Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

November 15, 2001

TVA-SQN-TS-01-08 10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 01-08, "INCREASE MAXIMUM ALLOWED REACTOR POWER LEVEL TO 3455 MEGA-WATT THERMAL (MWt)"

In accordance with the provisions of 50.90, TVA is submitting a request for an amendment to SQN's Licenses DPR-77 and 79 to change the TSs for Units 1 and 2. The proposed license amendment would increase the full core thermal power rating by 1.3 percent from 3411 MWt to 3455 MWt, based on planned installation of the improved Caldon, Incorporated (Caldon) Leading Edge Flow Meter, LEFM✓™ (LEFM) feedwater flow measurement instrumentation. This change affects Operating License Condition 2.C.(1) and Definition 1.26 for Rated Thermal Power. In addition, changes are necessary to the reactor power limits of TS Table 3.7.1 with an inoperable main steam safety valve for both units and, for Unit 2 only, the interval for which the pressure and temperature curves and the low temperature over pressure protection curves (TS Figures 3.4-2, 3.4-3, and 3.4-4) are valid. A change to the Bases for TS Section 3/4.7.1.1 is also included to address the changes in main steam safety valve capabilities.

Rec'd at NRC/DCD 12/12/01
AP01

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of SQN Units 1 and 2, in accordance with the proposed change, will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains copies of the appropriate TS pages from Units 1 and 2 marked up to show the proposed change. Enclosure 3 forwards the revised TS pages for Units 1 and 2 which incorporate the proposed change.

As addressed in Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the (LEFM✓™) System," the LEFM will enable determination of core power level with improved measurement uncertainties, thereby allowing a power uprate. The Staff's review and approval of ER-80P is documented in NRC's Safety Evaluation for Texas Utilities' (TU) Comanche Peak Unit 2, dated March 8, 1999. TVA provided supplemental information specific to the Watts Bar Nuclear Plant (WBN) power uprate effort in Caldon Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System," and Caldon Engineering Report-160 (non-proprietary), "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System." NRC approved the WBN operating license and TS changes associated with the proposed uprate and accepted these engineering reports in NRC's issued amendment for WBN dated January 19, 2001. These reports have been verified by Caldon to be applicable and bounding to the SQN uprate effort and consistent with the application at WBN. In addition, SQN's nuclear steam supply system (NSSS) vendor, Westinghouse Electric Company, has performed a power calorimetric measurement uncertainty calculation for use of the LEFM. Proprietary and non-proprietary summaries of this information are provided in Enclosures 4 and 5, respectively.

Westinghouse and Framatome Advance Nuclear Power have performed specific evaluations and analyses for the proposed power uprate. This information has been compiled in Westinghouse Commercial Atomic Power WCAP-15726, Revision 0, "Sequoyah Units 1 and 2 1.3 Percent Power Uprate Program Licensing Report." This report is included in Enclosure 6.

TVA's amendment request is consistent with the power uprate license amendments granted to TU for Comanche Peak Unit 2, dated September 30, 1999, and TVA's WBN Unit 1, dated January 19, 2001. In order to facilitate NRC's review of the enclosed SQN application, TVA has addressed NRC Staff questions raised in the licensing process for the Comanche Peak and WBN power uprate efforts. Proprietary and non-proprietary versions of this information are provided in Enclosures 7 and 8, respectively, and incorporated into the license amendment request where practical. Enclosure 7 includes only those responses that contain proprietary information and Enclosure 8 contains all responses. TVA's implementation of the proposed license amendment is based upon the revised requirements of 10 CFR 50, Appendix K, "Emergency Core Cooling System Evaluation Models, ECCS," as approved by the NRC Commission and issued on June 1, 2000 (65 FR 34913), with an effective date of July 31, 2000.

Enclosures 4 and 7 contain information proprietary to Westinghouse. Accordingly, Enclosure 9 includes Westinghouse Applications for Withholding Proprietary Information from Public Disclosure, and accompanying Affidavits CAW-01-1486 and CAW-01-1489 signed by Westinghouse, the owner of the information. Also included are a Proprietary Information Notice and a Copyright Notice.

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

Correspondence regarding the proprietary aspects of the Westinghouse information listed above, the Copyright Notice, or the supporting affidavit, should reference Westinghouse letters CAW-01-1486 and CAW-01-1489 and be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

TVA requests that approval be provided as soon as possible, but no later than prior to completion of the Unit 2 Cycle 11 refueling outage for both SQN units. TVA plans to implement the proposed power uprate during the Unit 1 Cycle 12 operating cycle and during the start-up of Unit 2 for Cycle 12 operation. The Unit 2 Cycle 11 refueling outage is currently scheduled to begin in April 2002. TVA requests that the revision be made effective within 45 days of NRC approval.

There are no new regulatory commitments being made by this submittal. This letter is being sent in accordance with NRC RIS 2001-05. If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 15th day of November

Peggy M. Billingsley
Notary Public

My Commission Expires October 9, 2002

Enclosures

LIST OF ENCLOSURES

ENCLOSURE 1 - DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

ENCLOSURE 2 - PROPOSED TECHNICAL SPECIFICATIONS - MARKUPS

ENCLOSURE 3 - PROPOSED TECHNICAL SPECIFICATIONS - REVISED PAGES

ENCLOSURE 4 - WCAP-15669, REVISION 0 - WESTINGHOUSE POWER MEASUREMENT INSTRUMENT UNCERTAINTY FOR TENNESSEE VALLEY AUTHORITY SEQUOYAH UNITS 1 & 2 (1.3% UPRATE TO 3467 Mwt - NSSS POWER) (PROPRIETARY)

ENCLOSURE 5 - WCAP-15670, REVISION 0 - WESTINGHOUSE POWER MEASUREMENT INSTRUMENT UNCERTAINTY FOR TENNESSEE VALLEY AUTHORITY SEQUOYAH UNITS 1 & 2 (1.3% UPRATE TO 3467 Mwt - NSSS POWER) (NON-PROPRIETARY)

ENCLOSURE 6 - WCAP 15726, REVISION 0 - SEQUOYAH UNITS 1 AND 2 1.3-PERCENT POWER UPRATE PROGRAM LICENSING REPORT

ENCLOSURE 7 - APPLICABILITY OF COMANCHE PEAK AND WATTS BAR POWER UPRATE RAIs TO SEQUOYAH 1&2 UPRATE (PROPRIETARY)

ENCLOSURE 8 - APPLICABILITY OF COMANCHE PEAK AND WATTS BAR POWER UPRATE RAIs TO SEQUOYAH 1&2 UPRATE (NON-PROPRIETARY)

ENCLOSURE 9 - WESTINGHOUSE APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NOS. 327 AND 328

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE NO. 01-08
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

The proposed license amendment would revise the SQN Unit 1 and Unit 2 operating licenses and TSs to increase the core power level by 1.3 percent to 3455 mega-watt thermal (MWt). The power uprate is based on the use of the Caldon, Incorporated (Caldon) Leading Edge Flow Meter, LEFM✓™ (LEFM) for determination of main feedwater flow and the associated determination of reactor power through the performance of a daily calorimetric, currently required by the SQN TSs. Specifically, as illustrated by the markup of the current SQN Units 1 and 2 operating licenses and TSs in Enclosure 2, the following changes are proposed:

(1) The Operating License for SQN Units 1 and 2 (DPR-77 and 79), Section 2.C.(1), identifies the maximum core thermal power level for which SQN is authorized to operate as 3411 MWt. TVA proposes changing the maximum core power level to 3455 MWt.

(2) The definition of RATED THERMAL POWER in TS 1.26 is changed to read:

"RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3455 MWt."

(3) TS Section 3/4.4.9.1, Pressure/Temperature Limits - Reactor Coolant System, Figures 3.4-2 and 3.4-3 for Unit 2 only are revised to change the effective full power year (EFPY) applicability from 16 EFPY to 14.5 EFPY. A corresponding change to the Bases has also been provided.

(4) TS Section 3/4.4.12, Low Temperature Over Pressure Protection Systems, Figure 3.4-4 for Unit 2 only is revised to change the EFPY applicability from 16 EFPY to 14.5 EFPY.

(5) TS Section 3/4.7.1.1, Safety Valves, Table 3.7-1, for inoperable steam line safety valves versus maximum allowable power range neutron flux high setpoint requirements is revised. With one inoperable main steam safety valve (MSSV) per loop, the setpoint is

lowered from 63 percent RTP to 62 percent RTP. Requirements for additional inoperable safety valves are not affected by the proposed power uprate.

In addition to the above TS changes, a change to the Bases for TS Section 3/4.7.1.1, "Turbine Cycle, Safety Valves," is necessary. This change revises the percentage between total secondary steam flow to safety valve relieving capacity from 127 to 106.4 percent. This also affects the total secondary steam flow that changes from 1.493×10^7 pounds per hour (lbs/hr) to 1.514×10^7 lbs/hr. The safety valve relieving capacity will also be changed from 1.9×10^7 to 1.6×10^7 lbs/hr. The change to the safety valve relieving capacity is a correction to this value and is not a result of the power uprate effort.

II. REASON FOR THE PROPOSED CHANGE

TVA has operated the SQN units at a maximum operating power level of 3411 MWt since the initial start-up of each unit. This reactor power level was based on the assumption of a 2 percent inaccuracy in the instrumentation used to determine true reactor power. Advanced instrumentation is currently available that improves the accuracy of determining feedwater flow and therefore, improves the ability to determine reactor power through heat balance analysis. TVA is installing the Caldon feedwater flow measurement instrumentation in the SQN units that will improve the accuracy of the heat balance analysis to an inaccuracy of no more than 0.7 percent. With the improved accuracy of this instrumentation, the power level of the SQN units can be increased by 1.3 percent to a power level of 3455 MWt. TVA is implementing this modification and requesting the associated Operating License and TS changes to increase reactor power output and improve the economic viability of the SQN units.

III. SAFETY ANALYSIS

Background

SQN Units 1 and 2 are presently licensed for a full core thermal power rating of 3411 MWt. The proposed license amendment would increase the core power level by 1.3 percent to 3455 MWt. TVA has evaluated the impact of a 1.3 percent uprate to 3455 MWt for applicable systems, structures, components, and safety analyses.

TVA's uprate is based on eliminating unnecessary analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in 10 CFR 50, Appendix K, "ECCS Evaluation Models," for SQN. The currently published regulation (as revised through 60 FR 24552, May 2, 1995), mandates consideration of an assumed reactor operating power level of 102 percent of the licensed power level for ECCS evaluation models of light water power reactors. However, the NRC has recently approved a change to the requirements of 10 CFR 50, Appendix K, whereby licensees are provided with the option of maintaining the current 2 percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin. For the latter case, the proposed alternative reduced margin must be demonstrated to account for uncertainties due to power level instrumentation error. Based on the proposed use of the improved Caldon LEFM instrumentation to determine core power level with a power measurement uncertainty of less than 0.7 percent, TVA proposes to reduce the licensed power uncertainty, required by 10 CFR 50, Appendix K, for modest increases of up to 1.3 percent in the licensed power level using current NRC approved methodologies.

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

Approach

The Sequoyah Units 1 and 2 Power Uprate Program was completed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," issued in 1983. Since its submittal to the NRC, the methodology has been successfully used as the basis for power uprate projects on over 20 pressurized water reactor (PWR) units, including Diablo Canyon Units 1 and 2, Turkey Point Units 3 and 4, Comanche Peak Unit 2, and Watts Bar Nuclear Plant (WBN) Unit 1.

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

For Sequoyah Units 1 and 2, Westinghouse Electric Corporation and Framatome Advance Nuclear Power (FRA-ANP) completed a comprehensive engineering review program that is consistent with the WCAP-10263 methodology to increase the licensed core power from 3411 MWt to 3455 MWt. The results of this review are contained in Enclosure 6 and is summarized as follows:

- Section 2.1 discusses the revised NSSS-design thermal and hydraulic parameters that were modified as a result of the 1.3 percent uprate and that serve as the basis for all of the NSSS analyses and evaluations.

- Section 2.2 concludes that no primary or secondary design transient modifications are required to accommodate the revised NSSS design conditions.

- Section 2.3 presents the evaluations and analyses performed in the NSSSs, e.g., safety injection system (SIS), residual heat removal (RHR) system, and control systems to support the revised design conditions.

- Section 2.4 presents the evaluations and analyses performed in the NSSS components, e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), steam generator, and NSSS auxiliary equipment to support the revised design conditions.

- Section 2.5 provides the evaluations of the accident analyses performed by Westinghouse which includes, evaluations performed for the loss-of-coolant accident (LOCA) and main steam line break mass and energy releases.

- Section 3.0 provides the evaluations of the fuel and the transient and accident analyses completed by FRA-ANP.

- Section 4.0 provides a review of plant operation and actions when the LEFM is inoperable or unavailable.

- Appendix A provides the calorimetric uncertainty calculations and is included separately in Enclosure 5.

- Appendix B provides input to support the 10 CFR 50.92 evaluation, in addition to any associated changes to the TSs.

The results of these analyses and evaluations demonstrate that all acceptance criteria continue to be met.

General Licensing Approach for Plant Analyses Using Plant Power Level

Most plant safety, component, and system analyses use the reactor and/or NSSS thermal power as inputs. These NSSS analyses generally model the core and/or NSSS thermal power in one of four ways, as described in the following paragraphs.

First, some analyses apply a 2 percent increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the 1.3 percent uprate conditions because the sum of increased core power level (1.3 percent) and the decreased power measurement uncertainty (less than 0.7 percent) fall within the previously analyzed conditions.

The power calorimetric uncertainty calculation, described in Enclosures 4 and 5, indicates that, with the Caldon LEFM installed, the power measurement uncertainty (based on a 95 percent probability, at a high confidence interval) is less than 0.7 percent. Thus, these analyses only need to reflect a 0.7 percent power measurement uncertainty. Accordingly, the existing 2 percent uncertainty can be allocated such that 1.3 percent is applied to provide sufficient margin to address the uprate to 3455 MWt, and 0.7 percent is retained in the analysis to still account for the power measurement uncertainty. In addition, for these types of analyses, it is shown that they still employ other conservative assumptions not affected by the 1.3 percent uprated power. Taken together, the use of the calculated 95/high confidence power measurement uncertainty and retention of conservative assumptions indicate that the margin of safety for these analyses would not be reduced.

Second, some analyses employ a nominal power level. These analyses have either been evaluated or re-performed for the 1.3 percent increased power level. The RHR cooldown analysis is the only analysis that was re-performed. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.3 percent uprate conditions.

Third, some of the analyses already employ a core power level in excess of the proposed 3455 MWt. These analyses were previously performed at a higher power level as part of prior plant programs. For these analyses, some of this available margin has been used to offset the 1.3 percent uprate. Consequently, the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the 1.3 percent uprate.

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

Westinghouse Results (Enclosure 6, Section 2.0)

This section summarizes the NSSS evaluations performed by Westinghouse for the uprating of SQN Units 1 and 2. These evaluations incorporated an increase in licensed core power from 3411 MWt to 3455 MWt.

NSSS Performance Parameters (Enclosure 6, Section 2.1)

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. These parameters provide the reactor coolant system (RCS) and secondary system conditions (steam generator temperatures, pressures, and flows) at the selected vessel average temperatures (Tavg), steam generator tube plugging (SGTP) levels, NSSS power levels, and RCS flowrates.

Due to the 1.3 percent increase in licensed core power from 3411 MWt to 3455 MWt, it was necessary to revise these parameters. The new parameters are identified in the following table and are incorporated, as required, into the applicable NSSS and component evaluations and into the safety analyses performed in support of the uprate.

NSSS design parameters are based on conservative inputs, such as a conservatively low thermal design flow (TDF) and bounding SGTP levels, which yield primary and secondary side conditions that bound the way the plant operates. The TDF is the conservatively low RCS flow value generally used in the safety analyses.

An increased NSSS power level of 3467 MWt (3455 MWt core power) is the only input assumption that changed from the current licensing basis.

The table provides the NSSS design parameter cases generated and used as the basis for the 1.3 percent power uprate. The 1.3 percent uprate resulted in changes to some of the NSSS design parameters, compared to the parameters that form the current licensing basis. The changes include the following RCS temperatures:

- Reactor Vessel Outlet (Thot) increased by 0.4 degree Fahrenheit (°F)

- Reactor Vessel Inlet (Tcold) decreased by 0.4°F

These small changes occurred because the Tavg was maintained at the current design value (578.2°F), while the core power was increased by 44 MWt to 3455 MWt. The temperature changes reflect the additional heat input from the uprated core. In addition, the 1.3 percent uprate resulted in the following changes to the secondary side parameters at 15 percent SGTP:

- Steam temperature decreased by 1.0°F

- Steam pressure decreased by 7 pounds per square inch

- Steam mass flow increased by 1.5 percent

These small changes are based on a calculation of the steam generator and secondary side performance, resulting from the increased power. The various Westinghouse analyses used the two cases of NSSS design parameters shown in the table in the evaluation of the effects of the 1.3 percent power uprate on SQN Units 1 & 2.

NSSS Performance Parameters		
Parameters	Current	Uprate
NSSS Power (MWt)	3423	3467
Reactor Power (MWt)	3411	3455
Thermal Design Flow (gpm/loop)	87,000(1)	87,000(1)
(total gpm)	348,000	348,000
Minimum Measured Flow (total gpm)	360,200(2)	360,200(2)
RCS Temperatures (°F)		
Core Outlet	616.0	616.4
Vessel Outlet (Thot)	611.2	611.6
Core Average	582.4	582.5
Vessel Average (Tavg)	578.2	578.2
Vessel/Core Inlet (Tcold)	545.2	544.8
Steam Generator Outlet (TSGout)	544.9	544.5
Zero Load Temperature	547	547
Reactor Coolant Pressure (psia)	2250	2250
Core Bypass (%)	7.5	7.5
Steam Generator		
Steam Pressure (psia)	802	795
Steam Temperature (Tsteam) (°F)	518.5	517.5
Steam Flow (Total, 106 lb/hr)	14.89	15.12
Feed Temperature (°F)	434.6	436.3
Tube Plugging (%)	15	15

Notes:

(1) TDF accommodates 15 percent SGTP.

(2) Reflects TS flow measurement uncertainty of 3.5 percent.

FRA-ANP Results (Enclosure 6, Section 3.0)

FRA-ANP has evaluated the impact of a 1.3 percent uprate to 3455 MWt core power on the applicable parameters analyzed in reload safety evaluations for SQN Units 1 and 2. All inputs to the accident analysis in the Final Safety Analysis Report (FSAR) related to fuel mechanical, system thermal-hydraulic, core thermal-hydraulic, neutronic, and Chapter 15 accident analyses were reviewed.

These reviews are detailed in Enclosure 6. For the fuel cycle design portion, FRA-ANP concluded that no significant adverse effect on the core power distribution analysis or core operating limits report (COLR) limits was expected as a result of implementation of the 1.3 percent power level uprate. Existing analytical methods remain adequate to evaluate reload cores, including error adjustments applied to COLR limits. Power peaking limits will not be significantly more restrictive. Although core monitoring margins may be slightly reduced, the reduction is within the variation seen as a result of fuel cycle design and final energy requirements. Core operating guidelines will continue to be applicable for power operation at the uprated thermal power.

Fuel mechanical analysis conclusions were that based on the fuel assembly and fuel rod mechanical evaluation, the 1.3 percent power level uprate for SQN Units 1 and 2 can be reached successfully. The 3455 MWt power level results in negligible changes to the hydraulic lift forces, thus the existing holddown margins remain applicable and acceptable. The increase in corrosion of the fuel assembly structural Zircaloy-4 components due to the slight increase in the core outlet temperature is small, thus acceptable structural margins for normal operating, faulted, and handling conditions exist. Changes in flow-induced vibration (FIV) forces and fuel assembly and fuel rod frequencies are negligible, thus the fuel assembly and fuel rod FIV performance remains acceptable. In addition, the existing fuel assembly faulted condition loading and analyses remain applicable and acceptable for the 3455 MWt power level. Existing fuel rod transient strain limits are shown to be applicable. Sufficient fuel rod creep collapse margin exists up to 65,000 mega-watt days/metric ton uranium. Although Zircaloy-4 fuel rod cladding corrosion remains limiting, fuel rod corrosion will remain acceptable at the 3455 MWt power level for the present cycle designs. Future fuel designs utilizing M5™ components will improve the safety margins compared to the Zircaloy-4 components, due to the inherent lower corrosion of the M5™ alloy.

The safety analysis and thermal hydraulic evaluation conclusion were based on the evaluation of the FSAR Chapter 15 safety analyses with respect to a 1.3 percent power uprate. All American Nuclear Society Condition II, III, and IV events were discussed with regard to the key parameters affecting the analyses and the role that the power uprate plays, if any, in each accident analysis.

The increase in power level will be accomplished by means of a decrease in the calorimetric uncertainty of the secondary side power measurement. A new main feedwater LEFM system will be installed that is designed to reduce the calorimetric uncertainty from ±2 percent to ±0.7 percent such that the 1.3 percent reduction in measurement uncertainty may be applied to power production. Because of this, all transient analyses that assumed an initial core power of 102 percent or greater were unaffected by the power uprate. In addition, the safety analyses performed at zero power conditions were also unaffected by the power uprate. The remainder of the Chapter 15 safety analyses were either insensitive to power level considerations or were bounded by other events.

It is the conclusion of this report that the key inputs used in the analysis of the FSAR Chapter 15 events continue to be applicable or bounding for a 1.3 percent power uprate coincident with a 1.3 percent decrease in calorimetric uncertainty. Therefore, there is no requirement that any of these events be reanalyzed.

Along with the proposed increase in reactor rated thermal power to 3455 MWt, TVA also proposes continued use of the topical reports identified in SQN Units 1 and 2 TS 6.9.1.14.a. These reports describe the NRC approved analytical methodologies used to determine the core protective and operating limits for Sequoyah Units 1 and 2, including the small break and large break LOCA analyses. In some of these topical reports, reference is made to the use of a 2 percent uncertainty applied to the reactor power, consistent with 10 CFR 50, Appendix K. This change in the power uncertainty does not constitute a significant change as defined in 10 CFR 50.46 and Appendix K. These topical reports reflect the 10 CFR 50, Appendix K rule before it was changed in June 2000. They do not reflect the current methodology as described in this license amendment request, and will not be updated at this time. If future methodology changes are made affecting the information in the topical reports, the references to the 10 CFR 50, Appendix K, 2% power uncertainty treatment will also be changed at that time.

TVA Evaluations

TVA has performed evaluations for the BOP systems and components and electrical distribution systems to verify the plant's ability to accommodate the proposed 1.3 percent power uprate. These evaluations have concluded that this increase in reactor core power will not create any adverse impact to plant systems. In some cases, plant instrumentation requirements have been modified to more appropriately accommodate the new power level requirements. Components for the turbine electro-hydraulic control system are being improved to provide more consistent control functions with the expected changes in turbine parameters associated with the increased core power level. These changes will provide additional confidence that the appropriate control and indication functions will adequately function with the uprated power.

The one required TS change associated with BOP systems is the requirements for reducing power when a MSSV is inoperable. TVA has reevaluated the calculations associated with the power range high flux trip setpoint limitations with inoperable safety valves. The requirements for two or more safety valves inoperable were acceptable and only the requirement for a single safety valve inoperable needs to be revised. This change is a reduction in the setpoint from 63 percent RTP to 62 percent RTP. This change is the result of additional core power that will have to be dissipated by the available safety valves under postulated accident conditions. Sufficient conservatism existed in the evaluations for two or more safety valves inoperable such that changes in the high flux trip setpoint were not necessary. This is a conservative change in the requirement that establishes a setpoint and power level that will ensure that the safety function of the safety valves is maintained. Therefore, the proposed change to the power range high flux trip setpoint will maintain the required safety functions for the MSSVs and will not adversely impact nuclear safety.

Conclusions

TVA is requesting a measurement uncertainty recapture power uprate similar to ones previously requested and approved by NRC. In particular, this request is patterned after TVA's WBN and Texas Utilities' Comanche Peak power uprate efforts. While there are some differences in the impacts to the SQN TSs from these efforts, these are primarily due to differences in the fuel vendor methodologies. For example, the FRA-ANP fuel vendor for SQN uses a statistical core design methodology which differs from the methodology used by Watts Bar's fuel vendor (Westinghouse). Other differences can be attributed to the Watts Bar TSs being formatted to the new improved standard TSs and Bases while SQN TSs have not been converted to the latest standard format. While differences do exist, the approach and intent of the SQN power uprate request is consistent with previous efforts and are based on equivalent evaluations and analysis techniques.

Enclosure 6 provides additional details regarding the above results for Westinghouse and FRA-ANP associated with the implementation of a 1.3 percent core power uprate for SQN Units 1 and 2. The evaluations performed for the proposed uprate verifies acceptability and that nuclear safety will not be adversely impacted. This is based on appropriate safety limits being utilized in response to the accuracy of the monitoring instrumentation being improved and the resulting change in operating margin that provides for an increase in reactor core power. The TVA evaluations support the proposed increase in core power and along with the change to the main steam safety valve requirements, provide acceptable provisions for the proposed power uprate. In addition, Enclosure 6 includes information about measures to be taken in the event the LEFM becomes inoperable and the Technical Requirements Manual (TRM) changes to accommodate this event. TVA has prepared a TRM revision that will implement the appropriate actions to compensate for the loss of the LEFM instrumentation. TVA is implementing this TRM change as part of the TS change implementation process for the proposed amendment. Therefore, the proposed TS changes are acceptable and nuclear safety functions will be maintained at appropriate levels for postulated plant events.

The following matrix provides a summary of the analysis impacts that resulted from the proposed power uprate. This matrix indicates the applicable section of the licensing summary report (Enclosure 6) that provides details for each conclusion.

MARGIN EVALUATION SUMMARY

NSSS SYSTEMS

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
RCS	Negligible. Minimum required pressurizer spray flow of 800 gpm is maintained.	2.3.1.1
Chemical and Volume Control System	Negligible. Existing margin available.	2.3.1.2
SIS	None.	2.3.1.3
RHR System - Cooldown	Some reduction in margin. However, system is still able to	2.3.1.4

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
	perform its design functions.	
Cold Overpressurization Mitigation System	None.	2.3.1.6
Control Systems (Condition I Transients)	None.	2.3.1.6
MSS - Main Steam Safety Valves (MSSVs)	Slight reduction. MSSVs can relieve 106.4% of rated steam flow, which exceeds 100% sizing criterion. Main steam pressure will therefore not exceed 110% of design pressure American Society of Mechanical Engineers code requirement.	2.3.2.2.1
Main Steam System (MSS) - Steam Dump	Slight reduction. Capacity is 41.5% of steam flow, which is still higher than design requirement of 40%.	2.3.2.3
Condensate Storage Tank	None. Analysis performed at 102%.	2.3.2.5.1

NSSS COMPONENTS

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
Reactor Vessel Structural (Fatigue)	Negligible. Substantial existing margin. Note: additional information included in Enclosure 7 (response to Watts Bar RAI 8/24/00 submittal, Mechanical Engineering Branch, Question 1).	2.4.1.1
Reactor Vessel (RV) Integrity	Minor decrease in margin due to revised fluence projections after power uprate. As a result, pressure-temperature limit curves changed from 16 EFPY to 14.5 EFPY for Unit 2. For other RV integrity issues (surveillance capsule withdrawal schedule, emergency response guideline limits, Pressurized thermal shock, and upper shelf energy), applicable limits are maintained.	2.4.1.3
Reactor Internals - bypass flow	Minimal effect. Design limit is still maintained.	2.4.1.4.1

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
Reactor Internals - Structural	Negligible based on minimal temperature changes. Gamma heating for the lower core plate increases but adequate fatigue margin still exists.	2.4.1.4.3
Reactor Internals - Rod Drop Time	Negligible. Increases by less than 0.010 seconds.	2.4.1.4.1
CRDMs	Negligible. Existing margin available.	2.4.2
Reactor Coolant Loop piping and supports	Negligible. Existing margin available.	2.4.3
Leak Before Break	Negligible. Existing margin available.	2.4.3
Reactor Coolant Pumps	Negligible. Existing margin available.	2.4.4
Steam Generator - Structural	Negligible. Existing margin available. Note: additional information included in Enclosure 7 (Response to Comanche Peak RAI in TXX-99115, Attachment 6, Question 5).	2.4.5.1
Steam Generator - U-Bend Fatigue	Some reduced margin. A few tubes are susceptible to fatigue and would have to be plugged if steam pressure falls below 800 psia.	2.4.5.3
Steam Generator - Moisture Carryover	Some decrease in margin. However, power uncertainty was calculated at a bounding value of 0.45%. At lower steam pressures associated with higher SGTP levels, MCO limit of 0.25% could be exceeded. This is not a licensing concern, but more of an economic concern.	2.4.5.4
Pressurizer	Negligible. Existing margin available.	2.4.6

ACCIDENT ANALYSIS

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
Steam Generator Tube Rupture - (overfill and dose releases)	None. Analysis performed at 102%.	3.3.9.4

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
Steam Line Break (SLB) - Long-Term M&E Releases Inside and Outside Containment	None. Analysis performed at 102%.	2.5.2.1, 2.5.2.2
Steam Releases (Dose)	None. Analysis performed at 104.5%.	2.5.2.4
Feedline Break M&E Releases	None. Analysis performed at 102%.	2.5.2.3
LOCA - LBLOCA	None. Analysis performed at 102%.	3.3.9.1
LOCA - SBLOCA	None. Analysis performed at 102%.	3.3.8.1
LOCA - Hydraulic Forcing Functions	Increased margin. More accurate break size assessment more than offsets decreased margin from uprate.	2.5.3
OTΔT, OPΔT, Power Range High/Low Neutron Flux Trip Setpoint	Slight decrease in operating margin. High flux trip setpoint redefined from 118% to 116.5%, with the same absolute power level. OTΔT and OPΔT setpoints are not affected.	3.3.7.3
LOCA M&E Releases - Long Term	None. Analysis performed at 102%.	2.5.1.1
LOCA M&E Releases - Short Term	Slight decrease in margin. But increase in M&Es due to uprate more than covered by margin in current analysis for loop subcompartment, pressurizer enclosure, and reactor cavity.	2.5.1.2

ACCIDENT ANALYSIS: DEPARTURE FROM NUCLEATE BOILING (DNB)

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
Core Limit Changes	Small operating margin reduction. Decrease in core average linear heat rate margin between 1.0-1.5%. Also 1-2% reduction in DNB peaking margins. No effect on peaking factor margin or LOCA peaking margins.	3.2.3.4
Rod Withdrawal from Subcritical	Negligible Departure from Nucleate Boiling Ratio (DNBR)	3.3.7.1

Evaluation/Analysis	Margin Impact	Section of WCAP-15726
	impact, since analysis starts from hot zero power and power increase is rapid. Effect of 'new' high neutron flux trip setpoint is negligible since absolute power value for setpoint does not change.	
Rod Withdrawal at Power	Small change. Adequate DNBR margin still exists.	3.3.7.2
Rod Control Cluster Assembly (RCCA) Misalignment (Dropped Rod)	Some reduction in margin. Adequate DNBR margin still exists.	3.3.7.3
Partial or Complete Loss of RCS Flow	Small change. Adequate DNBR margin still exists.	3.3.7.5 and 3.3.8.4
Loss of Load/Turbine Trip	Small change. Adequate DNBR margin still exists.	3.3.7.7
Feedwater System Malfunction - Excessive Heat Removal	Small change. Adequate DNBR margin still exists.	3.3.7.10
Accidental RCS Depressurization	Small change. Adequate DNBR margin still exists.	3.3.7.12
Inadvertent ECCS Actuation at Power	Limiting DNB occurs at event initiation and gets higher during event. Initial value based on 102% power. Adequate DNBR margin still exists.	3.3.7.14
Locked Rotor	Slight increase in number of fuel failures. Adequate DNBR margin still exists.	3.3.9.5
SLB Coincident With Rod Withdrawal at Power	Small change. Adequate DNBR margin still exists.	3.3.8.7
Rod Ejection	Small change. Adequate DNBR margin still exists.	3.3.9.7

Note: all other Non-LOCA accidents described in section 3.3 of Enclosure 6 explicitly modeled 102% power so there is no effect on the existing margin.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the technical specifications (TSs) and operating licenses, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

The following determination addresses the TSs and operating license changes associated with the increase in rated thermal power from 3411 to 3455 mega-watt thermal (MWt). The proposed TS changes also revise the effective duration of the reactor vessel pressure and temperature limits and the low temperature over-pressure requirements from 16 effective full power years (EFPY) to 14.5 EFPY for Unit 2. The final change lowers the required power range high flux trip setpoint with one main steam safety valve (MSSV) inoperable from 63 percent to 62 percent rated thermal power.

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

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

The primary loop components (reactor vessel, reactor internals, control rod drive mechanism, loop piping and supports, reactor coolant pump, steam generator and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. The rod control cluster assembly (RCCA) drop time remains within the current limits assumed in the accident analyses. Thus, there is no increase in the consequences of the accidents which credit RCCA drop. Several steam generator tubes may need to be plugged to preclude the potential for U-bend fatigue if the plant operates below certain steam pressure values. As long as these provisions are maintained, there is no increase in the probability of an steam generator tube rupture event. The leak before break analysis conclusions remain valid and thus the limiting break sizes determined in this analysis remain bounding.

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

The fuel has been completely analyzed to determine the effect of the 1.3 percent power uprate. The fuel assembly and fuel rod integrity have been evaluated. The change results in negligible changes to the hydraulic lift forces and the existing holddown margins remain acceptable. The increase in corrosion of the fuel assembly structural Zircaloy-4 components due to a slight increase in temperature is small, thus acceptable structural margin for normal operating, faulted, and handling conditions exist. The fuel assembly and fuel rod flow-induced vibration (FIV) performance remains acceptable. The existing fuel assembly faulted condition loading and analysis remain applicable and acceptable. The fuel rod strain, creep collapse, and corrosion performance were evaluated at the higher power level with acceptable results.

The fuel cycle design was evaluated and there was no significant effect caused by the 1.3 percent power uprate. The operational analysis of the core was evaluated for the change and found to remain applicable with acceptable results.

The thermal-hydraulic analysis was evaluated and found to remain applicable. The safety analysis addressed all Condition II, III, and IV events with the conclusion that current analyses remain applicable or bounding. The radiological consequences were evaluated and determined to be bounded by current analyses.

Additionally, the current licensing basis steamline break and LOCA mass and energy releases that are used to determine the peak containment pressure and temperature limits continue to remain bounding with the increase in power. Thus, there is no significant increase in the consequences of an accident previously evaluated.

The heatup and cooldown curves for Unit 2 are now applicable for 14.5 EFPY instead of 16 EFPY. The heatup and cooldown curves define limits that still ensure the prevention of nonductile failure for the SQN Units 1 and 2 reactor coolant system (RCS). The design-basis events that were protected have not changed. This modification does not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequences of an accident described in the Updated Final Safety Analysis Report. Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

The revised requirements for inoperable MSSVs provide limits for the power range high flux trip setpoint that ensure adequate relief capability for postulated accidents. This change does not alter any plant systems, components, or operating methods. Since the plant will continue to operate in the same manner with the same protective features, this change will not increase the possibility of an accident. The revised setpoint is a conservative change that provides additional margin considering the effect of the proposed power uprate. Since the revised setpoint continues to provide an equivalent level of safety function, this change will not significantly increase the consequences of an accident and the offsite dose impact will not be significantly increased.

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

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

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

Operation at the 3455 MWt core power does not involve a significant reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. The reduction in the EFPY for the Unit 2 heatup and cooldown curves does not reduce the margin of safety since the curves define the limits for ensuring the prevention of nonductile failure for the RCS and these curves remain unchanged.

The pressure and temperature safety limits will be the same as those for the current operating cycle, thus ensuring that the fuel will be maintained within the same range of safety parameters that form the basis for the Final Safety Analysis Report (FSAR) accident evaluations.

The power uprate represents a small increase in the energy production for the fuel cycle and is well within typical variations that occur as a result of increases in cycle length and capacity factor. The burnup of the fuel will increase proportionally with the increase in power, but will not challenge the current licensed burnup limit for Mark-BW fuel.

The slight increase in core average linear heat rate will result in a slight loss of operating margin, but will not affect safety margins. The centerline fuel melt and transient cladding strain limits will not be affected by the power level uprate, but the margin to these limits will decrease slightly. The LOCA F_Q limits will not be altered since the increase in core power is absorbed by reducing the power uncertainty used in determination of the limits.

The power peaking limits that provide DNB protection are slightly lower resulting in a proportional loss in DNB margins. The mechanical evaluation of the fuel demonstrates that the power level uprate can be successfully accomplished in compliance with all design criteria.

All FSAR Chapter 15 events have been evaluated and found to remain applicable for the power uprate. The radiological consequences analyses include an initial power assumption of 105 percent of 3411 MWt and remain bounding for the 1.3 percent power uprate.

The more restrictive limits for the power range high flux trip setpoint is based on calculations that ensure sufficient relief capacity to meet accident mitigation requirements. This change will appropriately limit reactor power levels, with inoperable MSSVs, such that the margin of safety is maintained at an equivalent level considering the proposed power uprate.

As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the NRC or that are in compliance with all applicable regulatory review guidance and standards. All of the fuel and safety evaluations for the 1.3 percent power uprate were performed with the Framatome-ANP approved methodology listed in TS Section 6.9.1.14 of the SQN TSs. Therefore, it is concluded that the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above information and on the analyses performed to support the proposed uprate conditions, the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

V. ENVIRONMENTAL IMPACT CONSIDERATION

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

The Proposed Decision

The Tennessee Valley Authority (TVA) proposes to install a Leading Edge Flow Measurement (LEFM) system for the feed water supply to the steam generators at Sequoyah Nuclear Plant (SQN). This installation and use would facilitate a power increase of 1.3 percent from 3423 Megawatt-thermal (MWt) to 3467 MWt.

Background

Chickamauga Reservoir on the Tennessee River, including the complex of TVA-controlled dams upstream of the plant intake (with Watts Bar Dam being the nearest upstream dam), and TVA's Chickamauga dam (nearest downstream dam) functions as the ultimate heat sink (UHS) for heat rejected by the turbogenerator cycle (via the condenser circulating water [CCW] system) primarily through the main condensers. The normal heat rejection path at Sequoyah Nuclear Plant (SQN) is through an open/helper loop circulating water system. Makeup water from the Tennessee River (CCW water) is used as a heat exchanger for the main condensers. Once heated, the CCW water is discharged through the CCW channel to the cooling towers (if needed) and then through the diffuser pond diffusers to the Tennessee River. This discharge is made under the conditions of National Pollutant Discharge Elimination System (NPDES) Permit No. TN 0026450.

Need for TVA's Action

The LEFM system measurement is very accurate and can substantially decrease the uncertainty associated with using existing venturi base flow measurement in the secondary side power calorimetric to determine thermal output of the core. The LEFM allows for a 0.7 percent power uncertainty associated with reactor power measurements.

Alternatives

Action

TVA proposes to implement the 1.3 percent power increase at SQN that is obtained by installation and use of an approved LEFM system for the feedwater supply to the steam generators. The flow measurement system utilizes ultrasonic transducers placed in a section of the main feedwater pipe and measure transient time of ultrasonic sound waves. All equipment would be installed within existing facility buildings.

The core power calculation, as determined by secondary side calorimetric, will be made using the LEFM inputs of feedwater mass flow and temperature. Control of feedwater flow will be by the existing controls from the nozzle venturi.

No Action

The No Action Alternative would be to continue to use the existing venturi base flow measurement. The existing measurement system consists of nozzle venturis placed in the feedwater lines to the individual steam generators.

Comparison of Alternatives

The No Action Alternative would utilize instrumentation that has a 2 percent uncertainty margin. The Proposed Action would utilize instrumentation that has a 0.7 percent uncertainty margin thus potentially increasing power output by 1.3 percent. As compared to the No Action Alternative, the Action Alternative would result in slightly increased heat rejection from the main condenser and thus slightly increased release of heat to the condenser cooling water (CCW) discharge from the plant. Under either alternative, compliance with river temperature, temperature rise and rate of temperature change limitations in the NPDES permit for SQN would be unaffected. The proposed uprate would remain bounded by Design Basis Accident doses and well within protective 10CFR100 limits. A minor, insignificant increase in frequency of processing of solid and liquid waste would result from action alternative over current rates which would also continue under the No Action alternative.

Affected Environment and Impacts Evaluated

Introduction

SQN is located in Hamilton County, Tennessee, 15 miles North of Chattanooga. The site is adjacent to the west shore of Chickamauga Lake, at Tennessee River Mile 484.5, on land operated by TVA. The impacts of construction and/or operation of the plant were addressed in the 1972 Environmental Statement for SQN Units 1 and 2. As noted in the environmental decision record, the media categories receiving further attention in this environmental assessment are wastewater, radioactivity, and solid waste.

Impacts Evaluated

<u>Waste Water Impacts</u>

Compliance with the river limitations (river temperature, temperature rise, and rate of temperature change), as stated in the NPDES permit, are monitored by means of a numerical model that solves thermohydrodynamic equations governing the flow and thermal conditions within the reservoir. This numerical model utilizes measured values of the upstream temperature profile, flow and temperature of the diffuser discharge, releases at Watts Bar and Chickamauga Dams, and the performance characteristics of the diffuser for SQN condenser circulating water (CCW). Compliance limitations for river temperature, temperature rise, and rate of temperature change are applicable at the edge of a mixing zone which is established as not to exceed the following dimensions: (1) a maximum length of 1500 feet downstream of the diffusers, (2) a maximum width of 750 feet, and (3) a maximum length of 275 feet upstream of the diffusers. The depth of the mixing zone measured from the surface varies linearly from the surface 275 feet upstream of the diffusers to the top of the diffuser pipes and extends to the bottom downstream of the diffusers. The NPDES temperature limitations at the edge of the mixing zone are as follows: (1) the maximum 24-hour average river temperature is limited to 30.5 degrees centigrade (°C) as a daily maximum, (2) the 24 hour average temperature rise shall be limited to 3.0°C during the months of April through October and 5.0°C during the months of November through March, and (3) the one hour average rate of temperature change shall be limited to 2°C per hour.

The component cooling water system (CCS) removes heat from various safety and non-safety related equipment and transfers it to the essential raw cooling water (ERCW) System, and then that heat is transferred to the ultimate heat sink (i.e., the Tennessee River and Chickamauga Reservoir). The CCS closed loop provides an intermediate barrier to contain radioactive or potentially radioactive sources, thus precluding direct leakage of radioactive fluids into the UHS.

The performance of the ERCW System is measured by its ability to remove heat from each ERCW-cooled component and transfer that heat to the UHS. The ability of the ERCW to remove heat from a component is a function of the Tennessee River (supply) temperature and the ERCW flow rate through the component. The CCS design is based on a maximum ERCW temperature of 84.5°F.

The Performance Evaluation of Power System Efficiencies software heat balances were performed for three different CCW inlet temperatures to the main condenser. The following table provides the results from the heat balances and information concerning heat rejected from the main condenser.

CCW Inlet		Before Uprate (3423 MWt - NSSS) CCW Outlet		After Uprate (3467 MWt - NSSS) CCW Outlet		Heat Rejected Difference Based on CCW Flow of 530600 gpm
Temp (°F)	Enthalpy (Btu/lbm)	Temp (°F)	Enthalpy (Btu/lbm)	Temp (°F)	Enthalpy (Btu/lbm)	(Btu/hr)
40.0	8.1	68.5	36.7	68.9	37.1	1.40%
62.0	30.1	90.5	58.6	91.0	59.0	1.40%
85.0	53.1	114.2	82.2	114.6	82.6	1.37%

Under the current operational conditions and permit limitations, SQN has only insignificant effects on the aquatic environment (TVA 1972) and no effect on the fish and benthic communities in the vicinity of SQN (D. S. Baxter, K. D. Gardner, and S. J. Fraley, "Results of Biological Monitoring in the Vicinity of Sequoyah Nuclear Plant, 2000)."

