermal design procedure has been implemented to obtain sufficient DNB margin to account for the TDF of 93,100 gpm, the new OTΔT/OPΔT setpoints and the Cycle 2 design features. All of the non-LOCA analyses have been re-analyzed or re-evaluated and all of the applicable acceptance criteria continue to be met.

The SLB radiological doses are unaffected and are still within the existing licensing basis limits. The margin to overfill during the SGTR event has been improved and the offsite doses during an SGTR have been re-calculated and shown to be well within the 10CFR100 guidelines. The plant control systems will still provide adequate response for the Condition 1 transients without causing a reactor trip on OTΔT and OPΔT.

Finally, the changes in the tolerances for ΔT_o, T' and T'' do not require any hardware modifications and only require changes to the Technical Specification Allowable Values for the OPΔT and OTΔT setpoints and for the vessel ΔT equivalent to power functions. Thus, there is no increase in the probability of an accident since the appropriate Allowable Values have been modified to determine channel operability for these functions.

(2) or create the possibility of a new or different kind of accident from any accident previously evaluated;

The proposed changes do not cause the initiation of any accident nor create any new limiting single failures. The OTΔT and OPΔT protection functions are used for accident mitigation and do not initiate any accidents. Also, the affected systems and components will still perform their intended design functions. All of the primary loop components (reactor vessel, reactor internals, CRDMs, loop piping and supports, reactor coolant pump, steam generator, and pressurizer) meet the applicable structural limits with the revised TDF of 93,100 gpm and will continue to perform their design functions. The RCCA drop time remains unaffected and the current design core bypass flow remains valid. No additional steam generator tubes need to be plugged to mitigate the potential for U-Bend fatigue. All of the NSSS systems will still perform their intended design functions. The pressurizer spray flow remains above the design value and the pressurizer relief system remains unaffected since the TDF is lower than the current design and the required pressure drop is lower. The design of the auxiliary system components remain bounding for the revised conditions. All of the NSSS/BOP interface systems will perform their intended design functions. The steam generator safety valves will provide adequate relief capacity to maintain the steam generator within applicable design limits. The ADVs will still relieve 20 percent of the maximum full load steam flow. The steam dump system will still relieve 40 percent of the maximum full load steam flow.

The proposed changes do not create any new failure modes for safety related equipment. The changes do not result in any original design specification, such as seismic requirements electrical separation requirements or equipment qualification being altered. The OTΔT and OPΔT setpoint changes do not require any hardware modifications and only require adjustments to the setpoint values. The setpoints are modeled in accident analyses which are used to demonstrate equipment and structural qualification during a SLB. With the setpoint changes and the TDF of 93,100 gpm, the current SLB break M&E releases inside containment remain bounding and thus there is no effect on the qualification of the equipment inside containment during a SLB. The SLB M&E releases outside containment have been re-calculated. The analysis of the impacts on equipment qualification outside containment has been completed by generating new temperature profiles. The application addresses and provides for continued qualification of equipment through the normal EQ program.

Also, with the reduced TDF of 93,100 gpm, the current LOCA M&E releases are still bounding, and thus there is no effect on the qualification of equipment inside containment during a LOCA. The OTΔT and OPΔT functions are not modeled in the LOCA analyses. Furthermore, all of the applicable compartments and subcompartments will maintain their integrity during the LOCA and the SLB since the mass and energy releases for these compartments and subcompartments remain unaffected.

In addition, the LOCA hydraulic forcing functions remain bounding for the TDF of 93,100 gpm. Thus, the applicable NSSS systems and components will still perform their structural functions during a LOCA.

Finally, the changes in the tolerances for ΔT_0, T′ and T″ do not require any hardware modifications and only require changes to the Technical Specification Allowable Values for the OPΔT and OTΔT setpoints and for the vessel ΔT equivalent to power functions. Thus, there is no increase in the probability of an accident different than any previously evaluated since the appropriate Allowable Values have been modified to determine channel operability for these functions.

(3) or involve a significant reduction in a margin of safety.

The margin of safety for the applicable safety analyses has not been reduced. The OPΔT and OTΔT setpoints have been incorporated into the affected safety analyses and all safety analysis criteria continue to be met. All of the applicable DNB limits continue to be met for the non-LOCA analyses. The LBLOCA input parameters do not require adjustment for the TDF of 93,100 gpm. The SBLOCA has been re-analyzed for the TDF of 93,100 gpm, and the SBLOCA PCT is well below the 2200°F limit. The affected NSSS systems and components will still meet the applicable design limits and perform their intended safety functions with the TDF of 93,100 gpm. Also, the SLB and LOCA M&E releases are still within the applicable equipment qualification limits. The SGTR doses remain within the applicable 10 CFR 100 limits, and the steam generator margin to overfill is maintained.

III. Summary

Based on the above, TVA has determined that operation of Watts Bar in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Therefore, operation of Watts Bar in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92.