Under the No Action alternative this condition is anticipated to continue. Based on the findings from the above table, an evaluation by Norris Engineering Laboratory concluded that the Action Alternative (1.3 percent power uprate) would not cause a thermally related NPDES permit violation, nor substantively affect the heat discharge characteristics of SQN outside the existing mixing zone. This minor addition of heat to the mixing zone would not result in significant effects to the aquatic environment of Chickamauga Reservoir nor cumulatively to the Tennessee River system.

Radiological Impacts

The existing baseline calculations have been evaluated to determine the potential impact on the radiological effluents from a 1.3 percent reactor power level uprating to 3467 MWt. The 1.3 percent calculations demonstrate that offsite dose from normal effluent releases remain significantly below bounding limits of 10CFR50, Appendix I. The gaseous waste processing system continues to meet its design basis under the uprated conditions, in that the gas storage tanks have sufficient capacity to store, for decay, the gases produced due to normal operation, including anticipated operational transients. The normal annual average gaseous release remains limited to a small fraction of 10CFR20 limits for identified mixtures. Continuation of normal operations under the No Action alternative, would result in no changes to existing baseline conditions for radiological effects. The 1.3 percent power uprate would not increase the potential for any additional personnel exposure. This power uprate is still bounded by the Design Basis Accident doses and would remain well within the 10CFR100 limits. Thus, no substantive additional impacts to human health or the environment are anticipated for the Action alternative.

Solid and Radioactive Waste Impacts

The solid waste management and liquid waste processing systems are designed to control, collect, process, store and dispose of radioactive wastes due to normal operation including anticipated operational transients. Operation of these systems are primarily influenced by the volume of waste processed. Under the No Action alternative the current frequency of operation for disposal systems would continue. Because these systems are typically operated in a batch mode, the only potential effect from the Action alternative would be about a 2 percent increase in batch releases each year. Thus, as a result of the uprate, the amounts of the solid waste and liquid waste processed are not expected to significantly change from those of current conditions.

Conclusions

The operating parameters associated with the power uprate were evaluated for the potential to affect thermal characteristics of CCW discharge, production of wastes, and radiological effluents and doses. These parameters either retain the same values as the original values evaluated in the Final Environmental Statement or are bounded by those values. No commitments or mitigation requirements were identified as necessary for undertaking the proposed action.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 01-08
MARKED PAGES

I. AFFECTED PAGE LIST

Unit 1

Operating License Page 3
1-5
3/4 7-2
B 3/4 7-1

Unit 2

Operating License Page 3
1-6
3/4 4-29
3/4 4-30
3/4 4-35
3/4 7-2
B 3/4 4-7
B 3/4 7-1

II. MARKED PAGES

See attached.

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of ~~3411~~ 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 270 are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria forany test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at a power level different from there described; and

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 DELETED

PURGE - PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411~~ 3455 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	~~63~~ ***62***
2	45
3	28

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

In Mode 1 above 28% RTP, the number of MSSVs per steam generator required to be operable must be according to Table 3.7-1 in the accompanying LCO. At or below 28% RTP in Modes 1, 2, and 3, only two MSSVs per steam generator are required to be operable.

In Modes 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be operable in these modes.

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~1.9~~ 1.6 x 10^7 lbs/hr at 1170 psig which is ~~127~~ 106.4 percent of the total secondary steam flow of ~~1.493~~ 1.514 x 10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

To calculate this setpoint, the governing equation is the relationship q =mΔh, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). Thus, an algorithm for use in defining the revised Technical Specification table setpoint values would be:

$$\text{Hi } \phi = (100/Q)\,\frac{(w_s h_{fg} N)}{K}$$

Where:

Hi ϕ = Safety Analysis power range high neutron flux setpoint, percent

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of ~~3411~~ megawatts thermal.

3455

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 258 are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at a power level different from there described; and

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of ~~3411~~ MWt. 3455

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.29 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1).

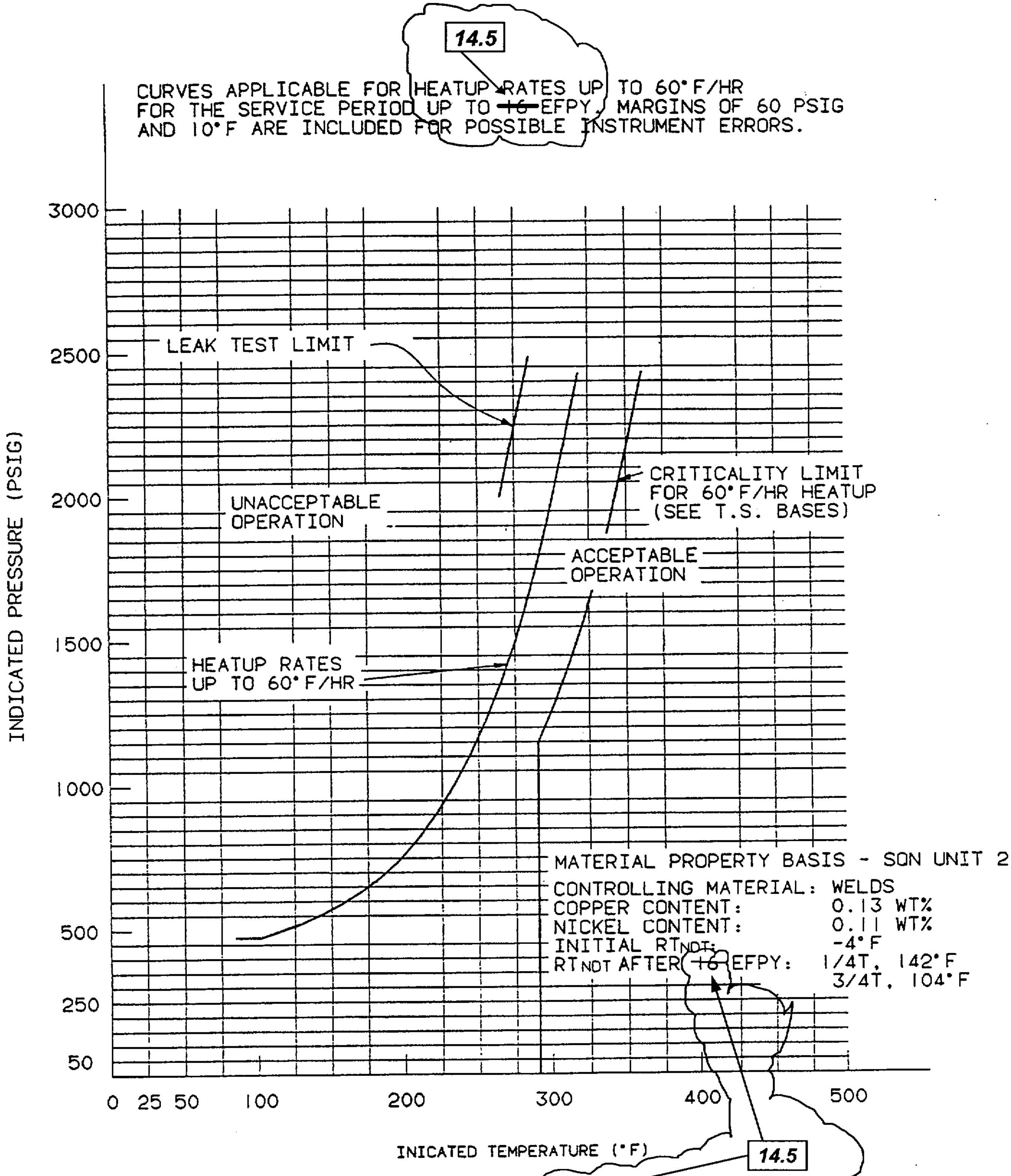


FIGURE 3.4-2 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO ~~16~~ EFPY

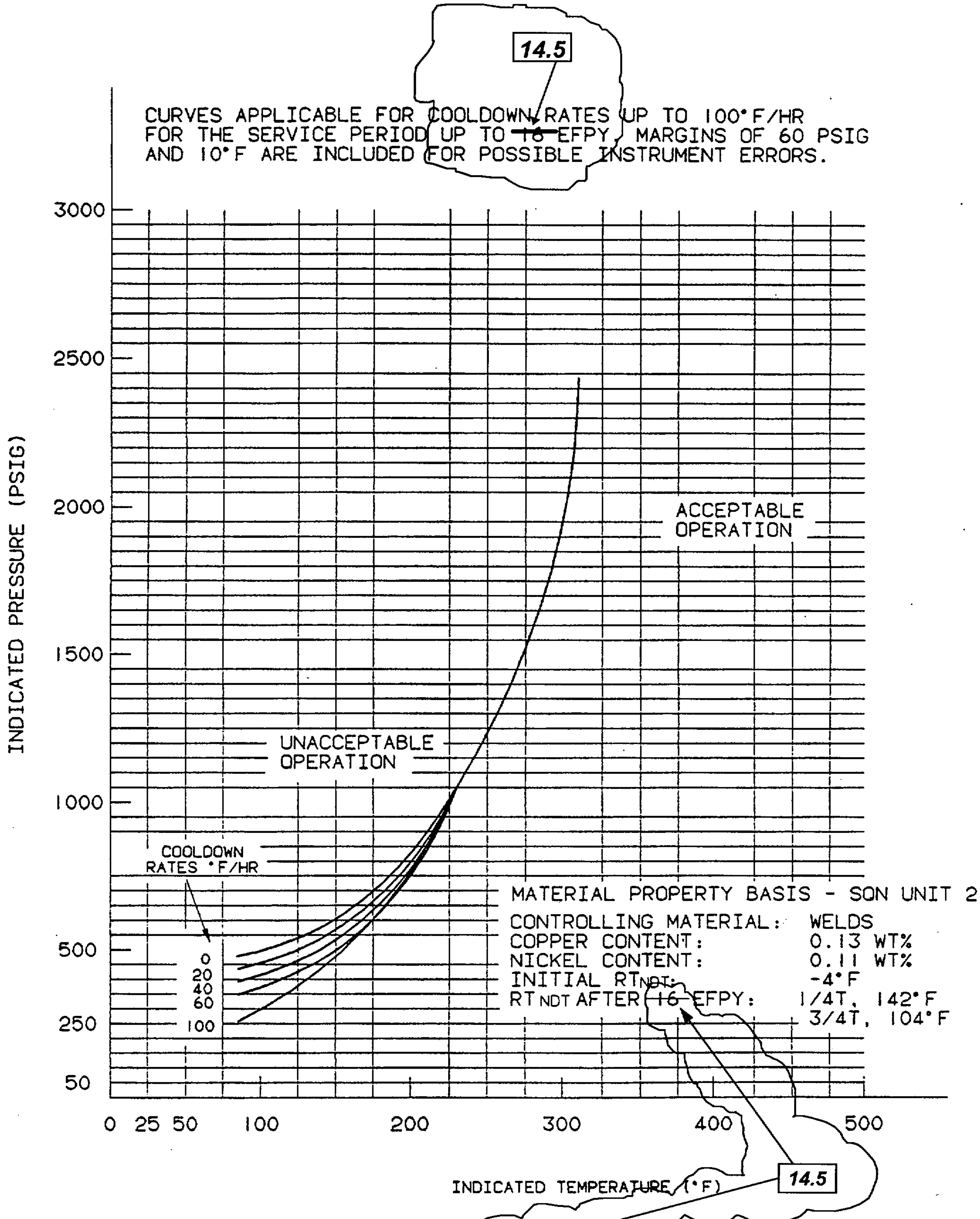
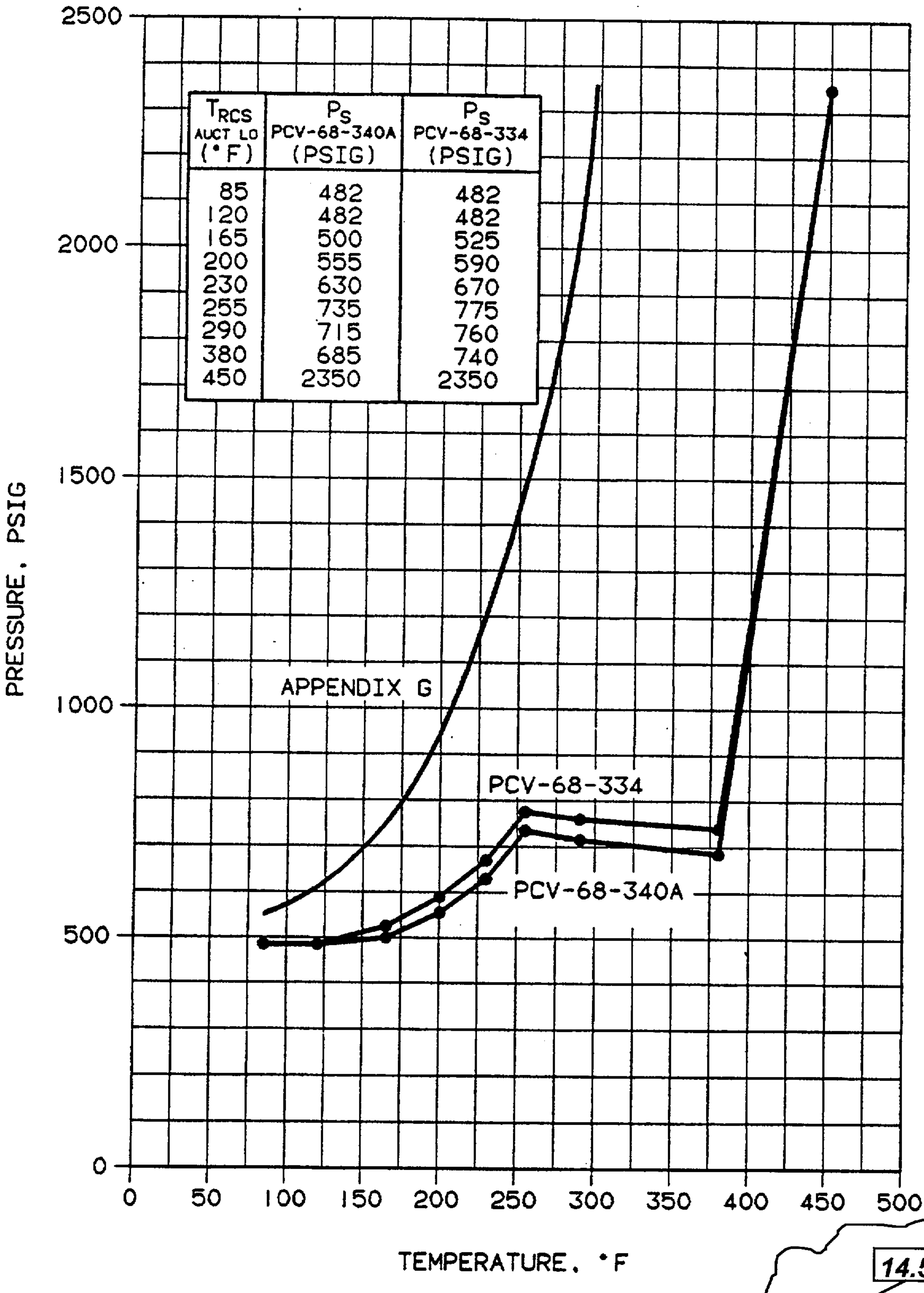


FIGURE 3.4-3 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPICICABLE UP TO ~~16~~ EFPY



T_{RCS} AUCT LO (°F)	P_S PCV-68-340A (PSIG)	P_S PCV-68-334 (PSIG)
85	482	482
120	482	482
165	500	525
200	555	590
230	630	670
255	735	775
290	715	760
380	685	740
450	2350	2350

PORV NOMINAL LIFT SETTINGS - APPLICABLE UP TO ~~16~~ 14.5 EFPY

FIGURE 3.4-4

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	~~63~~ ***62***
2	45
3	28

PRESSURE/TEMPERATURE LIMITS (Continued)

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

10 CFR 50, Appendix G, addresses metal temperature of the closure head flange and vessel regions. Appendix G states that the minimum metal temperature of the closure flange region should be at least 120 degrees Fahrenheit (F) higher than the limiting RT_{NDT} for this region when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (561 pounds per square inch gauge (psig) for Westinghouse Electric Corporation plants). For SQN, Unit 2, the minimum temperature of the closure flange and vessel flange regions is 117 degrees F since the limiting initial RT_{NDT} for the closure head flange is -13 degrees F (see Table B 3/4.4-1). These numbers (561 psig and 117 degrees F) include a margin for instrumentation error of 10 degrees F and 60 psig. The SQN Unit 2 heat up and cooldown curves shown in Figures 3.4-2 and 3.4-3 are not impacted by this regulation.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, and ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to 10 CFR 50 and Appendix G of the 1986 ASME Boiler and Pressure Vessel Code, Section III, Division 1 and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} at the end of 16 effective full power years of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2 and a peak surface fluence of 0.864×10^{19} n/cm^2 for 16 effective full power years (Reference WCAP 12971, "Heatup and Cooldown Limit Curves for Normal Operation," June 1991. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY~~, as well as~~ adjustments ~~for possible errors in the pressure~~ and temperature sensing instruments. The heatup and cooldown limits in WCAP-12971 were based on a core thermal power of 3411 MWt. The curves have been evaluated in WCAP-15725 to be effective for operation through the end of 14.5 EFPY for the uprated core thermal power of 3455 MWt.

Added

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

In Mode 1 above 28% RTP, the number of MSSVs per steam generator required to be operable must be according to Table 3.7-1 in the accompanying LCO. At or below 28% RTP in Modes 1, 2, and 3, only two MSSVs per steam generator are required to be operable.

In Modes 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be operable in these modes.

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~1.9 x~~ 1.6 x 10^7 lbs/hr at 1170 psig which is ~~127~~ 106.4 percent of the total secondary steam flow of ~~1.493 x~~ 1.514 x 10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

To calculate this setpoint, the governing equation is the relationship q = mΔh, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). Thus, an algorithm for use in defining the revised Technical Specification table setpoint values would be:

$$\text{Hi } \Phi = (100/Q) \frac{(W_s h_{fg} N)}{K}$$

where:

Hi Φ = Safety Analysis power range high neutron flux setpoint, percent

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 01-08
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1

Operating License Page 3
1-5
3/4 7-2
B 3/4 7-1

Unit 2

Operating License Page 3
1-6
3/4 4-29
3/4 4-30
3/4 4-35
3/4 7-2
B 3/4 4-7
B 3/4 7-1

II. REVISED PAGES

See attached.

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 270 are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

Amendment Number Affected by Previous TS Change Request

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at a power level different from there described; and

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 DELETED

PURGE - PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3455 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	62
2	45
3	28

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

In Mode 1 above 28% RTP, the number of MSSVs per steam generator required to be operable must be according to Table 3.7-1 in the accompanying LCO. At or below 28% RTP in Modes 1, 2, and 3, only two MSSVs per steam generator are required to be operable.

In Modes 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be operable in these modes.

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.6×10^7 lbs/hr at 1170 psig which is 106.4 percent of the total secondary steam flow of 1.514×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

To calculate this setpoint, the governing equation is the relationship $q = m\Delta h$, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). Thus, an algorithm for use in defining the revised Technical Specification table setpoint values would be:

$$\text{Hi } \phi = (100/Q)\,\frac{(w_s h_{fg} N)}{K}$$

Where:

Hi ϕ = Safety Analysis power range high neutron flux setpoint, percent

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 258 are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

Amendment Number Affected by Previous TS Change Requests

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 ofTVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at a power level different from there described; and

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3455 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.29 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1).

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR
FOR THE SERVICE PERIOD UP TO 14.5 EFPY. MARGINS OF 60 PSIG
AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERRORS.

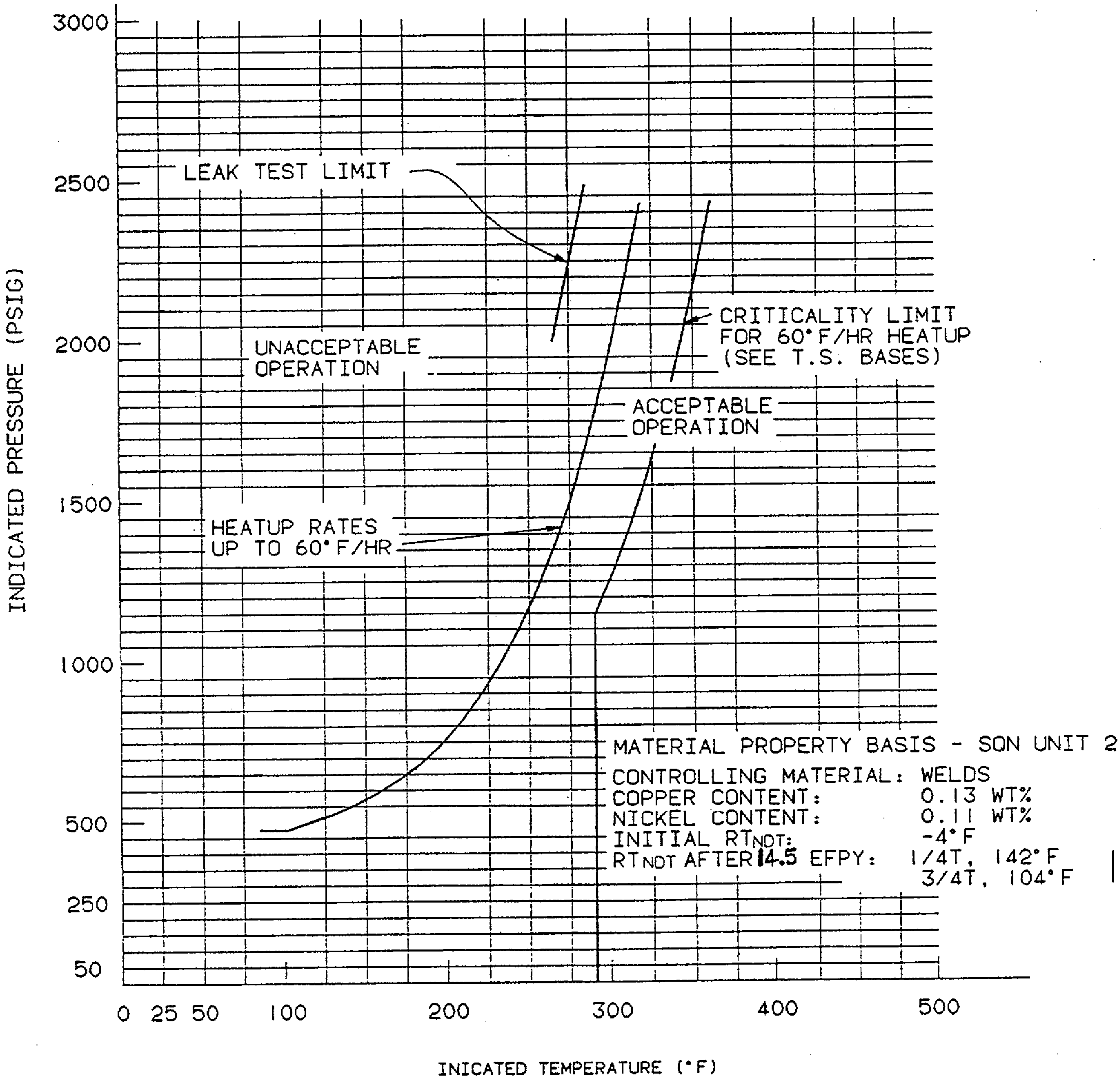


FIGURE 3.4-2 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 14.5 EFPY

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR
FOR THE SERVICE PERIOD UP TO 14.5 EFPY. MARGINS OF 60 PSIG
AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERRORS.

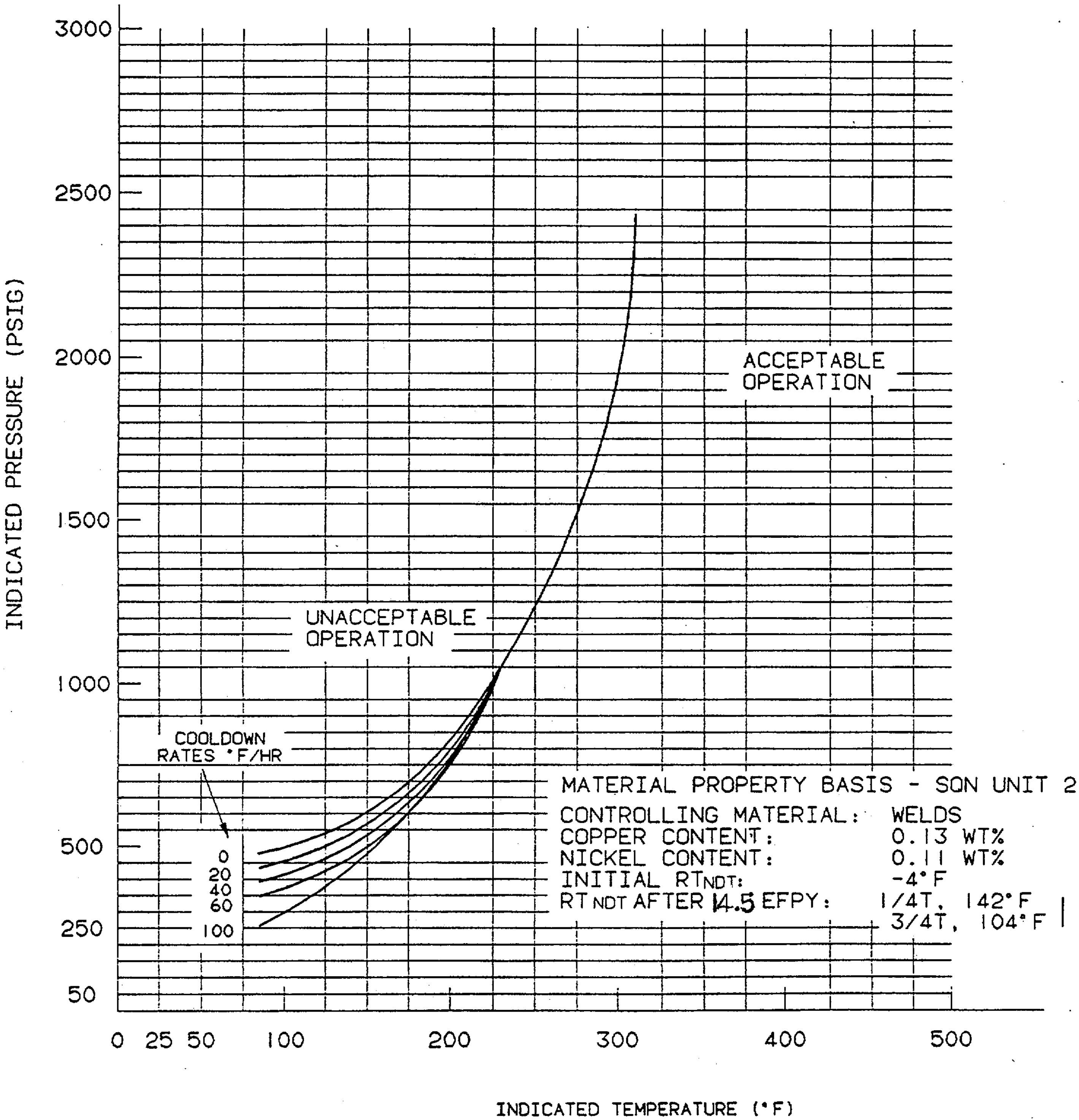
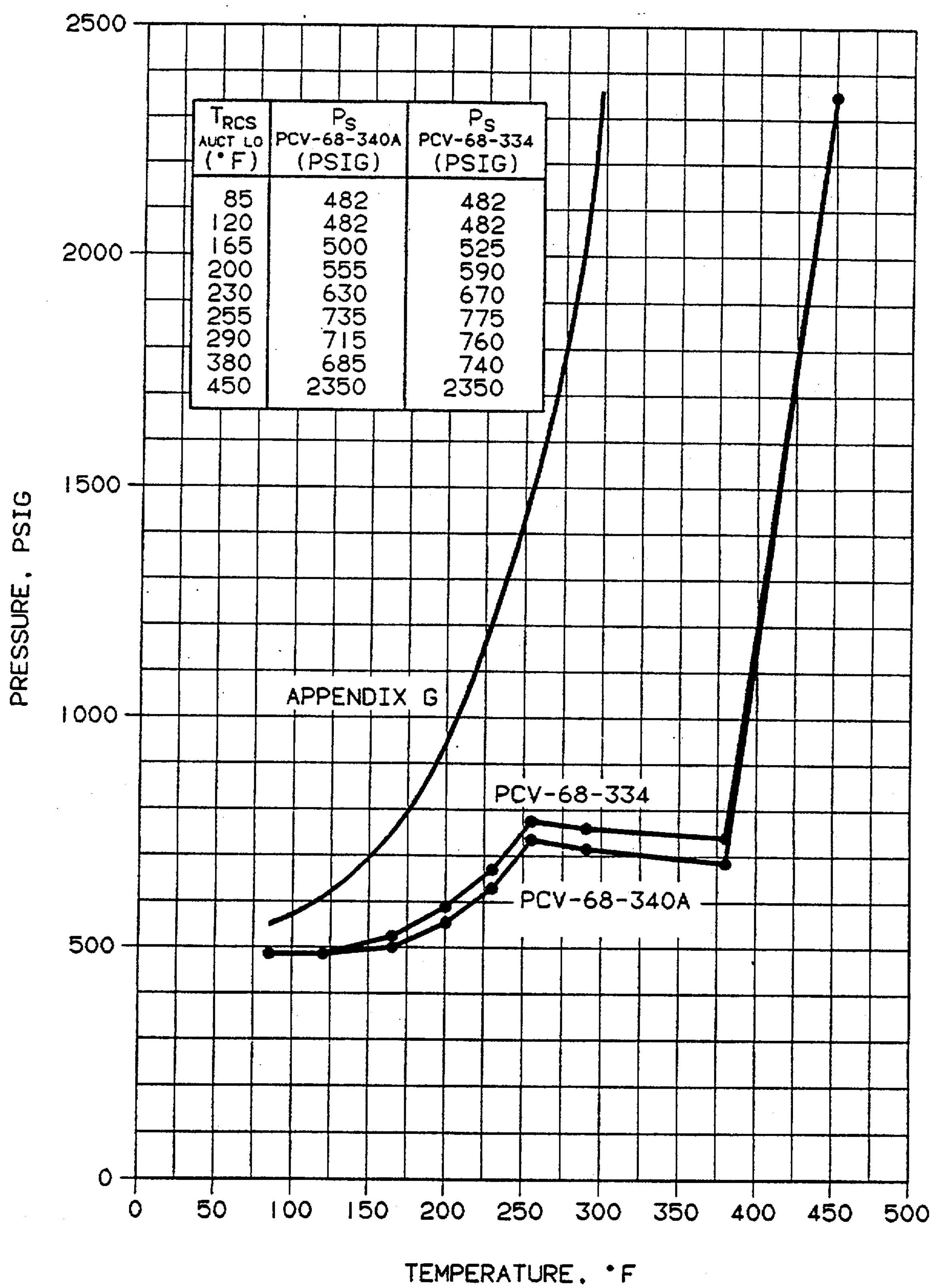


FIGURE 3.4-3 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE UP TO 14.5 EFPY



T_{RCS} AUCT LO (°F)	P_S PCV-68-340A (PSIG)	P_S PCV-68-334 (PSIG)
85	482	482
120	482	482
165	500	525
200	555	590
230	630	670
255	735	775
290	715	760
380	685	740
450	2350	2350

PORV NOMINAL LIFT SETTINGS - APPLICABLE UP TO 14.5 EFPY

FIGURE 3.4-4

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	62
2	45
3	28

PRESSURE/TEMPERATURE LIMITS (Continued)

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

10 CFR 50, Appendix G, addresses metal temperature of the closure head flange and vessel regions. Appendix G states that the minimum metal temperature of the closure flange region should be at least 120 degrees Fahrenheit (F) higher than the limiting RT_{NDT} for this region when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (561 pounds per square inch gauge (psig) for Westinghouse Electric Corporation plants). For SQN, Unit 2, the minimum temperature of the closure flange and vessel flange regions is 117 degrees F since the limiting initial RT_{NDT} for the closure head flange is -13 degrees F (see Table B 3/4.4-1). These numbers (561 psig and 117 degrees F) include a margin for instrumentation error of 10 degrees F and 60 psig. The SQN Unit 2 heat up and cooldown curves shown in Figures 3.4-2 and 3.4-3 are not impacted by this regulation.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, and ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to 10 CFR 50 and Appendix G of the 1986 ASME Boiler and Pressure Vessel Code, Section III, Division 1 and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} at the end of 16 effective full power years of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2 and a peak surface fluence of 0.864×10^{19} n/cm^2 for 16 effective full power years (Reference WCAP 12971, "Heatup and Cooldown Limit Curves for Normal Operation," June 1991. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments. The heatup and cooldown limits in WCAP-12971 were based on a core thermal power of 3411 MWt. The curves have been evaluated in WCAP-15725 to be effective for operation through the end of 14.5 EFPY for the uprated core thermal power of 3455 MWt.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of the system design pressure during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

In Mode 1 above 28% RTP, the number of MSSVs per steam generator required to be operable must be according to Table 3.7-1 in the accompanying LCO. At or below 28% RTP in Modes 1, 2, and 3, only two MSSVs per steam generator are required to be operable.

In Modes 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be operable in these modes.

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 1.6×10^7 lbs/hr at 1170 psig which is 106.4 percent of the total secondary steam flow of 1.514×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

To calculate this setpoint, the governing equation is the relationship $q = m\Delta h$, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). Thus, an algorithm for use in defining the revised Technical Specification table setpoint values would be:

$$\text{Hi } \Phi = (100/Q)\,\frac{(W_s h_{fg} N)}{K}$$

where:

Hi Φ = Safety Analysis power range high neutron flux setpoint, percent

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

WCAP-15670, REVISION 0
WESTINGHOUSE POWER MEASUREMENT INSTRUMENT UNCERTAINTY FOR TENNESSEE VALLEY AUTHORITY SEQUOYAH UNITS 1 & 2
(1.3% UPRATE TO 3467 Mwt - NSSS POWER)
(NON-PROPRIETARY)

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15670
Rev.0

Westinghouse Power Measurement Instrument Uncertainty Methodology for Tennessee Valley Authority Sequoyah Units 1 & 2 (1.3% Uprate to 3467 Mwt - NSSS Power)

July 2001

W.H.Moomau

Westinghouse Electric Company LLC
4350 Northern Pike
Monroeville, Pennsylvania 15146-2886

ACKNOWLEDGEMENT

The author would like to acknowledge Lynn Cowan, Sherry Cutts, Don Elkins, Duane Morris and Ed Turner of TVA for their review of this report.

TABLE OF CONTENTS

LIST OF TABLES

WESTINGHOUSE POWER MEASUREMENT
INSTRUMENT UNCERTAINTY METHODOLOGY FOR TENNESSEE VALLEY AUTHORITY
SEQUOYAH UNITS 1 & 2
(1.3% Uprate to 3467 MWT - NSSS POWER)

I. INTRODUCTION

A reactor power measurement is used in the Sequoyah safety analysis. Power is monitored by the performance of a secondary side heat balance (power calorimetric measurement) once every 24 hours to comply with the Sequoyah Technical Specifications. The uncertainty associated with the daily power measurement is used in the Sequoyah safety analysis for an initial plant condition assumption and for the development of reactor trip setpoints. This report provides the power measurement uncertainty and is applicable for 18 month fuel cycles.

The uncertainty calculation in this report is for the Sequoyah Units 1 & 2 1.3% Uprate to 3467 Mwt NSSS power. The uncertainty calculation is based on instrument channel uncertainties provided by TVA for the instrument channels used in the calorimetric power measurement. The evaluation of calorimetric measurement uncertainties includes the power calorimetric measurement used for the daily nuclear instrumentation channel normalization.

II. METHODOLOGY

The total instrument channel uncertainties used in this power measurement uncertainty calculation have been provided by TVA. The TVA methodology used to combine the uncertainty components for an instrument channel is the square root of the sum of the squares of those groups of components that are statistically independent. Those uncertainties that are dependent are combined arithmetically into independent groups, which are then combined by the square root of the sum of the squares. The uncertainties are considered to be random, two sided distributions. The sum of both sides is equal to the range for that parameter, e.g., rack drift allowance is [ ]$^{+a,c}$, the range for this parameter is [ ]$^{+a,c}$. This technique has been utilized before, and has been endorsed by the NRC staff[1,2,3,4] and industry standards[5,6].

The methodology used to combine the power measurement uncertainty components is the square root of the sum of the squares of those groups of components which are statistically independent. Those uncertainties that are dependent are combined arithmetically into independent groups, which are then combined by the square root of the sum of the squares. This technique has been utilized before, and has been endorsed by the NRC staff[1,2,3,4] and industry standards[5,6].

III. INSTRUMENTATION UNCERTAINTIES

1. POWER MEASUREMENT UNCERTANTY

(Using LEFM on Feedwater Header)

Sequoyah Units 1 & 2 Technical Specifications require a secondary side heat balance (or calorimetric) power measurement every 24 hours when power is above 15% of Rated Thermal Power. This heat balance is used to verify that the plant is operating within the limits of the Operating License and to adjust the Power Range Nuclear Instrumentation System (NIS) channels when the difference between the NIS and the heat balance is greater than that required by the Technical Specifications. Since it is necessary to make this determination daily, the plant computer is used for the measurements. The following calculation determines the computer power measurement uncertainty for the Sequoyah safety analysis.

Assuming that the primary and secondary sides are in equilibrium, the core power is determined by summing the thermal output of the steam generators, correcting the total secondary power for steam generator blowdown (if not secured), subtracting the RCP heat addition, adding the primary side system losses, and dividing by the core rated Btu/hr at full power. The equation for this calculation is:

$$RP = \frac{N\{Q_{SG} - Q_P + (Q_L/N)\}(100)}{H} \qquad \text{Eq. 1}$$

where;

RP	=	Core power (% RTP)
N	=	Number of primary side loops
Q_{SG}	=	Steam Generator thermal output (BTU/hr)
Q_p	=	RCP heat adder (Btu/hr)
Q_L	=	Primary system net heat losses (Btu/hr)
H	=	Core rated Btu/hr at full rated power.

For the purposes of this uncertainty analysis (and based on H noted above) it is assumed that the plant is at 100% RTP when the measurement is taken.

The thermal output of the Steam Generator is determined by a secondary side calorimetric measurement, which is defined as:

$$Q_{SG} = (h_s - h_f)W_f + (h_{sgbd} - h_s)W_{sgbd} \qquad \text{Eq. 2}$$

where;
h_s = Steam enthalpy (Btu/lb)
h_f = Feedwater enthalpy (Btu/lb)
h_{sgbd} = Steam generator blowdown enthalpy (Btu/lb)
W_f = Feedwater flow (lb/hr)
W_{sgbd} = Steam generator blowdown flow (lb/hr).

The steam enthalpy is based on measurement of steam generator outlet steam pressure, assuming saturated conditions. The feedwater enthalpy is based on the measurement of feedwater temperature and feedwater pressure. The feedwater flow is determined by a Leading Edge Flow Meter (LEFM) measurement on the main feedwater header, and it is assumed that the loop feedwater flows are equal.

The steam generator blowdown flow is the outlet flow from the steam generators used to control water chemistry, and is determined by measurement from the steam generator blowdown header flow orifice and the following calculation:

$$W_{sgbd} = (K)(F_a)(a)\ \{(2)(g_c)(p_f)(d/p)\}^{1/2} \qquad \text{Eq.3}$$

where;
K = Steam generator blowdown header flow orifice coefficient
F_a = Steam generator blowdown header flow orifice correction for thermal expansion
a = Steam generator blowdown header flow orifice area
g_c = Gravitational constant (32.174 ft/sec^2)
p_f = Steam generator blowdown header flow density (lb/ft^3)
d/p = Steam generator blowdown header flow orifice pressure drop (inches H_2O).

The steam generator blowdown orifice flow coefficient is the product of a number of constants including as-built dimensions of the orifice and pipe internal diameter. The thermal expansion correction is based on the coefficient of expansion of the orifice material and the difference between steam generator blowdown temperature and calibration temperature. Steam generator blowdown density and enthalpy are based on the measurement of steam generator steam pressure. The blowdown liquid enthalpy is assumed to be equal to that of a saturated liquid at the measured steam pressure. The orifice pressure drop is obtained from the output of the differential pressure transmitter.

RCP heat addition is determined by calculation, based on the best estimate of coolant flow, pump head, and pump hydraulic efficiency.

The primary system net heat losses are determined by calculation, considering the following system heat inputs and heat losses:

Charging flow
Letdown flow
Seal injection flow
RCP thermal barrier cooler heat removal
Pressurizer spray flow
Pressurizer surge line flow
Component insulation heat losses
Component support heat losses
CRDM heat losses.

A single calculated sum for 100% of Rated Thermal Power operation is used for these losses or heat inputs.

The power measurement is thus based on the following plant measurements:

Steamline pressure (P_s)
Feedwater temperature (T_f)(LEFM)
Feedwater pressure (P_f)
Feedwater header flow (LEFM)
Steam generator blowdown header flow orifice differential pressure (d/p)(if not secured)

and on the following calculated values:

Steam generator blowdown header flow orifice coefficient (K)
Steam generator blowdown header flow orifice thermal expansion correction (F_a)
Steam generator blowdown header flow orifice area (a)
Feedwater density (ρ_f)
Feedwater enthalpy (h_f)
Steam enthalpy (h_s)
Steam generator blowdown enthalpy (h_{sgbd})
Steam generator blowdown density (p_f)

Moisture carryover (impacts h_s)

Primary system net heat losses (Q_L)

RCP heat addition (Q_p)

The derivation of the measurement uncertainties and the calorimetric power measurement uncertainties on Table 3 are noted below.

Secondary Side

The secondary side uncertainties are in four principal areas, feedwater flow, feedwater enthalpy, steam enthalpy and net pump heat addition. These four areas are specifically identified on Table 3.

For the measurement of feedwater flow, a LEFM is installed on the feedwater header with an accuracy provided by TVA and Caldon, Inc. The feedwater temperature is also measured by the LEFM, thus no uncertainties are associated with feedwater temperature. The overall uncertainty of the LEFM is given as [ ]$^{+a,c}$.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 1 notes the instrument uncertainties for the hardware used to perform the measurements. Table 2 lists the various sensitivities that are specific to the operating conditions at 100% of Rated Thermal Power and are affected by the magnitudes of the instrument uncertainties noted in Table 1. As can be seen on Table 3, feedwater temperature uncertainties have an effect on feedwater density and feedwater enthalpy. Feedwater pressure uncertainties affect feedwater density and feedwater enthalpy.

Using the NBS/NRC Steam Tables again, it is possible to determine the sensitivity of steam enthalpy to changes in steam pressure and steam quality. Table 1 notes the uncertainty in steam pressure and Table 2 provides the sensitivity. For steam quality, the NBS/NRC Steam Tables were used to determine the sensitivity for a moisture content range of [ ]$^{+a,c}$, and the value associated with the limiting power measurement uncertainty is noted on Table 2.

The net pump heat addition uncertainty is derived from the combination of the uncertainties for the primary system net heat losses and the reactor coolant pump heat addition, and are summarized as follows:

= Net Heat input to RCS +12.0 MWt (difference between rated reactor power and rated NSSS power)

The uncertainty on system heat losses, which is essentially all due to charging and letdown flows, has been estimated to be [ ]$^{+a,c}$ of the calculated value. Since direct measurements are not possible, the uncertainty on component conduction and convection losses has been assumed to be [ ]$^{+a,c}$ of the calculated value. Reactor coolant pump hydraulics are known to a relatively high confidence level, supported by system hydraulics tests performed at Prairie Island Unit 2 and by input power measurements from several plants. Therefore, the uncertainty for the pump heat addition is estimated to be [ ]$^{+a,c}$ of the best estimate value. Considering these parameters as one quantity, which is designated the net pump heat addition uncertainty, the combined uncertainties are less than [ ]$^{+a,c}$ of the total, which is [ ]$^{+a,c}$ of core power.

Parameter dependent effects are identified on Table 3. Westinghouse has determined the dependent sets in the calculation and the direction of interaction, i.e., whether components in a dependent set are additive or subtractive with respect to a conservative calculation of power. The same work was performed for the instrument bias values. As a result, the calculation explicitly accounts for dependent effects and biases with credit taken for sign (or direction of effect).

Using the power uncertainty values noted in Table 3, the 4 loop uncertainty equation (with biases) is as follows:

[ ]$^{+a,c}$

Based on four (4) loops, and the instrument uncertainties for the measured parameters, the power uncertainty is:

# of loops	power measurement uncertainty (% power)
4	±0.65

TABLE 1

POWER MEASUREMENT INSTRUMENTATION UNCERTAINTIES
(USING LEFM ON FEEDWATER HEADER)
FOUR LOOP OPERATION

(% SPAN)		FW TEMP	FW PRESS	FW FLOW (header)	STM PRES	SG BLOWDOWN FLOW
LEFM	=			0.482%flow		
FLOW ORIFICE	=					10 gpm
CSA	=	*	5.00	0.482%flow	4.02	55 gpm
BIAS	=	0.00	0.00	0.000%flow	0.00	0 gpm

Channel uncertainty and bias values have been determined by TVA.

NUMBER OF INSTRUMENTS USED

	1/	1/	1/	1/	1/
	HEADER	LOOP	HEADER	LOOP	HEADER
	°F	psi	%flow	psi	% flow
INST SPAN =		1200		1200	1.0%rated feedwater flow (rfwf)
INST UNC (RANDOM) =	*	60.0	0.482	48.2	0.021% rfwf
INST UNC (BIAS) =	0.0	0.0	0.0	0.0	0.0% rfwf
NOMINAL =	436.3	895 psia	100%flow	795 psia	20-270gpm**

* Effects are included in the LEFM supplied feedwater flow uncertainty.

** The conditions analyzed for steam generator blowdown for the measurement uncertainty are based on a nominal flow of 5.0 to 67.5 gpm per loop, equivalent to a total system steam generator blowdown flow of 20 to 270 gpm. The instrument range is 0-360 gpm.

All parameters are read by the plant computer.

TABLE 2

POWER MEASUREMENT SENSITIVITIES
(USING LEFM ON FEEDWATER HEADER)
FOUR LOOP OPERATION

			+g
FEEDWATER DENSITY			
TEMPERATURE	=		
PRESSURE	=		
			+a,c
FEEDWATER ENTHALPY			
TEMPERATURE	=		
PRESSURE	=		
h_s	=		
h_f	=		
$\Delta h(SG)$	=		
STEAM ENTHALPY			
PRESSURE	=		
MOISTURE	=		
S.G. BLOWDOWN FLOW			
F_a			
TEMPERATURE	=		
MATERIAL	=		
DENSITY			
PRESSURE	=		
ΔP	=		
S .G. BLOWDOWN ENTHALPY			
PRESSURE	=		

* Supplied by TVA.
** Incorporated into the feedwater flow uncertainty supplied by TVA.

TABLE 3

POWER MEASUREMENT UNCERTAINTIES
(USING LEFM ON FEEDWATER HEADER)
FOUR LOOP OPERATION

COMPONENT		INSTRUMENT UNCERTAINTY	POWER UNCERTAINTY (% POWER)
FEEDWATER FLOW (HEADER)			[ ]+a,c
LEFM		0.482 % flow	
FEEDWATER DENSITY			
TEMPERATURE		***	
PRESSURE		60.0 psi #	
FEEDWATER ENTHALPY			
TEMPERATURE		***	
PRESSURE		60.0 psi #	
STEAM ENTHALPY			
PRESSURE		48.2 psi #	
MOISTURE		0.45 %Moisture	
NET PUMP HEAT ADDITION		20.0 %	
STEAM GENERATOR BLOWDOWN HEADER FLOW			
ORIFICE (FLOW COEFFICIENT)		2.8 % flow	
THERMAL EXPANSION COEFFICIENT			
TEMPERATURE		3.5 °F (PRESS. EQUIV)	
MATERIAL		5.0 %	
DENSITY			
PRESSURE		48.2 psi #	
DELTA P		0.021 % flow	
STEAM GENERATOR BLOWDOWN ENTHALPY			
PRESSURE		48.2 psi #	
BIAS VALUES			
STEAM PRESSURE	ENTHALPY		
SG BLOWDOWN	LIQUID ENTHALPY		
SG BLOWDOWN	LIQUID DENSITY		
POWER BIAS	TOTAL VALUE		
4 LOOP UNCERTAINTY	(WITHOUT BIAS VALUES)		
4 LOOP UNCERTAINTY	(WITH BIAS VALUES)		

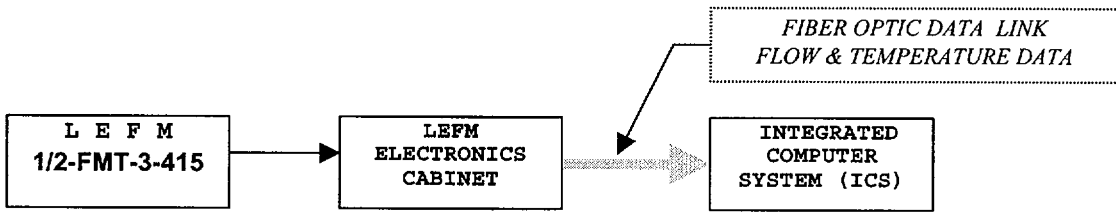
*, ** Indicate sets of dependent parameters.

*** Effects are included in the feedwater flow uncertainty provided by TVA.

These values are single loop values. These parameters are averaged over the 4 loops of the plant.

Table 4
INPUTS REQUIRED FOR LEFM POWER CALORIMETRIC MEASUREMENT CALCULATION
SEQUOYAH NUCLEAR PLANT UNITS 1 & 2

Main Feedwater Flow Leading Edge Flow Meter (total of 1)(LEFM check System 2000FC)

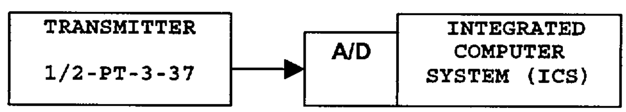


1/2- FMT-3-415

Feedwater Temperature (included in the Leading Edge Flow Meter)

Feedwater Pressure Transmitter (total of 4) (Foxboro E11GM transmitter):

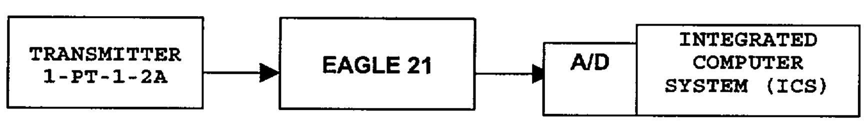
0 - 1200 psig



1/2-PT-3-37	1/2-PT-3-92
1/2-PT-3-50	1/2-PT-3-105

Steam Pressure Transmitter (total of 12) (Foxboro N-E11GM transmitters, Foxboro E11GM transmitters):

0 - 1200 psig



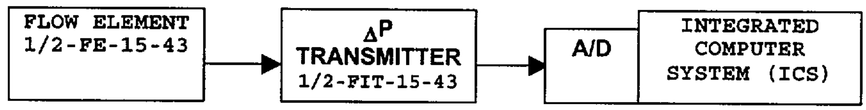
1/2-PT-1-2A	(Loop 1)	[Foxboro N-E11GM(U1&2)]
1/2-PT-1-2B	(Loop 1)	[Foxboro E11GM(U1) & N-E11GM(U2)]
1/2-PT-1-5	(Loop 1)	[Foxboro E11GM(U1&2)]
1/2-PT-1-9A	(Loop 2)	[Foxboro E11GM(U1&2)]
1/2-PT-1-9B	(Loop 2)	[Foxboro E11GM(U1&2)]
1/2-PT-1-12	(Loop 2)	[Foxboro E11GM(U1&2)]
1/2-PT-1-20A	(Loop 3)	[Foxboro E11GM(U1&2)]
1/2-PT-1-20B	(Loop 3)	[Foxboro E11GM(U1&2)]
1/2-PT-1-23	(Loop 3)	[Foxboro E11GM(U1&2)]
1/2-PT-1-27A	(Loop 4)	[Foxboro N-E11GM(U1&2)]
1/2-PT-1-27B	(Loop 4)	[Foxboro E11GM(U1) & N-E11GM(U2)]
1/2-PT-1-30	(Loop 4)	[Foxboro E11GM(U1&2)]

Table 4 (continued)

Steam Generator Blowdown Flow transmitter

PRIMARY INPUT **Header Flow** (total of 1)(ROSEMOUNT TRANSMITTER MODEL 1151DP):

0 - 360 gpm



1/2-FE-15-43 1/2-FIT-15-43

IV. CONCLUSIONS

The preceding sections provide the methodology to account for the power measurement uncertainty. The uncertainty calculation has been performed for Sequoyah with plant-specific instrumentation and calibration procedures. The following table summarizes the result and the uncertainty that is used in the Sequoyah safety analysis.

Parameter	Calculated Uncertainty	Uncertainty Used in Safety Analysis
Power	±0.7% RTP (random)	±2.0% RTP (random)

REFERENCES

1. NRC letter, S. A. Varga to J. Dolan, Indiana and Michigan Electric Company, dated 2/12/81.

2. NUREG-0717 Supplement No. 4, Safety Evaluation Report related to the operation of Virgil C. Summer Nuclear Station Unit No. 1, Docket 50-395, August, 1982.

3. Regulatory Guide 1.105 Rev. 2, "Instrument Setpoints for Safety-Related Systems", dated 2/86.

4. NUREG/CR-3659 (PNL-4973), "A Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors", 2/85.

5. ANSI/ANS Standard 58.4-1979, "Criteria for Technical Specifications for Nuclear Power Stations".

6. ISA Standard S67.04, Part 1, 1994, "Setpoints for Nuclear Safety-Related Instrumentation".

7. TVA Letter N9830, D.L. Lundy to J.W. Irons, dated May 29, 2001.

8. TVA Letter N9850, D.L. Lundy to J.W. Irons, dated June 28, 2001.

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

WCAP-15726, REVISION 0
SEQUOYAH UNITS 1 AND 2
1.3-PERCENT POWER UPRATE PROGRAM LICENSING REPORT

Westinghouse Non-Proprietary Class 3

WCAP-15726

Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program Licensing Report

Westinghouse Electric Company LLC

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15726

Sequoyah Units 1 and 2
1.3-Percent Power Uprate Program
Licensing Report

November 2001

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

LIST OF TABLES

ACRONYMS

AFD	axial flux difference
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
ANS	American Nuclear Society
ARC	alternate repair criteria
ART	adjusted reference temperature
ARV	atmospheric relief valve
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
BOP	balance of plant
BPR	burnable poison rod
C&FS	condensate and feedwater system
CCS	component cooling system
CF	chemistry factor
CHF	critical heat flux
COLR	Core Operating Limit Report
COMS	cold overpressure mitigation system
CRDM	control rod drive mechanism
CSL	core safety limit
CST	condensate storage tank
CVCS	chemical and volume control system
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EFPD	effective full-power day
EFPY	effective full-power year
EGTS	emergency gas treatment system
EOL	end of license
ERG	Emergency Response Guideline
ESF	engineered safety feature
FF	fluence factor
FIV	flow-induced vibration
FRA-ANP	Framatome ANP
FSAR	Final Safety Analysis Report
HFP	hot full power
HZP	hot zero power

ACRONYMS (Cont'd)

IR	inner radius
LBB	leak before break
LEFM	Leading Edge Flow Meter
LOCA	loss-of-coolant accident
LOEL	loss of external electrical load
LOOP	loss-of-offsite power
MAP	maximum allowable peaking
MCO	moisture carryover
MDNBR	minimum departure from nucleate boiling ratio
MFIV	main feedwater isolation valve
MFRV	main feedwater regulator valve
MSIV	main steam isolation valve
MSS	main steam system
MSSV	main steam safety valve
NDR	Nuclear Design Report
NIS	nuclear instrumentation system
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OBE	Operating basis earthquake
ODSCC	outside diameter stress corrosion cracking
OPΔP	overpower delta T
OTΔT	overtemperature delta T
P&I	proportional and integral
PCT	peak cladding temperature
PORV	power-operated relief valve
PRT	pressurizer relief tank
P-T	pressure-temperature
PTLR	Pressure-Temperature Limit Report
PTS	pressurized thermal shock
PWR	pressurized water reactor
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RHRS	residual heat removal system
RSER	Reload Safety Evaluation Report
RSR	relative stability ratio

ACRONYMS (Cont'd)

RTDP	Revised Thermal Design Procedure
RTP	rated thermal power
RTS	reactor trip system
RV	reactor vessel
RWAP	rod withdrawal at power
RWST	refueling water storage tank
SC	surveillance capsule
SCD	statistical core design
SDL	statistical design limit
SFP	spent fuel pit
SFPCS	spent fuel pit cooling system
SG	steam generator
SGBS	steam generator blowdown system
SGTP	steam generator tube plugging
SI	safety injection
SIS	safety injection system
SLB	steam line break
SR	surveillance requirement
SRP	Standard Review Plan
SSE	safe shutdown earthquake
TDF	thermal design flow
TDL	thermal design limit
TMD	transient mass distribution
TMI	Three Mile Island
TPBAR	tritium-producing burnable absorber rods
TRM	Technical Requirements Manual
TSP	tube support plate
TTD	trip-time delay
TU	Texas Utilities
TVA	Tennessee Valley Authority
USAS	United States of American Standard
USE	upper shelf energy
V5H	Vantage 5 Hybrid
VCT	volume control tank

1.0 INTRODUCTION

1.1 BACKGROUND

The Sequoyah Units 1 and 2 are presently licensed for a full core power rating of 3411 MWt. Through the use of more accurate feedwater flow measurement equipment, the Tennessee Valley Authority (TVA) is seeking approval to increase this core power by 1.3-percent, to 3455 MWt. Westinghouse evaluated the effect of a 1.3-percent core power uprate on the nuclear steam supply system (NSSS) systems, components, and safety analyses.

In addition, Framatome ANP (FRA-ANP) evaluated the effect of the 1.3-percent uprate on the Final Safety Analysis Report (FSAR) accident analyses related to fuel and the Chapter 15 accident analyses. FRA-ANP holds the current fuel design for Sequoyah Units 1 and 2.

This document summarizes these evaluations and analyses for use in the Sequoyah Nuclear Plant licensing documentation. This document provides a brief description of the evaluations and analyses performed and states the appropriate conclusions. This document also provides information to support answers to the three questions required for the Code of Federal Regulations (CFR) 10 CFR 50.92 submittal.

1.2 APPROACH

The Sequoyah Units 1 and 2 Power Uprate Program was completed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," issued in 1983. Since its submittal to the Nuclear Regulatory Commission (NRC), the methodology has been successfully used as the basis for power uprate projects on over 20 pressurized water reactor (PWR) units, including Diablo Canyon Units 1 and 2, Turkey Point Units 3 and 4, Comanche Peak Unit 2, and Watts Bar Unit 1.

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

For Sequoyah Units 1 and 2, Westinghouse and FRA-ANP completed a comprehensive engineering review program that is consistent with the WCAP-10263 methodology to increase the licensed core power from 3411 MWt to 3455 MWt. The results of this review are summarized in this report, as follows:

- Section 2.1 discusses the revised NSSS-design thermal and hydraulic parameters that were modified as a result of the 1.3-percent uprate and that serve as the basis for all of the NSSS analyses and evaluations.

- Section 2.2 concludes that no design transient modifications are required to accommodate the revised NSSS design conditions.

- Section 2.3 presents the systems (e.g., safety injection system (SIS), residual heat removal system (RHRS), and control systems) evaluations completed for the revised design conditions.

- Section 2.4 presents the components (e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), steam generator (SG), and NSSS auxiliary equipment) evaluations completed for the revised design conditions.

- Section 2.5 provides the evaluations of the accident analyses and evaluations performed for the loss-of-coolant accident (LOCA) and main steam line break mass and energy releases.

- Section 3.0 provides the evaluations of the fuel and the transient and accident analyses completed by FRA-ANP.

- Appendix A provides the calorimetric uncertainty calculations.

- Appendix B provides input to support the 10 CFR 50.92 evaluation, in addition to any associated changes to the Technical Specifications.

The results of these analyses and evaluations demonstrate that all acceptance criteria continue to be met.

1.3 GENERAL LICENSING APPROACH FOR PLANT ANALYSES USING PLANT POWER LEVEL

Most plant safety, component, and system analyses use the reactor and/or NSSS thermal power as inputs. These NSSS analyses generally model the core and/or NSSS thermal power in one of four ways, as described in the following paragraphs.

First, some analyses apply a 2-percent increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the 1.3-percent uprate conditions because the sum of increased core power level (1.3 percent) and the decreased power measurement uncertainty (less than 0.7 percent) falls within the previously analyzed conditions.

The power calorimetric uncertainty calculation described in Appendix A indicates that, with the Caldon Leading Edge Flow Meter (LEFM) installed, the power measurement uncertainty (based on a 95-percent probability, at a high confidence interval) is less than 0.7 percent. Thus, these analyses only need to reflect a 0.7-percent power measurement uncertainty. Accordingly, the existing 2-percent uncertainty can be allocated such that 1.3 percent is applied to provide sufficient margin to address the uprate to 3455 MWt, and 0.7 percent is retained in the analysis to still account for the power measurement uncertainty. In addition, for these types of analyses, it is shown that they still employ other conservative assumptions not affected by the 1.3-percent uprated power. Taken together, the use of the calculated 95/high confidence power measurement uncertainty and retention of conservative assumptions indicate that the margin of safety for these analyses would not be reduced.

Second, some analyses employ a nominal power level. These analyses have either been evaluated or re-performed for the 1.3-percent increased power level. The RHRS cooldown analysis is the only analysis that was re-performed. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.3-percent uprate conditions.

Third, some of the analyses already employ a core power level in excess of the proposed 3455 MWt. These analyses were previously performed at a higher power level as part of prior plant programs. For these analyses, some of this available margin has been used to offset the 1.3-percent uprate. Consequently, the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the 1.3-percent uprate.

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

1.4 TECHNICAL SPECIFICATION CHANGES

The Technical Specification changes are included in Appendix B.

2.0 RESULTS

This section summarizes the NSSS evaluations performed by Westinghouse for the uprating of Sequoyah Units 1 and 2. These evaluations incorporated an increase in licensed core power from 3411 MWt to 3455 MWt.

2.1 NSSS PERFORMANCE PARAMETERS

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. These parameters provide the reactor coolant system (RCS) and secondary system conditions (steam generator temperatures, pressures, and flows) at the selected vessel average temperatures (T_{avg}), steam generator tube plugging (SGTP) levels, NSSS power levels, and RCS flowrates.

Due to the 1.3-percent increase in licensed core power from 3411 MWt to 3455 MWt, it was necessary to revise these parameters. The new parameters are identified in Table 2.1-1 and are incorporated, as required, into the applicable NSSS system and component evaluations and into the safety analyses performed in support of the uprate.

NSSS design parameters are based on conservative inputs, such as a conservatively low thermal design flow (TDF) and bounding SGTP levels, which yield primary- and secondary-side conditions that bound the way the plant operates. The TDF is the conservatively low RCS flow value generally used in the safety analyses.

An increased NSSS power level of 3467 MWt (3455 MWt core power) is the only input assumption that changed from the current licensing basis.

Table 2.1-1 provides the NSSS design parameter cases generated and used as the basis for the 1.3-percent power uprate. The 1.3-percent uprate resulted in changes to some of the NSSS design parameters, compared to the parameters that form the current licensing basis. The changes include the following RCS temperatures:

- Reactor vessel (RV) outlet (T_{hot}) increased by 0.4°F
- RV inlet (T_{cold}) decreased by 0.4°F

These small changes occurred because the vessel average temperature (T_{avg}) was maintained at the current design value (578.2°F), while the core power was increased by 44 MWt to 3455 MWt. The temperature changes reflect the additional heat input from the uprated core.

In addition, the 1.3-percent uprate resulted in the following changes to the secondary-side parameters at 15-percent SGTP:

- Steam temperature decreased by 1.0°F
- Steam pressure decreased by 7 psi
- Steam mass flow increased by 1.5 percent

These small changes are based on a calculation of the steam generator and secondary-side performance, resulting from the increased power.

The various Westinghouse analyses used the two cases of NSSS design parameters shown in Table 2.1-1 in the evaluation of the effects of the 1.3-percent power uprate on Sequoyah Units 1 and 2.

Table 2.1-1 NSSS Performance Parameters

Parameters	Current	Uprate
NSSS Power (MWt)	3423	3467
Reactor Power (MWt)	3411	3455
Thermal Design Flow (gpm/loop)	87,000(1)	87,000(1)
Thermal Design Flow (total gpm)	348,000	348,000
Minimum Measured Flow (total gpm)	360,200(2)	360,200(2)
RCS Temperatures (°F)		
Core Outlet	616.0	616.4
Vessel Outlet (T_{hot})	611.2	611.6
Core Average	582.4	582.5
Vessel Average (T_{avg})	578.2	578.2
Vessel/Core Inlet (T_{cold})	545.2	544.8
Steam Generator Outlet (T_{SGout})	544.9	544.5
Zero Load Temperature	547	547
Reactor Coolant Pressure (psia)	2250	2250
Core Bypass (%)	7.5	7.5
Steam Generator		
Steam Pressure (psia)	802	795
Steam Temperature (T_{steam}) (°F)	518.5	517.5
Steam Flow (Total, 10^6 lb/hr)	14.89	15.12
Feed Temperature (°F)	434.6	436.3
Tube Plugging (%)	15	15

Notes:

(1) TDF accommodates 15% SGTP.

(2) Reflects Technical Specification flow measurement uncertainty of 3.5%.

2.2 DESIGN TRANSIENTS

2.2.1 NSSS Design Transients

The revised design conditions in Table 2.1-1 and the NSSS design transients applicable to the uprated conditions serve as primary inputs to the evaluation and analysis of the NSSS systems and components. Current primary- and secondary-side design transients were reviewed in order to determine their continued applicability for the revised design conditions.

Primary-Side Transients

The review of the primary-side design conditions listed in Table 2.1-1 indicates that the full-power values of vessel outlet and vessel inlet (hot leg and cold leg) vary by 0.4°F from the previously applicable design values. Also, the vessel average temperature was not changed. Given the conservative assumptions used to develop the current design transients (e.g., initial conditions, unavailability of control systems during certain transients), a 0.4°F change in primary-side full-power temperatures is considered insignificant during all transient conditions. Therefore, the revised conditions have negligible impact on the primary-side design transients, and the previously applicable NSSS design transients for the primary side continue to apply, without modification, at the revised design conditions.

Secondary-Side Transients

With regard to secondary-side design parameters, the revised design conditions in Table 2.1-1 indicate that the plant may operate with slightly lower full-power values for steam temperature and steam pressure and slightly higher values for feedwater temperature. Lower nominal steam temperatures (e.g., from 522.8 to 517.5°F) and pressures (e.g., from 832 to 795 psia) result in relatively small changes from initial conditions than those currently reflected in the current NSSS design transients. Similarly, a 1.7°F increase in feedwater temperature (i.e., 434.6 to 436.3°F) is insignificant in comparison to the analyzed feedwater temperature transients.

The small variations in these parameters were either shown to be enveloped by the existing transient curves or encompassed by the conservative assumptions used to develop the design transients. Therefore, it was determined that the existing secondary-side transients remained valid for the 1.3-percent uprate conditions.

2.2.2 Auxiliary Equipment Design Transients

The review of the NSSS auxiliary equipment design transients was based on a comparison between the revised operating conditions in Table 2.1-1 and the parameters that make up the current auxiliary equipment design transients. A review of the current auxiliary equipment transients determined that the only transients potentially affected by the revised conditions are those temperature transients affected by full-load NSSS operating temperatures, namely T_{hot} and T_{cold}. These transients are currently based on an assumed full-load NSSS worst case T_{hot} of 630°F and worst case T_{cold} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

A comparison of the limiting operating values for T_{hot} and T_{cold} of 611.6°F and 544.8°F, respectively, with the existing values indicates that they are still within the design. Therefore, the 1.3-percent uprate does not require any changes to these transients.

2.3 NSSS SYSTEMS

This section presents the results of the evaluations and analyses performed in the NSSS systems area to support the revised design conditions in Table 2.1-1. The systems addressed in this section include fluid systems and NSSS/BOP interface systems. The results and conclusions of each analysis are presented within each subsection.

2.3.1 NSSS Fluid Systems

2.3.1.1 Reactor Coolant System

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

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

It was demonstrated that the minimum required pressurizer spray flow of 800 gpm can be achieved for the 1.3-percent uprate conditions defined in Table 2.1-1. Also, the maximum expected T_{hot} at the revised design conditions is 611.6°F. This temperature is well below the RCS loop design temperature of 650°F.

With respect to the PRT discharge analysis, the nominal full-load pressurizer steam volume is essentially unaffected by the uprate since the RCS average temperature of 578.2°F has not changed. Therefore, the existing discharge analysis is unaffected.

2.3.1.2 Chemical and Volume Control System (CVCS)

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

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

With regards to the CVCS thermal performance, the T_{cold} of 544.8°F is still lower than the design system inlet temperature of 560°F. Also, it is much lower than the shell-side design temperature of 650°F for the regenerative heat exchanger. The excess letdown path is used to process excess effluents associated with fluid expansion during plant heatup and, therefore, is unaffected by the revised T_{cold} at full-power conditions. If operated during power conditions, the excess letdown heat exchanger outlet flow is throttled to maintain the desired outlet temperature and efflux. Therefore, operation of the CVCS is unaffected by the temperature change.

2.3.1.3 Safety Injection System

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

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

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

2.3.1.4 Residual Heat Removal System

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

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

A normal cooldown analysis and an Appendix R cooldown analysis were performed to address the uprated reactor power (3455 MWt). The increased reactor power causes an increase in the decay heat load. Normal cooldown is defined as cooldown assuming all equipment available. Appendix R cooldown requires that the plant be able to be cooled to 140°F in less than 72 hours using specific requirements associated with Appendix R.

The results of the normal cooldown show that the plant can be cooled to 140°F within 33 hours using limiting conditions.

The results of the Appendix R cooldown show that the plant can be cooled to 140°F within 72 hours.

The ability of the RHRS to perform its design functions is unaffected by the uprated conditions.

2.3.1.5 Spent Fuel Pit Cooling System

The function of the spent fuel pit cooling system (SFPCS) is to remove decay heat from spent fuel stored in the spent fuel pit. The SFPCS is comprised of two cooling trains, each containing a pump and a heat exchanger. The pump circulates spent fuel pit (SFP) fluid through a heat exchanger where SFP heat can be transferred to the component cooling system (CCS).

The SFPCS is not connected to the RCS and its operation is totally independent of the RCS. Therefore, the revised plant conditions have no effect on the operation of the SFPCS and the design basis analyses remain bounding and conservative for the revised plant conditions. However, the uprate conditions result in slightly higher fuel decay heat rates. This will have a minimal impact on the total SFPCS decay heat load and will not challenge the system cooling capabilities.

2.3.1.6 Control Systems Response

As part of the Sequoyah 1.3-percent uprating, an analysis was performed on the plant operability and margin to trip for normal operability transients. These include such transients as 5-percent unit loading and unloading, 10-percent step-load increase or decrease, and large-load rejection. The transient chosen for analysis was the large-load rejection. This was a step-load decrease from 100-percent to 50-percent turbine load with all control systems operable. For conservatism, the beginning of core cycle life conditions were chosen for analysis.

The results showed a very smooth transient response with no oscillatory or diverging parameter responses. No reactor trip setpoint was approached during the transient. The overtemperature (OTΔT) setpoint is the limiting reactor trip setpoint noted during this transient. In this analysis, the margin to the trip setpoint was approximately 10-percent.

Based on these results, no plant operability concerns are expected for the 1.3-percent uprating. The Sequoyah plants can accommodate the design basis operability transients, without challenging the reactor protection system setpoints or suffering oscillatory control system response, assuming that no control system failures or abnormal operation occur and all control systems are operating in automatic control.

The current cold overpressure mitigation system (COMS) setpoints were also evaluated for the effect of the power uprate. Since the Appendix G heatup and cooldown curves do not change, the COMS setpoints also remain applicable.

2.3.2 NSSS/BOP Fluid Systems Interfaces

2.3.2.1 Introduction and Background

As part of the Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program, the following BOP fluid systems were evaluated to assess compliance with applicable NSSS/BOP interface guidelines at the revised design conditions in Table 2.1-1:

- Main steam system
- Steam dump system
- Condensate and feedwater system
- Auxiliary feedwater system
- Steam generator blowdown system

2.3.2.2 Main Steam System

The following summarizes the Westinghouse evaluation of the major steam system components relative to the power uprate conditions. The major components of the MSS are the steam generator main steam safety valves (MSSVs), the steam generator power-operated atmospheric relief valves (ARVs), and the main steam isolation valves (MSIVs).

2.3.2.2.1 Steam Generator Main Steam Safety Valves

The setpoints of the MSSVs are based on the design pressure of the steam generators (1085 psig) and the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the steam generators is not changed with the power uprate, there is no need to revise the setpoints of the safety valves.

The MSSVs must have sufficient capacity so that 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. Each Sequoyah unit has 20 safety valves with a total capacity of 16.105×10^6 lb/hr, which provides 106.4 percent of the maximum uprated full-load steam flow of 15.14×10^6 lb/hr (based on 0-percent SGTP). Therefore, based on the range of NSSS parameters for the uprating, the capacity of the installed MSSVs meets the Westinghouse sizing criterion.

2.3.2.2.2 Steam Generator Power-Operated Atmospheric Relief Valves

The primary function of the ARVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the ARVs, in conjunction with the auxiliary feedwater system (AFWS), permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs, (with a ±3-percent tolerance) to the point where the RHRS can be placed in service. During cooldown, the ARVs are either automatically or manually controlled. In automatic, each ARV proportional and integral (P&I) controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

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

Each steam generator ARV is required to have a capacity at least equal to 64,000 lb/hr at 100-psia inlet pressure. At maximum calculated power, this capacity permits a plant cooldown rate of 50°F/hr to RHRS operating conditions, assuming a minimum of two hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFWS. This is based on one train of auxiliary feedwater (AFW) operating and flow going through two steam generators. An evaluation of the installed capacity (69,413 lb/hr per valve, at 100 psia inlet pressure) indicates that the original design-basis cooldown capability can still be achieved over the full range of NSSS design parameters for the 1.3-percent power uprate.

2.3.2.2.3 Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs and ARVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and the containment pressure to within acceptable limits, following a main steam line break. To accomplish this function, the design requirements specify that the MSIVs must be capable of closure, within 5 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 MSS piping and piping supports, in the area of the MSIVs. The worst cases for differential pressure increase and thrust loads are controlled by the effects of the steam line break area (i.e., mass flowrate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and the no-load operating pressure are not affected by the uprating, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change. Consequently, the power uprate has no significant effect on the interface requirements for the MSIVs.

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

2.3.2.3 Steam Dump System

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The Westinghouse sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow, at full-load steam pressure, to permit the NSSS to withstand an external load reduction of up to 50 percent of plant-rated electrical load, without a reactor trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the RCS, which accommodates 10 percent of the load reduction. A steam dump capacity of 40 percent of rated steam flow, at full-load steam pressure, also prevents MSSV lifting following a reactor trip from full power. Each Sequoyah unit is provided with 12 condenser steam dump valves. Each valve is specified to have a flow capacity of 5.53×10^5 lb/hr, at a valve inlet pressure of 796.7 psia. This total capacity provides a steam dump capability of about 41.5-percent times 15.12×10^6 lb/hr (rated steam flow at uprated conditions), or 6.27×10^6 lb/hr, at a full-load steam pressure equal to 795 psia. These operating conditions are based on an NSSS power level of 3467 MWt, an assumed SGTP level of 15-percent, and a T_{avg} (578.2°F) equal to the original design value. Therefore, the condenser steam dump valves are adequate for the NSSS operating conditions proposed for the 1.3-percent power uprate.

2.3.2.4 Condensate and Feedwater System

The condensate and feedwater system (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The range of NSSS parameters will result in a required feedwater volumetric flow increase and the major components of the C&FS are the main feedwater isolation valves (MFIVs), the main feedwater regulator valves (MFRVs), and the C&FS pumps.

2.3.2.4.1 Main Feedwater Isolation Valves/Main Feedwater Regulator Valves

The MFRVs are located outside containment and upstream of the MFIVs. The valves function, in conjunction with the primary isolation signals to the MFIVs 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 RCS cooldowns. To accomplish this function, the MFRVs and the backup MFIVs must be capable of closure within 7.0 seconds and 13 seconds, respectively, after receipt of a closure signal under all operating and accident conditions. This includes a maximum flow condition with all main feedwater pumps delivering to one steam generator.

The quick-closure requirements imposed on the MFRVs and the backup MFIVs 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 line 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 affected by the uprating, the design loads and associated stresses resulting from rapid closure of these valves will not change.

2.3.2.4.2 Condensate and Feedwater System Pumps

The C&FS available head, in conjunction with the MFRV characteristics, must provide sufficient margin for feedwater control, to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous steady feedwater flow should be maintained at all loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the MFRVs 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 feedwater heaters, feed flow meter, and steam generator). In addition, adequate margin should be available in the MFRVs at full-load conditions, with the MFRVs fully open. However, based on the Sequoyah MFRV design and the system layout, the present pump speed control program was set to provide a MFRV pressure drop of about 128 psi to achieve about a 50-percent valve lift at full load.

For the range of NSSS design parameters approved for the uprate, the present speed control program results in a negligible change in MFRV pressure drop and a corresponding negligible change in valve lift, at 100-percent power. Therefore, based on the NSSS design parameters approved for the 1.3-percent uprate, 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 7.0 seconds, after receipt of a trip close signal, to mitigate certain transients and accidents. These requirements are still applicable at the uprated conditions.

2.3.2.5 Auxiliary Feedwater System

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The system provides feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations and also functions as an engineered safeguards system. In the latter function, the AFWS is required to prevent core damage and system overpressurization during transients and accidents, such as a loss of normal feedwater or a secondary-system pipe break. The minimum flow requirements of the AFWS are dictated by accident analyses; and since the uprating affects safety analyses performed at the nominal 100-percent power rating, evaluations were performed (see Section 3.0) to confirm that the AFWS performance is acceptable at the uprated conditions.

2.3.2.5.1 Auxiliary Feedwater Storage Requirements

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

The limiting transient, with respect to CST inventory requirements, is the loss-of-offsite power (LOOP) transient. In the event of a LOOP, sufficient CST useable inventory must be available to bring the unit from full power to hot standby conditions, maintain the plant at hot standby for two hours, and then cool down the RCS to the RHRS cut-in temperature (350°F) in four hours. In light of these design basis requirements, the Sequoyah Units 1 and 2 Analysis of Record concluded that the tank should be designed to accommodate a minimum inventory of 190,000 gals. The minimum CST inventory of 190,000 gals is based on reactor trip from 102 percent of the original rated reactor power, or 3479.2 MWt. Since the proposed power uprate is based on improved calorimetric error, no change in the minimum CST volume is required for operation at the uprated power level.

2.3.2.6 Steam Generator Blowdown System

The steam generator blowdown system (SGBS) controls the chemical composition of the steam generator secondary-side water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator secondary side.

The blowdown flowrates required during plant operation are based on chemistry control and tubesheet sweep requirements to control the buildup of solids. The blowdown flowrate required to control chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant service water, allowable primary-to-secondary leakage, and the performance of the condensate polishers. Since these variables are not affected by power uprate, the blowdown required to control secondary chemistry and steam generator solids is not be affected by power uprate.

Based on the revised range of NSSS design parameters for power uprate, the no-load steam pressure (1020 psia) remains the same and the minimum full-load steam pressure (795 psia) decreases about 7 psi or less than 1-percent. This small decrease in blowdown system inlet pressure will not significantly affect the required maximum lift of the blowdown flow control valves. Therefore, the range of design parameters approved for power uprate is not expected to affect blowdown flow capability.

2.4 NSSS COMPONENTS

2.4.1 Reactor Vessel

2.4.1.1 Reactor Vessel Structural Evaluation

The evaluation assessed the effects of the revised operating parameters (see Table 2.1-1) on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress report and addenda. The design transients are not modified as a result of the revised parameters. However, the normal vessel outlet temperature increases from 611.2°F to 611.6°F with the 1.3-Percent Power Uprate Program. This increases the T_{hot} variation in the outlet nozzles during normal plant loading and plant unloading. Also, the normal vessel inlet temperature decreases from 545.2°F to 544.8°F, slightly increasing the temperature change in all the other vessel regions during normal plant loading and plant unloading. Therefore, the normal plant loading and plant unloading are considered to be more severe transients for all of the various vessel regions in the evaluation. The evaluation considers a worst-case set of operating parameters from the current design basis parameters and the 1.3-percent uprate parameters. As a result of these worst-case considerations, the uprate parameter cases and the current design basis parameters are fully covered by the evaluation. Also, reactor vessel operation, in accordance with the Sequoyah 1.3-Percent Power Uprate Program for the remainder of the current operating licenses for Units 1 and 2, is justified.

The Sequoyah Units 1 and 2 reactor vessels were evaluated for the structural and fatigue effects of the Sequoyah 1.3-Percent Power Uprate Program. The evaluation concludes that the reactor vessel structural analyses are not significantly affected by the 1.3-percent uprating. The design inputs related to the 1.3-percent uprate program either exhibit small changes that do not affect the analysis results to a significant degree, or the inputs remain bounded by the parameters previously considered in the reactor vessel stress report. The Sequoyah reactor vessel stress report remains valid with the implementation of the 1.3-Percent Power Uprate Program.

The 0.4°F increase in the vessel outlet temperature from 611.2°F to 611.6°F and the 0.4°F decrease in the vessel inlet temperature from 545.2°F to 544.8°F due to the Sequoyah 1.3-Percent Power Uprate Program have no effect on the maximum ranges of stress intensity and maximum cumulative fatigue usage factors previously reported in the reactor vessel stress report. Furthermore, the existing NSSS design transients from the original design bases and the 1998 Sequoyah 15-Percent SGTP Program remain valid for the 1.3-percent uprating. Finally, the previous LOCA and seismic reactor vessel/reactor internals interface loads remain valid for the 1.3-percent power uprate conditions. The maximum ranges of primary-plus-secondary stress intensity and maximum cumulative fatigue usage factors reported for the Sequoyah reactor vessels are unchanged and continue to satisfy the applicable limits of Section III of the 1968 ASME B&PV Code.

The faulted condition stress analysis for the Sequoyah Units 1 and 2 reactor vessels does not change as a result of the 1.3-Percent Power Uprate Program since no changes in the faulted condition reactor vessel/reactor internals interface loads or other faulted condition loads were identified as a result of the uprating.

2.4.1.2 Reactor Vessel Neutron Exposure Projections

Neutron exposure projections for Sequoyah Units 1 and 2 have been made to reflect a 1.3-percent power uprating. Fluence values for the reactor vessels of Sequoyah Units 1 and 2 were previously evaluated at the end of Cycle 9 and projections were made to operating times of 20, 32, and 48 effective full-power years (EFPY). These projections were made at a rated power level of 3411 MWt. These projections were made using both calculated fluence values and best-estimate fluence values, which take into account the results of dosimetry measurements from four surveillance capsules for each plant. Updated fluence projections have been made to reflect an uprated power of 3455 MWt. The results are presented in Tables 2.4.1.2-1 and 2.4.1.2-2. These fluence projections use fixed numbers of EFPY for the projection and, therefore, do not depend on when the uprating occurs. The projections would all be exactly 1.3-percent higher due to the uprating, except for the difference between the past and future cycles in the fluence per EFPY.

Table 2.4.1.4-1 gives results for the calculated neutron exposures at the vessel inner radius (IR) for Sequoyah Unit 1. Values for neutron fluence (E > 1.0 MeV), neutron fluence (E > 0.1 MeV), and displacements per atom (dpa) are presented in the table. The values at 9.90 EFPY are the values at the end of Cycle 9. The EFPY value has been adjusted, i.e., reduced by 1.3-percent from 10.03 EFPY to 9.90 EFPY, to reflect the redefinition of full power from 3411 MWt to 3455 MWt. Values at 11.18 EFPY are the projection to the end of Cycle 10 based on operation at 3411 MWt and a total power generation equal to Cycle 9. The neutron flux values used for projections beyond Cycle 9 are the average of Cycles 5 to 9, which have similar low-leakage cores. These low-leakage fuel designs are assumed to be used for all future cycles. Other fluence and dpa projections to 20 EFPY, 32 EFPY, and 48 EFPY (extended life) are also given. These projections assume operation at 3455 MWt to produce this amount of thermal power during the life of the plant. The values at 20, 32, and 48 EFPY are close to 1.3-percent higher, reflecting the uprated power.

Table 2.4.1.2-2 provides similar calculated neutron exposure estimates for Sequoyah Unit 2.

Table 2.4.1.2-1 Azimuthal Variations of the Neutron Exposure Projections on the Reactor Vessel Clad/Base Metal Interface at Core Midplane-Sequoyah Unit 1 Uprated Power 3455 MWt – Calculated

Calculated

	0°	**15°**	**30°**	**45°(a)**
9.90 EFPY(b) E>1.0 MeV	2.05E+18	3.21E+18	4.09E+18	6.37E+18
11.18 EFPY(b) E>1.0 MeV	2.29E+18	3.57E+18	4.55E+18	7.08E+18
20 EFPY E>1.0 MeV	3.91E+18	6.04E+18	7.70E+18	1.20E+19
32 EFPY E>1.0 MeV	6.13E+18	9.40E+18	1.20E+19	1.86E+19
48 EFPY E>1.0 MeV	9.07E+18	1.39E+19	1.77E+19	2.75E+19

Notes:

(a) Maximum neutron exposure projection.

(b) 9.90 EFPY is the exposure at the end of Cycle 9 renormalized to full power of 3455 MWt and 11.18 EFPY is the projected exposure at the end of Cycle 10 normalized to 3455 MWt.

Table 2.4.1.2-2	Azimuthal Variations of the Neutron Exposure Projections of the Reactor Vessel Clad/Base Metal Interface at Core Midplane-Sequoyah Unit 2 Uprated Power 3455 MWt – Calculated			
Calculated				
	0°	**15°**	**30°**	**45°[(a)]**
10.41 EFPY[(b)] E>1.0 MeV	2.11E+18	3.36E+18	4.26E+18	6.37E+18
11.78 EFPY[(b)] E>1.0 MeV	2.36E+18	3.75E+18	4.76E+18	7.13E+18
20 EFPY E>1.0 MeV	3.84E+18	6.07E+18	7.82E+18	1.17E+19
32 EFPY E>1.0 MeV	6.00E+18	9.46E+18	1.23E+19	1.85E+19
48 EFPY E>1.0 MeV	8.89E+18	1.40E+19	1.82E+19	2.74E+19

Notes:

(a) Maximum neutron exposure projection.

(b) 10.41 EFPY is the exposure at the end of Cycle 9 renormalized to full power of 3455 MWt and 11.78 EFPY is the projected exposure at the end of Cycle 10 normalized to 3455 MWt.

2.4.1.3 Reactor Vessel Integrity

Reactor vessel integrity is affected by any changes in plant parameters that affect neutron fluence levels or temperature and pressure transients. The changes in neutron fluence resulting from the proposed Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program have been evaluated to determine the effect on reactor vessel integrity. This assessment included a review of the current material surveillance capsule withdrawal schedules, applicability of the plant heatup and cooldown pressure-temperature limit curves, applicability of the plant heatup and cooldown pressure-temperature limit curves, applicability of the Emergency Response Guideline (ERG) limits, the effect on the pressurized thermal shock (RT_{PTS}) values (10CFR50.61, known as the Pressurized Thermal Shock (PTS) Rule), and a review of the updated inlet temperature.

Surveillance Capsule Withdrawal Schedule

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel, to effectively monitor the condition of the reactor vessel materials under actual operating conditions. The American Society for Testing and Materials (ASTM) E185-82 defines the recommended number of surveillance capsules and the recommended withdrawal schedule, based on the vessel material predicted transition temperature shifts (ΔRT_{NDT}). The surveillance capsule withdrawal schedule is in terms of EFPY of plant operation, with a design life of 32 EFPY.

The capsules removed from the Sequoyah Units 1 and 2 vessel to date meet the intent of ASTM E185-82. However, since the revised fluence projections after the power uprating have exceeded the fluence projections used in development of the current withdrawal schedules for Sequoyah Units 1 and 2, then a calculation of ΔRT_{NDT} at 32 EFPY must be performed to determine if the increase fluences alters the number of capsule to be withdrawn for Sequoyah Units 1 and 2. This calculation is documented in Tables 2.4.1.3-1 and 2.4.1.3-2. It shows that the maximum ΔRT_{NDT} using the uprated fluences for Sequoyah Units 1 and 2 at 32 EFPY is 189°F and 122°F, respectively. Per ASTM E185-82, these ΔRT_{NDT} values would require four capsules to be withdrawn from each unit. This is unchanged from the current withdrawal schedule. Therefore, the only update to the current withdrawal schedules would be to the referencing of future fluence values. The updated withdrawal schedules are documented in Tables 2.4.1.3-3 and 2.4.1.3-4.

Heatup and Cooldown Pressure-Temperature Limit Curves

Sequoyah Units 1 and 2 are currently operating to 16 EFPY pressure-temperature (P-T) limit curves per WCAP-12970 (Unit 1) and WCAP-12971 (Unit 2). In 1999, new P-T curves for 32 EFPY were included in a Pressure-Temperature Limit Report (PTLR). It is assumed that the 32 EFPY curves (WCAP-15293 and WCAP-15321) will be used in the near future. However, for completeness, this evaluation will also cover the 16 EFPY curves.

A review was completed of the current heatup and cooldown curve applicability dates for Sequoyah Units 1 and 2. This review indicates that the revised fluence projections after the power uprating have exceeded the fluence projections used in developing the current adjusted reference temperature (ART) values for Sequoyah Unit 1 at 32 EFPY and Sequoyah Unit 2 at 16 and 32 EFPY. Therefore, new applicability dates have been calculated, which are documented in Table 2.4.1.3-5. It is shown in this table that only the 16 EFPY curves for Sequoyah Unit 2 are affected, reducing them to 14.5 EFPY. The change for the other curves is so minimal that the current applicability remains valid.

ERG Limits

For Sequoyah Units 1 and 2, the current peak inside surface RT_{NDT} values at end of license (EOL) and license renewal were calculated to be 231°F and 241°F (Unit 1), and 155°F and 164°F (Unit 2). The limiting material for Sequoyah Unit 1 was the lower shell forging, while the limiting material at Sequoyah Unit 2 was the intermediate shell forging. Comparing these values with the limits provided in Table 2.4.1.3-6 would currently (pre-uprating) put Sequoyah Unit 1 in Category II and Sequoyah Unit 2 in Category I. Even though the revised fluence projections after the power uprating have exceeded the fluence projections used in development of the current peak inside surface RT_{NDT} values at EOL or license renewal, Sequoyah Units 1 and 2 will still remain in the same ERG categories through license renewal, respectively. See Table 2.4.1.3-7 for the revised peak RT_{NDT} (or RT_{PTS}) values.

Pressurized Thermal Shock

The PTS calculations were performed for Sequoyah Units 1 and 2 in WCAP-15293 and WCAP-15321 using the fluence projections documented in WCAP-15224 and WCAP-15320 and the latest procedures specified by the NRC in the PTS Rule. The calculated neutron fluence values for the uprated condition for Sequoyah Units 1 and 2 have exceeded the current fluences. Therefore, to evaluate the effects of the uprating, the PTS values for the most limiting material from each unit will be re-evaluated using the uprated fluences. This evaluation is presented in Table 2.4.1.3-7. Based on this evaluation, all RT_{PTS} values remain below the NRC screening criteria values using the projected uprated fluence values through 32 and 48 EFPY for Sequoyah Units 1 and 2.

Upper Shelf Energy

Based on WCAP-15224 (Unit 1) and WCAP-15320 (Unit 2), all beltline materials are expected to have an upper shelf energy (USE) greater than 50 ft-lb through EOL (32 EFPY) as required by 10 CFR 50, Appendix G. The EOL (32 EFPY) USE was predicted using the EOL 1/4T fluence projection.

The revised fluence projections after the power uprating have exceeded the fluence projections used in developing the predicted EOL USE values. However it has only affected the 1/4T fluence by less than 1 percent. This small amount has no measurable effect on percent decrease in USE. Therefore, the current predicted USE values for Sequoyah Units 1 and 2 remain valid.

Inlet Temperature

Table 2.1-1 indicates that the inlet temperature is 544.8°F. This inlet temperature is within the range of 530°F and 590°F. Therefore, all current analyses remain valid.

Table 2.4.1.3-1 EOL (32 EFPY) ΔRT_{NDT} Values for all Sequoyah Unit 1 Beltline Materials				
Material	**CF(a)**	**f @ 32 EFPY(b)**	**FF(c)**	**ΔRT_{NDT}(d)**
Intermediate Shell Forging 05	115°F	1.86	1.17	135°F
Lower Shell Forging 04	95°F	1.86	1.17	111°F
– Using Surveillance Capsule (S/C) Data	105.85°F	1.86	1.17	124°F
Intermediate to Lower Shell Circumferential Weld Seam W05	161.3°F	1.86	1.17	189°F
– Using S/C Data	135.0°F	1.86	1.17	158°F

Notes:

(a) The chemistry factors (CFs) are from WCAP-15293.

(b) f @ 32 EFPY is the 32 EFPY uprated fluence at the clad/base metal interface ($x\ 10^{19}$ n/cm^2, E > 1.0 MeV).

(c) Fluence factor (FF) = $f^{(0.28 - 0.1 \log f)}$, where f is the clad/base metal interface fluence.

(d) ΔRT_{NDT} = CF * FF

Table 2.4.1.3-2 EOL (32 EFPY) ΔRT_{NDT} Values for all Sequoyah Unit 2 Beltline Materials				
Material	**CF(a)**	**f @ 32 EFPY(b)**	**FF(c)**	**ΔRT_{NDT}(d)**
Intermediate Shell Forging 05	95°F	1.85	1.169	111°F
– Using S/C Data	89.7°F	1.85	1.169	105°F
Lower Shell Forging 04	104°F	1.85	1.169	122°F
Inter. to Lower Shell Circumferential Weld	63°F	1.85	1.169	74°F
– Using S/C Data	77.8°F	1.85	1.169	91°F

Notes:

(a) The chemistry factors are from WCAP-15321.

(b) f @ 32 EFPY is the 32 EFPY fluence at the clad/base metal interface ($x\ 10^{19}$ n/cm^2, E > 1.0 MeV).

(c) Fluence factor (FF) = $f^{(0.28 - 0.1 \log f)}$, where f is the clad/base metal interface fluence.

(d) ΔRT_{NDT} = CF * FF

Table 2.4.1.3-3 Recommended Surveillance Capsule Withdrawal Schedule for Sequoyah Unit 1				
Capsule	**Capsule Location**	**Lead Factor [a]**	**Withdrawal EFPY [b]**	**Fluence (n/cm^2) [a]**
T	40°	3.39	1.03	2.61×10^{18} [c]
U	140°	3.47	3.00	7.96×10^{18} [c]
X	220°	3.47	5.27	1.32×10^{19} [c]
Y	320°	3.43	10.03	2.19×10^{19} [c, d]
S	4°	1.08	Standby	[e]
V	176°	1.08	Standby	[e]
W	184°	1.08	Standby	[e]
Z	356°	1.08	Standby	[e]

Notes:

(a) Updated in Capsule Y dosimetry analysis.

(b) EFPY from plant startup.

(c) Plant-specific evaluation.

(d) This fluence is not less than once or greater than twice the peak EOL fluence.

(e) Capsules S, V, W, and Z will reach a fluence of **2.75×10^{19}** n/cm^2 (E > 1.0 MeV), the 48 EFPY peak vessel fluence, at approximately **45 EFPY**. If vessel fluence data are needed at the EOL for life extension, one or more of the standby capsules will be moved to a higher flux location within the next few cycles of operation.

Table 2.4.1.3-4 Recommended Surveillance Capsule Withdrawal Schedule for Sequoyah Unit 2				
Capsule	**Capsule Location**	**Lead Factor [a]**	**Withdrawal EFPY [b]**	**Fluence (n/cm^2) [a]**
T	40°	3.33	1.04	2.61×10^{18} [c]
U	140°	3.40	2.93	6.92×10^{18} [c]
X	220°	3.39	5.36	1.22×10^{19} [c]
Y	320°	3.35	10.54	2.14×10^{19} [c, d]
S	4°	1.09	Standby	[e]
V	176°	1.09	Standby	[e]
W	184°	1.09	Standby	[e]
Z	356°	1.09	Standby	[e]

Notes:

(a) Updated in Capsule Y dosimetry analysis.

(b) EFPY from plant startup.

(c) Plant-specific evaluation.

(d) This fluence is not less than once or greater than twice the peak EOL fluence.

(e) Capsules S, V, W, and Z will reach a fluence of **2.74×10^{19}** n/cm^2 (E > 1.0 MeV), the 48 EFPY peak vessel fluence, at approximately **45 EFPY**. If vessel fluence data are needed at the EOL for life extension, one or more of the standby capsules will be moved to a higher flux location within the next few cycles of operation.

Table 2.4.1.3-5 Applicability Dates for Heatup and Cooldown Curves at Sequoyah Units 1 and 2			
Unit	**Current Applicability Date**	**Reference**	**Applicability Date Using Uprated Fluences**
Sequoyah Unit 1	16 EFPY	WCAP-12970	16 EFPY[a]
Sequoyah Unit 1	32 EFPY	WCAP-15293	32 EFPY[b]
Sequoyah Unit 2	16 EFPY	WCAP-12971	14.5 EFPY
Sequoyah Unit 2	32 EFPY	WCAP-15321	32 EFPY[c]

Notes:

(a) Uprated fluences are lower than fluences used to develop the current 16 EFPY curves.

(b) Actual applicability date is 31.3, which is essentially 32 EFPY.

(c) Actual applicability date is 31.8, which is essentially 32 EFPY.

Table 2.4.1.3-6 ERG Pressure-Temperature Limits	
Applicable RT_{NDT} (ART) Value[a]	**ERG P-T Limit Category**
$RT_{NDT} < 200°F$	Category I
$200°F < RT_{NDT} < 250°F$	Category II
$250°F < RT_{NDT} < 300°F$	Category IIIb

Note:

(a) Longitudinally oriented flaws are applicable only up to 250°F, the circumferentially oriented flaws are applicable up to 300°F.

Table 2.4.1.3-7 RT_{PTS} Calculations for the Sequoyah Units 1 and 2 Limiting Region Material at 32 and 48 EFPY							
Material	**Fluence (n/cm^2, E>1.0 MeV)**	**FF**	**CF (°F)**	**ΔRT_{PTS}[c] (°F)**	**Margin (°F)**	**$RT_{NDT(U)}$[a] (°F)**	**RT_{PTS}[b] (°F)**
Sequoyah Unit 1 (32 EFPY)							
Lower Shell Forging 04[d] **(Current)**	1.84	1.167	105.9	123.6	34	73	**231**
Lower Shell Forging 04[d] **(Uprating)**	1.86	1.17	105.9	123.9	34	73	**231**
Sequoyah Unit 1 (48 EFPY)							
Lower Shell Forging 04[d] **(Current)**	2.72	1.267	105.9	134.2	34	73	**241**
Lower Shell Forging 04[d] **(Uprating)**	2.75	1.270	105.9	134.5	34	73	**242**
Sequoyah Unit 2 (32 EFPY)							
Intermediate Shell Forging 05 **(Current)**	1.82	1.164	95	110.6	34	10	**155**
Intermediate Shell Forging 05 **(Uprating)**	1.85	1.169	95	111.1	34	10	**155**
Sequoyah Unit 2 (48 EFPY)							
Intermediate Shell Forging 05 **(Current)**	2.71	1.266	95	120.3	34	10	**164**
Intermediate Shell Forging 05 **(Uprating)**	2.74	1.269	95	120.6	34	10	**165**

Notes:

(a) Initial RT_{NDT} values are measured values.

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F).

(c) $\Delta RT_{PTS} = CF * FF$.

(d) Using surveillance capsule data. This is higher than Position 1.1.

2.4.1.4 Reactor Internals Evaluation

The reactor internals support and orient the fuel and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support in-core instrumentation. Changes in the RCS temperatures produce changes in the boundary conditions experienced by the reactor internal components. The increase in core power increases the nuclear heating rates seen by reactor internal components close to the core. This section describes the evaluations that demonstrate that the reactor internals can perform their intended design function at the 1.3-percent uprate condition.

2.4.1.4.1 Thermal-Hydraulic Systems Evaluations

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region. This flow is not considered effective in the core heat transfer process. The principal core bypass components are the barrel-baffle region flow, vessel head cooling spray nozzles flows, vessel outlet nozzle gap flow, baffle plate cavity gap flow, and the thimble tubes flow.

The present Vantage 5 Hybrid (V5H) design core bypass flow limit is 7.5 percent of the total reactor vessel flow. Since the Sequoyah units currently have non-Westinghouse fuel, the design core bypass flow limit was confirmed by the fuel vendor to be less than the design value.

Upper Head Fluid Temperature

The average temperature of the primary coolant fluid that occupies the reactor vessel closure head volume is an important initial condition for certain dynamic LOCA analyses. Therefore, it is necessary to determine the upper head temperature at the revised RCS conditions. The thermal-hydraulic evaluation modeled the interaction among the different flow paths into and out of the closure head region. Based on this interaction, the model calculates the core bypass flow into the head region and the average head fluid temperature.

Using the Westinghouse 17x17 V5H fuel assembly hydraulic characteristics, the results show that the best-estimate temperature in the closure head (544.8°F) will not differ from the T_{cold} value (544.8°F) associated with the 1.3-percent uprate conditions.

Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large-diameter belleville-type, rectangular cross-section spring. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and upper internals flange, and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation of the hydraulic lift forces on the various reactor internal components showed that the reactor internals assembly would remain seated and stable for all conditions. Using the Westinghouse 17x17 V5H fuel assembly hydraulic characteristics, the hydraulic lift forces on the reactor internal components for the 1.3-percent power uprate did not change significantly (i.e., less than 0.2-percent).

Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability, or vibration, of fuel rods caused by a high-velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates that surround the core.

A number of experimental tests have been performed to study the interaction between baffle joint jetting and the response of the fuel rod. These tests indicate that there are two vibration levels that can result in fuel rod damage. Lower levels of vibration amplitude can inflict damage in the form of vibration wear at the rod/grid interface. Large amplitude vibration (whirling), caused by fluid elastic instability, can result in fuel rod damage due to cladding fatigue failure, rod-to-rod contact, or even rod-to-baffle-plate wall contact.

To guard against fuel rod failures from flow-induced vibration, the cross-flow emanating from baffle joint gaps must be limited to a specific momentum flux, V^2h. V^2h is the product of the gap width, h, and the square of the baffle joint jet velocity, V^2. This momentum flux varies from point to point along the baffle plate, due to changes in pressure differential across the plate and the local gap-width variations. In addition, the modal response of the vibrating fuel rod must be considered. That is, a large value of local momentum flux impinging near a grid is much less effective in causing vibration than the same V^2h impinging near the mid span of a fuel rod.

The effective momentum flux at the revised design conditions was determined. The calculations assume the Westinghouse 17x17 V5H fuel assembly hydraulic characteristics and the baffle/barrel region in the original design configuration.

The results show that the momentum flux did not increase by more than 0.04-percent above the current design base, as a result of the 1.3-percent uprate conditions. This is expected due to the small change in temperature associated with the 1.3-percent uprate.

Rod Cluster Control Assembly Drop Time Analyses

An evaluation determined the effect of 1.3-percent power uprate on the rod cluster control assembly (RCCA) drop time. The evaluation assumed the Westinghouse 17x17 V5H fuel assembly hydraulic characteristics. The RCCA drop time evaluation determined that the RCCA drop time would increase less than 0.010 seconds, with the 1.3-percent power uprate. This is expected due to the decrease in core inlet temperature (T_{cold}) associated with the 1.3-percent uprate.

2.4.1.4.2 Mechanical Evaluations

The 1.3-percent uprate conditions do not affect the current design basis for seismic and LOCA loads. Therefore, it was not necessary to re-evaluate the structural affects from seismic operating basis earthquake (OBE) and safe shutdown earthquake (SSE) loads and the LOCA hydraulic loads.

The 1.3-percent power uprate design conditions will slightly alter the T_{cold} and T_{hot} fluid densities, which will slightly change the forces induced by flow. The corresponding T_{cold} and T_{hot} fluid densities increase by 0.05-percent and decrease by 0.09-percent, respectively, for the 1.3-percent uprate conditions. The changes in temperature and fluid density are judged to be negligible when compared to the current design basis temperatures. This is shown in Table 2.4.1.4.2-1.

The evaluation is also based on the mechanical design flow rate (101,600 gpm), which is not changed by the 1.3-percent power uprate. Mechanical design flow is defined in Section 5.1 of the Sequoyah FSAR.

The temperature change from the current design basis to the 1.3-percent power uprate has been shown to be negligible. The mechanical design flow did not change at all. Therefore, the effect on the flow-induced vibration of the reactor internals is negligible.

Table 2.4.1.4.2-1 Changes in Temperature and Fluid Density

	Current Design Basis	1.3% Uprate	Fluid Density % Change
T_{cold}	545.2°F	544.8°F	+0.05
T_{hot}	611.2°F	611.6°F	-0.09
Temp. Range	66.0°F	66.8°F	---

2.4.1.4.3 Structural Evaluations

Evaluations were required to demonstrate that the structural integrity of the reactor components is not adversely affected by the 1.3-percent uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be considered in the design and analysis of various components. The core support structures affected by the 1.3-percent uprate conditions are discussed in the following subsections. The primary inputs to the evaluations are the design parameters (see Table 2.1-1) and the gamma heating rates.

Baffle-Barrel Region Evaluations

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

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

An evaluation of the baffle-barrel region was previously performed as part of the RCS flow reduction, due to a 15-percent SGTP. The structural assessment determined that the reactor internals were structurally adequate for the revised plant operating condition associated with the 15-percent SGTP. The heating rates and design transients have not changed from those for the 15-Percent SGTP Program. Therefore, the evaluation performed to determine the effects of the 15-percent SGTP covers the 1.3-percent uprate conditions. The ability to provide structural restraint and direct coolant flow (i.e., maintain coolable core geometry) of the baffle-barrel region is maintained.

Lower Core Plate Structural Analysis

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

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

A structural evaluation demonstrated that the 1.3-percent power uprate does not adversely affect the structural integrity of the lower core plate.

The 1.3-percent power uprate causes an increase in the heat generation seen by the lower core plate. However, the existing Sequoyah Units 1 and 2 design transients remained valid for the 1.3-percent power uprate.

The conclusion of the evaluation is that the structural integrity of the lower core plate is maintained. The new RCS conditions, which are due to the 1.3-percent uprate, produced acceptable margins of safety and fatigue utilization factors, under all loading conditions.

Upper Core Plate Structural Analysis

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

The normal and upset stresses in the upper core plate are mainly due to hydraulic, seismic, and thermal loads. The total thermal stresses are due to secondary membrane stress and surface skin stress. The bending stress is negligible. The secondary membrane stress results mainly from the average temperature difference between the perforated region and the rim region. The surface stress results from the average temperature of the metal and the surface temperature. The main contributor to the metal temperature is the fluid temperature. Gamma heating effects are small and only contribute a few degrees, whereas the individual core outlet temperature is the major contributor to the perforated region metal temperatures. The rim region is primarily heated by the fuel assembly peripheral flow and the baffle-barrel region exit flow. For this evaluation, it is assumed that the rim region temperature is equal to the core inlet temperature. Since the temperature effects on the upper core plate remained the same compared to the previous fuel cycles and the existing design transients remained valid, the upper core plate is structurally adequate for the 1.3-percent power uprate.

2.4.2 Control Rod Drive Mechanisms

The effect of the NSSS performance parameters shown previously in Table 2.1-1 and the NSSS design transients on the control rod drive mechanisms (CRDMs) has been evaluated. The maximum decrease in the vessel/core inlet temperature is from 545.2°F to 544.8°F. The current stress and fatigue evaluations are based on an operating temperature of at least 550°F. This evaluation showed that the full-length L-106A CRDMs and the part-length CRDMS remain acceptable for the 1.3-percent power uprate conditions. The code version for the CRDM design is unchanged from the original version.

2.4.3 Reactor Coolant Loop Piping and Supports

The parameters for the 1.3-percent power uprating were reviewed for effects on the existing design basis for the reactor coolant loop (RCL) piping system. The parameters associated with the 1.3-percent power uprating were reviewed for effects on the existing design basis analysis for the following components: the RCL piping, the primary equipment supports, the primary equipment nozzles, the RCL branch nozzles, and the Auxiliary Class 1 pressurizer surge line piping. The temperature changes associated with the 1.3-percent power uprating have been evaluated, and the results are discussed below.

RCL Piping, Equipment and Branch Nozzles, and Equipment Supports

The existing design-basis parameters were compared with the parameters for the 1.3-percent power uprating. The comparison showed changes in the temperatures of the hot leg, crossover leg, and cold leg. The changes to the temperature of the hot leg, crossover leg, and cold leg are less than or equal to 0.4°F. The temperature changes will have insignificant effect on the material properties, thermal expansion, support gaps, and allowable stresses. Therefore, the effect is insignificant on the thermal analysis of the RCL piping system due to the 1.3-percent power uprating.

As part of the uprating analysis effort, it was determined that the NSSS design transients as they presently exist remain valid for the 1.3-percent uprated condition. The RCL analysis is based on the Power Piping United States of America Standard (USAS) B31.1 Piping Code and does not require a fatigue analysis for the RCL piping system.

As part of the 1.3-percent power uprating, the RCL LOCA hydraulic forces associated with the defined postulated breaks and the reactor pressure vessel dynamic LOCA displacements associated with the defined postulated breaks were evaluated. It was determined that the existing design-basis RCL LOCA hydraulic forces and the reactor pressure vessel LOCA displacements remain valid for the 1.3-percent power uprating.

In summary, the 1.3-percent power uprating will have insignificant effect on the existing design-basis analysis of the RCL piping system. Therefore, the effect on the primary equipment nozzle loads, the RCL branch nozzle loads, and the input loads for the RCL leak-before-break (LBB) analysis are also insignificant. There will be no significant change to the primary equipment support loads, since a new analysis was not required.

Therefore, the existing analysis design basis of the RCL piping system remains valid for the 1.3-percent power uprating.

Pressurizer Surge Line Piping

As part of the uprating analysis effort it was determined that the NSSS design transients as they presently exist remain valid for the 1.3-percent uprated condition. Therefore, there is no effect on the fatigue evaluation of the Auxiliary Class 1 pressurizer surge line piping.

The increase of 0.4°F in the hot leg temperature for the 1.3-percent power uprating over the existing design basis temperature will have an insignificant effect on the material properties, thermal expansion, support gaps, and allowable stress. The change will have an insignificant effect on the stratification in the Auxiliary Class 1 pressurizer surge line and the nozzles. Actually, the increase in the hot leg temperature is a benefit to the stratification analysis since ΔT between the RCL hot leg and the pressurizer becomes smaller. Therefore, no changes to the pressurizer surge line analysis are required and the existing design-basis analysis still remains valid.

Leak-Before-Break 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-12011.

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 that 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-12011.

The temperature increases in the critical location, which is in the hot leg, from 611.2°F to 611.6°F. This results in insignificant effect on the loads of the RCL piping. The effect of material properties due to the changes in temperature will have negligible effect on the LBB margins. Therefore, the LBB analysis remains applicable for the 1.3-percent power uprating for Sequoyah Units 1 and 2.

2.4.4 Reactor Coolant Pumps and Motors

Structural Analysis

The RCPs are affected by the reactor coolant pressure, steam generator outlet temperature, and primary-side cold leg NSSS design transients.

The reactor coolant pressure did not change from the original parameters. Since the reactor coolant pressure remains the same as originally specified (2250 psia) for the RCPs, the RCPs remain satisfactory for the 1.3-percent uprate reactor coolant pressure condition.

For the 1.3-Percent Power Uprate Program parameters, the steam generator outlet reactor coolant temperatures decreased by 0.4°F from the 15-Percent SGTP Program parameters. Likewise, the 15-Percent SGTP Program parameter (steam generator outlet reactor coolant temperature) decreased by 1.5°F from the original NSSS parameters. Therefore, for the 1.3-Percent Power Uprate Program parameters, the steam generator outlet reactor coolant temperatures decreased by 1.9°F from the original NSSS parameters.

The steam generator outlet reactor coolant temperatures are used in the analysis of the RCPs as a starting temperature for the normal and upset conditions transients and for determining material properties. Typically, a higher temperature value results in a greater actual stress and in a lower allowable stress. A change of 1.9°F is insignificant to the analysis. However, the 1.9°F decrease in temperature from the original conditions and the 0.4°F decrease in temperature from the 15-Percent SGTP Program are actually slightly less severe conditions. Since the RCPs were evaluated at normal temperatures greater than the 1.3-percent uprate parameter temperatures, the analyses of the RCPs remain conservatively valid. Therefore, the RCPs remain structurally satisfactory for use at the 1.3-percent uprate program steam generator outlet temperature.

The NSSS design transients for the 1.3-Percent Power Uprate Program did not change from the transients previously analyzed. Since the NSSS design transients have not changed, the RCPs remain satisfactory for the 1.3-Percent Power Uprate Program transient conditions.

RCP Motor Analysis

The NSSS parameters provided previously in Table 2.1-1 were used to generate worst-case loading of the RCP motors at Sequoyah Units 1 and 2 for a power uprating program. The new RCP motor loads are encompassed under the bounding loads of the previous analysis. The RCP motors at Sequoyah Units 1 and 2 are acceptable for operation for the revised operating conditions.

2.4.5 Steam Generators

2.4.5.1 Structural Integrity Evaluation

The uprating of Sequoyah Units 1 and 2 to 3467 MWt NSSS power will incorporate SGTP in the range from 0-percent to 15-percent maximum in any steam generator. The uprating will consider a slightly higher feedwater temperature of 436.3°F versus the value of 434.6°F considered for the design-basis analysis. These parameters were used for the structural evaluation of steam generators. The design transients modified by the 1.3-percent uprating were used in this evaluation.

Comparisons of the primary-side transients and RCS parameters were performed to determine the scale factors that would be applied to the baseline analyses for maximum stress ranges and fatigue usage factors. The baseline analysis results for various components were updated for uprated conditions.

For the primary-side components (notably, the divider plate, the tubesheet and shell junctions, and the tube-to-tubesheet weld), the scale factors were the ratios of the primary-to-secondary ΔPs for the baseline and uprated scenarios. These have been calculated over the entire time span of the applicable transients.

For the secondary-side components (such as the main steam and feedwater nozzles), the stress ranges involving transients that originate from or lead to full power are increased from the baseline values due to a decrease in secondary pressure. The increased stress ranges have been addressed in the evaluations of the main steam and feedwater nozzles.

The primary stress analyses are unchanged from the baseline values. Results of the analyses performed on the Sequoyah Units 1 and 2 steam generators show that the ASME Code Section III limits are met for the proposed uprated conditions for up to 15-percent SGTP.

2.4.5.2 Steam Generator Thermal-Hydraulic Performance

Secondary-side steam generator performance characteristics such as circulation ratio, moisture carryover (MCO), hydrodynamic stability, heat flux, and others are affected by increases in thermal power. Steam pressure is, in turn, determined by the power, the primary temperature, and the tube plugging level. The following paragraphs assess the magnitude and importance of changes in the secondary-side thermal-hydraulic performance characteristics at the 1.3-percent power uprate conditions.

Circulation Ratio

The circulation ratio is a measure of bundle flow in relation to the steam flow. It is primarily a function of steam flow (power). The 1.3-percent increase in power from current rating causes the circulation ratio to decrease by less than 2 percent. The bundle liquid flow, given by the product (Circulation Ratio -1) times steam flow, changes by less than 1 percent. The bundle liquid flow minimizes the accumulation of contaminants on the tubesheet and in the bundle. Therefore, the uprating and other operating condition changes have minimal effect on this function. No effects on sludge accumulation or local concentrations are expected.

Damping Factor

The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit. That is, small perturbations of steam pressure or circulation ratio will die out rather than grow in amplitude. This parameter remains at nearly the same high negative value for all conditions. The steam generators will continue to operate in a hydrodynamically stable manner for all operating conditions.

Heat Flux

The value of heat flux will increase with power and tube plugging. For uprating, increased total heat load is passed through the same bundle heat transfer area, increasing the heat flux. For increased plugging, the same heat load is passed through a smaller heat transfer area, also increasing the heat flux. The effect of assumed plugging (15 percent) is much larger than the effect of the power increase (1.3 percent), as could be expected. In all cases, the maximum calculated peak heat flux is well within nucleate boiling limits and is comparable to values for other steam generators currently operating in the field.

Secondary-Side Steam Generator Pressure Drop

The increase in total secondary-side pressure drop, resulting from the uprating, is less than 1 psi. This increase is very small in relation to the total feedwater system pressure drop and should have no significant effect on the feedwater system operation.

Summary

In summary, the thermal-hydraulic operating characteristics of the Sequoyah Units 1 and 2 steam generators are within acceptable ranges for the uprated power.

2.4.5.3 U-Bend Vibration and Fatigue

A U-bend fatigue evaluation was performed in 1998 for Sequoyah Units 1and 2 as part of the 15-percent tube plugging study. The evaluation calculated the maximum allowable relative stability ratios (RSRs) for the most susceptible tubes, and they are included in Table 2.4.5-1. If the actual RSR for a given tube is below this value, the tube will not be susceptible to high-cycle fatigue. The seven most susceptible tubes and their respective maximum allowable RSR values were determined. Those maximum allowable RSR values remain valid for the current U-bend fatigue evaluation at the uprated condition.

One-dimensional RSR values were calculated for three power levels: current rating, 1.3-percent power uprate, and 1.02 times the 1.3-percent power uprate. The last case is to account for the loop-to-loop power skew. Table 2.4.5-2 shows the relative steam flow rate for the four loops normalized to the average steam flow per loop for Sequoyah Units 1 and 2, derived from the plant data obtained in 1998. Steam Generator 1 in Unit 1 is typically operating at a higher thermal power than all the other steam generators, but has no susceptible tubes. Steam Generator 4 in Unit 2 operates with steam flow about 2 percent above the average value. This steam flow envelopes the steam flow rate for all susceptible tubes.

Tubes R8C60 in Steam Generator 4 and R9C35 in Steam Generator 1 of Unit 2 are expected to become susceptible to high-cycle fatigue if the steam pressure falls below 780 to 800 psia. Three more tubes are also expected to become susceptible to high-cycle fatigue if the steam pressure falls below 650 to 700 psia. These tubes have to be repaired if it is necessary to decrease the steam pressure below the range indicated.

2.4.5.4 Moisture Carryover

The performance of moisture separator packages, like those in the Sequoyah Units 1 and 2 steam generators, is primarily determined by three operating parameters: steam flow (power), steam pressure, and water level.

At the current power level with 0-percent plugging, the MCO is expected to remain below 0.25 percent. At the lower steam pressure corresponding to the 1.3-percent uprating with 15-percent tube plugging, the best-estimate MCO is expected to exceed 0.25-percent. Based on evaluations of the conditions for the uprating, the calorimetric uncertainty calculations were performed with an assumed MCO of 0.25 percent to 0.45 percent. This increased MCO results in a total calorimetric uncertainty less than 0.7 percent.

2.4.5.5 Steam Generator Tubing Corrosion Rates

The minor change in temperature and secondary-side pressure will have non-quantifiable effects on degradation rates, structural integrity, and/or leakage integrity. Steam generator inspections are driven by unit-specific steam generator degradation assessments and repair decisions are driven by operational assessments performed for each inspection. Temperature and pressure are inputs into calculations performed to ensure tube integrity. The current methodology of performing condition monitoring and operational assessments will not change due to this minor uprate. If the temperature change affects degradation growth rates, the repair limit will be assessed during operational assessments.

2.4.5.6 Steam Generator Tubing Wear Rates

The slight increase in steam flow due to the uprate should not result in a significant increase in wear rates at the expected operating conditions. Anti-vibration bar wear rates are evaluated after each inspection and an operational assessment is performed considering growth rates. Repair limits are adjusted if necessary based on the operational assessment.

2.4.5.7 Uprating Evaluation of Mechanical and Weld Plugs, Laser-Welded Sleeves, Tube Stabilizers, and Tube Support Plate Alternate Repair Criteria

Evaluations were performed to determine the effect of the power uprating conditions (as provided in Table 2.1-1) on the structural evaluations of the steam generator hardware changes and additions, which are discussed in the following subsections.

Steam Generator Tube Mechanical Plug

The tube mechanical plug is adequately anchored in the tube for all steady-state and transient conditions. There is adequate friction to prevent dislodging of the plug, and there is adequate leakage resistance for the limiting steady-state and transient loadings. All of the stress/allowable ratios are less than unity, indicating that all primary stress limits are satisfied for the plug shell wall, between the top land and the plug end cap. The plug shell meets the Class 1 fatigue exemption requirements, per Article N-415.1 of the 1968 ASME Code, equivalent to NB-3222.4 of the 1989 Edition of the Code.

Steam Generator Weld Plug

The weld plug and the associated welded attachment have been shown to be adequate for installation in the steam generator tubesheet. The primary stress analysis was reviewed, and all of the stress/allowable ratios are less than unity, indicating that all primary stress limits are satisfied for the weld, between the weld plug and tubesheet cladding. The cumulative fatigue usage remains at less than unity.

Laser-Welded Sleeves

All ASME Code structural limits for pressure, stress-range ($3S_m$) and fatigue remain satisfied with positive margins for the 1.3-percent uprated loading conditions with 15-percent tube plugging for all Westinghouse laser-welded sleeves types that may be installed in Sequoyah Units 1 and 2.

Secondary Side Loose Parts

Eddy current data indicates that there is a loose part on the secondary side of steam generator 1 in Unit 2. Calculations have been performed to determine the amount of time that would be required to wear a tube down to a minimum acceptable tube wall thickness with the steam generator operating in the 1.3-percent uprate condition. These calculations indicated that there is only a small change in wear rate after uprate. The amount of time required to wear a tube down to 60-percent remaining tube wall thickness is approximately 23 years. This value has been calculated assuming that an initial 20-percent deep wear scar is present and that the steam generators are operating at the uprated condition. In addition, a review of the eddy current data indicates that the orientation of the object has not changed since 1997, and that no additional tubes are being contacted by this object if the object remains near the 1997 location.

As a result of this evaluation, it was concluded that, although the projected wear rates increase after the uprate, the increase does not significantly change the projected rate of tube wear. The amount of time required for the object to wear a tube down to a minimum tube wall thickness of 60 percent (40-percent wear depth) is still significantly larger than the period of time between eddy current inspections that occur at the end of each fuel cycle.

Tube Stabilizers

Westinghouse-designed cable stabilizers installed in Sequoyah Units 1 and 2 are not adversely affected by the changes in fluid conditions associated with the power uprating. Therefore, there is no effect on those stabilizers due to uprating.

Tube Support Plate (TSP) Alternate Repair Criteria (ARC)

Because the outside diameter stress corrosion cracking (ODSCC) ARC at the support plates are only affected while the corrosion-degraded portion of the tube is exposed during faulted conditions (due to

TSP deflection) and faulted conditions are unaffected by the uprate, the TSP ARC are not affected by the uprate.

Table 2.4.5-1 Minimum Allowable Steam Pressures for Tubes Susceptible to U-Tube Fatigue

Susceptible Tubes				Minimum Allowable Steam Pressure (psia)		
Unit	**Steam Generator**	**Tube**	**Maximum RSR**	**1.3% Uprate Nominal**	**1.02 Times 1.3% Uprate**	**Current Rating**
2	4	R8C60	1.024		801	763
2	1	R9C35	1.025	781		762
2	2	R8C35	1.098	698		671
2	2	R8C60	1.121	670		642
2	4	R9C35	1.122		680	641
1	4	R8C36	1.149	632		608
1	2	R8C35	1.157	623		598

Table 2.4.5-2 Relative Loop Steam Flow

	SG #1	SG #2	SG #3	SG #4	Average
Unit 1	1.036	0.991	1.007	0.966	1.00
Unit 2	0.994	1.005	0.982	1.019	1.00

2.4.6 Pressurizer

An analysis was performed to assess the effect 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 analyzed for the pressurizer is 653°F. Table 2.4.6-1 is a summary of the temperature changes incurred (see Table 2.1-1 for T_{hot} and T_{cold} data).

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 41.8° to 41.4°F (i.e., actual ΔT and associated stress is lower). 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 108.2°F.

No changes were made to the design transients. Therefore, the transients specified in the current design specification are still applicable. For this reason, it was concluded that the revised parameters would not have any effect on the pressurizer stress analysis and fatigue analysis.

Table 2.4.6-1 Summary of Temperature Changes for Pressurizer

Parameter	Current Stress Analysis Value (15% SGTP)	Revised Value for 1.3% Uprate Conditions	$\Delta T_{Current\ Stress\ Report}$	ΔT_{Uprate}
Vessel Outlet (T_{hot}°F)	611.2	611.6	653-611.2 = 41.8	653-611.6 = 41.4
Vessel/Core Inlet (T_{cold}°F)	545.2	544.8	653-545.2 = 107.8	653-544.8= 108.2

2.4.7 Auxiliary Equipment

As part of the Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program, the auxiliary equipment was reviewed to verify applicability of the original design conditions. This equipment includes the auxiliary pumps, tanks, valves, and heat exchangers supplied by Westinghouse. The NSSS uprate transients and design conditions for the auxiliary valves, subjected to these transients and conditions, are bounded by the original design parameters. The auxiliary system transients for the auxiliary system valves, which are subject to the auxiliary transients, and the tanks, pumps and heat exchangers are bounded by the original design conditions.

Based on review of the auxiliary equipment design conditions, versus those for the 1.3-Percent Power Uprate Program, the auxiliary pumps, valves, tanks, and heat exchangers supplied by Westinghouse are qualified for the Sequoyah Units 1 and 2 1.3-percent power uprate conditions.

2.5 ACCIDENT ANALYSES COMPLETED BY WESTINGHOUSE

2.5.1 LOCA Evaluation for Containment Integrity, Compartment Analyses, and Equipment Qualification

2.5.1.1 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 methodology for the licensing basis analysis is contained in WCAP-10325-P-A. The Analysis of Record, which is documented in WCAP-12455 Revision 1, Supplement 1R, assumes a reactor power of 3411 MWt. In addition, the analysis applies an extra 2-percent calorimetric uncertainly to this power to account for power measurement uncertainty. The analyzed power including the current uncertainty is 3479 MWt.

The use of the Caldon LEFM will increase the accuracy of the measured main feedwater flow rate, thereby reducing error in the calorimetric power measurement process. The improvement is such that the uncertainty is reduced to 0.7 percent instead of 2 percent. Therefore, the existing 2-percent calorimetric uncertainly is reallocated such that 1.3 percent is applied to accommodate the 1.3-percent uprating and 0.7 percent is retained as calorimetric uncertainty. The margin of safety would not be reduced because the analyzed power including current uncertainty remains at 3479 MWt, the loop average temperature remains unchanged (578.2°F), and the initial energy content of the RCS fluid remains unchanged. The current Analysis of Record remains bounding.

2.5.1.2 Short-Term LOCA Mass and Energy Release Analysis

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

The short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses. The use of the lower temperatures maximizes the critical mass flux in the Zaloudek correlation. The effect of the minimum composite RCS T_{hot} and T_{cold} that are calculated for the 1.3-percent uprate conditions are evaluated and discussed in the following three subsections.

Loop Subcompartment Analysis

The loop subcompartment analysis was 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) that accompanies a LOCA. Also, this analysis verifies the adequacy of the ice condenser performance.

The Transient Mass Distribution (TMD) Program described in Section 6.2.1.3.3 of the Sequoyah FSAR was used for the current licensing basis subcompartment analysis. This licensing basis analysis considered a vessel outlet temperature of 594.7°F and a vessel/core inlet temperature of 523.7°F, both conservatively bounded low for short-term considerations. These values bound those for the 1.3-percent uprating. The licensing basis analysis also concluded that there are margins in the current subcompartment calculations that would offset the increase in mass and energy releases. For example, from a subcompartment perspective, it is more realistic to split the break flow between nodes 1 and 2, or at a minimum, to increase the ease of flow from TMD node 1 to TMD node 2. This is because the limiting break is in the hot leg, which is right at the boundary between TMD nodes 1 and 2. Increasing the flow from node 1 to node 2 by less than 20 percent results in the demonstration that the current loadings remain bounding. Analysis margins, therefore, offset any penalty associated with the uprate program. Additionally, and more importantly, analysis was conducted on a similar ice condenser design that showed the temperature reduction was actually a benefit. Therefore, the current licensing basis loop subcompartment analysis remains bounding.

Reactor Cavity Analysis

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

The 100 sq. in. reactor vessel inlet break currently forms the licensing basis for this subcompartment. It was estimated that the peak releases would conservatively increase by less than 10 percent for the 1.3-Percent Power Uprate Program. However, based on results of the structural analysis of the RCS, a better estimate of the break size is <90 sq. in. The reduced release rates from this reduced break size more than offset the predicted 10-percent increase. For example, the releases are approximately proportional to the break size, and as such, the releases would be reduced by a factor of (>100/90 = 1.11). The margin of safety would not be reduced.

Pressurizer Enclosure Analysis

The pressurizer enclosure analysis was performed to ensure that the walls in the immediate proximity of the pressurizer enclosure can maintain their structural integrity. Loadings acting across the pressurizer were 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 22 percent. The difference in break sizes leads to greater than 22-percent margin in the mass and energy releases. This more than offsets the increases in mass and energy releases predicted for the uprating program. Therefore, the current mass and energy releases remain bounding, and the current pressurizer enclosure pressure analysis remains bounding. The margin of safety would not be reduced.

2.5.1.3 Maximum Reverse Pressure Differential Analysis

Following a LOCA, the pressure and temperature in the lower compartment of containment increase. 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. This 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 6.2.1.3.12. 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 thereby 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. Therefore, this would reduce the reverse differential pressure. The mass and energy releases are extracted from a model used to maximize the LOCA peak cladding temperature (PCT) and not from a model used to maximize the peak containment pressure. It is judged that the RCS temperature changes and the resulting effects would not affect the results of the maximum reverse pressure differential calculation.

The purpose of this analysis is to show that significant margin exists in the design. The existing peak calculated differential pressure of 0.65 psi is significantly lower than the structural design and load carrying capability of the operating deck. Therefore, the 1.3-percent uprating will have a minimal effect, if any, on the analysis and there is significant analysis margin available. The current Analysis of Record remains bounding. The margin of safety would not be reduced.

2.5.2 Steam Line Break Mass and Energy Releases

An evaluation of the effect of a 1.3-percent power uprate on the Sequoyah Units 1 and 2 licensing-basis safety analyses related to secondary-side mass and energy releases is provided in this section. The events included in this evaluation are steam line mass and energy releases inside and outside containment, feed line break mass and energy releases inside containment, and radiological steam releases for dose evaluations.

2.5.2.1 Long-Term Steam Line Break Mass and Energy Releases Inside Containment

Critical parameters for the long-term steam line break event relate to the following conditions on the primary and secondary sides: NSSS power level, reactivity feedback characteristics including the moderator density coefficient and minimum plant shutdown margin at the end of core life, initial and trip values for the steam generator water mass, main feedwater flow, auxiliary feedwater flow, main and auxiliary feedwater enthalpy, and the times at which steam line and feed line isolation occur. The input assumptions that relate to these critical parameters dictate the quantity and rate of the mass and energy releases.

The 1.3-percent power increase for Sequoyah Units 1 and 2 will be offset by an equivalent reduction in the calorimetric uncertainty. The Analyses of Record for the inside containment long-term steam line breaks assume a 2-percent power calorimetric uncertainty on the 3423 MWt NSSS power. A 0.7-percent power calorimetric uncertainty applied to a 1.3-percent power increase is equivalent to the 2-percent uncertainty in the licensing-basis safety Analyses of Record for Sequoyah.

However, the TDF, the steam temperature, and the full-power feedwater temperature for the uprated condition are not the same as assumed in the Analyses of Record for Sequoyah. The reduction in the value for the TDF (associated with 15-percent SGTP) provides a slight benefit in the calculation of the mass and energy releases. In general, however, small differences in the RCS flowrate used in the safety analyses have no effect upon the results. The nominal value for the steam temperature used in the Analyses of Record is a conservatively high value of 526.2°F. Also, the initial value for the feedwater enthalpy used in the Analyses of Record is a conservatively high value based on a full-power feedwater temperature of 436.9°F. Therefore, there is no effect on either the current licensing-basis analyses of the long-term steam line break mass and energy releases inside containment or the FSAR conclusions, as a result of:

- The 1.3-percent power increase
- A reduced TDF
- The 3.4°F margin between the nominal steam temperature for the uprating (522.8°F at 0-percent SGTP and 517.5°F at 15-percent SGTP) and the steam temperature in the Analyses of Record (526.2°F)
- The 0.6°F margin between the nominal full-power main feedwater temperature for the uprating (436.3°F) and the Analyses of Record (436.9°F)

In general, limiting values of reactivity coefficients are used in the analyses of the long-term steam line break mass and energy releases inside containment to bound the transient over a wide range of core conditions. It was determined that the future operating moderator temperature coefficient at end of life is less limiting than the safety analysis value for the moderator density coefficient. Also, the minimum value for the shutdown margin is the same as the safety analysis value for the limiting steam line breaks inside containment.

2.5.2.2 Long-Term Steam Line Break Mass and Energy Releases Outside Containment

Critical parameters for the long-term steam line break event relate to the following conditions on the primary and secondary sides: NSSS power level, reactivity feedback characteristics including the moderator density coefficient and minimum plant shutdown margin at end of life, initial and trip values for the steam generator water mass, main feedwater flow, auxiliary feedwater flow, main and auxiliary feedwater enthalpy, and the times at which steam line and feed line isolation occur. The input assumptions that relate to these critical parameters dictate the quantity and rate of the mass and energy releases.

The 1.3-percent power increase for Sequoyah Units 1 and 2 will be offset by an equivalent reduction in the calorimetric uncertainty. The Analyses of Record for the outside containment long-term steam line breaks assume a 2-percent power calorimetric uncertainty on the 3423 MWt NSSS power. A 0.7-percent power calorimetric uncertainty applied to a 1.3-percent power increase is equivalent to the 2-percent uncertainty in the licensing-basis safety Analyses of Record for Sequoyah.

The TDF, the steam temperature, and the full-power feedwater temperature for the uprated conditions are not the same as assumed in the Analyses of Record for Sequoyah. The reduction in the value for the TDF (associated with 15-percent SGTP) provides a slight benefit in the calculation of the mass and energy releases. In general, however, small differences in the RCS flowrate used in the safety analyses have no effect on the results. The nominal value for the steam temperature used in the Analyses of Record is a conservatively high value of 524.7°F. The temperature used to establish the initial value for the feedwater enthalpy used in the Analyses of Record is 1.7°F less than the full-power feedwater temperature of 436.3°F at the uprated power. However, the decrease in the steam temperature more than compensates for the feedwater temperature increase, particularly when main feedwater is quickly isolated in the steam line break Analyses of Record for Sequoyah. Therefore, the net result is that there is no effect on either the current licensing-basis analyses of the long-term steam line break mass and energy releases outside containment or the FSAR conclusions, as a result of:

- The 1.3-percent power increase
- A reduced TDF
- The 1.9°F margin between the nominal steam temperature for the uprating (522.8°F at 0-percent SGTP and 517.5°F at 15-percent SGTP) and the steam temperature in the Analyses of Record (524.7°F)
- The 1.7°F margin between the nominal full-power main feedwater temperature for the uprating (436.3°F) and the main feedwater temperature in the Analyses of Record (434.6°F)

In general, limiting values of reactivity coefficients are used in the analyses of the long-term steam line break mass and energy releases outside containment to bound the transient over a wide range of core conditions. It was determined that the future operating moderator temperature coefficient at end of life is less limiting than the safety analysis value for the moderator density coefficient. Also, the minimum value for the shutdown margin is the same as the safety analysis value for the spectrum of steam line breaks outside containment.

2.5.2.3 Long-Term Feed Line Break Mass and Energy Releases

This analysis was performed to confirm the adequacy of the trip-time-delay (TTD) logic associated with the low-low steam generator water level reactor trip signal following a feed line break event at initial conditions less than 50-percent power. Critical parameters for this event relate to the following conditions on the primary and secondary sides: initial and trip values for the steam generator water mass, auxiliary feedwater flow and enthalpy, the core reactivity feedback characteristics including the moderator density coefficient and minimum plant shutdown margin at end of life, and decay heat. The input assumptions that relate to these critical parameters dictate the transient progression of the feed line break and the effect of the superheated energy releases on the TTD logic.

Since this analysis was performed at part-power operation, when the TTD logic is utilized, the effect of the 1.3-percent power increase for Sequoyah Units 1 and 2 is not an important parameter. The Analysis of Record for the long-term feed line breaks assumes initial powers of 59 percent, 39 percent, and 9 percent of the 3423 MWt NSSS power. The effect of the 1.3-percent power increase is to create an alternate partial power initial condition for which the analysis results have been calculated.

The reduction in the value for the TDF (associated with 15-percent SGTP) provides a slight benefit in the calculation of the mass and energy releases. In general, however, small differences in the RCS flowrate used in the safety analyses have no effect upon the results. The nominal value for the steam temperature used in the Analyses of Record is a conservatively high value of 524.7°F. No feedwater flow is assumed in the analysis, so there is no effect caused by any difference in the feedwater temperature. Therefore, the net result is that there is no effect on either the current licensing-basis analysis of the long-term feed line break mass and energy releases or the FSAR conclusions, as a result of:

- The 1.3-percent power increase
- A reduced TDF
- The 1.9°F margin between the nominal steam temperature for uprating (522.8°F at 0-percent SGTP and 517.5°F at 15-percent SGTP) and the steam temperature in the Analyses of Record (524.7°F)
- No effect due to feedwater, since loss of feedwater is assumed

In general, limiting values of reactivity coefficients are used in the analyses of the long-term feed line break mass and energy releases to bound the transient over a wide range of core conditions. It was determined that the future operating moderator temperature coefficient at end of life is less limiting than the safety analysis value for the moderator density coefficient. Also, the minimum value for the shutdown margin is the same as the safety analysis value for the feed line break mass and energy releases.

2.5.2.4 Radiological Steam Releases for Dose Calculations

Critical parameters for calculations of the radiological steam releases used as inputs to the dose evaluation model include the NSSS power, the RCS average temperature, and the steam temperature and pressure. Each of the primary-side inputs is conservatively calculated assuming the engineered safeguards design power, which is equivalent to a 4.5-percent uprated power. The current Analyses of Record assume primary- and secondary-side design parameters that are conservatively high with respect to the actual Sequoyah Units 1 and 2 operating conditions. Therefore, there is no effect on either the current licensing-basis radiological steam release analysis or the FSAR conclusions, as a result of the 1.3-percent power increase and the increase in the nominal full-power main feedwater temperature.

Fuel-related reactivity coefficients are not used in the analysis of the radiological steam releases for dose. Therefore, this licensing-basis calculation is not affected by the changes in the Sequoyah Units 1 and 2 cores.

2.5.3 LOCA Hydraulic Forces

In support of the proposed 1.3-percent uprating conditions for Sequoyah Units 1 and 2, an assessment of the impact of uprated RCS conditions on the LOCA forces was performed. This assessment consisted of evaluating the LOCA forces Analysis of Record to reflect the more limiting uprated RCS conditions while incorporating a less conservative break model using LBB methodology. The estimated increase to the LOCA forces due to the change in RCS conditions for the uprate was then compared to the estimated decrease in LOCA forces due to the break area reduction. The comparison showed that the force reduction from the break area margin more than offset the increase in forces associated with the uprated conditions. It is, therefore, concluded that the existing LOCA forces Analysis of Record supporting Sequoyah remains conservative.

3.0 FUEL AND ACCIDENT ANALYSES COMPLETED BY FRAMATOME

3.1 INTRODUCTION AND SUMMARY

3.1.1 Introduction

FRA-ANP has evaluated the impact of a 1.3-percent uprate to 3455 MWt core power on the applicable parameters analyzed in reload safety evaluations for Sequoyah Units 1 and 2. All inputs to the accident analysis in the FSAR related to fuel mechanical, system thermal-hydraulic, core thermal-hydraulic, neutronic, and Chapter 15 accident analyses were reviewed. The power uprate described in this report is based on eliminating unnecessary analytical margin originally required in emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in 10 CFR 50, Appendix K (Emergency Core Cooling System Evaluation Models, ECCS) for Sequoyah Units 1 and 2. Prior to June 1, 2000, NRC regulations required an assumed power level of 102 percent of the licensed power level for application in ECCS evaluation models of light water reactors. The Federal Register of June 1, 2000 included an amendment to the introductory paragraph of I.A. of Appendix K to 10 CFR 50. The amendment allows the use of an assumed power level lower than 102 percent with the provision that the alternative value has been demonstrated to account for power level uncertainties. A power level uncertainty of 0.7 percent has been justified for Sequoyah Units 1 and 2. This allows the Sequoyah units to operate at the increased power level of 1.013 of 3411 MWt and remain within the assumptions of the existing ECCS evaluations.

Based on the proposed use of the improved Caldon LEFM instrumentation, the licensed power uncertainty required by 10 CFR 50, Appendix K may be reduced for modest increases of up to 1.3-percent in the licensed power level using current NRC approved methodologies. The basis for the amendment request is that the Caldon instrumentation provides a more accurate indication of feedwater flow (and correspondingly reactor thermal power) than assumed during the development of Appendix K requirements. Complete technical support for this conclusion is discussed in detail in Caldon Topical Report ER-80P. This report is approved in the NRC's Safety Evaluation for Texas Utilities (TU) Electric, dated March 8, 1999, and supplemented by Caldon Engineering Report 160P. The improved thermal power measurement accuracy obviates the need for the full 2-percent power margin assumed in Appendix K, thereby increasing the thermal power available for electrical generation.

Along with the proposed increase in reactor rated thermal power (RTP) to 3455 MWt, TVA also proposes continued use of the topical reports identified in Sequoyah Units 1 and 2 Technical Specification 6.9.1.14.a. These reports describe the NRC approved analytical methodologies used to determine the core protective and operating limits for Sequoyah Units 1 and 2, including the small-break and large-break LOCA analyses. In some of these topical reports, reference is made to the use of a 2-percent uncertainty applied to the reactor power, consistent with 10 CFR 50, Appendix K. These topical reports reflect the 10 CFR 50 Appendix K rule before it was changed in June 2000. They do not reflect the current methodology as described in this license amendment request, and will not be updated at this time. This change in the power uncertainty does not constitute a significant change as defined in 10 CFR 50.46 and Appendix K. If any future methodology changes are made affecting the information in the topical reports, the references to the 10 CFR 50, Appendix K, 2-percent power uncertainty treatment will also be changed at that time.

The following sections describe the reviews performed for each of the disciplines analyzed in the core reload safety evaluation.

3.2 FUEL ANALYSIS

3.2.1 Fuel Cycle Design

The following discussion provides a fuel cycle design assessment of the 1.3-percent power level uprate from 3411 MWt to 3455 MWt for the Sequoyah units.

The uprate of 1.3-percent power from a fuel cycle design perspective represents a small increase in the energy production of the fuel cycle. This difference in energy production is well within the typical variation of energy production from recent fuel cycles at the Sequoyah units which have varying energy production due to the typical changes in cycle length that are experienced at all nuclear power plants. The only difference is that this change represents a consistent increase in energy production, just as increases in capacity factor have resulted in greater energy production for each fuel cycle.

The following discussions describe the expected impact on each fuel cycle design parameter.

3.2.1.1 Core Average Burnups

Core average burnups will increase slightly due to the power level increase and may have some small impacts on global parameters such as power deficit, moderator coefficients, or excess shutdown margin.

3.2.1.2 Fuel Assembly Burnups

Fuel assembly burnups will increase approximately proportional to the overall power level increase of 1.3-percent power. The increase in burnup is about the same as would be experienced in increasing the fuel cycle design length by 7 effective full-power days (EFPD) every cycle. This is considered within the flexibility that could be accommodated with normal fuel cycle design variables such as feed batch size, enrichment, and burnable absorbers. Current fuel cycles and feed batch sizes are not challenging the technical burnup-related fuel assembly limits and a feed batch size change due to the power level increase alone would not be required.

3.2.1.3 Fuel Rod Burnups

The fuel rod burnups will increase slightly. Current rod burnups are not challenging the licensed Mark-BW burnup limit and the increase in power level will not change this situation. There are specific instances where special accommodation is needed in fuel cycle design because of availability of fuel assemblies and rod pressure considerations, but these situations are not expected to increase because of this power level uprate.

3.2.1.4 Batch Size

Based upon the margin to the current fuel rod burnup limit and enrichment margin to 5.00 wt% ^{235}U, no increase in batch size will be needed for this magnitude of additional energy requirement.

3.2.1.5 Fuel Enrichments

With no increase in batch size, the 1.3-percent power uprate will tend to require slightly increased fuel enrichments. This is expected to be about 0.06 wt% ^{235}U increase for the initial transition feed batch and about 0.03 wt% ^{235}U on an equilibrium core-wide basis.

3.2.1.6 Power Peaking

The higher feed batch reactivity required to accomplish the power level uprate will cause a slight increase in the relative power density differences between the fresh and carry-over fuel assemblies. However, this effect is expected to be very small and well within the variability typically seen from one fuel cycle to the next. The linear heat rate will increase proportionally and result in less operating margin. Cycle-specific licensing analyses ensure compliance with fuel protection limits.

3.2.1.7 Fuel Shuffle Impact

The increase in RTP level does not result in any modifications to the fuel shuffle.

3.2.1.8 Conclusion

No significant impact on the fuel cycle design is expected as a result of the change in the rated power level. The differences in fuel cycle designs are equivalent to the plant consistently operating with a higher capacity factor. This change is well within the normal variation in fuel cycle energy requirements typically accommodated at Sequoyah Units 1 and 2.

3.2.2 Nuclear Licensing Parameters

3.2.2.1 Transient Review and Evaluation

The nuclear licensing parameters and safety analysis checklist parameters were evaluated for the Sequoyah Units 1 and 2 core power uprate from 3411 MWt to 3455 MWt. The effect on specific accidents and Technical Specification compliance are addressed for each reload cycle in the Reload Safety Evaluation Report (RSER). The RSER events evaluated for each reload fuel cycle include: post-LOCA sump boron concentration, steam line break, rod ejection, rod misoperation, locked rotor, and boron dilution. The impact of the 1.3-percent power level uprate on the post-LOCA sump boron concentration is provided below. The impact of the 1.3-percent power level uprate on the rest of these events is provided in Sections 3.3.7, 3.3.8, and 3.3.9.

3.2.2.1.1 Post-LOCA Subcritical Boron

This evaluation is performed to verify that the core remains subcritical after a LOCA event. The verification for each reload core design relies on determining the post-LOCA sump boron concentration versus pre-LOCA RCS boron concentration, at the limiting time in the cycle. This limiting point is compared to a bounding curve determined from RCS and ECCS volumes and boron concentrations, which generally remains fixed from cycle to cycle. Since the power level increase would only slightly affect the pre-LOCA RCS boron concentration and none of the other parameters, there should be no significant effect on the post-LOCA sump concentration. Nonetheless, a specific calculation was performed at the uprated power level pre-LOCA critical boron concentration. The results were nearly indistinguishable from the current power level results. Therefore, post-LOCA sump boron concentration will not be affected by the power uprate. The cycle-by-cycle checks remain valid at nominal conditions.

3.2.2.2 Conclusion

The nuclear licensing parameters and safety analysis checklist parameters were evaluated for the Sequoyah Units 1 and 2 power level uprate from 3411 MWt to 3455 MWt. Overall, the power level uprate from 3411 MWt to 3455 MWt causes no significant impact on the reload cycle licensing and safety evaluation.

3.2.3 Power Distribution Analysis

3.2.3.1 Methodology Review

The FRA-ANP approved methodology for power distribution analysis used to set the core operating and protective limits appears in Sequoyah Technical Specification 6.9.1.14.a. This methodology was examined and found to be applicable for the power uprate conditions without any further modifications.

3.2.3.2 Peaking Factor Evaluation

When core protective and operating limits are set, the assembly-by-assembly limiting peaking factors are augmented to account for design tolerances, calculational uncertainties, and modeling simplifications. For calculation of margin to power peaking limits, the total calculated peak (F_Q) and radial calculated peak ($F_{\Delta H}$) are augmented as described in Section 4 of BAW-10163P-A.

The impact of the power uprate at Sequoyah Units 1 and 2 on all of the radial and total peaking augmentation factors that are used to set the core limits was evaluated. The results showed that none of these augmentation factors are affected by the power uprate. Therefore, the peaking augmentation factors are appropriate for application to reload safety evaluations at the uprated power level.

3.2.3.3 Error Adjustment

The $f_1(\Delta I)$ limits used in the OTΔT trip function, the $f_2(\Delta I)$ limits used in the overpower ΔT (OPΔT) trip function, and the axial flux difference (AFD) limits contain adjustments to account for process measurement accuracy and measurement system uncertainty. The $f_1(\Delta I)$, $f_2(\Delta I)$, and AFD limits are set or validated in the maneuvering analysis on a cycle-by-cycle basis. Once set, they are adjusted to account for instrument uncertainties. The instrument uncertainties are not affected by the slight power uprate.

3.2.3.4 Effects on the Core Limits

The potential effect on peaking margin due to the maximum allowable peaking limits and the possible effects on the core power distribution as a result of fuel assembly burnup gradients expected in the uprated core were evaluated. The effects on the core power distribution will be analyzed by FRA-ANP during the cycle-specific reload safety evaluation. The effects of possible change to the core allowable peaking limits are described below.

The core safety linear heat rate limits based on centerline fuel melt and cladding strain criteria will not be affected by the power uprate. Since the core average linear heat rate will increase by 1.3 percent, a slight margin loss of between 1.0 percent and 1.5 percent will be observed as a consequence of the power uprate. Since the Sequoyah cores tend to have a substantial amount of margin to these limits, the power uprate will not affect the OPΔT trip reset function ($f_2(\Delta I)$) limits.

The LOCA F_Q limits will not be altered as a result of the power uprate since the increase in core power is absorbed by reducing the power uncertainty used in determination of the limit. Therefore, LOCA peaking margins will not be affected by the power uprate.

Maximum allowable peaking (MAP) limits have been generated specifically for Sequoyah Units 1 and 2 at the power uprate conditions based on departure from nucleate boiling (DNB) criteria. These MAP limits are slightly more restrictive, by various amounts depending on the magnitude and location of the axial peaks, than the current MAP limits at 3411 MWtRTP. This slight reduction in the MAP limits will result in about 1-percent to 2-percent loss in DNB peaking margins.

The reduction in the DNB peaking margin will not affect the core AFD limits because these limits tend to be set by the LOCA peaking margins. Since the LOCA margins will not be affected by the power uprate, as discussed above, the AFD limits will not be affected by the power uprate. The slight reduction in DNB peaking margin is not expected to affect the OTΔT trip reset function ($f_1(\Delta I)$) limits, but is expected to result in a slight reduction in the available DNB peaking margins for core monitoring.

3.2.3.5 Core Power Distribution Monitoring

Sections 6 and 7 of BAW-10163P-A describe the procedures for generating the software database used by the plant computer to perform the F_Q and $F_{\Delta H}$ surveillance requirements specified by the Sequoyah Technical Specifications. The process of generating the software database is not affected by the power uprate. In addition to the peaking augmentation factors, discussed previously, the database requires the use of an F_Q Deviation Allowance (DAQ) factor and an $F_{\Delta H}$ Deviation Allowance (DAH) factor. The DAQ is defined as "the amount that the measured power can exceed the predicted value and still be within the design." The definition for the DAH is similar to the definition for DAQ with the exception of using radial comparisons instead of total peak comparisons. Review of the supporting references for these factors shows that they are not dependent on RTP. Therefore, these factors are not affected by the power uprate at Sequoyah.

3.2.3.6 Operating Guidelines

Operating guidelines are reviewed on a cycle-by-cycle basis and are specified in Appendix C of the Nuclear Design Report (NDR) for each cycle. Appendix C contains a set of guidelines for appropriate plant operation and the steps to be taken in case plant operation deviates from the cycle design parameters by pre-specified amounts, such as extended operation at low power. It also contains cycle-specific information such as the predicted (target) ΔI for steady-state operation. Operating guidelines for current fuel cycles specify a maximum burnup value that, if exceeded, would result in the potential need for an evaluation. This value was based on the change in LOCA peaking margin that could occur as a result of operation at reduced power. Since LOCA peaking margins will not be significantly affected by the power uprate (Section 3.2.3.4), the burnup value remains applicable for the power uprate conditions. Therefore, no changes to the guidelines are required other than the cycle-specific updates that are provided for current cycles.

3.2.3.7 Core Operating Limit Report Changes

None of the Core Operating Limit Report (COLR) parameters will be significantly affected by the power uprate except for the DNB MAP limits (Table 1 of the COLR), which will be revised to include MAP limits applicable for power uprate conditions. Changes in other cycle-specific parameters, such as AFD limits, will be well within the normal variations observed as a result of fuel cycle designs and energy requirements.

3.2.3.8 Conclusion

No significant adverse effect on the core power distribution analysis or COLR limits is expected as a result of implementation of the 1.3-percent power level uprate. Existing analytical methods remain adequate to evaluate reload cores, including error adjustments applied to COLR limits. Power peaking limits will not be significantly more restrictive. Although core monitoring margins may be slightly reduced, the reduction is within the variation seen as a result of fuel cycle design and final energy requirements. Core operating guidelines will continue to be applicable for power operation at the uprated thermal power.

3.2.4 Fuel Mechanical Analysis

3.2.4.1 Fuel Assembly Mechanical Analysis

The evaluation of fuel assembly structural components indicates that a 1.3-percent power level uprate for the Sequoyah Units 1 and 2 can be implemented successfully.

3.2.4.1.1 Hydraulic Lift Analysis

The uprating effect on core conditions for this evaluation is a change of less than 0.5°F in the vessel T_{out} and T_{in} respectively, which corresponds to an increase of less than 1°F in the core outlet temperature. Thermal-hydraulic analyses of the 3411 MWt and 3455 MWt power levels provide bounding inputs for the mechanical evaluation. The analyses consist of two classes of input. One class consists of changes in hydraulic lift force, including the total hydraulic lift for the fuel assembly and individual hydraulic lift for spacer grids. The other class consists of changes in coolant temperature, both globally for the core and locally along the length of the fuel assembly at spacer grid elevations. The results of the thermal-hydraulic analyses, discussed in Section 3.2.4.3, indicate a negligible effect on fuel assembly and grid lift forces and coolant conditions at the grid elevations for the increase in the nominalRTP from 3411 MWt to 3455 MWt. As a result, the hydraulic loads remain unchanged for the fuel assembly mechanical evaluation. Therefore, the existing fuel assembly hold-down margins remain applicable and acceptable.

3.2.4.1.2 Corrosion Evaluation

The evaluation of fuel assembly structural Zircaloy-4 components, including guide tubes and intermediate spacer grids, addresses the effect of the small differences in the coolant core outlet and inlet temperatures. The slight increase in the core outlet temperature is considered in the corrosion allowance for the structural components and the resulting structural margins for normal operating, faulted, and handling conditions. The oxide formation on the guide tube and intermediate grids is maximized near the top end of the fuel assembly. The model used to determine the maximum guide tube oxide thickness bounds the current database, which includes PIE oxide measurements at TMI-1, Oconee 2, Davis-Besse, and Catawba 2. The corresponding range of maximum core outlet temperatures for these plants is 604.0°F to 620.0°F, which envelopes that of the 3455 MWt core power level for Sequoyah Units 1 and 2. Hydrogen pickup in the guide tube increases slightly due to the small increase in oxide formation but remains acceptable. No structural material corrosion problems have been observed for FRA-ANP fuel including that operating in plants with the highest T_{out} temperatures. The predicted increase in corrosion is similarly small for the intermediate spacer grids. The resulting effects on relative grid strength and growth are negligible. The fuel assembly structural components, primarily consisting of Zircaloy-4 materials, including guide tubes and intermediate spacer grids, are evaluated for the effect of a slightly increased corrosion allowance with no resulting specific limitation on fuel assembly performance for operation at the 3455 MWt core power level.

3.2.4.1.3 Flow Induced Vibration

Flow-induced vibration (FIV) response can be affected by changes in fluid forcing function and structural frequency and damping. The slight decrease in the core inlet temperature has a negligible effect on the fluid density, which can affect the fluid forcing function on the fuel rod and assembly. In addition, the small differences in temperature have a negligible effect on the fuel assembly and fuel rod frequency and response. Since the changes in FIV forces and fuel assembly and fuel rod frequencies are negligible, any change in turbulent flow and fuel assembly and rod vibration response is considered negligible. Present fuel performance shows no FIV fretting-induced failures.

3.2.4.1.4 Faulted Condition Loads

Core temperatures change the fluid density and subsequently the core pressure drop, which can affect the resulting vertical LOCA forces that the fuel assembly experiences. The slight changes in core inlet and outlet temperatures have a negligible effect on the core pressure drop and the faulted condition force time histories. Thus, the existing faulted condition loading remains applicable for the 3455 MWt core power level, which shows acceptable structural margins.

3.2.4.1.5 Material Considerations

Note that future fuel designs will utilize M5™ components. Likewise, there should be no limitation on fuel assembly performance due to the 3455 MWt power uprate since the safety margins for the Zircaloy-4 components envelope those for M5™ components, due to the inherent lower corrosion of the M5™ alloy.

3.2.4.2 Fuel Rod Mechanical Analysis

The fuel rod mechanical evaluation indicates that a 1.3-percent power level uprate for Sequoyah Units 1 and 2 can be implemented successfully. The fuel rod mechanical analyses performed for each fuel rod design (UO_2 and gadolinia) include: cladding fatigue, transient strain, stress, creep collapse, and corrosion. Cladding fatigue and stress margins remain unaffected by the 3455 MWt power level uprate. Transient strain, creep collapse, and corrosion are the primary parameters that may be affected by the power uprate.

3.2.4.2.1 Transient Strain

The fuel rod cladding transient strain limits were verified using bounding power histories for fuel cycle designs with UO_2 and gadolinia fuel rods considering the 3455 MWt power level uprate. Use of tritium-producing burnable absorber rods (TPBARs) was also considered for the resulting local variations in rod power within the fuel assembly. The design power histories were shown to bound those for fuel cycles considering the power level uprate. Therefore, the existing transient strain limits remain applicable. The transient strain limits are verified on a cycle-specific basis.

3.2.4.2.2 Creep Collapse

The creep collapse lifetime was shown to be 65,000 MWD/MTU for an enveloping power history for fuel cycle designs considering the 3455 MWt core power level and use of TPBARs. The design power histories were shown to bound those for fuel cycles considering the power level uprate. Therefore, the existing creep collapse margins remain applicable and acceptable.

3.2.4.2.3 Corrosion

Fuel rod cladding corrosion analysis is adversely affected by increases in coolant temperature. Since limited corrosion margin exists in the current plant operating analyses (for Zircaloy-4 cladding), it is apparent that cladding corrosion will remain a limiting constraint for fuel cycle designers as long as Zircaloy-4 fuel rod cladding is used. It is expected that batch-specific fuel rod corrosion analyses will be performed using the NRC approved COROSO2 corrosion model within the KOROS code. The analysis predicts the thickness of the corrosion layer for the highest burnup pin within each resident sub-batch of fuel for any given cycle. This predicted corrosion thickness is compared to a best-estimate limit of 100 microns. All sub-batches within each fuel cycle must stay below the 100 micron limit. Given the small increase of 0.5°F in the outlet temperature, the lower relative outlet temperature compared to other Mark-BW plants, and the comparable power histories, Zircaloy-4 fuel rod cladding corrosion will be acceptable at the 3455 MWt power level. Corrosion analyses for reload licensing will continue to be performed on a cycle-by-cycle basis using actual rod power histories. Use of M5™ fuel rod cladding with its improved corrosion performance will remove the imposition of the Zircaloy-4 fuel rod corrosion-related burnup constraints in the future.

3.2.4.3 Conclusions

Based on the fuel assembly and fuel rod mechanical evaluation, the 1.3-percent power level uprate for Sequoyah Units 1 and 2 can be reached successfully. The 3455 MWt power level results in negligible changes to the hydraulic lift forces. Therefore, the existing hold-down margins remain applicable and acceptable. The increase in corrosion of the fuel assembly structural Zircaloy-4 components due to the slight increase in the core outlet temperature is small. Therefore, acceptable structural margins for normal operating, faulted, and handling conditions exist. Changes in FIV forces and fuel assembly and fuel rod frequencies are negligible. Therefore, the fuel assembly and fuel rod FIV performance remains acceptable. In addition, the existing fuel assembly faulted condition loading and analyses remain applicable and acceptable for the 3455 MWt power level. Existing fuel rod transient strain limits are shown to be applicable. Sufficient fuel rod creep collapse margin exists up to 65,000 MWD/MTU. Although Zircaloy-4 fuel rod cladding corrosion remains limiting, fuel rod corrosion should remain acceptable at the 3455 MWt power level for the present cycle designs. Future fuel designs utilizing M5™ components will improve the safety margins compared to the Zircaloy-4 components, due to the inherent lower corrosion of the M5™ alloy.

3.3 SAFETY ANALYSIS AND THERMAL-HYDRAULIC EVALUATION

After some preliminary comments regarding key input parameters of the safety analyses, each accident evaluation is described. A brief description of the event is given first, followed by its categorization in the Standard Review Plan (SRP), and the acceptance criteria the event must meet. The key parameters and the changes in these caused by the power uprate are given next, followed by the evaluation of the event. The justification for the acceptability is provided in each case. In most cases, the FSAR analysis was performed at a power level at or above a core power of 3455 MWt. The order chosen is consistent with the order that accidents are presented in the FSAR.

3.3.1 Initial Power Assumptions

The current FSAR guaranteed core thermal power is 3411 MWt with an additional 12 MWt generated by the RCPs. Where initial power operating conditions are assumed in accident analyses, the guaranteed NSSS thermal power output (core power plus RCP power), 3423 MWt, plus allowance for errors in steady-state power determination is assumed. Where demonstration of adequacy of the containment and ESFs are concerned, the ESFs design rating (which includes RCP power) plus allowance for error is assumed.

The initial core power assumption is also modeled indirectly in the OPΔT, OTΔT, and high neutron flux trip setpoints as described in the next section.

3.3.2 Reactor Trip System Setpoints

The reactor trip system (RTS) maintains the reactor in a safe operating region by tripping the reactor whenever a safety limit is approached. The RTS maintains surveillance on several process variables that are directly related to equipment mechanical limitations and reactor heat transfer capability. Whenever a process or calculated variable exceeds a setpoint, the reactor is shut down to prevent damage to fuel cladding or loss of system integrity.

Three RTS trips are used extensively in the FSAR Chapter 15 safety analyses. The power range high/low flux, OPΔT, and OTΔT trips are modeled for various transients and have been evaluated for the power uprate. High and Low power range neutron flux trip setpoints allow for a 2-percent calorimetric uncertainty, bounding the lower uncertainty (0.7-percent) associated with the power uprate. For the power uprate condition, the safety analysis high flux trip setpoint will be redefined to be 116.5 percent of 3455 MWt. This value is equivalent, in terms of total megawatts, to the current licensing basis at 3411 MWt (i.e., 116.5 percent of 3455 MWt = 118-percent of 3411 MWt = 4025 MWt) and pre-empts the necessity of additional analysis. An evaluation of the existing accuracy calculations for the high flux trip indicates that the current Technical Specification setpoints (respective trip setpoint and allowable value of 109.0 percent and 111.4 percent of RTP) do not need to be changed. There is adequate existing margin to the trip to accommodate the power uprate. However, the margin between the actual and allowable uncertainties will be reduced as a result. The safety analysis low flux trip setpoint will be similarly redefined without a subsequent Technical Specification change.

The OPΔT and OTΔT trips do not include a calorimetric uncertainty directly. However in the FRA-ANP safety analyses, the ΔT_0 used to model the OPΔT and OTΔT trips is the ΔT_0 at 102 percent of RTP (3479 MWt). After the power uprate, the OPΔT and OTΔT trips will be modeled with the ΔT_0 at 100.7-percent RTP (3479 MWt). Thus, the high flux trip setpoint and the modeling of the transients are applicable for the power uprate.

FRA-ANP reassessed the OPΔT and OTΔT limit constants ("K" terms) currently in use at 3411 MWt. The current limit constants were found to be bounding for 3455 MWt. The limit lines formed by these constants will change slightly due to the change in ΔT_0 from 3411 MWt to 3455 MWt. FRA-ANP recommends no changes in the OPΔT/OTΔT constants currently in use.

3.3.3 BWCMV-A CHF Correlation

Departure from nucleate boiling ratio (DNBR) predictions performed for the power uprate used the BWCMV-A CHF correlation. The BWCMV-A correlation has been specifically developed to reflect the high critical heat flux (CHF) performance of the Mark-BW mixing vane grids. This results in increased margin to thermal limits during accident analyses.

3.3.4 Statistical Core Design Methodology

FRA-ANP uses a method called statistical core design (SCD) that generates a statistical design limit (SDL) that treats the thermal-hydraulic uncertainties statistically in a manner that assumes that all the worst-case conditions are unlikely to occur simultaneously. The SDL for the Mark-BW fuel design at Sequoyah using the BWCMV-A CHF correlation is 1.345. A higher design limit, known as the thermal design limit (TDL), was used for Sequoyah Units 1 and 2. The DNB margin between the SDL and the TDL is treated as retained DNB margin that can be used to offset cycle-specific penalties (e.g., transition core penalties).

This methodology incorporates a 2-percent power uncertainty. The new LEFM equipment allows a reduction in the power uncertainty to 0.7 percent. FRA-ANP has determined that recalculating the SDL with the reduced power uncertainty would reduce the SDL to approximately 1.34. Since the benefits of such a small increase in margin do not outweigh the cost of recalculating the SDL, the SDL of 1.345 with a 2-percent power uncertainty has been conservatively retained.

3.3.5 Core Safety Limits

FRA-ANP reassessed the core safety limits (CSLs) and determined that the current CSLs implemented in the Sequoyah Technical Specifications are bounding at 3455 MWt. No changes in the CSL curves are recommended.

3.3.6 Condition I Events

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Since Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation. Since Condition I events are not "fault" conditions, they are not evaluated by FRA-ANP for the effect of the power uprate at Sequoyah Units 1 and 2. A control/protection system interaction study has been performed to confirm that the reactor control system will not challenge any reactor protection system setpoints during normal operation at the uprated power. (See Section 2.3.1.6).

3.3.7 Condition II Events

Condition II events are expected to occur with moderate frequency at the plant. These faults at worst result in a reactor shutdown with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate and cause a more serious fault, i.e., Condition III or IV category. In addition, Condition II events are not expected to result in fuel rod failures or RCS overpressurization.

3.3.7.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." An RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs resulting in a power excursion. The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. The RCS pressurizes until the reactor trips on power range neutron flux level (low setting).

The following safety concerns exist for this event:

- Reactivity and power excursion
- RCS pressurization
- Reduced margin to DNB

The analysis of this event was performed at hot zero-power (HZP) conditions so the uprated power level has no direct effect on the core power assumption. However, the core power assumption is indirectly modeled as part of the power range high neutron flux setpoint (low setting). Following the power uprate, the power range high neutron flux setpoint (high and low settings) will remain such that the reactor trips at the same value in absolute megawatts. Some additional parameters that affect the plant response to the RCCA withdrawal from subcritical conditions are reactivity insertion due to rod motion, initial axial power distribution, moderator temperature reactivity coefficient, and Doppler reactivity coefficient. None of these parameters are affected by the increase in power level.

The Uncontrolled RCCA Withdrawal From a Subcritical Condition analysis documented in Section 15.2.1 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." An uncontrolled withdrawal of an RCCA bank when the reactor is at full power results in an increase in core heat flux. The primary-to-secondary heat transfer rate lags behind the core heat generation rate causing the temperature of the reactor coolant to increase. The rising temperature would eventually result in DNB unless terminated manually or automatically. The reactor protection system automatically terminates this transient before the DNBR falls below the safety limit. For a rapid RCCA withdrawal starting from 102-percent of RTP, the reactor trips on high neutron flux very early in the transient and maintains a large margin to DNB. For a slow RCCA withdrawal starting from 102-percent RTP, the reactor trips on OTΔT and still maintains acceptable margin to DNB.

The following safety concerns exist for this event:

- Reactivity and power excursion
- RCS pressurization
- Reduced margin to DNB

The uncontrolled rod withdrawal at power is initiated at RTP plus 2-percent measurement uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3-percent and equipment changes that reduce measurement uncertainty to 0.7-percent. The reactor trip on high neutron flux is unaffected by the power uprate because the power range high neutron flux setpoint (high and low settings) will remain such that the reactor trips at the same value in absolute megawatts. The reactor trip on OTΔT is also unaffected by the power uprate because the ΔT_0 used in the modeling of the OTΔT setpoint is the ΔT_0 at 102-percent power.

Although the system response for this event was not re-analyzed, the minimum DNBR (MDNBR) was recalculated using the revised thermal power level and calorimetric uncertainty because the increase in RTP has an adverse effect on the MDNBR. The limiting cases were re-analyzed to verify that the design basis is still met. The analyses show that the MDNBR is still at an acceptable level.

The Uncontrolled RCCA Withdrawal at Power analysis documented in Section 15.2.2 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.3 Rod Cluster Control Assembly Misalignment

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." Rod cluster control assembly misalignment accidents include: (1) a dropped full-length assembly, (2) a dropped full-length assembly bank, and (3) a statically misaligned full-length assembly. In the limiting case of one or more dropped RCCAs in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action, after which the control system will insert the control bank to restore nominal power.

The following safety concerns exist for this event:

- Reactivity and power excursion
- Reduced margin to DNB
- RCS pressurization

The dropped RCCA event is analyzed to assure that DNB does not occur. The dropped RCCA event is hypothesized to overshoot normal 100-percent power conditions under asymmetric peaking conditions. Asymmetric peaking is not accounted for in the safety limits. The overshoot occurs because the control rod controller withdraws bank D when it senses both core average temperature and neutron flux are low. The low flux from the quadrant having the lowest power (dropped RCCA quadrant) is assumed to be used by the rod controller even though the controller samples the largest detector flux. Bank D will continue to withdraw until nominal core average temperature is achieved or the indicated 100-percent power flux is achieved (or the actual 118-percent power trip setpoint is achieved).

The core is assumed to trip at 118-percent power, either by high flux or by the OPΔT setpoint. The 118-percent power level is the maximum credible power level based on the safety limits. For the power uprate, the trip setpoint has been redefined to be 116.5-percent power so that the original safety analysis remains valid, i.e., 116.5-percent of 3455 MWt equals 118-percent of 3411 MWt. The power level is also limited by the reactivity balance between the dropped RCCA(s) worth, the inserted bank D rod worth, and the power coefficient. NEMO core calculations were performed for all the uniquely possible dropped RCCA configurations, including dropped banks, at limiting xenon and time-in-life conditions, to determine the maximum peaking and power level. The MDNBR margin for the uprated conditions was then determined for each dropped RCCA configuration and confirmed to be below the limiting criteria. Therefore the dropped RCCA event continues to meet the limiting criteria and the cycle-by-cycle checks with the uprated DNBR models remains valid.

With respect to system pressurization, the Loss of External Electrical Load (LOEL) is most limiting. System pressures resulting from a dropped RCCA are well bounded by the results of the LOEL.

A statically misaligned single RCCA of control bank D was also evaluated. Previous reload cycle evaluations have demonstrated that the dropped RCCA peaking margins are always significantly more limiting that the misaligned single RCCA. Since the dropped RCCA event is acceptable, the misaligned single RCCA event is also acceptable. Therefore, the RCCA Misalignment analysis documented in Section 15.2.3 of the FSAR remains applicable following the power uprate.

The RCCA Misalignment event has been evaluated relative to the proposed power level uprate. All relevant acceptance criteria for the event continue to be met. In addition, specific checks are performed for each fuel cycle, verifying continued margin to DNB.

3.3.7.4 Uncontrolled Boron Dilution

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." An uncontrolled boron dilution is a consequence of injecting non-borated water (primary grade water) into the RCS via the reactor makeup portion of the CVCS. The boron dilution event adds positive reactivity to the core that could result in power excursion and challenge core thermal margins.

The following safety concern exists for this event:

- Loss of shutdown margin
- RCS pressurization

The FSAR analysis considers a boron dilution event to occur during reactor refueling, startup, and full-power operation. This event evolves slowly and resembles a low rod worth withdrawal event. The parameters that dominate in the plant response to this event are the dilution flow rate, initial boron concentration, final boron concentration, boron worth, and RCS volume. The power level increase does not measurably affect any of these parameters. Concerning system pressurization limits, the results associated with the LOEL event bound those that could be postulated for the boron dilution event. Each reload fuel cycle design is evaluated for acceptability by reviewing the predicted cycle values of boron concentrations and reactivity worth.

The Uncontrolled Boron Dilution analysis documented in Section 15.2.4 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.5 Partial Loss of Forced Reactor Coolant Flow

This event is categorized in the SRP under the heading, "Decrease in Reactor Coolant System Flow Rate." A partial loss of coolant flow accident can result from a mechanical or electrical failure in an RCP or a fault in the pump power supply. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

The following safety concerns exist for this event:

- RCS pressurization
- Reduced margin to DNB as a result of increased fluid temperature and core flow reduction

The FSAR analysis considers loss of two of the four RCPs from 102-percent of RTP, which includes the 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent. Furthermore, this event is bounded by the complete loss of flow accident, FSAR Section 15.3.4. Even though the complete loss of flow is a Condition III event, it is analyzed to Condition II acceptance criteria. The DNB margin associated with the complete loss of flow is much smaller than that associated with the partial loss of flow because of the more severe reduction in core flow. Because the MDNBR for the complete loss of flow is acceptable for a 1.3-percent power uprate, the partial loss of flow event is also acceptable.

Because this event is bounded by the Complete Loss of Forced Reactor Coolant Flow, an evaluation was unnecessary. The evaluation for the Complete Loss of Forced Reactor Coolant flow is contained in Section 3.3.8.4 of this report.

3.3.7.6 Startup of an Inactive Reactor Coolant Loop

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." The Sequoyah Technical Specifications require that all reactor coolant loops be in operation during plant startup and power operations. However, an analysis was performed in the FSAR with the event initiated at partial power, three RCPs operating, and an inadvertent startup of the inactive pump. Startup of an idle RCP without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a rapid reactivity insertion and subsequent power increase.

The FSAR analysis of this event was performed at 72-percent power that includes a 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3-percent and equipment changes that reduce measurement uncertainty to 0.7 percent. The important parameter for this event is not reactor power, but reactor coolant temperature and is not affected by the power uprate.

The Startup of an Inactive Reactor Coolant Loop analysis documented in Section 15.2.6 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.7 Loss of External Electrical Load and/or Turbine Trip

This event is categorized in the SRP under the heading, "Decrease in Heat Removal by the Secondary System." Major load loss on the plant can result from loss of external electrical load or from a turbine trip. Power to plant components is available and RCPs continue to operate. The loss of load results in an abrupt reduction in RCS heat sink and subsequent RCS heatup. The RCS heat removal is established when the secondary safety valves open. On the primary side, the pressure will increase and the pressurizer safety valves will open to maintain the pressure within the design limits. TheRTS will act to trip the reactor prior to a loss of DNB margin.

The following safety concerns exist for this event:

- RCS pressurization
- Reduced margin to DNB

The MDNBR for a total loss of load transient is bounded by the value calculated for a complete loss of forced reactor coolant flow. Consequently, the analysis of total loss of load is performed to show the adequacy of the pressure relieving devices on the primary and secondary systems. The FSAR analysis includes two loss-of-load cases—one at 102-percent power and one at 52-percent power. Both of these cases include a 2-percent calorimetric uncertainty in the initial core power assumption. The current analysis, therefore, can accommodate a power uprate of 1.3-percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Loss of External Electrical Load and/or Turbine Trip analysis documented in Section 15.2.7 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.8 Loss of Normal Feedwater

This event is categorized in the SRP under the heading, "Decrease in Heat Removal by the Secondary System." A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. As the secondary liquid inventory is depleted, the RCS heats up and the primary system pressurizes. If the reactor is not promptly tripped during this accident, primary plant damage could occur due to the loss of heat sink.

The following safety concerns exist for this event:

- RCS pressurization
- Reduced margin to DNB
- Sufficient long-term cooling capacity
- Reduced margin to pressurizer fill

The RCS and steam generator secondary pressure responses for a loss of normal feedwater are bounded by the LOEL event. The peak RCS pressure is less than that for the LOEL because the reactor and turbine trip at about the same time for a loss of feedwater, minimizing the mismatch in primary-to-secondary heat transfer. The peak secondary pressure is less than that for the LOEL because, unlike the LOEL event, the reactor trips at the time of turbine trip, providing less heat transfer to the secondary. The RCPs are assumed to continue to operate for the duration of the loss of normal feedwater transient. The complete loss of reactor coolant flow event, which assumes immediate pump trip from an at-power plant configuration, therefore, bounds the loss of normal feedwater with respect to DNBR. The FSAR analysis of this event was performed at 102-percent of NSSS thermal power, including 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3-percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Loss of Normal Feedwater analysis documented in Section 15.2.8 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.9 Loss of Offsite Power to the Station Auxiliaries

This event is categorized in the SRP under the heading, "Decrease in Heat Removal by the Secondary System." This event is initiated by a complete loss of normal feedwater resulting from pump failures or valve malfunctions with a subsequent LOOP. As secondary liquid inventory is depleted, the RCS heats up and the primary system pressurizes. If the reactor is not promptly tripped during this accident, primary plant damage could occur due to the loss of heat sink.

The following safety concerns exist for this event:

- Increase in RCS temperature
- RCS pressurization
- Reduced margin to DNB
- Sufficient long-term cooling capacity under natural circulation RCS flow conditions
- Reduced margin to pressurizer fill

Prior to reactor trip, the LOOP to the station auxiliaries event is identical to the loss of feedwater event. At reactor trip, the RCPs trip. The remainder of the event tests the capability of auxiliary feedwater to remove decay heat via natural circulation RCS flow. The RCS and steam generator secondary pressure responses for a LOOP to the station auxiliaries are bounded by the LOEL event. The peak RCS pressure is less than that for the LOEL because the reactor and turbine trip at about the same time for a LOOP to the station auxiliaries, minimizing the mismatch in primary-to-secondary heat transfer. The peak secondary pressure is less than that for the LOEL because, unlike the LOEL event, the reactor trips at the time of turbine trip, providing less heat transfer to the secondary. Both the reactor and RCPs trip coincidentally in the LOOP event. In comparison, the pumps trip prior to reactor trip in the complete loss of coolant flow event. Since the power-to-flow ratio is greater for the latter event, DNB margins associated with a LOOP are bounded by those predicted for a complete loss of coolant flow event. The FSAR analysis for this event was performed at 102 percent of RTP, which includes the 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Loss of Offsite Power to the Station Auxiliaries analysis documented in Section 15.2.9 of the FSAR has been evaluated and determined to remain applicable following the power uprate. The environmental consequence analysis of this event is contained in FSAR Section 15.5.1 and evaluated in Section 3.3.10.1 of this report.

3.3.7.10 Excessive Heat Removal Due to Feedwater System Malfunctions

This event is categorized in the SRP under the heading, "Increase in Heat Removal by the Secondary System." Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Excessive feedwater flow could be caused by a full opening of one or more feedwater regulator valves due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generators. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and, thus, a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

The following safety concerns exist for this event:

- Reduction in RCS temperature, causing an increased core power
- RCS pressurization (note that, since this is an overcooling event, RCS pressurization is not really a consideration and the RCS pressure boundary is not threatened)
- Reduced margin to DNB

The RCS and steam generator secondary pressure responses for a feedwater malfunction event are bounded by the LOEL event. The peak RCS pressure is less than that for the LOEL because the reactor and turbine trip at about the same time for a feedwater malfunction event, minimizing the mismatch in primary-to-secondary heat transfer. The peak secondary pressure is less than that for the LOEL because, unlike the LOEL event, the reactor trips at the time of turbine trip, providing less heat transfer to the secondary.

The FSAR analysis of this event considers both zero- and full-power cases of feedwater malfunctions. The zero-power analyses are unaffected by the power uprate. The full-power analyses assumed an initial core power level of 102 percent of 3411 MWt, including a 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Excessive Heat Removal Due to Feedwater System Malfunctions analysis documented in Section 15.2.10 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.11 Excessive Load Increase

This event is categorized in the SRP under the heading, "Increase in Heat Removal by the Secondary System." This event is characterized by a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control. The cooling of the reactor primary system fluid causes an increase in reactor power due to negative end-of-cycle moderator temperature coefficient and causes a reduction in primary system pressure due to the contraction of the reactor coolant. The increase in power and decrease in primary system pressure produce a reduction in DNBR.

The following safety concerns exist for this event:

- Reduction in RCS temperature, causing an increased core power

- RCS pressurization (note that, since this is an overcooling event, RCS pressurization is not really a consideration and the RCS pressure boundary is not threatened)

- Reduced margin to DNB

The FSAR analysis includes four cases to demonstrate the plant behavior following a 10-percent step load increase from rated load: (1) manually controlled reactor at beginning-of-life, (2) manually controlled reactor at end-of-life, (3) reactor in automatic control at beginning-of-life, and (4) reactor in automatic control at end-of-life. The excessive load causes an overcooling and depressurization of the RCS and steam generator systems so the system design pressures are not challenged. The analysis assumes an initial core power of 102 percent of RTP, which includes the 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Excessive Load Increase analysis documented in Section 15.2.11 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.12 Accidental Depressurization of the Reactor Coolant System

This event is categorized in the SRP under the heading, "Decrease in Reactor Coolant Inventory." The accidental depressurization of the RCS is initiated by the inadvertent opening of a pressurizer safety valve or by the failure of a valve to close following an overpressurization transient. The event can cause a reduction in reactor coolant inventory and subsequent reduction in RCS pressure. If the valve is not closed, the continuing depressurization leads to a reactor trip on low RCS pressure or OTΔT.

The following safety concern exists for this event:

- Reduced margin to DNB

The plant response to an accidental RCS depressurization event is dictated by the RCS pressure, RCS temperature, moderator density, Doppler reactivity feedback, and pressurizer safety valve capacity. None of these parameters are affected by the power uprate. The FSAR analysis conservatively used an initial core power assumption of 102 percent of RTP, including 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent. The analysis of this event modeled a reactor trip on OTΔT. The reactor trip on OTΔT is unaffected by the power uprate because the initial core power assumption of 102 percent of RTP was indirectly modeled in the trip setpoint.

The Accidental Depressurization of the Reactor Coolant System analysis documented in Section 15.2.12 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.13 Accidental Depressurization of the Main Steam System

This event is categorized in the SRP under the heading, "Increase in Heat Removal by the Secondary System." The spurious opening or failure of a steam generator relief, safety, or steam dump valve represents the most severe overcooling moderate frequency event. Increased steam flow resulting from the opened valve results in a depressurization of the secondary system with an attendant reduction in RCS temperature and pressure. The RCS cooling could produce a positive reactivity insertion via moderator feedback and a power increase that could challenge the fuel thermal limits. Because the primary and secondary systems depressurize during this event, the system design pressures are not challenged.

The following safety concerns exist for this event:

- Reduction in RCS temperature, causing an increased core power
- RCS pressurization (note that, since this is an overcooling event, RCS pressurization is not really a consideration and the RCS pressure boundary is not threatened)
- Reduced margin to DNB

The accidental depressurization of the MSS event analysis is initiated at HZP so the initial power assumption is unaffected by the power uprate. The critical parameters that affect the system and core responses to this event are relief, safety, or steam dump valve flow capacity and core reactivity coefficients. None of these parameters are affected by the power uprate. Furthermore, this event is initiated from the same operational conditions as the main steam line break documented in FSAR Section 15.4.2.1. Modeling considerations, such as heat removal capabilities and reactivity feedback, are also identical. The main steam line break, however, presents a much greater overcooling event and resulting power excursion. In addition, the larger break results in a more extensive reduction in RCS pressure. Even though the main steam line break is a Condition IV event, it is analyzed to Condition II acceptance criteria. This allows for a relative comparison of the accidental depressurization and main steam line break event, in terms of margin to acceptance criteria. The DNB margin associated with the large break is much smaller than that associated with the accidental depressurization both because of the higher core power and the lower RCS pressure. The large break, in effect, bounds the smaller. Therefore, no fuel pins will experience DNB for the failure of a steam generator relief, safety, or steam dump valve.

The Accidental Depressurization of the Main Steam System analysis documented in Section 15.2.13 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.7.14 Spurious Operation of the Safety Injection System at Power

This event is categorized in the SRP under the heading, "Increase in Reactor Coolant Inventory." An error by the operator or a false actuation signal could produce spurious operation of the ECCS during full-power operation. The actuation of the SIS will result in delivery of highly borated water to the RCS. A reactor trip on the actuation of a spurious SI actuation cannot be guaranteed. A spurious SI, therefore, is capable of causing a negative reactivity excursion and power reduction. Two cases of spurious SI were analyzed: (1) reactor trip at the same time spurious injection starts and (2)RTS produces reactor trip later in the transient.

The following safety concerns exist for this event:

- Reactivity and power excursion
- RCS pressurization

- Reduced margin to DNB

Boration of the RCS, resulting from a spurious SI, causes a reduction in core power. A reduction in RCS temperature and pressure occurs as a result of the core heat production and secondary heat removal mismatch. Secondary steam flow is reduced and the secondary pressure drops relative to a reduction in steaming rate. The primary and secondary system design pressure limits are not challenged. The FSAR analysis conservatively assumed an initial core power of 102 percent of RTP, which includes a 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent. The FSAR analysis shows that the margin to DNB is never reduced below the initial DNB margin for the duration of the SI event because the core power decreases due to boron addition. This result will remain applicable following the power uprate.

The Spurious Operation of the Safety Injection System at Power analysis documented in Section 15.2.14 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.8 Condition III Events

By definition, Condition III occurrences are faults that may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault nor result in a consequential loss of function of the RCS or containment barriers.

3.3.8.1 Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes, Which Actuates Emergency Core Cooling System

This event is categorized in the SRP under the heading, "Decrease in Reactor Coolant Inventory." A small-break LOCA, by definition, is a break of a small pipe or a crack in a larger pipe that can trigger the emergency safety feature actuation system. The break flow for this event cannot be accommodated by the normal makeup system. A small break causes a reduction in liquid inventory and a depressurization of the RCS. Pumped ECCS injection, and the possible injection of the passive cold leg accumulators, is required to mitigate the event and prevent prolonged core uncovery. This event is actually analyzed as a spectrum of break sizes larger than breaks that are adequately mitigated with normal makeup.

The following safety concerns exist for this event:

- Potential for core uncovery
- Fuel heatup, cladding oxidation, and potential fuel failure
- Capability of long-term cooling

The FSAR analysis of this event assumes that, at the time of break initiation, the plant is operating at 102 percent of full power, including a 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent. Furthermore, the FSAR analysis shows that the small-break LOCA is not limiting with respect to large-break LOCA results. This result will continue to be applicable following the power uprate.

The Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes, Which Actuates Emergency Core Cooling System analysis documented in Section 15.3.1 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.8.2 Minor Secondary-System Pipe Breaks

This event is categorized in the SRP under the heading, "Increase in Heat Removal by the Secondary System." The rupture of the steam system with a corresponding steam release rate equivalent to a 6-inch diameter break or smaller initiates this event. These breaks must be accommodated with a limited failure of fuel elements.

This event is not analyzed for Sequoyah Units 1 and 2 in the FSAR. The text, instead, defers to the more limiting steam line break, the Major Secondary System Pipe Break event documented in FSAR Section 15.4.2.1. Although the major secondary breaks are Condition IV events, they are analyzed to Condition II acceptance criteria. Assurance that all of the acceptance criteria are met for the major secondary breaks ensures that all the criteria are met for the minor secondary breaks. The bounding of the minor secondary-system pipe breaks by the major secondary-system pipe breaks will remain applicable following the power uprate.

The discussion of Minor Secondary-System Pipe Breaks documented in Section 15.3.2 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.8.3 Inadvertent Loading of a Fuel Assembly Into an Improper Position

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." The arrangement of assemblies with different fuel enrichments in the core will determine the power distribution of the core during normal operation. The loading of fuel assemblies into improper core positions or the incorrect preparation of the fuel assembly enrichment could alter the power distribution of the core, leading to potentially increased power peaking and possible violation of fuel thermal limits.

Particular misloading scenarios examined in the FSAR include:

- Interchange of two assemblies in an interior core location
- Interchange of a burnable poison rod (BPR) fuel assembly with a non-BPR fuel assembly
- Misplacement of a BPR cluster in a non-BPR fuel assembly
- Enrichment error in an interior core location
- Enrichment error on the core periphery

The following safety concerns exist for this event:

- Core power distortion
- Reduced margin to DNB
- Centerline fuel melt
- Tensile strain limits

The FSAR analyses concluded that fuel misloadings are low probability events, owing to administrative controls regarding fuel pellet loading in a fuel pin, fuel pin loading in an assembly, and fuel assembly manufacture. The analyses also confirm that power distribution effects resulting from misloading events will either (1) be readily detected by the in-core moveable detector system or (2) be of a sufficiently small magnitude to remain acceptable and within the design peaking limits. None of these conclusions will be affected by the power uprate.

The Inadvertent Loading of a Fuel Assembly Into an Improper Position analysis documented in Section 15.3.3 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.8.4 Complete Loss of Forced Reactor Coolant Flow

This event is categorized in the SRP under the heading, "Decrease in Reactor Coolant System Flow Rate." A complete loss of forced reactor coolant flow accident can be caused by a coincident loss of electrical power to all of the RCPs. If the reactor is at power at the time of the accident, the immediate effect of the loss of coolant flow is a rapid increase in coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not properly tripped. The necessary protection against the loss of coolant flow accident is provided by the low primary coolant flow reactor trip signal, which is actuated by redundant low flow signals. Although the original analysis of this event was performed under the assumptions of a Condition III event, as of the Cycle 9 reload, this event is analyzed with Condition II acceptance criteria (no fuel failures).

The following safety concerns exist for this event:

- Increased RCS temperature
- RCS pressurization
- Reduced margin to DNB as a result of increased fluid temperature and core flow reduction

The FSAR analysis of this event assumed a loss of four pumps with four loops in operation. Following the loss of supply to all pumps at power, a reactor trip is actuated by bus undervoltage. The reactor trip on bus undervoltage is unaffected by the power uprate. The complete loss of coolant flow analysis assumed an initial core thermal power of 102 percent of 3411 MWt, which includes 2-percent measurement uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

Although the system response for this event was not re-analyzed, the MDNBR was recalculated using the revised thermal power level and calorimetric uncertainty because the increase in RTP has an adverse effect on the MDNBR. The analysis confirmed that the DNB design basis was satisfied for this accident.

The Complete Loss of Forced Reactor Coolant Flow analysis documented in Section 15.3.4 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.8.5 Waste Gas Decay Tank Rupture

This event is categorized in the SRP under the heading, "Postulated Radioactive Release Due to Liquid-Containing Tank Failures." The waste gas decay tanks contain the gases vented from the RCS, the volume control tank, and the liquid holdup tanks. Sufficient volume is provided in each of four tanks to store the gases evolved during a reactor shutdown. The waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton gases that are stored in the waste gas storage system.

This event is analyzed in the FSAR for environmental consequences in Section 15.5.2. The evaluation of the waste gas decay tank rupture environmental consequences for the power uprate is contained in Section 3.3.10.2 of this report.

3.3.8.6 Single Rod Cluster Control Assembly Withdrawal at Full Power

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." The withdrawal of a single RCCA from its inserted bank results in both a reactivity increase and increased power peaking in the region of the core surrounding the withdrawn RCCA. The reactivity increase causes the neutron flux to increase and produces a localized increase in peaking. Subsequently, thermal power, coolant and fuel temperature, and system pressure increase. Reactor trip on OTΔT provides protection for this event. The peaking asymmetry associated with the withdrawn RCCA can, however, lead to localized fuel failures.

The following safety concerns exist for this event:

- Reactivity and power excursion
- Failed pins in DNB

Two cases were considered in the FSAR analysis of this event. The first case assumed the reactor was in manual control mode with a continuous withdrawal of a single RCCA. The second FSAR case assumed the reactor was in automatic control mode with a withdrawal of a single RCCA resulting in the immobility of the other RCCAs in the controlling bank. The RCCA bank withdrawal at power analyses consists of a spectrum of withdrawal rates (reactivity per time) that encompass that expected from a single RCCA. Therefore, in terms of core power and RCS response, the consequences of a single RCCA withdrawal event are properly characterized by the RCCA bank withdrawal analyses.

The FSAR analysis of this event assumes an initial core thermal power equal to 100-percent power. However, a pin census is conducted during each reload safety evaluation that conservatively estimates the number of fuel rods that experience DNB as a result of a single RCCA withdrawal event. The maximum peaking associated with a steady-state, pre-transient core is determined. A peaking map is then developed for the core, simulating the full withdrawal of the maximum worth assembly at full power with the bank D RCCAs at the insertion limit. The pins exhibiting power peaks in excess of the maximum peaking of the steady-state core are considered to be in DNB. Subsequent DNB checks will be made for this event, using actual fuel cycle designs to confirm that less than 5 percent of the fuel rods experience a DNBR less than the limit value. This will allow the FSAR analysis to remain applicable following the power uprate.

Although the system response for this event was not re-analyzed, the current limiting DNBR case for this event was re-analyzed to confirm that the 1.3-percent power uprate would have a negligible effect. The analysis results showed that the DNBR design basis was satisfied for the limiting case. The results of this analysis were found to be acceptable and the conclusions documented in the FSAR remain valid.

The Single Rod Cluster Control Assembly Withdrawal at Full Power analysis documented in Section 15.3.6 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.8.7 Steam Line Break Coincident With Rod Withdrawal at Power (SLB c/w RWAP)

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." The steam line break coincident with rod withdrawal occurs at full-power conditions, initiated by a steam line break that causes rod withdrawal to occur. Since a steam line break may occur inside containment in the vicinity of the excore detectors or outside containment in the vicinity of the turbine impulse pressure transmitters, the automatic rod control system may be exposed to an adverse environment. Due to this adverse environment, the high neutron flux and the OTΔT reactor trips are disabled in this event. The OPΔT and low steam line pressure trips provide the necessary protection for the reactor.

The following safety concerns exist for this event:

- Reactivity and power excursion
- RCS pressurization
- Reduced margin to DNB

The FSAR analysis of this event considered a spectrum of steam line break sizes coincident with the withdrawal of control bank D at 102 percent of RTP, which includes the 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent. The FSAR analysis showed that the limiting case with respect to MDNBR was for a 0.5825 ft^2 break that tripped on OPΔT. The reactor trip on OPΔT is unaffected by the power uprate because the initial core power assumption of 102 percent of 3411 MWt is indirectly modeled in the OPΔT trip setpoint.

The reactivity assumption associated with the rod withdrawal was based on the maximum speed of the rod speed controller and the maximum differential rod worth of control bank D at hot full-power (HFP) conditions. The maximum differential rod worth is verified each fuel cycle. Cycle-specific checks are performed for each reload and are very similar to the checks performed for the main steam line break (see FSAR Section 15.4.2.1.2). Sophisticated three-dimensional neutronics calculations are conducted with NEMO. The calculations demonstrate that for the statepoints characterizing the least margin to DNB resulting from the accident, the k-effective predicted by the systems (RELAP5) analysis is conservative.

Although the system response for this event was not re-analyzed, the SCD case for DNBR was re-analyzed using the revised thermal power level and calorimetric uncertainty since the increase in power has an adverse effect. The analysis shows that there is no DNBR concern for a rod withdrawal accident coincident with a steam line break for the uprate conditions.

The Steam Line Break Coincident With Rod Withdrawal at Power analysis documented in Section 15.3.7 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.9 Condition IV Events

Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic events which must be designed against and, thus, represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the ECCS and the containment.

3.3.9.1 Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

This event is categorized in the SRP under the heading, "Decrease in Reactor Coolant Inventory." Large-break LOCAs are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system. This accident is due to breaks in pipes of the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS.

The following safety concerns exist for this event:

- RCS depressurization
- Core uncovery
- Fuel heatup, cladding oxidation, and fuel failure
- Capability of long-term core cooling

The FSAR analysis of the large-break LOCA event is initiated at full power, equivalent to the RTP plus 2-percent for measurement uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The previous set of LOCA peaking limits are based on 100 percent of RTP (3411 MWt), and the LOCA analysis used 102 percent ofRTP (3479 MWt). Therefore, a 2-percent margin was retained in the LOCA analysis. With the power uprate, the new LOCA limits will be based on 100 percent of RTP (3455 MWt), and the existing LOCA analysis, which was performed at 100.7 percent ofRTP (3479 MWt), will still be applicable. A margin of 0.7 percent will be retained instead of a 2-percent margin and the existing LOCA analysis will remain bounding for the power uprate.

The Major Reactor Coolant System Pipe Ruptures analysis documented in Section 15.4.1 of the FSAR has been evaluated and determined to remain applicable following the power uprate. The environmental consequence analysis of this event is contained in FSAR Section 15.5.3 and evaluated in Section 3.3.10.3 of this report.

3.3.9.2 Major Secondary System Pipe Rupture - Rupture of a Main Steam Line

This event is categorized in the SRP under the heading, "Increase in Heat Removal by the Secondary System." A major secondary system pipe rupture is generally defined as a guillotine break of the main steam line. A steam line break results in the blowdown of the affected steam generator and severe overcooling of the primary system. The event is initiated from HZP, the worst operational mode for overcooling. With a negative moderator temperature reactivity coefficient, the primary system cooldown results in a reduction in core shutdown margin and possible return to power. If the most reactive control rod is assumed stuck in its fully withdrawn position, the core can become critical and return to power. The return to power, with the large local flux peak in the region of the stuck control rod, could result in fuel pins experiencing DNB. Even though this is a Condition IV event, it is successfully analyzed to the acceptance criteria of a Condition II event - no fuel failures are predicted.

The consequences of this event are directly affected by the magnitude of the return to power, and the severity of the localized peaking. The magnitude of the return to power is governed by a reactivity balance between the positive reactivity inserted by moderator cooldown, and the negative reactivity due to Doppler feedback and control rod insertion. These parameters were not significantly affected by the increase in power level. Also, localized peaking was not significantly affected by the uprate to 3455 MWt.

The following safety concerns exist for this event:

- Reduction in RCS temperature, causing an increased core power
- RCS pressurization (note that, since this is an overcooling event, RCS pressurization is not really a consideration and the RCS pressure boundary is not threatened)
- Reduced margin to DNB

The main steam line break event results in a cooling and depressurization of both the secondary system and the primary RCS. The event does not, therefore, pose a threat to system pressure limits. The LOEL event remains bounding for system pressure effects. The critical parameters that affect the system and core responses to the steam line break are heat transfer surface area of the steam generator tubes and break size. Neither of these parameters is affected by the power uprate. This event is initiated from an HZP condition. Therefore, power uprate considerations are not relevant to the main steam line depressurization event.

The Main Steam Line Rupture analysis documented in Section 15.4.2.1 of the FSAR has been evaluated and determined to remain applicable following the power uprate. The environmental consequence analysis of this event is contained in FSAR Section 15.5.4 and evaluated in Section 3.3.10.4 of this report.

3.3.9.3 Major Secondary System Pipe Rupture - Major Rupture of a Main Feedwater Pipe

This event is categorized in the SRP under the heading, "Decrease in Heat Removal by the Secondary System." A major rupture of a main feedwater pipe represents a rapid decrease in heat removal capability of the secondary system because it reduces the availability of normal feedwater to the steam generators. Cross-connects in the feedwater system will cause most of the feedwater to spill into the containment through the break, without reaching the steam generators. Therefore, main feedwater is assumed to be lost to all steam generators in this event. All steam generators initially blow down through the broken feedwater line. Depletion of secondary inventory will rapidly cause the water level in the steam generators to reach the low-low level trip setpoint. This generates reactor and turbine trips and auxiliary feedwater initiation. Following the reactor trip, the main feedwater line break event is characterized by excess heat removal from the primary as the steam generators continue to blow down. This overcooling phase can result in losing pressurizer liquid level, and impossible voiding in the reactor vessel. When the steam line low-pressure setpoint is reached, the main steam isolation valves close, thus isolating the faulted steam generators from the intact steam generators. The main steam isolation valve actuation terminates the overcooling phase of the event and starts the recovery of secondary steam pressure in the intact steam generators. Consequently, the primary system undergoes a heatup phase, ultimately terminated when the auxiliary feedwater to the intact generators can effectively remove the decay heat. Even though this is a Condition IV event, it is successfully analyzed to the acceptance criteria of a Condition II event - no fuel failures are predicted.

The following safety concerns exist for this event:

- RCS pressurization
- Reduced margin to DNB
- Sufficient long-term cooling capacity
- Reduced margin to pressurizer fill

The feedwater line break is initiated at full power, equivalent to the RTP plus 2 percent for measurement uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Main Feedwater Line Break analysis documented in Section 15.4.2.2 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.9.4 Steam Generator Tube Rupture

This event is categorized in the SRP under the heading, "Decrease in Reactor Coolant Inventory." The steam generator tube rupture is a design basis accident that considers postulated failure of a single steam generator tube. After the rupture, the RCS depressurizes via mass transfer from the primary system to the steam generator secondary. The reactor is tripped, main feedwater flow is isolated, and the SIS is actuated on the low-pressurizer pressure reactor protection signal. The primary-system event is effectively terminated when the injected flow of the ECCS matches the rate of coolant loss through the failed steam generator tube. Tube leakage is terminated when the operator depressurizes the primary system below the steam pressure of the affected steam generator.

The following safety concerns exist for this event:

- Mass transfer from the RCS to the Steam System
- Environmental consequences

The steam generator tube rupture event is analyzed for Sequoyah Units 1 and 2 to provide the basis for environmental consequences. A simplistic transient progression is assumed and a conservative estimate is conducted to determine the mass of RCS transferred to the steam generator secondary and the subsequent steam release via steam line safety valves to the environment. The important parameters in the system response to this event are break size, low pressurizer pressure reactor trip and SI setpoints, ECCS operation and capacity, and steam line safety valve setpoint and capacity. None of these parameters are affected by the power uprate. Furthermore, the analysis of this event was performed at 102 percent of RTP including the 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Steam Generator Tube Rupture analysis documented in Section 15.4.3 of the FSAR has been evaluated and determined to remain applicable following the power uprate. The environmental consequence analysis of this event is contained in FSAR Section 15.5.5 and evaluated in Section 3.3.10.5 of this report.

3.3.9.5 Single Reactor Coolant Pump Locked Rotor

This event is categorized in the SRP under the heading, "Decrease in Reactor Coolant System Flow Rate." The locked rotor accident is analyzed considering a postulated seizure of an RCP rotor. The locked rotor is more limiting than an RCP shaft break as it presents a greater resistance to RCS flow. The RCS flow is rapidly reduced as a result of a locked rotor and the reactor trips on low flow. The rapid flow reduction results in a decrease in the DNBR. Following reactor trip and subsequent rod insertion, the DNBR increases.

The following safety concerns exist for this event:

- RCS pressurization
- Fuel failures greater than 10 percent (environmental consequence limit)

The locked rotor event results in loss of reactor coolant flow, with an increase in temperature prior to reactor trip. The FSAR bases assume the event will occur during the most adverse steady-state operating conditions at beginning of cycle when it may be possible to have a positive moderator coefficient, due to high boron concentration, and produce an increase in power prior to reactor trip. Some pin failures are assumed to occur due to exceeding the DNB limit at the locked rotor pressure and temperature conditions.

A pin census is performed on the limiting NEMO power distribution to verify that the number of failed pins remains less than the 10-percent failure limit identified in the FSAR. The results showed a slight increase in the number of pins failed but with sufficient margin to the 10-percent failure limit to ensure that future reload designs at the uprated power will not be affected by the locked rotor DNB limits. The locked rotor is initiated at full power, equivalent to the RTP plus 2 percent for measurement uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Single Reactor Coolant Pump Locked Rotor analysis documented in Section 15.4.4 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.9.6 Fuel Handling Accident

This event is categorized in the SRP under the heading, "Radiological Consequences of a Fuel Handling Accident." This event takes place in the spent fuel pit floor. A spent fuel assembly is dropped on the pit floor and results in the rupture of the cladding of all fuel rods. Because a fuel handling accident takes place outside the RCS, the Sequoyah FSAR does not contain a system analysis for this event. The environmental consequence analysis of this event is contained in FSAR Section 15.5.6 and evaluated in Section 3.3.10.6 of this report.

3.3.9.7 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

This event is categorized in the SRP under the heading, "Reactivity and Power Distribution Anomalies." This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of an RCCA and drive shaft. The result of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The following safety concerns exist for this event:

- Reactivity and power excursion
- RCS pressurization
- Fuel heatup and failure

The primary parameters that affect the fuel pellet enthalpy, RCS pressure, and centerline fuel melting are time in core life, ejected rod worth, and Doppler temperature/power coefficient. These parameters are unaffected by the power uprate. The FSAR analysis of this event considered an RCCA ejection from 0 percent of 3411 MWt HZP and an RCCA ejection from 102 percent of 3411 MWt HFP. The HZP case is unaffected by the power uprate. However, recent reload core evaluations have shown the ejected rod worth and peaking to be close to the limits, and may be adversely affected by minor operational differences at the higher power level, particularly at end of cycle (EOC). The HZP ejected rod calculations were performed at EOC based upon core operation at 3455 MWt and compared to equivalent results at 3411 MWt. There was a very small increase in ejected rod peaking and worth which could be easily accommodated in the reload core design process.

The HFP case included a 2-percent calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7-percent. Cycle-specific checks will also be performed to verify that the RCCA ejection analysis remains applicable for the power uprate. The trip setpoint for the event is unchanged (116.5-percent power of 3455 is equivalent to 118-percent power of 3411 MWt). Therefore the FSAR conclusions remain valid.

Although the system response for this event was not re-analyzed, the MDNBR was recalculated using the revised thermal power level and calorimetric uncertainty because the power uprate adversely affects the SCD DNBR analysis at hot full power. The analysis shows that the limiting case does not result in fuel damage beyond the 10-percent fuel melt limit for the limiting case.

The Rod Cluster Control Assembly Ejection analysis documented in Section 15.4.6 of the FSAR has been evaluated and determined to remain applicable following the power uprate. The environmental consequence analysis of this event is contained in FSAR Section 15.5.7 and evaluated in Section 3.3.10.7 of this report.

3.3.10 Environmental Consequences

This section contains a brief discussion of the FSAR Chapter 15 environmental consequence analyses with respect to the 1.3-percent power uprate. The current FSAR environmental consequence analyses used source terms based on a core thermal power of 105 percent of 3411 MWt or 3582 MWt. The core thermal power level of 3582 MWt used in the current FSAR analyses bounds the uprated core thermal power of 3479 MWt, including calorimetric uncertainty.

3.3.10.1 Loss of AC Power to Plant Auxiliaries

The analysis of the environmental consequences of a postulated loss of AC power to plant auxiliaries is presented in FSAR Section 15.5.1. The key parameters affecting this analysis are primary-to-secondary leakage, primary coolant activity, iodine partition factor, and steam generator blowdown rate. The analysis does not allow variability in the plant response to the transient, but uses conservative values to bound the plant response. None of the key input parameters for this event are affected by the power uprate. The current FSAR analysis of this event used an initial primary coolant activity and core thermal power based on 3582 MWt (105 percent of 3411 MWt). The current analysis can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Loss of AC Power to Plant Auxiliaries environmental consequence analysis documented in Section 15.5.1 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.10.2 Waste Gas Decay Tank Rupture

The analysis of the environmental consequences of a postulated waste gas decay tank rupture is presented in FSAR Section 15.5.2. The FSAR analysis of this event assumes the rupture of a single waste gas decay tank. The parameters important to the dose calculations for a waste gas decay tank rupture are the tank activity concentration and site-specific dispersion factors. Site dispersion factors are a function of meteorology of the site and are not affected by the power uprate. The tank activity assumed at event initiation is conservatively determined, based on the RCS volume and the RCS volume is unaffected by the power uprate. The current FSAR analysis of this event used an initial primary coolant activity and core thermal power based on 3582 MWt (105 percent of 3411 MWt). The current analysis can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent

The Waste Gas Decay Tank Rupture environmental consequence analysis documented in Section 15.5.2 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.10.3 Loss-of-Coolant Accident

The analysis of the environmental consequences of a postulated LOCA is presented in FSAR Section 15.5.3. The analysis assumes a prescribed dose release to the containment based on core fission product inventory. The key parameters affecting this analysis are modeling of the fission product removal process, performance of the ice condenser, primary containment leak rate, auxiliary building ventilation, and performance of the emergency gas treatment system (EGTS). The analysis does not allow variability in the plant response to the transient, but uses conservative values to bound the plant response. None of the key input parameters for this event are affected by the power uprate. The current FSAR analysis of this event used an initial primary coolant activity and core thermal power based on 3582 MWt (105-percent of 3411 MWt). The current analysis can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Loss-of-Coolant Accident environmental consequence analysis documented in Section 15.5.3 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.10.4 Steam Line Break

The analysis of the environmental consequences of a postulated steam line break is presented in FSAR Section 15.5.4. The key parameters affecting this analysis are primary-to-secondary leakage, primary coolant activity, iodine partition factor, and steam generator blowdown rate. The analysis does not allow variability in the plant response to the transient, but uses conservative values to bound the plant response. None of the key input parameters for this event are affected by the power uprate. The current FSAR analysis of this event used an initial primary coolant activity and core thermal power based on 3582 MWt (105 percent of 3411 MWt). The current analysis can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Steam Line Break environmental consequence analysis documented in Section 15.5.4 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.10.5 Steam Generator Tube Rupture

The analysis of the environmental consequences of a postulated steam generator tube rupture is presented in FSAR Section 15.5.5. The key parameters affecting this analysis are primary-to-secondary leakage, primary coolant activity, iodine partition factor, and steam generator blowdown rate. None of the key input parameters for this event are affected by the power uprate. The current FSAR analysis of this event used an initial primary coolant activity and core thermal power based on 3582 MWt (105 percent of 3411 MWt). The current analysis can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

The Steam Generator Tube Rupture environmental consequence analysis documented in Section 15.5.5 of the FSAR has been evaluated and determined to remain applicable following the power uprate.

3.3.10.6 Fuel Handling Accident

The analysis of the environmental consequences of a postulated fuel handling accident is presented in FSAR Section 15.5.6. A fuel handling accident takes place in the spent fuel pit when a spent fuel assembly is dropped on the pit floor and results in the rupture of the cladding of all fuel rods. The analysis of this event considered the damaged assembly to be the highest powered assembly in the core region. The fission gas activity of the damaged assembly was based on full power operation at the end of core life immediately preceding shutdown. The current FSAR analysis of this event used a fission gas activity and core thermal power based on 3582 MWt (105 percent of 3411 MWt). The current analysis can accommodate a power uprate of 1.3 percent and equipment changes that reduce measurement uncertainty to 0.7 percent.

3.3.10.7 Rod Ejection Accident

The analysis of the environmental consequences of a postulated rod ejection accident is presented in FSAR Section 15.5.7. The environmental consequences of a rod ejection accident are bounded by the environmental consequences of a LOCA. Since the LOCA environmental consequence analysis was found to be unaffected by the power uprate, the rod ejection accident environmental consequence analysis is also unaffected.

3.3.11 Conclusions

This section has evaluated the FSAR Chapter 15 safety analyses with respect to a 1.3-percent power uprate. All American Nuclear Society (ANS) Condition II, III, and IV events were discussed with regard to the key parameters affecting the analyses and the role that the power uprate plays, if any, in each accident analysis.

The increase in power level will be accomplished by means of a decrease in the calorimetric uncertainty of the secondary side power measurement. A new main feedwater LEFM System will be installed that is designed to reduce the calorimetric uncertainty from ±2 percent to ±0.7 percent such that the 1.3-percent reduction in measurement uncertainty may be applied to power production. Because of this, all transient analyses that assumed an initial core power of 102 percent or greater were unaffected by the power uprate. In addition, the safety analyses performed at zero-power conditions were also unaffected by the power uprate. The remainder of the Chapter 15 safety analyses were either insensitive to power level considerations or were bounded by other events.

It is the conclusion of this report that the key inputs used in the analysis of the FSAR Chapter 15 events continue to be applicable or bounding for a 1.3-percent power uprate coincident with a 1.3-percent decrease in calorimetric uncertainty. Therefore, there is no requirement that any of these events be re-analyzed.

4.0 LEFM OPERABILITY CONSIDERATIONS

4.1 METHOD FOR PLANT OPERATION WHEN LEFM IS INOPERABLE

Sequoyah Units 1 and 2 will be operated in accordance with the safety analyses and the applicable power calorimetric uncertainty analysis. When the improved LEFM-based calorimetric measurement is available, the plant will be operated at a nominal core power of 3455 MWt. This section provides a suggested method for plant operation when the LEFM becomes inoperable or is not performing as designed, thus ceasing to provide the required accuracy on feedwater flow measurement for input to the heat balance calculation.

The power calorimetric uncertainty is shown to be less than 0.7-percent RTP based on the use of the improved LEFM. However, this uncertainty calculation is not applicable to the case where the power calorimetric is based on venturi-based feedwater flow indication, even if the improved LEFM is used to correct the venturi-based feedwater flow indications for effects such as fouling. The reactor operators will be provided procedural guidance for those occasions when the improved LEFM is not available. As summarized below, for those instances a new section of the Sequoyah Technical Requirements Manual (TRM) will specify the appropriate actions to be taken when the LEFM is unavailable.

The Sequoyah TRM and other appropriate plant procedures will specify that if the LEFM becomes unavailable during the interval between daily performances of the calorimetric heat balance comparison with the nuclear instrumentation system (NIS) (Technical Specification Surveillance Requirement (SR) 4.3.1.1.1), plant operations may remain at a thermal power of 3455 MWt while continuing to use the power indications from the NIS power range channels. However, in order to remain in compliance with the bases for operation at an RTP of 3455 MWt, the LEFM system must be returned to service prior to the performance of SR 4.3.1.1.1. If the LEFM has not been returned to service prior to the performance of SR 4.3.1.1.1, the procedural guidance/TRM would require that the reactor power be reduced to, and maintained at or below, a power level of 3411 MWt prior to performing SR 4.3.1.1.1. This power level is consistent with the uncertainty previously assumed for the venturi-based indication of feedwater flow. The surveillance would then be performed using the venturi-based feedwater flow indications in the case where the LEFM is unavailable. Once SR 4.3.1.1.1 is performed using the corrected venturi-based feedwater flow indications, the assumed power uncertainty is 2-percent RTP even though the actual uncertainty is significantly less than this. In order to maintain compliance with the safety analyses, it would be necessary to operate the plant at a maximum core thermal power of 3411 MWt, until the LEFM is restored. Once the LEFM is restored, performance of SR 4.3.1.1.1 is required prior to increasing thermal power above 3411 MWt using the LEFM indication of feedwater flow. Upon completion of SR 4.3.1.1.1, the plant could again be operated at 3455 MWt.

4.2 ADJUSTMENTS REQUIRED WHEN LEFM IS INOPERABLE OR UNAVAILABLE

Section 4.1 presented an overall framework for continued power operation when the LEFM becomes inoperable or is not performing as designed, thus ceasing to provide the required accuracy on feedwater flow measurement for input to the heat balance calculation. In addition, key core parameters and limits were reviewed to identify additional adjustments necessary to ensure that unit power operation would continue to comply with the safety analyses when the LEFM is not available. The additional adjustments will be specified in the COLR, plant procedures, or other documents, as necessary and appropriate.

The plant nuclear instrumentation must be calibrated at regular intervals (daily) per the requirements of SR 4.3.1.1.1. To maintain the RTP at 3455 MWt during LEFM operability, the maximum allowable thermal power is limited to 98.7 percent of 3455 MWt and NIS power range channel recalibrations are performed. In addition, to preserve margin, the adjustments referred to below are required during the time that the LEFM is not available.

4.2.1 Thermal Power

The TRM will specify that upon loss of the LEFM, power operation may continue at a thermal power of 3455 MWt while continuing to use the power indications from the NIS power range channels. If the LEFM has not been returned to service prior to the performance of SR 4.3.1.1.1, the procedural guidance/TRM would require that reactor power be reduced to 3411 MWt (98.7 percent of 3455 MWt). This power level is consistent with the uncertainty previously assumed for the venturi-based indication of feedwater flow. The power reduction would be performed prior to SR 4.3.1.1.1 being performed. The surveillance would then be performed using the venturi-based feedwater flow indications, and the assumed power uncertainty would be increased from 0.7-percent to 2-percent RTP. Power operation would then continue at a maximum core thermal power of 3411 MWt (98.7 percent of 3455 MWt), until the LEFM is restored. This action preserves the maximum real core power that is assumed as the initial condition for the plant safety analyses.

4.2.2 Nuclear Overpower (High Flux) Trip Setpoint

Based on FRA-ANP's evaluation of the power level uprate, it was determined that the nuclear overpower (high flux) safety analysis trip setpoint would be redefined to be 116.5 percent of 3455 MWt. This preserves the total absolute power in megawatts that forms the current licensing basis (4025 MWt). As noted in Section 3 of this report, an evaluation of the existing accuracy calculations for the high flux trip indicates that the current Technical Specification setpoints (respective trip setpoint and allowable value of 109.0 percent and 111.4 percent ofRTP) remain applicable for operation at 3455 MWt. That is, the Technical Specification setpoint, in percent power, remains the same but is increased in terms of MW thermal. Although reduced, adequate margin between the Technical Specification and analysis trip setpoints was identified to accommodate the power uprate. Preliminary evaluation of the trip margin shows that the setpoint does not have to be adjusted upon loss of the LEFM function, but that the margin is reduced by a return to the original 2-percent calorimetric error.

4.2.3 Overpower Delta-Temperature and Overtemperature Delta-Temperature Trip Setpoints

While not specifically identified as power calorimetric uncertainty values, allowances for power measurement uncertainty are included in the overall instrument channel accuracy calculations for the OPΔT and OTΔT trip functions. The uncertainty allowances for the process measurement accuracy term associated with hot leg temperature streaming, the temperature sensor calibration term, and temperature sensor drift term are included in the demonstrated accuracy calculation to account for a maximum 2-percent power measurement uncertainty. The FRA-ANP safety analyses model the ΔT_0 in the OPΔT and OTΔT trips as the ΔT_o at 102 percent of RTP (3479 MWt). After the power uprate, the OPΔT and OTΔT trips will be modeled with the ΔT_o at 100.7 percent of RTP (3479 MWt). There will be a loss of approximately 1.3-percent margin between the safety limits and the trip limits compared to the margin at an RTP of 3411 MWt. However, as long as margin exists between the safety limits and the unerror adjusted setpoints, the current K factors in the Technical Specifications for the OPΔT and OTΔT trip functions will remain valid for operation at the uprated condition.

FRA-ANP re-assessed the OPΔT and OTΔT limit constants ("K" terms) currently in use at 3411 MWt. The current limit constants were found to be bounding for 3455 MWt. The limit lines formed by these constants will change slightly due to the change in ΔT_o from 3411 MWt to 3455 MWt. Because the error adjustments for the K factor methodology were not reduced from 2.0 percent to 0.7 percent, the K factors remain valid when the LEFM is unavailable. FRA-ANP recommends no changes in the OPΔT/OTΔT constants currently in use. The ultimate responsibility for assuring that the drift error considered in the demonstrated accuracy calculation accommodates either the 0.7-percent or 2.0-percent uncertainty with adequate margin for these trips remains with TVA.

4.2.4 $f_1(\Delta I)$ Limits (OTΔT) and $f_2(\Delta I)$ Limits (OPΔT)

Since the K factors in the OTΔT and OPΔT trip equations contain the error uncertainties for power, no adjustments are needed for application to the $f_1(\Delta I)$ limits (OTΔT) or $f_2(\Delta I)$ limits (OPΔT) when the LEFM is unavailable.

4.2.5 Axial Flux Difference Limits

The AFD limits are based on LOCA and initial condition DNB peaking margins. The LOCA margins at 3411 MWt with 2.0-percent calorimetric uncertainty and 3455 MWt with 0.7-percent calorimetric uncertainty will be the same for the same power distribution. There is no change in the F_Q limit, therefore the LOCA linear heat rate limit is increased by the increase in core average linear heat rate. Since the calculated linear heat rate is increased by the same amount due to the uprate, the effect cancels, and the margin is the same. Thus, the axial offset limit at 98.7 percent of 3455 MWt should be maintained at the same value as 100 percent of 3455 MWt when the LEFM is not being used. This preserves the margin at the same assumed actual core thermal power (.987 X 3455 X 1.02 = 1 .00 X 3455 X 1.007 = 3479 MWt). This means that each point on the AFD limit lines (in terms of offset) should be reduced by the difference between 100-percent and 98.7-percent power (1.3 percent) when the flow meter is not operable. Reducing the allowable power level by 1.3 percent and making the AFD limit lines more restrictive by 1 percent in AFD bounds the required changes.

A secondary effect occurs when the offset is converted to AFD or ΔI. The AFD calculation relies on RTP, however the unit would be operating at a lower thermal power. Therefore, the calculated value of AFD would be slightly smaller in magnitude for the same power distribution. The magnitude of the effect is dependent upon power level and AFD. At full power the error is 0.13 AFD at 10-percent AFD, and at 50-percent power the error is 0.65 at 25-percent AFD. These error variations with AFD and power are well within the conservatisms of the methodology and can be ignored, so that the 1.3-percent reduction in power for the AFD limit lines is sufficient.

4.2.6 Control Rod Insertion Limits

The control rod insertion limits in the COLR preserve power peaking criteria and minimum required shutdown margin. Upon loss of the LEFM function, the allowable power at each point along the insertion limit would be reduced by approximately 1.3 percent. Reducing the allowable power level by 1.3 percent and increasing the rod insertion limit lines by 3 steps withdrawn bounds the required changes.

4.2.7 Interpretation of MONITOR Results

For MONITOR, the LOCA and ICDNB margin factors defined in the analysis for the power uprate would continue to be applicable upon loss of the LEFM. The ICDNB margin factors at power uprate conditions are reduced relative to those at current RTP conditions due to the reduction in DNB MAP limits for the power uprate. Although the design power distribution would change slightly, it is acceptable to operate for a short time without having to revise the design power distribution. Changes to TVA procedures will be required to document the required actions to execute for periods when the LEFM is not available. If an incore flux map is taken during a period when the LEFM is unavailable, the power level entered into the INCORE/MONITOR system should be the NIS power divided by 0.987, i.e. (NIS power / 0.987). Any Technical Specification action to reduce allowable power when the LEFM is not operable should be calculated by [98.7 percent (percent power reduction calculated by MONITOR)], so that the power reduction would be taken from 98.7 percent of 3455 MWt.

4.3 SUMMARY

The actions specified in this section are based upon a reduction of the maximum allowable thermal power from 100 percent of 3455 MWt to 98.7 percent of 3455 MWt without changing the RTP level at the calibration during the period when the LEFM is not operable.

To maintain a RTP of 3455 MWt, adjustments to the COLR AFD limits and control rod insertion limits in addition to the reduction in thermal power are required. TVA will develop procedures for Sequoyah Units 1 and 2 to implement correct actions and interpretations of the core limits and documents. For example, the plant procedures must clearly indicate that 100 percent of RTP would not be the maximum allowable thermal power level when the LEFM is unavailable. In addition, retained margin between the existing trip setpoints and the allowable setpoints will be reduced.

A summary of the actions required upon loss of the LEFM function, as described in Section 4.2, is provided in Table 4-1.

Table 4-1 Treatment of Parameters when LEFM Inoperable		
Section	**Parameter**	**Action**
4.2.1	Thermal Power	Reduce thermal power to 98.7% RTP prior to next performance of SR 4.3.1.1.1.
4.2.2	Nuclear Overpower Trip Setpoint	No adjustments are required.
4.2.3	OPΔT and OTΔT Trip Functions	No adjustments to the K constants for the ΔT trip setpoints are required to support continued power operation. However, the margin between the safety limits and trip limits is reduced as a result of assuming the increase in RTP from 3411 MWt to 3455 MWt, and the drift errors must be assured to be applicable for a 2.0% power calibration uncertainty by TVA.
4.2.4	COLR $f_1(\Delta I)$ Limits (OTΔT) and $f_2(\Delta I)$ Limits (OPΔT)	No adjustments to the $f_1(\Delta I)$ limits (OTΔT) or $f_2(\Delta I)$ limits (OPΔT) are required, since the K factors in the OTΔT and OPΔT trip equations contain the error uncertainties for power. Also, no adjustments to the CFM or SSDNB margin factors in the MONITOR database would be required to support peaking factor surveillance if an incore flux map were required.
4.2.5	COLR AFD Limits	Each point on the AFD limit lines should be reduced by 1.3% in allowable power. In addition, the maximum thermal power should be limited to 98.7% of 3455 MWt (RTP).
4.2.6	COLR Rod Insertion Limits	Each point on the control rod insertion limit lines should be reduced by 1.3% in allowable power. In addition, thermal power should be limited to 98.7% of 3455 MWt RTP.
4.2.7	MONITOR database: LOCA and ICDNB margin factors	No adjustments are required for the LOCA and ICDNB margin factors in the MONITOR database. Assuming the period of time that the unit operates with the LEFM inoperable is limited, no adjustment of the design power distribution would be required. MONITOR results generated at 98.7% of RTP (3455 MWt) should be interpreted as if the unit were operating at 100% of RTP. Changes to the TVA procedures will be required to document the required actions to execute for periods when the LEFM is not available.

Appendix A
RTDP Power Calorimetric Uncertainty

REFER TO ENCLOSURE 4 OF TVA TECHNICAL SPECIFICATION CHANGE REQUEST TS 01-08

Appendix B
Technical Specification Markups and 10 CFR 50.92 Evaluation

REFER TO ENCLOSURES 1 AND 2 OF TVA TECHNICAL SPECIFICATION CHANGE REQUEST TS 01-08

ENCLOSURE 8

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

APPLICABILITY OF COMANCHE PEAK AND WATTS BAR
POWER UPRATE RAIs TO SEQUOYAH 1&2 UPRATE
(NON-PROPRIETARY)

APPLICABILITY OF COMANCHE PEAK AND WATTS BAR POWER UPRATE RAIs TO SEQUOYAH 1&2 UPRATE (NON-PROPRIETARY)

TVA has addressed NRC Staff written questions [Request for additional information (RAIs)] raised in the licensing process for the power uprate license amendments granted to TU Electric for Comanche Peak Unit 2 and to TVA for Watts Bar Nuclear Plant (WBN) Unit 1. This information is incorporated into the Sequoyah Units 1 and 2 license amendment request (i.e., Enclosure 6) where practical. The RAI questions were taken from TU Electric and WBN transmittals to the NRC. The reference number of each RAI question and the associated transmittal letter are provided for cross-reference. A list of the subject TU Electric and TVA WBN transmittal letters and the associated responses is provided below:

Comanche Peak Transmittals:

A. TXX-99105	-	April 23, 1999
B. TXX-99115	-	May 14, 1999 (Attachments 3, 6, and 7)
C. TXX-99195	-	August 13, 1999
D. TXX-99164	-	July 9, 1999
E. TXX-99203	-	August 25, 1999
F. TXX-98274	-	December 17, 1998 (Response to selected questions)

Watts Bar Transmittals:

G. Follow-up submittal dated August 24, 2000
H. Follow-up submittal dated October 6, 2000

A. TXX-99105

Question 1 (TXX-99105):

Provide a discussion that addresses the impact of the proposed power uprate on the load, voltage, and short circuit values for all levels of the station auxiliary electrical distribution system. Include in this discussion any impact on the direct current power systems.

Response:

As a result of this uprate, no dc or ac auxiliary load ratings are expected to change. In addition, the dc loads are not expected to experience additional demands above their ratings but some secondary ac pump motor loads are expected to experience a slight increase in demand. The ac auxiliary loads expected to increase in demand will be the No.3 Heater Drain Tank Pumps, the No. 7 Heater Drain Tank Pumps, the Hotwell Pumps, and the Condensate Booster Pumps. The increase is expected to be less than 0.5% of their current load and remain bounded by the current analysis. Therefore, the plant ac or dc auxiliary electrical loads will not increase above their ratings. The main generator electrical parameters remain the same, and the uprate capacity remains within the generator rating. The voltage controls and grid source impedance at the SQNP 500-kV and 161-kV grid will not be affected by this uprate; therefore, the evaluated voltages and short circuit values at different levels of station auxiliary electrical distribution system will not change as a result of this uprate.

Question 2 (TXX-99105):

For the power uprated conditions, discuss environmental qualification for the safety related electrical equipment located in harsh environmental areas. For this safety-related electrical equipment, address the continued environmental qualification and the process for establishing qualification for any increased temperature, pressure, humidity, and radiation values.

Response:

The normal environments for the plant buildings were assessed. The 1.3% uprate has an insignificant effect on process fluid temperatures in the auxiliary and control buildings. With the exception of the main feedwater, the increase in the heat loads is caused by the increase in the decay heat load as it is transferred to the Component Cooling System and Essential Raw Water Cooling System. The increase in these system temperatures has been evaluated and found to have an insignificant impact. The main feedwater temperature is changing by approximately 1.7°F with the Steam Generators at the maximum plugged tubes of 15%. This small change in fluid temperatures has an insignificant affect on the area temperatures. Similar conclusions were reached following the evaluations of the normal environmental conditions in the containment building.

The post-accident thermal environmental parameters were generated from computer models of the building structures that calculate the environment created by mass and energy releases during postulated pipe breaks. Evaluations concluded that through the use of the reduced 0.7% power calorimetric uncertainty to offset the 1.3% increase in reactor power, the existing mass and energy releases used in the environmental analyses for both inside and outside containment would remain valid. Because the mass and energy releases are not changed, the resulting environments are also unchanged. Therefore, the 1.3% power uprate has no impact on the Sequoyah non-radiological equipment qualification program.

The current design basis was performed in accordance with RG 1.49 which requires the normal power level to be 1.02% of the licensed power. For both post-accident and normal-operating , the Sequoyah source terms were based on 104.5% of the licensed power or greater.

The effects of post-accident radiological consequences on equipment qualification were evaluated. The source term used in the original analyses was generated for operation at a thermal power of 3565 MWt (i.e., 104.5% RTP). Generally, postulated radiation doses impacting equipment qualification depend primarily on post-accident contributions; however, normal-operating dose rate contributions are included in the design basis calculations. These normal-operating contributions are, in all cases, based on source terms which were originally generated for a power level of 105% RTP (i.e., 3582 MWt). Therefore, in regard to cases where equipment qualification dose rate contributions may be significant, it can safely be concluded that a power uprating of 1.3% would not cause dose rates or integrated doses to exceed design basis values.

In summary, the 1.3% thermal power uprate has a negligible effect on normal environmental conditions and no effect on the environmental conditions currently used for equipment qualification.

Question 3 (TXX-99105):

Discuss and verify the assumptions for the station blackout analysis are valid for the power uprate conditions, particularly as they relate to issues such as the heat-up analysis, equipment operability, and battery capacity.

Response:

Equipment

To provide for an orderly and safe cooldown of the unit during and following a Station Blackout (SBO) event, the following conditions must be met:

The turbine driven auxiliary feedwater pump must operate to provide feedwater to the steam generators (SGs), a slight repositioning of the discharge valves (air operated) may be necessary, the main steam safety valves (MSSVs) are utilized to relieve steam generator secondary side pressure to maintain hot standby conditions, the SG power operated relief valves (PORVs) must cycle open to relieve steam for unit cooldown (after 4 hour SBO event), and an adequate supply of water from the condensate storage tank must be available to maintain adequate water level in the steam generators.

Control power from batteries has been provided to the turbine driven auxiliary feedwater pump for operation during an SBO. The turbine is supplied steam via the number 1 or 4 SG. A flow path is ensured from at least one of the two steam generators for pump operation.

The MSSVs have been evaluated and found acceptable - see Enclosure 6, Section 2.3.2.

Specific air operated valves in the main steam system and the auxiliary feedwater system must be able to be operated from accumulators (a backup source of compressed nitrogen has been provided to allow manual control of the LCVs for SBO that have sufficient capacity to cycle the valves as needed during the controlled unit cooldown. In each case, the required number of valve cycles was established independent of and was determined to be reasonably insensitive to the actual power level. Nitrogen bottles have been included with each of these valves to meet operational requirements during the SBO event. The accumulator (i.e., nitrogen bottles) sizes are therefore sufficient to provide a safe cooldown during a SBO event.

An evaluation was performed in which it was concluded that the current minimum available condensate inventory in the condensate storage tank is sufficient for the 1.3% uprate condition (see Enclosure 6, Section 2.3.2.5).

The existing calculations used to demonstrate the capability to withstand an SBO event of four hours duration without uncovering the core were reviewed for the 1.3% uprate conditions. The later stages of the existing analysis credit operator action to maintain the RCS temperature and pressure below specified limits; the SG PORVs are used to accomplish this action. The capacity of the SG PORVs was evaluated and determined to be sufficient to accommodate the 1.3% uprated condition (see Enclosure 6, Section 2.3.2). The conclusions of the calculation remain valid, i.e., the time to uncover the core following a SBO event is greater than four hours.

Environmental

The existing loss of ventilation analyses for an SBO at Sequoyah is a 4 hour transient. The SBO room temperatures in vital areas were calculated using transient heatup computer models. The temperatures identified were the peak temperatures calculated for the 4 hour coping period. Equipment operability was assessed at those peak temperatures and no required operations were compromised by overheating.

The containment environment during a 4 hour SBO event is significantly less than the thermal profiles considered for LOCA/MSLB events. A small increase in decay heat and initial process temperatures cannot result in a change of such magnitude that the calculated LOCA/MSLB environment will be exceeded. Therefore, it was concluded that a small change in RCS temperature, decay heat, main steam and feedwater temperatures would have no effect on the equipment as evaluated for the SBO event.

The primary heat loads in the main steam and feedwater piping areas are from the main steam and feedwater piping. The power uprate results in a lower operating steam temperature at full load and no change to the no-load steam temperature. The slight increase in feedwater temperature realized from the 1.3% uprate would be insignificant during an SBO since feedwater heating would be terminated upon turbine trip.

The primary heat load in the turbine-driven auxiliary feedwater pump room is from the main steam piping feeding the turbine and the turbine casing. The power uprate results in a lower operating steam temperature and no change to the no-load steam temperature. Therefore, the current heat load resulting from the main steam lines bound the expected heat loads following the 1.3% uprate.

Based on the preceding discussions, it is concluded that the small changes in main steam and feedwater temperatures do not adversely impact the environment and equipment previously evaluated for the SBO event.

Battery Capacity

As a result of this uprate, no ac or dc auxiliary load ratings are expected to change, and the loads that would impact the station batteries are not expected to experience additional demands above their ratings. Since the plant auxiliary ac/dc electrical load will not change, there is no impact on the station battery capacity due to the 1.3% uprate.

Question 4 (TXX-99105):

Provide a discussion addressing the impact of the CPSES Unit 2 power uprate on the turbine/generator, isophase bus, main transformers, and switchyards. Address in detail any non hardware changes for these items as a result of the CPSES Unit 2 power uprate.

Response:

Turbine/Generator

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

The Units 1 and 2 steam turbine-driven polyphase generator is a four pole machine rated at 1,356 MVA, with an operating point of 1221 MWe at a 0.9 power factor. This rating is based upon 75 psig hydrogen pressure, which is supplemented with water cooling for the stator and rotor.

At the current thermal rating of Units 1 and 2 of 3411 MWt, the Units 1 and 2 main generator electrical output is typically 1186.3 MWe with an approximately 4 MWe increase periodically observed during the colder winter months. The anticipated net increase of approximately 12 MWe lies well within the nameplate rating of the generator of 1221 Mwe at 0.9 power factor. Therefore there will be no generator limitations to prevent operation at a core power of 3455 MWt.

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

To deliver electrical power provided by the generator to the transmission system, the unit is equipped with an isolated phase bus, a main transformer, and switchyard breakers and switches. The components are rated to deliver electrical power at or in excess of the main generator nameplate rating of 1356 MVA.

Isophase Bus

The isophase bus is designed to standards ANSI/IEEE C37.20 and C37.23, IEEE Guide for Metal-Enclosed Bus and Calculating Losses in Isolated Phase Bus, with a forced cooling rating of 34,300 amps (along the main bus section) and forced cooled rating of 19,800 amps each phase (at the generator and transformer terminals). These ratings are greater than the Units 1 and 2 Main Generator rating of 32,625 stator amps at 1356 MVA and are well in excess of the anticipated generator output. The Isophase Bus will support the power increase with no modifications.

Main Transformers

The Main Bank Transformers for SQN Unit 1 has a manufacturer's nameplate output rating of 415 MVA per phase(1,245 MVA for the bank) with a 55° C winding temperature rise above ambient; or 465 MVA per phase(1,395 MVA for the bank) with a 65° C winding temperature rise above ambient. The Main Bank Transformers for SQN Unit 2 has a manufacturer's nameplate output rating of 420 MVA per phase(1,260 MVA for the bank) with a 55° C winding temperature rise above ambient; or 470 MVA per phase(1,410 MVA for the bank) with a 65° C winding temperature rise above ambient. The nameplate rating of the unit 1 Main Bank Transformer, being the most limiting of the two units of 1395 MVA, will remain above the anticipated maximum net output after the uprate of 1337 MVA. This includes the net increase of 13.3 MVA (12 MWe / 0.9 PF) due to the 1.3% uprate plus the maximum nominal output of 1323 MVA (1190.3 MWe/0.9 PF) which includes the ~4 MWe increase periodically observed during the colder winter months. The main bank transformers, at the 65° C winding temperature rise above ambient, are also rated above the main generator nameplate rating of 1356 MVA, therefore, the main bank transformers will operate within all applicable limits at the 1.3% power uprate conditions.

Switchyard

The switchyard equipment exceeds the nameplate rating of the main generator. All 500kV switches and breakers that interface with the Unit 1 main generator are rated at 3000 amperes, which exceeds the Unit 1 main generator maximum output current at its nameplate rating of 1356 MVA. All 161kV switches and breakers that interface with the Unit 2 main generator are rated at 5000 amperes, which exceeds the Unit 2 main generator maximum output current at its nameplate rating of 1356 MVA. All other 161kV switches and breakers are rated at 3000 amperes. The switchyard will accept the additional load without the need for any hardware modifications.

Question 5 (TXX-99105):

Discuss the impact of the CPSES Unit 2 power uprate electrical conditions on the current grid stability and reliability analysis. Describe in this discussion, how the station continues to be in conformance with General Design Criterion 17 with CPSES Unit 2 at the power uprated electrical conditions.

Response:

Grid Stability

The current main generator electrical output is typically 1186.3 MWe. After the anticipated 12 MWe uprate, the generator output will be approximately 1198.3 MW range with an approximately 4 MWe increase periodically observed during the colder winter months. The current grid study has analyzed the safe shutdown of the plant with SQN U1 @ 1198.7 MW and SQN U2 @ 1198.5 MW. This analysis is approximately at the same generation output level as the proposed uprate. (The difference is considered negligible and within the accuracy range of the calculation.) An update to the study is currently being performed to determine the impact on the stability of 161kV and 500kV grid for a 1.3% increase in SQN U1&2 generation. This study considers a line out pre-event and a subsequent simultaneous LOCA of the SQN unit and a fault and trip of another line. This study will also consider additional impacts to the grid anticipated for the next three years and will not have an impact on the validity of the current study due to the 1.3% uprate.

161kV OFFSITE POWER SUPPLY

The SQN units receive shutdown power from the 161 kV system through two physically and functionally independent circuits. This power is normally supplied through two 161 kV transmission lines from the switchyard to the plant common station service transformers A and C with a third line connected to common station service transformer B that may be used in place of either A or C. Additionally, the 161 kV system is interconnected to the 500 kV system through a 1200 MVA, 500-161 kV inter-tie transformer bank. There is no change in the shutdown loads at SQN or the voltage requirements of these loads associated with the 1.3% upgrade. Therefore, the extra 1.3% of power generated into the 161 kV and 500 kV systems has no significant impact on the 161kV system at SQN and the ability of the unit to safely shut down, and therefore Sequoyah will continue to be in conformance with GDC 17.

Question 6 (TXX-99105):

Provide a pressurized thermal shock evaluation for the CPSES Unit 2 reactor vessel before implementing the power uprate and after implementing the power uprate.

Response:

Refer to Enclosure 6, Section 2.4.1.3 and Table 2.4.1.3-7 for this evaluation. The evaluation concluded that the existing RT_{PTS} values remain valid and conservative.

Question 7 (TXX-99105):

What is the calculated end-of-life fluence in the current vessel design of CPSES Unit 2? What is the expected fluence for pressurized thermal shock with the revised design conditions/power uprate for CPSES Unit 2?

Response:

Refer to Section 2.4.1.2 and Tables 2.4.1.2-1 and 2.4.1.2-2 of Enclosure 6 for the calculation of neutron exposure projections. The evaluation concluded that the neutron fluences increased slightly for the uprated power conditions.

Question 8 (TXX-99105):

Does the power uprate for CPSES Unit 2 change the cold leg temperature? If so, please provide details.

Response:

Refer to Section 2.1 of Enclosure 6 for the change in T_{cold}. It indicates that T_{cold} decreases by 0.4°F with the increase in core power.

Question 9 (TXX-99105):

Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?

Response:

The modest 1.3% power uprate is not expected to have any significant effect on the manner in which the operators control the plant, either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Main Turbine Impulse Pressure, Steam Generator Pressure and Main Feedwater and Steam Flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be included in Operator Training accordingly.

Question 10 (TXX-99105):

Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the

required manual actions in the times allowed. Discuss results of simulator observations regarding operator response times for operator actions that are potentially sensitive to power uprate.

Response:

The modest 1.3% power uprate is not expected to have any significant effect on the manner in which the operators control the plant (including operator response times), either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Main Turbine Impulse Pressure, Steam Generator Pressure and Main Feedwater and Steam Flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be included in Operator Training accordingly.

Question 11 (TXX-99105):

Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-or-tolerance range)? If changes will occur, discuss how they will be addressed.

Response:

No changes to control room control functions are required and a minimal change to the annunciator system and ICS display screens will be required. When the power uprate is put in place, the Nuclear Instrumentation System will simply be adjusted to indicate the new 100% RTP in accordance with Technical Specification requirements and plant administrative controls. Because this power uprate is predicated on the availability of the LEFM, procedural guidance, supplemented by plant computer displays, will be developed to facilitate operation when the LEFM is unavailable. The plant computer system will provide an audible and visual alarm for LEFM system failure or if maintenance is required. The plant computer system will also provide for the actuation of a main control board Beta annunciator window for operator notification whenever the computer point ID that calculates the LEFM reactor thermal power value indicates unreliable data. No other changes to control room indicators or controls are required as a direct result of the power uprate. There are no new operator tasks required for safe shutdown by implementing this uprate. The operator's response has not changed. The reactor operators will be trained on the changes in a manner consistent with any other design modification.

Question 12 (TXX-99105):

Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.

Response:

The SPDS is unaffected by the proposed 1.3% increase in Reactor Thermal Power.

Question 13 (TXX-99105):

Describe all changes the power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.5-1985, Section 5.4.1.

Specifically, please propose a license condition and/or commitment that stipulates the following:

(a) **Provide classroom and simulator training on all changes that effect operator performance caused by the power uprate modification.**

(b) **Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be re-validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator revalidation will include comparison of individual simulated systems and components and simulated integrated plant steady state and transient performance with reference plant responses using similar startup test procedures.**

(c) **Complete all control room and plant process computer system changes as a result of the power uprate.**

(d) **Modify operator training and the plant simulator, as required, to address all related issues and discrepancies that are identified during the startup testing program.**

Response:

The modest 1.3% power uprate is not expected to have any significant effect on the manner in which the operators control the plant, either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Main Turbine Impulse Pressure, Steam Generator Pressure and Main Feedwater and Steam Flows. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be included in Operator Training accordingly.

In addition, the modest 1.3% power uprate is not expected to have a significant effect on any simulated systems. Changes associated with the power uprate will be treated in a manner consistent with any other plant modification, and will be tested and documented accordingly. The SQN Simulator will be modified to match predicted plant values for 101.3% rated power. Following plant implementation, startup and operation at the uprated power, plant data will be collected and incorporated as the reference plant data for Simulator Steady State Performance Tests in accordance with the Simulator Certification annual testing program.

Question 14 (TXX-99105):

The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurement and plant operation.

Response:

New procedures for maintenance and calibration of the LEFM system will be developed per the design control process based on the vendor's recommendations.

Current Operations procedures are used to perform a calorimetric heat balance measurement for the purpose of calibrating the Power Range NIS channels. Contingencies and instructions will be added to the procedure in the event that the LEFM system becomes unavailable. This procedure will be revised per the design change control process to incorporate the requirements for the new LEFM system. In addition, more formal guidance, including routine surveillance requirement(s) for the LEFM and appropriate contingency actions, will be provided in the Technical Requirements Manual (TRM). Appropriate contingency actions for continued operation with inoperable LEFM instrumentation are described in Section 4.0 of Enclosure 6. Refer to response for Question 2 (TXX-99203) response for additional information.

Question 15 (TXX-99105):

For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the installed installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Response:

The LEFM 8300 strap-on system that is currently installed at SQN Units 1 & 2 is only used as a basis for determining the correction factor for feedwater venturi fouling. The existing LEFM 8300 strap-on system is not as accurate as the new LEFM Check system and therefore will not be used as a basis for the 1.3% uprate. A complete new system will be installed at SQN that is bounded by the analysis and assumptions set forth in the Caldon Topical Report ER-80P. The new LEFM Check system is the same LEFM system that formed the basis of the analysis in the Topical Report. Commissioning of the system will be completed following the installation and prior to the uprate that will document that the new system is bounded by the Topical Report. This documentation will be available for inspection.

Question 16 (TXX-99105):

The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response:

The methodology used to calculate the combined feedwater mass flow and feedwater temperature uncertainty for the improved LEFM system is exactly the same as the methodology presented in Caldon Topical Reports ER-80P and ER-160P. This value is then utilized to calculate the total power measurement uncertainty described in WCAP-15669, Rev.0, "Westinghouse Power Measurement Instrument Uncertainty Methodology for Tennessee Valley Authority, Sequoyah 1 & 2 (1.3% Uprate to 3467 MWt-NSSS Power)".

Question 17 (TXX-99105):

Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), should provide additional justification for use. This justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response:

The LEFM Check systems to be installed at Sequoyah Units 1 and 2 will be calibrated to a plant/unit specific piping configuration prior to installation. The results of the calibration will provide a meter factor representative of the plant/unit specific configuration. In addition, the accuracy with which the meter factor is determined will be incorporated into the uncertainty analysis of record for the Sequoyah LEFM Check system. Therefore, additional justification for use will not be required.

Question 18 (TXX-99105):

Based on the above, the staff finds that feedwater flow measurement using the LEFM can provide a thermal power measurement that will remain bounding within an uncertainty of 1% of rated thermal power. This is premised on the assumption that no additional uncertainties beyond those included in Topical Report ER-80P are assumed to be included in the 10 CFR Part 50, Appendix K 102% thermal power margin requirement.

Response:

Refer to Enclosure 4 for a discussion of the power measurement uncertainty calculation.

Question 19 (TXX-99105):

The amendment request proposes to reduce the margin for assumed power level for non-LOCA accident and transient analysis on the same basis as the proposed exemption to the Appendix K ECCS evaluation requirement. Staff consideration of the related Appendix K exemption request was in part based on the premise that the power level requirement is one of several conservative features that, taken together, provide substantial conservatism in ECCS analyses.

Justify the proposed margin reduction for non-LOCA analyses that currently assume 102% power. The justification should include a quantitative or qualitative discussion of conservative analysis assumptions for the non-LOCA accidents and transients and the safety margin they provide relative to the power level margin assumption.

Response:

The Sequoyah 1.3% power level uprate proposed in the current submittal is based on eliminating unnecessary analytical margin originally required in ECCS Evaluation Models performed in accordance with the requirements set forth in 10CFR50 Appendix K. The Federal Register of June 1, 2000 included an amendment to the introductory paragraph of 1.A of Appendix K to 10CFR50. The amendment allows the use of an assumed power level lower than 102% with the provision that the alternative value has been demonstrated to account for power level uncertainties. The basis for the current amendment request is that the Caldon instrumentation provides a more accurate indication of feedwater flow (and correspondingly reactor thermal power) than assumed during the development of the 10CFR50 Appendix K requirements. Complete technical support for this conclusion is discussed in detail in Caldon Topical Report ER-80P. This report is approved in the NRC's Safety Evaluation for TU Electric, dated March 8, 1999, and supplemented by Caldon Engineering Report 160P. The improved thermal power measurement accuracy eliminates the need for the full 2% power margin assumed in 10CFR50 Appendix K, thereby increasing the thermal power available for electrical generation. There is no margin reduction associated with Sequoyah operation at 101.3% RTP with a 0.7% calorimetric uncertainty when using the improved LEFM instrumentation.

The text of the current submittal for Sequoyah contains a comprehensive discussion of each non-LOCA transient (Enclosure 6, Section 3). For each transient, the text contains a qualitative discussion of the conservative analysis assumptions and the applicable safety margins. Some of the conservative assumptions used in the non-LOCA analyses include: core power distribution, peaking factors, moderator and Doppler fuel temperature reactivity feedbacks, trip reactivity worth and reactivity insertion characteristics. With respect to available equipment and instrumentation, the beneficial effects of some control systems are not credited in the analysis and a single failure of equipment or instrumentation required to mitigate the transient is assumed. Additional conservatisms are inherently present in the models and methods used in the analyses. In addition to these generic assumptions, additional conservative assumptions were made on an event-specific basis. The conservative assumptions, other than power level uncertainty, used in the accident analyses are unaffected by the change in the uncertainty allowance applied to the initial power level.

The allowance provided for the power calorimetric uncertainty is but one of several conservative assumptions that are applied to each of the safety analyses. Through the use of the improved LEFM instrumentation, the use of a smaller value of the power calorimetric uncertainty does not result in reduction of analytical margin in the safety analyses.

Question 20 (TXX-99105):

Increasing licensed power level would result in an increased heat source that could affect the progression of certain accidents. Discuss the potential impact of plant operation at the higher proposed power level on ATWS progression, containment integrity analyses, and on overall IPE results.

Response:

Sections 7.7.1.12 and 10.4.7.2 of the Sequoyah FSAR document that TVA installed the AMSAC system to comply with the ATWS rule. Unlike CPSES, the Sequoyah FSAR does not include a section addressing an ATWS analysis. The 1.3% increase in core power will not affect the ability of the AMSAC to perform its intended functions stated in FSAR Sections 7.7.1.12 and 10.4.7.2.

Refer to Enclosure 6, Section 2.51 for an evaluation of the mass and energy releases used as input to the containment integrity analysis. Since the current mass and energy releases remained bounding for the 1.3% uprate conditions, it was not necessary to re-perform the containment integrity analysis.

The Sequoyah PRA model includes both level 1 systems analysis and level 2 containment analysis. The systems analysis is not impacted by the additional hardware installed for the LEFM project. Success criteria for the level 1 analysis would be minimally impacted by the minor power uprate. This is evidenced by the fact that the Large Break (LB) LOCA ECCS analysis presented in Sequoyah's FSAR (Section 15.4) remains bounding and was not affected by the power uprate (see Enclosure 6, Section 2.51). Many of the PRA success criteria are based on traditional analyses (e.g., 3 out of 4 accumulators for LBLOCA success is derived from the design basis analysis). The power uprate would also have minor impacts on the containment analysis. A slight decrease in time for onset of the hydrogen/zirc water reaction, earlier time to rupture for the pressurizer relief tank rupture disk, for transients, etc., would be expected although not significant in terms of damage progression. An PRA re-analysis is therefore not warranted prior to the next PRA update.

Question 21 (TXX-99105):

Discuss the impact on LOCA and non-LOCA analysis results (e.g., main steam line break) of the revised values for RCP heat addition and RCS flow rate included in the amendment request.

Response:

The license amendment for Sequoyah does not involve a change to the design basis RCS flow rate or RCP heat addition values.

Question 22 (TXX-99105):

Provide the detailed calculational basis to substantiate the statement made in the amendment request that a 10-percent SG tube plugging level supports a peak plugging level of 15% in any one SG, provided that the average level of plugging of all four SGs is no greater than 10 percent. Explain the difference between the plugging level used in the analysis discussed in the amendment request and the plugging level assumed in the current LOCA analysis?

Response:

Refer to Enclosure 6, Section 2.1. The analyzed SGTP level is 15% and does not presently address the use of asymmetric tube plugging levels above 15%.

Question 23 (TXX-99105):

Plant response to SGTR and other events depends on SG atmospheric relief valve operation. Reactor operation at higher power levels may cause these valves to operate more often in the event of certain events, thereby affecting their reliability. Discuss the effects of operation at the proposed new power level on the possible increased challenge to these valves and their expected failure frequency during a SGTR event (and other events requiring their operation).

Response:

While it is true that transients initiated from a higher power level may present more challenges to the ARVs, the frequency of such challenges is not considered to be significant. The proposed increase in reactor power is very modest. The capacity and reliability of the Steam Dump System are such that the ARVs are generally not anticipated to be operated any more frequently than they are currently cycled. See also Section 2.3.2.2.2 of Enclosure 6.

Question 24 (TXX-99105):

When considered in terms of core power, the proposed changes in power range neutron flux, and overpower N-16 nominal and allowable reactor power trip levels appear slightly non-conservative. Explain the basis for the proposed revision to the N-16 overpower and power range neutron flux trip set points given in the amendment request. Provide justification for the apparently non-conservative set point changes.

Response:

The Sequoyah design does not employ an N-16 reactor trip function. The Sequoyah High and Low power range neutron flux trip setpoints allow for a 2% calorimetric uncertainty, bounding the lower uncertainty (0.7%) associated with the power uprate. For the power uprate condition, the safety analysis high flux trip setpoint (high setting) will be redefined to be 116.5% of 3455 MWt. This value is equivalent, in terms of total megawatts, to the current licensing basis at 3411 MWt (116.5% of 3455 MWt = 118% of 3411 MWt) and pre-empts the necessity of additional analysis. An evaluation of the existing accuracy calculations for this trip indicates that the current technical specification setpoints (respective trip setpoint and allowable value of 109.0% and 111.4% of rated thermal power) would not need to be changed. There is adequate existing margin in the trip to accommodate the power uprate. However, the margin between the actual and allowable uncertainties will be reduced as a result. The safety analysis high flux trip setpoint (low setting) will be similarly redefined without a subsequent technical specification change. See Section 3.3.2 of Enclosure 6 for further discussion of the Sequoyah trip setpoints.

Question 25 (TXX-99105):

The N-16 overtemperature trip setpoint was not changed in the amendment request, based on the statement that it was previously analyzed at the power level requested in the proposed amendment. Confirm that the other proposed changes to plant parameters such as RCS flow and coolant temperatures do not result in a change to the N-16 overtemperature trip setpoint. Explain how the proposed changes in core flow rate and coolant temperatures affect the calculation of the N-16 overtemperature trip setpoint.

Response:

The Sequoyah design does not employ an N-16 reactor trip function. Refer to Enclosure 6, Section 3.3.2 - which confirms that no changes were required to the reactor trip and engineered safety feature actuation system setpoints as a result of the slight changes to the RCS temperatures for the 1.3% uprate conditions.

B. TXX-99115

Question 1 (TXX-99115 - Attachment 3):

Part 1) In Attachment 2 of the submittal, the licensee states that the Balance of Plant (BOP) fluid systems were reviewed for compliance with the Westinghouse Nuclear Steam Supply System (NSSS)/BOP Interface guidelines. How does the power uprate affect the design basis of the following systems: main steam, steam dump system, feedwater and condensate system, and auxiliary feedwater system?

Part 2) In Section C of Attachment 2, the licensee states that design documentation and instrumentation and control setpoint changes are required. Which, if any, of the following systems and items would exceed the design basis: circulating water, turbine plant cooling, spent fuel pool cooling, component cooling, station service water, station blackout, spent fuel storage, HVAC systems, turbine/generator? In any, provide the new limits and explain why the new design basis is acceptable.

Part 3) In Table IV-1 of Attachment 2, "NSSS Revised Design Parameters," the licensee describes three limiting cases. Explain which case(s) was (were) used in the evaluation of the above listed BOP systems and the NSSS/BOP interfaces. If only one was used, explain why it provides conservative results.

Response to Part 1
Refer to Enclosure 6, Section 2.3.2.

Response to Part 2
Using the revised NSSS parameters (see Enclosure 6, Table 2.1-1), TVA has performed heat balances at 101.3% reactor thermal power for the proposed uprate. The secondary side plant systems were originally designed to support the operation of the Westinghouse supplied turbine/generator. At the valves wide open or stretch condition the turbine/generator is rated at 1,229,701 kW with a steam flow of 15,629,181 lb/hr. This equates to operation at approximately 104.5% reactor thermal power. Therefore, no major impacts to the balance of plant were expected. Comparison of the uprate heat balances with the current 100% heat balance revealed no significant differences in pressures, temperatures, or flows for the secondary side plant systems (See Table A-1).

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

- Extraction Steam - Each of the turbine extraction lines will realize an increase in pressure of 1.92% or less. The mass flow changes will be within +4% of the current flows. These increases and/or changes are within the design parameters as given in the design basis documentation.

- Condensate - The condensate system does not require an increase in storage capacity of the condensate storage tank, has a modest temperature increase (≤1.5°F), no pressure increase, and a slight increase in flow rate. All of the preceding have been evaluated against the design bases and are judged to be bounded by existing analysis and are adequate for the 1.3% power increase.

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

- Condensate Polishing - The condensate polishing system will experience a small increase in temperature (this is not reflected within Table 7-1 but would occur if the condenser backpressure is allowed to float based on steam loading at a specified cooling water temperature and flow) which is well within the capacity of the system. An increase in the total flow of less than two percent is within the capacity of the system. The purity of the condensate is not expected to be significantly different with the power uprate and the review of the design bases documents for the condensate demineralizers indicates that the power uprate is acceptable for this system.

- Turbine/Generator Cooling - The generator is designed for ~104.5% of nameplate rated power. The hydrogen cooling system was also designed for this thermal loading. The incoming raw water is required to have sufficient capacity to cool this thermal load imposed on the hydrogen. Therefore, no impacts are expected for the cooling system.

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

- Main Turbine Electro-Hydraulic Control (EHC) System - Plant modifications are required associated with the main turbine governor valve AW3 servo cards (one card per each of the four governor valves). The existing Westinghouse Analog Electro-Hydraulic (AEH) Control system has design limitations that will limit operation once the 1.3% power uprate is completed. The existing AW3 servo cards have only two breakpoints making it difficult to match the electronic characteristics to the actual flow characteristics of the steam valve. This limitation will be further exacerbated after the power uprate because the valves will be operating in a more fully open position. Further, the current Speed Error Amplifier B card does not have a deadband which results in valve movement at full power due to grid frequency deviation. The purpose of this planned modification is to improve these AEH Control system design limitations and provide additional operational enhancements. The four AW3 servo cards associated with the governor valves will be replaced by AVP servo cards. Each AVP card provides 15 curve segments which will allow a better match of the electronics to the mechanical flow characteristics of the governor valves. The new Speed Error Amplifier B card has a programmable deadband which allows more precise control of valve position while reducing oscillations. Further system enhancements made by this modification include the addition of 4 test point cards and a data acquisition system.

Based on TVA's preliminary evaluation and planned EHC modifications, the Balance of Plant systems are deemed adequate for the increase in thermal loads produced by the power uprate.

TABLE A-1

FIELD DESCRIPTION	UNITS	SEQUOYAH 100.0% RTP	SEQUOYAH 101.3% RTP	% DIFF
STEAM GENERATOR STEAM OUTLET FLOW	#/HR	14,911,680	15,122,570	1.39%
GSC/SGBD HX BYPASS INLET FLOW	#/HR	6,314,178	6,442,846	2.00%
2ND STAGE REHEATER TUBE SIDE INLET FLOW	#/HR	661,008	626,906	-5.44%
MOISTURE REMOVAL STAGE 1 OUTLET FLOW	#/HR	24,815	24,070	-3.09%
NO 1 EXTRACTION STEAM OUTLET FLOW	#/HR	609,830	633,797	3.78%
1ST STAGE REHEATER TUBE SIDE INLET	#/HR	506,423	520,198	2.65%

TABLE A-1				
FIELD DESCRIPTION	**UNITS**	**SEQUOYAH 100.0% RTP**	**SEQUOYAH 101.3% RTP**	**% DIFF**
FLOW				
MOISTURE REMOVAL STAGE 2 OUTLET FLOW	#/HR	34,328	33,578	-2.23%
NO 2 EXTRACTION STEAM OUTLET FLOW	#/HR	605,951	619,022	2.11%
NO 3 EXTRACTION STEAM OUTLET FLOW	#/HR	915,501	933,700	1.95%
MOISTURE SEPARATOR DRAIN OUTLET FLOW	#/HR	1,294,034	1,290,967	-0.24%
REHEAT STEAM TO MFPT LP INLET FLOW	#/HR	132,609	138,503	4.26%
NO 4 EXTRACTION STEAM OUTLET FLOW	#/HR	376,189	385,505	2.42%
NO 5 EXTRACTION STEAM OUTLET FLOW	#/HR	543,482	556,164	2.28%
MOISTURE REMOVAL STAGE 3 OUTLET FLOW	#/HR	69,373	72,841	4.76%
NO 6 EXTRACTION STEAM OUTLET FLOW	#/HR	388,576	397,078	2.14%
MOISTURE REMOVAL STAGE 4 OUTLET FLOW	#/HR	113,802	117,732	3.34%
NO 7 EXTRACTION STEAM OUTLET FLOW	#/HR	417,052	425,565	2.00%
MOIST REMOVAL STAGE 5 WATER OUTLET FLOW	#/HR	140,164	144,020	2.68%
MOIST REMOVAL STAGE 5 STEAM OUTLET FLOW	#/HR	61,488	62,450	1.54%
MAIN STEAM THROTTLE FLOW	#/HR	14,236,430	14,481,430	1.69%
HP TURBINE EXHAUST FLOW	#/HR	12,449,080	12,644,680	1.55%
MAIN STEAM AT LP TURBINE INLET FLOW	#/HR	10,106,940	10,281,510	1.70%
LP TURBINE EXHAUST STAGE FLOW	#/HR	8,006,062	8,129,412	1.52%
CONDENSER HOTWELL DRAIN FLOW	#/HR	8,234,185	8,362,846	1.54%
NO. 1 FWH DRAIN FLOW	#/HR	1,292,479	1,281,484	-0.86%
NO. 2 FWH DRAIN FLOW	#/HR	2,435,979	2,451,018	0.61%
NO. 3 HDT DRAIN FLOW	#/HR	4,640,937	4,671,016	0.64%
NO. 4 FWH DRAIN FLOW	#/HR	374,308	383,578	2.42%
NO. 5 FWH DRAIN FLOW	#/HR	915,073	936,961	2.34%
NO. 6 FWH DRAIN FLOW	#/HR	1,370,732	1,404,530	2.41%
NO. 7 HDT DRAIN FLOW	#/HR	2,036,558	2,088,711	2.50%
MFPT CONDENSER INLET FLOW	#/HR	8,314,178	8,442,846	1.52%
NO. 7 FWH TUBE INLET FLOW	#/HR	8,083,014	8,213,792	1.59%
NO. 6 FWH TUBE INLET FLOW	#/HR	10,119,570	10,302,500	1.78%
NO. 4 FWH TUBE OUTLET FLOW	#/HR	10,350,740	10,531,560	1.72%
NO. 1 FWH TUBE OUTLET FLOW	#/HR	14,991,670	15,202,570	1.39%
STEAM GENERATOR BLOWDOWN FLOW	#/HR	80,000	80,000	0.00%
MAIN TURBINE CONTROL VALVE INLET MOISTURE	%	0.36	0.35	-2.74%
MAIN TURBINE CONTROL VALVE INLET TEMP	°F	522.6	519.2	-0.65%
MOISTURE SEP SHELL SIDE OUTLET TEMP	°F	366.1	367.2	0.32%
1ST ST REHEATER SHELL SIDE OUTLET TEMP	°F	424.3	426.0	0.39%
2ND ST REHEATER SHELL SIDE OUTLET TEMP	°F	502.0	498.6	-0.68%
LP TURBINE REHEAT STEAM INLET TEMP	°F	501.3	497.9	-0.68%
MAIN CONDENSER DRAIN OUTLET TEMP	°F	101.1404	101.1404	0.00%
MFPTC TUBE SIDE INLET TEMP	°F	101.9	101.9	0.00%
MFPTC TUBE SIDE OUTLET TEMP	°F	116.9	117.3	0.31%
NO 7 FWH OUTLET TEMP	°F	166.0	166.5	0.29%
NO 6 FWH INLET TEMP	°F	167.3	167.8	0.32%
NO 6 FWH OUTLET TEMP	°F	208.0	208.6	0.31%
NO 5 FWH OUTLET TEMP	°F	261.6	262.4	0.30%
NO 4 FWH INLET TEMP	°F	261.2	262.0	0.29%
NO 4 FWH OUTLET TEMP	°F	295.7	296.6	0.31%
NO 3 FWH OUTLET TEMP	°F	362.4	363.5	0.30%
NO 2 FWH INLET TEMP	°F	363.1	364.2	0.31%
NO 2 FWH OUTLET TEMP	°F	398.9	400.2	0.33%

TABLE A-1				
FIELD DESCRIPTION	UNITS	SEQUOYAH 100.0% RTP	SEQUOYAH 101.3% RTP	% DIFF
NO 1 FWH INLET TEMP	°F	400.1	401.5	0.35%
NO 1 FWH OUTLET TEMP	°F	434.5	436.0	0.35%
NO 1 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	4.36	4.49	2.96%
NO 1 FWH DRAIN COOLER APPROACH (DCA)	°F	10.0	9.9	-0.50%
NO 2 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	5.34	5.52	3.21%
NO 2 FWH DRAIN COOLER APPROACH (DCA)	°F	9.26	9.39	1.36%
NO 3 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	2.39	2.49	3.96%
NO 4 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	1.11	1.16	4.31%
NO 4 FWH DRAIN COOLER APPROACH (DCA)	°F	0.49	0.58	15.59%
NO 5 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	6.20	6.39	2.97%
NO 5 FWH DRAIN COOLER APPROACH (DCA)	°F	10.3	10.6	2.47%
NO 6 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	6.08	6.24	2.57%
NO 6 FWH DRAIN COOLER APPROACH (DCA)	°F	11.4	11.6	2.06%
NO 7 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	7.23	7.42	2.55%
NO 1 FWH DRAINS OUTLET TEMP	°F	410.1	411.4	0.33%
NO 2 FWH DRAINS OUTLET TEMP	°F	372.3	373.6	0.33%
NO 3 HDT DRAINS OUTLET TEMP	°F	364.4265	365.6131	0.32%
NO 4 FWH DRAINS OUTLET TEMP	°F	261.7	262.6	0.32%
NO 5 FWH DRAINS OUTLET TEMP	°F	218.3	219.2	0.41%
NO 6 FWH DRAINS OUTLET TEMP	°F	178.7	179.5	0.43%
NO 7 HDT DRAINS OUTLET TEMP	°F	172.1	172.8	0.39%
MAIN TURBINE CONTROL VALVE INLET ENTHALPY	BTU/LB	1196.0	1196.8	0.06%
MAIN CONDENSER BACKPRESSURE	IN-HG	2.00	2.00	0.00%
MFPT EXHAUST STEAM OUTLET PRESS	IN-HG	3.98	4.06	1.83%
MAIN FEED PUMP TURBINE OUTPUT	KW	9387	9826	4.48%
TOTAL GENERATOR OUTPUT	MWe	1187.456	1198.991	0.96%
STEAM GENERATOR STEAM OUTLET PRESS	PSIA	857.0	832.0	-3.00%
MAIN TURBINE CONTROL VALVE INLET PRESS	PSIA	831.1	807.2	-2.97%
HP TURBINE IMPULSE PRESS	PSIA	628.0	639.7	1.83%
HP TURBINE 1ST EXTRACTION OUTLET PRESS	PSIA	396.8	403.7	1.71%
HP TURBINE 2ND EXTRACTION OUTLET PRESS	PSIA	273.0	277.6	1.68%
HP TURBINE EXHAUST STEAM OUTLET PRESS	PSIA	171.1	173.6	1.47%
MOISTURE SEP SHELL SIDE INLET PRESS	PSIA	169.4	171.9	1.47%
MOISTURE SEP SHELL SIDE OUTLET PRESS	PSIA	164.3	166.8	1.47%
1ST ST REHEATER SHELL SIDE OUTLET PRESS	PSIA	162.9	165.3	1.47%
2ND ST REHEATER SHELL SIDE OUTLET PRESS	PSIA	161.2	163.6	1.47%
LP TURBINE REHEAT STEAM INLET PRESS	PSIA	157.9	160.3	1.47%
LP TURBINE 4TH EXTRACTION OUTLET PRESS	PSIA	67.2	68.2	1.44%
LP TURBINE 5TH EXTRACTION OUTLET PRESS	PSIA	42.5	43.2	1.58%
LP TURBINE 6TH EXTRACTION OUTLET PRESS	PSIA	16.1	16.4	1.56%
LP TURBINE 7TH EXTRACTION OUTLET PRESS	PSIA	6.79	6.89	1.53%
MOISTURE REMOVAL STAGE 5 OUTLET PRESS	PSIA	3.10	3.15	1.52%
2ND STAGE REHEATER TUBE SIDE INLET PRESS	PSIA	822.8	799.1	-2.97%
1ST STAGE REHEATER TUBE SIDE INLET PRESS	PSIA	392.8	399.6	1.71%

TABLE A-1				
FIELD DESCRIPTION	**UNITS**	**SEQUOYAH 100.0% RTP**	**SEQUOYAH 101.3% RTP**	**% DIFF**
NO 1 FWH SATURATION PRESS	PSIA	376.9	383.5	1.71%
NO 2 FWH SATURATION PRESS	PSIA	259.3	263.8	1.68%
NO 3 FWH SATURATION PRESS	PSIA	162.5	165.0	1.47%
NO 4 FWH SATURATION PRESS	PSIA	63.8	64.8	1.44%
NO 5 FWH SATURATION PRESS	PSIA	40.4	41.0	1.58%
NO 6 FWH SATURATION PRESS	PSIA	15.3	15.6	1.56%
NO 7 FWH SATURATION PRESS	PSIA	6.45	6.55	1.53%
MAIN FEED PUMP TURBINE SPEED	RPM	4719	4778	1.25%

In addition, as part of the design change process for the power uprate, additional heat balance studies have been performed at higher ambient conditions to assess potential impacts on individual BOP components. Heat balances were performed at the current NSSS thermal input of 3423 MWt and at the proposed uprate thermal input of 3467 MWt with the following boundary conditions:

BOUNDARY CONDITION	UNITS	100% RTP	101.3% RTP
REACTOR THERMAL POWER (+12 MWT RCP INPUT)	MWT	3423	3467
STEAM GENERATOR OUTLET PRESSURE	PSIA	857.0	832.0
MAIN STEAM THROTTLE PRESSURE	PSIA	832.0	807.0
CONDENSER CIRCULATING WATER (CCW) TEMP	°F	85.0	85.0
CONDENSER CLEANLINESS	%	80.0	80.0
STEAM GENERATOR BLOWDOWN (SGBD) FLOW	GPM	270	270

The boundary conditions are representative of the following:

- CCW temperature experienced during summer operation
- Condenser cleanliness representative of the condenser following a long continuous run with macro-fouling on the tubesheets
- Maximum allowable SGBD flow

The results of the two heat balances are shown in Table A-2. As in the previous heat balance comparison, no notable differences existed which would warrant further investigation. However, areas of consideration that were explored further included the main condenser backpressure, main feed pump turbine and associated condenser, high pressure turbine impulse pressure, flow instrumentation range limitations, heater drain pump/control valve capacity, and the high pressure reheater operating vent line. The following paragraphs discuss each of these items:

Main Condenser Backpressure

Sequoyah's Low Pressure Turbines currently have a backpressure limitation of 5.0 in-HgAbs. Current operation at the listed boundary conditions would result in a backpressure of 3.77 in-HgAbs. Likewise, for the uprate conditions, the backpressure is expected to increase to 3.83 in-HgAbs. Since the predicted backpressure remains below the limit, the main condenser can support the proposed uprate.

Main Feedwater Pump Turbine (MFPT) and Associated Condenser

Sequoyah has two turbine-driven feedwater pumps available to supply feedwater to the steam generators. The MFPTs driving the feedwater pumps exhaust to MFPT Condensers which are cooled by condensate. The MFPTs are designed for normal conditions of 7000 horsepower at 4990 rpm with low pressure (LP) inlet steam conditions of 160 psia, 494°F and 5.0 in-HgAbs. These units have a nameplate rating of 11,700 horsepower at 6000 rpm at the LP conditions. When the CCW inlet temperature to the main condenser approaches the higher values, the condensate temperatures supplied to the MFPT condenser also rise resulting in higher backpressures on the MFPTs. An Original Equipment Manufacturer (OEM) evaluation was performed for the Sequoyah MFPTs and MFPT Condensers (Reference 1). The OEM technical evaluations performed concluded that there are no hardware changes required to accommodate the power uprate of 1.3%.

<u>High Pressure (HP) Turbine Impulse</u>
As a result of the proposed uprate, the steam flow through the HP turbine increases, such that the Impulse Pressure also must increase. An increase of approximately 12 psi is anticipated following the uprate. This increase will result in the re-calibration of the Impulse Pressure transmitters. In addition, runback setpoints and AMSAC arming setpoints associated with the Impulse Pressure have been evaluated and no setpoint modification to these instruments are required. Applicable instrumentation changes are being included in the engineering design package which is currently being finalized to support the proposed 1.3% uprate.

<u>Instrumentation Range Limitations</u>
The change in measured parameters (i.e., affected by the uprate) did not impact the instrumentation supporting BOP operation (except as noted above - Impulse Pressure). However, the total power calorimetric uncertainty using LEFM was evaluated by Westinghouse and resulted in the uncertainties for several BOP instrument channels having to be re-calculated using current Westinghouse methodology. This required several BOP instrument loop accuracies to be revised to comply with the Westinghouse calculation.

<u>Heater Drain Pumps/Control Valve Capacity</u>
Sequoyah typically operates all BOP pumps within the condensate, feedwater, and heater drain systems at full power. This provides the greatest margin of NPSHA at the suction of the main feed pumps. The pumps operated include: (3) hotwell, (3) demineralizer, (3) condensate booster, (2) No. 7 heater drain, and (3) No. 3 heater drain pumps. During winter operation when the CCW temperature is at its minimum, the condensate temperature entering the tube side of the last stage feedwater heaters (Nos. 7A, 7B & 7C) is significantly reduced when compared with operation at a condenser backpressure of 2 inches HgAbs. This lower heater inlet temperature increases the amount of steam condensed in the heater shell; thereby increasing the drain flows to the heater drain system (by approximately 8%). There is currently (i.e., before the uprate) insufficient head/capacity in the No. 7 heater drain pumps to supply the increased flow against the backpressure generated by the condensate system with all of the listed pumps in operation. During periods of peak drain flow, Sequoyah currently shuts off one of the (3) parallel demineralizer pumps to reduce the pressure at the point in the condensate system at which the No. 7 heater drains are pumped in. An evaluation has determined that this same operational configuration will be adequate to permit operation at the uprated conditions.

<u>HP Reheater (Second Stage) Operating Vent Line</u>
The HP reheater operating vent lines pass two-phase flow from the exit of the each second stage reheater 4th tube pass to the number 1 extraction piping. The design limit of these lines indicate that they would not be able to handle any significant increase in flow. Steam to the second stage reheaters is supplied off main steam. Due to the reduction in steam generator pressure (thereby directly reducing the saturation temperature of the steam) the achievable reheat temperature is reduced along with the steam demand to the reheater. Therefore, since the steam supply to the HP reheater is reduced with the uprate, the operating vent line is not impacted.

Based on TVA's evaluations, the Balance of Plant systems are deemed adequate for the increase in thermal loads produced by the power uprate.

TABLE A-2

FIELD DESCRIPTION	UNITS	SEQUOYAH 100.0% RTP	SEQUOYAH 101.3% RTP	% DIFF
STEAM GENERATOR STEAM OUTLET FLOW	#/HR	14,901,560	15,112,830	1.40%
GSC/SGBD HX BYPASS INLET FLOW	#/HR	6,557,682	6,694,221	2.04%
2ND STAGE REHEATER TUBE SIDE INLET FLOW	#/HR	660,446	626,395	-5.44%
MOISTURE REMOVAL STAGE 1 OUTLET FLOW	#/HR	24,825	24,078	-3.10%
NO 1 EXTRACTION STEAM OUTLET FLOW	#/HR	612,626	636,716	3.78%
1ST STAGE REHEATER TUBE SIDE INLET	#/HR	506,606	520,451	2.66%

TABLE A-2				
FIELD DESCRIPTION	UNITS	SEQUOYAH 100.0% RTP	SEQUOYAH 101.3% RTP	% DIFF
FLOW				
MOISTURE REMOVAL STAGE 2 OUTLET FLOW	#/HR	34,345	33,600	-2.22%
NO 2 EXTRACTION STEAM OUTLET FLOW	#/HR	609,794	623,023	2.12%
NO 3 EXTRACTION STEAM OUTLET FLOW	#/HR	917,255	935,393	1.94%
MOISTURE SEPARATOR DRAIN OUTLET FLOW	#/HR	1,293,688	1,290,851	-0.22%
REHEAT STEAM TO MFPT LP INLET FLOW	#/HR	148,748	155,838	4.55%
NO 4 EXTRACTION STEAM OUTLET FLOW	#/HR	372,868	382,213	2.45%
NO 5 EXTRACTION STEAM OUTLET FLOW	#/HR	527,945	540,100	2.25%
MOISTURE REMOVAL STAGE 3 OUTLET FLOW	#/HR	68,666	72,117	4.79%
NO 6 EXTRACTION STEAM OUTLET FLOW	#/HR	354,313	361,185	1.90%
MOISTURE REMOVAL STAGE 4 OUTLET FLOW	#/HR	112,194	116,062	3.33%
NO 7 EXTRACTION STEAM OUTLET FLOW	#/HR	248,963	250,370	0.56%
MOIST REMOVAL STAGE 5 WATER OUTLET FLOW	#/HR	142,176	146,179	2.74%
MOIST REMOVAL STAGE 5 STEAM OUTLET FLOW	#/HR	62,910	63,931	1.60%
MAIN STEAM THROTTLE FLOW	#/HR	14,226,880	14,472,200	1.70%
HP TURBINE EXHAUST FLOW	#/HR	12,432,670	12,628,250	1.55%
MAIN STEAM AT LP TURBINE INLET FLOW	#/HR	10,072,990	10,246,170	1.69%
LP TURBINE EXHAUST STAGE FLOW	#/HR	8,192,208	8,323,268	1.57%
CONDENSER HOTWELL DRAIN FLOW	#/HR	8,422,695	8,559,221	1.60%
NO. 1 FWH DRAIN FLOW	#/HR	1,294,710	1,283,884	-0.84%
NO. 2 FWH DRAIN FLOW	#/HR	2,442,233	2,457,675	0.63%
NO. 3 HDT DRAIN FLOW	#/HR	4,648,588	4,679,243	0.66%
NO. 4 FWH DRAIN FLOW	#/HR	371,003	380,302	2.45%
NO. 5 FWH DRAIN FLOW	#/HR	896,308	917,702	2.33%
NO. 6 FWH DRAIN FLOW	#/HR	1,317,170	1,348,838	2.35%
NO. 7 HDT DRAIN FLOW	#/HR	1,830,278	1,874,369	2.35%
MFPT CONDENSER INLET FLOW	#/HR	8,557,682	8,694,221	1.57%
NO. 7 FWH TUBE INLET FLOW	#/HR	8,123,746	8,261,979	1.67%
NO. 6 FWH TUBE INLET FLOW	#/HR	9,954,022	10,136,350	1.80%
NO. 4 FWH TUBE OUTLET FLOW	#/HR	10,387,960	10,568,590	1.71%
NO. 1 FWH TUBE OUTLET FLOW	#/HR	15,036,550	15,247,830	1.39%
STEAM GENERATOR BLOWDOWN FLOW	#/HR	135,000	135,000	0.00%
MAIN TURBINE CONTROL VALVE INLET MOISTURE	%	0.36	0.35	-2.72%
MAIN TURBINE CONTROL VALVE INLET TEMP	°F	522.6	519.3	-0.65%
MOISTURE SEP SHELL SIDE OUTLET TEMP	°F	365.8	367.0	0.32%
1ST ST REHEATER SHELL SIDE OUTLET TEMP	°F	424.2	425.9	0.39%
2ND ST REHEATER SHELL SIDE OUTLET TEMP	°F	502.0	498.6	-0.68%
LP TURBINE REHEAT STEAM INLET TEMP	°F	501.4	498.0	-0.68%
MAIN CONDENSER DRAIN OUTLET TEMP	°F	123.258	123.8197	0.45%
MFPTC TUBE SIDE INLET TEMP	°F	124.2	124.7	0.44%
MFPTC TUBE SIDE OUTLET TEMP	°F	140.5	141.6	0.71%
NO 7 FWH OUTLET TEMP	°F	170.5	171.2	0.39%
NO 6 FWH INLET TEMP	°F	171.1	171.8	0.40%
NO 6 FWH OUTLET TEMP	°F	208.8	209.5	0.32%
NO 5 FWH OUTLET TEMP	°F	261.7	262.5	0.30%
NO 4 FWH INLET TEMP	°F	261.5	262.2	0.28%
NO 4 FWH OUTLET TEMP	°F	295.5	296.4	0.30%
NO 3 FWH OUTLET TEMP	°F	362.1	363.2	0.30%
NO 2 FWH INLET TEMP	°F	362.8	363.9	0.30%
NO 2 FWH OUTLET TEMP	°F	398.7	400.0	0.33%

TABLE A-2				
FIELD DESCRIPTION	UNITS	SEQUOYAH 100.0% RTP	SEQUOYAH 101.3% RTP	% DIFF
NO 1 FWH INLET TEMP	°F	399.9	401.3	0.35%
NO 1 FWH OUTLET TEMP	°F	434.3	435.9	0.35%
NO 1 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	4.39	4.52	2.96%
NO 1 FWH DRAIN COOLER APPROACH (DCA)	°F	10.0	10.0	-0.47%
NO 2 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	5.39	5.57	3.22%
NO 2 FWH DRAIN COOLER APPROACH (DCA)	°F	9.32	9.45	1.40%
NO 3 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	2.40	2.50	3.94%
NO 4 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	1.10	1.15	4.34%
NO 4 FWH DRAIN COOLER APPROACH (DCA)	°F	0.44	0.53	16.97%
NO 5 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	5.94	6.12	2.95%
NO 5 FWH DRAIN COOLER APPROACH (DCA)	°F	10.0	10.2	2.42%
NO 6 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	5.37	5.50	2.30%
NO 6 FWH DRAIN COOLER APPROACH (DCA)	°F	10.1	10.3	1.74%
NO 7 FWH TERMINAL TEMP DIFFERENCE (TTD)	°F	3.76	3.79	0.98%
NO 1 FWH DRAINS OUTLET TEMP	°F	409.9	411.3	0.33%
NO 2 FWH DRAINS OUTLET TEMP	°F	372.1	373.3	0.33%
NO 3 HDT DRAINS OUTLET TEMP	°F	364.1601	365.3397	0.32%
NO 4 FWH DRAINS OUTLET TEMP	°F	261.9	262.7	0.32%
NO 5 FWH DRAINS OUTLET TEMP	°F	218.8	219.7	0.42%
NO 6 FWH DRAINS OUTLET TEMP	°F	181.2	182.1	0.47%
NO 7 HDT DRAINS OUTLET TEMP	°F	173.4	174.1	0.41%
MAIN TURBINE CONTROL VALVE INLET ENTH	BTU/LB	1196.0	1196.8	0.06%
MAIN CONDENSER BACKPRESSURE	IN-HG	3.77	3.83	1.53%
MFPT EXHAUST STEAM OUTLET PRESS	IN-HG	7.45	7.70	3.33%
MAIN FEED PUMP TURBINE OUTPUT	KW	9448	9877	4.35%
TOTAL GENERATOR OUTPUT	MWe	1139.532	1150.052	0.91%
STEAM GENERATOR STEAM OUTLET PRESS	PSIA	857.0	832.0	-3.00%
MAIN TURBINE CONTROL VALVE INLET PRESS	PSIA	831.3	807.3	-2.97%
HP TURBINE IMPULSE PRESS	PSIA	614.6	639.3	3.87%
HP TURBINE 1ST EXTRACTION OUTLET PRESS	PSIA	396.4	403.3	1.71%
HP TURBINE 2ND EXTRACTION OUTLET PRESS	PSIA	272.6	277.3	1.68%
HP TURBINE EXHAUST STEAM OUTLET PRESS	PSIA	170.5	173.1	1.46%
MOISTURE SEP SHELL SIDE INLET PRESS	PSIA	168.8	171.3	1.46%
MOISTURE SEP SHELL SIDE OUTLET PRESS	PSIA	163.8	166.2	1.46%
1ST ST REHEATER SHELL SIDE OUTLET PRESS	PSIA	162.3	164.7	1.46%
2ND ST REHEATER SHELL SIDE OUTLET PRESS	PSIA	160.6	163.0	1.46%
LP TURBINE REHEAT STEAM INLET PRESS	PSIA	157.4	159.8	1.46%
LP TURBINE 4TH EXTRACTION OUTLET PRESS	PSIA	67.0	68.0	1.43%
LP TURBINE 5TH EXTRACTION OUTLET PRESS	PSIA	42.4	43.1	1.58%
LP TURBINE 6TH EXTRACTION OUTLET PRESS	PSIA	16.2	16.4	1.56%
LP TURBINE 7TH EXTRACTION OUTLET PRESS	PSIA	6.95	7.06	1.59%
MOISTURE REMOVAL STAGE 5 OUTLET PRESS	PSIA	3.18	3.23	1.58%
2ND STAGE REHEATER TUBE SIDE INLET PRESS	PSIA	823.0	799.2	-2.97%
1ST STAGE REHEATER TUBE SIDE INLET PRESS	PSIA	392.4	399.2	1.71%
NO 1 FWH SATURATION PRESS	PSIA	376.6	383.1	1.71%

TABLE A-2				
FIELD DESCRIPTION	UNITS	SEQUOYAH 100.0% RTP	SEQUOYAH 101.3% RTP	% DIFF
NO 2 FWH SATURATION PRESS	PSIA	259.0	263.4	1.68%
NO 3 FWH SATURATION PRESS	PSIA	162.0	164.4	1.46%
NO 4 FWH SATURATION PRESS	PSIA	63.6	64.6	1.43%
NO 5 FWH SATURATION PRESS	PSIA	40.3	40.9	1.58%
NO 6 FWH SATURATION PRESS	PSIA	15.4	15.6	1.56%
NO 7 FWH SATURATION PRESS	PSIA	6.60	6.71	1.59%
MAIN FEED PUMP TURBINE SPEED	RPM	4727	4786	1.23%

References

1 Main Feed Pump Turbine and Condenser OEM Evaluations, Sequoyah Unit 1 SGR Project - Bechtel Job No. 24370, January 12, 2001 (B38 010112 810) which includes Siemens-Westinghouse Power Corporation's "Feasibility Study to Uprate EMM-25A1N Steam Generator Feedwater Pump turbines, Serial No.'s 15A3146-1, 2, 3, 4".

Response to Part 3
Refer to Enclosure 1, "NSSS Performance Parameters" for the bounding 1.3% uprate parameters used in this evaluation (also see Enclosure 6, Table 2.1-1).

Question 2 (TXX-99115 - Attachment 3):

Solid, liquid, and gaseous radioactive waste activity are influenced by the reactor coolant activity which is a function of the reactor core power. What is the impact of these systems by the increase in power?

Response:

Offsite doses from normal effluent releases remain below referenced bounding results, which are within 10CFR50 Appendix I limits. Further, the capabilities of the plant radioactive waste processing systems were evaluated to assess the effects of the 1.3% power uprate. Thus, the capability to process and store effluents will not be significantly impacted by the 1.3% uprate.

The solid waste management and liquid waste processing systems are designed to control, collect, process, store and dispose of radioactive wastes due to normal operation including anticipated operational transients. Operation of these systems are primarily influenced by the volume of waste processed, which is not expected to change as a result of the 1.3% uprate condition. Thus, the capability of the solid waste management and liquid waste processing systems are not significantly impacted by the 1.3% uprate.

In summary, the 1.3% power uprate has no significant effect on any of the waste subsystems or components of these subsystems. Because these systems are typically operated in a batch mode, the only potential effect is a slight increase in the frequency at which the batches may be processed. These systems continue to meet current design bases.

Question 3 (TXX-99115 - Attachment 3):

Discuss why the current containment analysis remains appropriate for use at power uprate conditions.

Response:

Refer to Enclosure 6, Sections 2.5.1 and 2.5.2. These sections concluded that the current LOCA and Steamline Break Mass and Energy Releases remained bounding for the 1.3% power uprate conditions, so the current containment analysis is also bounding.

Question 1 (TXX-99115 - Attachment 6):

In regard to Section B.4 of Attachment 2 to the reference transmittal, provide the maximum-calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of the RPV and internals (such as RPV nozzles, lower and core plates, core barrel, baffle/barrel, control rod drive mechanism, and fuel assembly, etc.), the allowable code limits, the Code and Code edition used in the evaluation for the power uprate. If different from the Code of Record, provide the necessary justification. Also, provide an assessment of flow-induced vibration of the reactor internal components due to power uprate.

Response:

As noted in Enclosure 6, Section 2.1, the 1.3% uprate conditions resulted in very small changes to the NSSS design conditions (e.g. - T_{cold} and T_{hot} changed by 0.4 °F). In addition, Section 2.2 indicates that there were no changes required for the NSSS design transients. As a result, in most cases, an evaluation was performed to confirm that the existing fatigue usage factors and maximum stress intensities were either negligibly affected or bounded by margin in the existing calculations. Thus, in most cases, revised fatigue usage factors and stress intensities did not need to be calculated.

Refer to Enclosure 6, Section 2.4.1.1 for a discussion of the RPV structural evaluation. The Code version used in the evaluation is the 1968 Edition of Section III of the ASME Code (no addenda), which is the same as the current Code of record for these components.

Refer to Enclosure 6, Section 2.4.1.4 for a discussion of the RV Internals evaluation. The reactor internals are not licensed to a Code version and were originally designed based on sound engineering practice.

Refer to Enclosure 6, Section 2.4.2 for a discussion of the CRDM evaluation. The Code version is the same as the current Code of record.

The flow-induced vibration analysis for the internals was unaffected since the power uprate did not require a change to the plant mechanical design flow.

An evaluation also concluded that the existing fuel assembly fatigue usage factors and maximum stress intensities were either negligibly affected or bounded by margin in the existing calculations (see section 3.2.4 of Enclosure 6). Thus, for the fuel assembly, revised fatigue usage factors and stress intensities did not need to be calculated.

In addition, on a cycle-specific basis, the mechanical design of the fuel assemblies is verified to meet all current design criteria. The fuel vendor performs the required analyses using methods specific to the fuel type. This evaluation is documented in the cycle-specific Reload Safety Evaluation, performed in accordance with 10 CFR 50.59.

Question 2 (TXX-99115 - Attachment 6):

On page 22 of Attachment 2 to the reference transmittal, provide the methodology and assumptions used for evaluating the reactor coolant piping systems, equipment nozzles, and supports for the increased hot leg and cold leg temperatures, increased dynamic hydraulic forcing functions, and the affected design transients due to the power uprate, as stated in the transmittal. Also, provide the calculated maximum stress, critical locations, allowable stress limits, and the Code and Code edition used in the evaluation for the power uprate.

Response:

Refer to Enclosure 6, Section 2.4.3 for a discussion of the RCL piping related evaluations performed for the 1.3% uprate. Enclosure 6, Section 2.2 indicates that none of the NSSS design transients, which include those for the reactor coolant system piping and nozzles, are affected by the uprate conditions.

Section 2.5.3 indicates that the current LOCA hydraulic forcing functions remained bounding for the uprate conditions.

For the Reactor Coolant System (RCS) temperature changes, an evaluation demonstrated that the current analyses for the reactor coolant loop piping, primary equipment nozzles, primary equipment supports and pressurizer surge line piping remained bounding for the uprated conditions due to the conservative nature of inputs for the current analyses. Thus, there were no new calculated maximum stresses, critical locations, and loads.

In addition to the above, an evaluation was performed to demonstrate that the existing fatigue usage factors for the reactor coolant loop piping, nozzles and auxiliary lines remained bounding. The uprated design conditions only impacted the starting and ending temperatures associated with cooldown and heat-up events. The potential slight increase in fatigue was offset by existing margin in the current analysis.

Furthermore, the evaluation performed to address the effects on the pressurizer surge line stratification analysis included a review of the fatigue analysis and the stratification loadings that were transmitted to the pressurizer nozzle from the surge line piping. The potential load increases (from the increased T_{hot}) were determined to be bounded by the current analysis since the analysis used conservative envelopes that lumped various transients under a reduced number of bounding thermal cases. Therefore, the current analysis results remain unchanged for the 1.3% uprate conditions.

Finally, by taking credit for the conservative nature of the existing inputs, it was not necessary to re-calculate the stresses and CUF values in accordance with the applicable Code versions. Thus, it was not necessary to change or review the existing Code versions.

Question 3 (TXX-99115 - Attachment 6):

Were the analytical computer codes used in the power uprate evaluation different from those used in the original design-basis analyses? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

Response:

The analytical computer codes used in the Westinghouse NSSS analysis are either the latest revisions of the computer codes presently described in the FSAR and/or are the same computer codes used in the original design basis analyses.

The analytical computer codes used in the Framatome fuel and safety analyses are also either the original computer codes or latest revisions of the computer codes presently described in the following reports:

1) BAW-10220P, Rev 0, "Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2."

Codes listed:
RELAP/MOD2-BAW
REFLOOD3B
BEACH
LYNXT
TACO3
GDTACO3
ANSYS
STARS
CASMO/NEMO

2) BAW-10084P-A, Rev. 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," B&W Fuel Company, Lynchburg, Virginia, July 1995.

Codes listed:

CROV

3) BAW-10186P-A, Rev. 1, "Extended Burnup Evaluation," Framatome Cogema Fuels, Lynchburg, Virginia, April 2000.

Codes listed:
KOROS

Question 4 (TXX-99115 - Attachment 6):

In reference to the reactor coolant pump (RCP) structural analysis on page 23 of Attachment 2 to the reference transmittal, you stated that "an analysis was performed to determine the impact of the revised design conditions on the stresses and fatigue usage of the RCP ("CRDM" stated in your report should be "RCP") components and the results indicated that the stress and fatigue usage remain within ASME Code limits. Describe the analysis methodology and assumptions (if any), used for evaluating RCP. Also provide the maximum-calculated stress and CUF for the RCP, the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide justification.

Response:

Refer to Enclosure 6, Section 2.4.4. The SG outlet temperature only changed by 0.4°F for the Sequoyah 1.3% uprate conditions. This small change will have a minimal effect on the current stress and fatigue analyses. As a result, it was concluded that the stress intensities remain below applicable limits and the fatigue usage is less than 1.0. The Code version used in the analysis is the same as the Code of record.

Question 5 (TXX-99115 - Attachment 6):

On page 23 of Attachment 2 to the reference transmittal, provide a comparison of the design parameters (i.e., steam pressure, temperature, primary-to-secondary pressure differential, etc.) and transients for the steam generators (SGs) Model D5 against the power uprate condition. Also, provide the maximum calculated stress and CUF for the critical locations (such as the vessel shell, secondary manway bolts, and nozzles), the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record provide justifications. Also, provide an evaluation on the flow-induced vibration of the SG U-bends tubes due to power uprate regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated cross flow velocity.

Response:

Refer to Enclosure 6, Section 2.4.5.1 for a discussion of the maximum calculated stress and CUF values at the critical locations in the model 51 SGs. The results found that CUF values remain less than unity. Table 1 below provides the results of the SG stress and CUF calculations at the critical locations. The Code version used in the evaluation is the 1971 Edition of Section III of the ASME Code through the Summer 1972 Addendum, which is the same as the current Code of record for these components.

Table 1: Maximum Stress Intensity Range/Allowable, and Cumulative Fatigue Usage Factors for Normal/Upset Conditions in the Steam Generators						
Component	**Section or Location**	**Maximum Stress Range/ Allowable (Baseline)**	**Maximum Stress Range/ Allowable (Uprated)**	**Fatigue Usage Factor (Baseline)**	**Fatigue Usage Factor (Uprated)**	**Comments**
Divider Plate	Junction of Tube Sheet			[ ]a,c	[ ]a,c	Plastic analysis performed
Tube Sheet and Shell Junction	Center of Tube Sheet	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	
Tube to Tube Sheet Weld	Hot Side Center Hole OD	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	
Main Feedwater Nozzle	Shell Near Knuckle (Max. Range)	[ ]a,c	[ ]a,c	[ ]a,c	[ ]a,c	Fatigue usage at knuckle reduced due to transient delumping

Section 2.4.5.3 of Enclosure 6 contains a discussion of the effect of the 1.3% uprate on flow-induced vibration of the SG U-bend tubes. It was found that several tubes will be susceptible to high-cycle fatigue if the operating steam pressure is below a certain value. The pressure at which the tubes become susceptible is much less than the normal operating pressure such that no immediate actions are required. Tubes will be evaluated on a cycle-to-cycle basis in accordance with the TVA Steam Generator inspection program.

Question 6 (TXX-99115 - Attachment 6):

On page 25 of Attachment 2 to the reference transmittal, you stated that the pressurizer structural evaluation was performed by comparing the key inputs in the current pressurizer stress report with the revised design conditions in Table IV-1 and that the results indicated that the design condition used in the current analysis remain bounding for the revised design conditions. Provide a comparison of the design parameters (i.e., RCS pressure hot let temperature, cold leg temperature, temperature differential, etc.), the stratification and cyclic design transients for the CPSES pressurizer against the power uprate condition. Also, provide the maximum calculated stress and CUF at the critical locations (such as surge nozzle, skirt support, spray nozzle, safety and relief nozzle, upper head/upper shell and instrument nozzle) of the pressurizer, the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide justification.

Response:

Refer to Enclosure 6, Section 2.4.6 for the discussion of the power uprate effect on the Sequoyah Pressurizer. Minor changes in temperature result from the 1.3% uprate conditions, and the primary side design transients were also found to be unaffected. It was therefore concluded that the current stress and fatigue results for the Pressurizer continue to remain bounding.

The Code version used in the evaluation is the 1968 Edition of Section III of the ASME Code, which is the same as the current Code of record for this component.

Question 7 (TXX-99115 - Attachment 6):

Discuss the operability of safety-related mechanical components (i.e., valves and pumps) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed.

Response:

The safety-related pumps are designed for the 10CFR50 App "K" required power level (102%). Refer to Enclosure 6, Section 2.3.1 for a discussion concerning the 1.3% uprate effect on systems for residual heat removal, chemical and volume control, and safety injection. Also see Enclosure 6, Section 2.4.7 for the effect on the NSSS auxiliary equipment. The flow requirements of the Auxiliary Feedwater Pumps (motor and turbine driven) are not effected by this modest power uprate. The steam generator MSSV setpoints will also remain the same; therefore, the steam generator pressures at which the equipment is required to pump against will be unchanged.

The air operated valves that are required to be operable are powered from the accumulators and are unchanged from present design limits. For further discussion of BOP valves, refer to Enclosure 6, Section 2.3.2.

No changes to the TVA procedures that address the MOV program are required as a result of this 1.3% power increase. The methodology used to document the requirements of the MOVATS and MOV program are standard for the three nuclear sites at TVA. Maximum differential temperatures and pressures (design) are used for sizing requirements for normal operation and worst case conditions are used for the accident required actions. As these conditions bound the power increase conditions, no reduction in margin of safety results from the power increase.

Question 8 (TXX-99115 - Attachment 6):

(This question has been subdivided in order to provide clearer responses.)

a) In reference to Section C on page 26 of Attachment 2 to the reference transmittal, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate.

b) Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports.

c) Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code editions used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

d) Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

Response:

The existing Code of record analyses of the BOP piping systems remain valid for the 1.3-percent power uprating. The BOP fluid systems design process conditions (i.e., pressures, temperatures) for remain bounding and are unaffected by the power uprate. Therefore, the operating modes used in the piping system qualification of piping systems remain applicable for the power uprate and no reevaluation of the BOP piping systems was required.

Question 9 (TXX-99115 - Attachment 6):

Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of evaluation for power uprate effects on the high energy line break analysis, jet impingement and pipe whip loads for the power uprate conditions.

Response:

Flow-induced vibration potential is a function of the shell side flow rates (i.e., flow velocities) in the various NSSS heat exchangers. Shell side flow rates in these heat exchangers are not significantly affected by the uprating. In addition, all of these heat exchangers have been designed to withstand up to 2 times the shell side design flow without encountering damaging tube vibrations. Therefore, flow-induced vibration is not a concern following the uprating.

The only area for potential problems with flow induced vibration is the Steam Generator - due to the increased steam flow and the decreased pressure. The vendor has concluded that the tube lengths and spacers are adequate for the small increase in steam velocity on the shell side of the tubes and flow induced vibration is not a concern due to the power uprate. Other heat exchangers (i.e., feedwater heaters, main condenser, etc.) on the secondary side are bounded by their design conditions.

The primary side pressures and flows are not affected by the 1.3% uprate as discussed in Enclosure 6, Section 2.1. Therefore, the current high energy line break analyses are bounding.

The high energy line breaks on the secondary side of the plant are discussed in Enclosure 6, Section 2.5.2.

Question 1 (TXX-99115 - Attachment 7):

Provide a description, references, and standards to describe CPSES configuration management/procedures including software.

Response:

The LEFM system is designed as a Quality Related system for TVA-SQN and thus configuration management of the LEFM system is maintained by TVA Standard Programs and Processes (SPP)-9.0, "Engineering." The Software and Firmware Verification and Validation Report by Caldon is described in Topical Report ER-80P, Section 6.4, "Quality Measures in Design, Fabrication and Factory Acceptance Testing of the LEFM." TVA software control for LEFM is in accordance with SPP-2.6, "Computer Software Control."

Question 2 (TXX-99115 - Attachment 7):

In response to Question 16 the methodology used to calculate calorimetric uncertainty is referenced as ASME PTC 19.1 - 1985, Measurement Uncertainty and Is the same methodology as used to determine the uncertainty using the LEFM✓ system.

A review of the CPSES FSAR and TS shows, the following Information:

* **Chapter 15 Page 15.0-16. Section 15.0.7, Instrumentation Drift and Calorimetric errors - Power range neutron Flux" Is deleted but references Section 15.0.6, "Trip Setpoints and Time Delays to Trip Assumed in Accident Analysis" references Section 7.1.2.1.9 and the CPSES Technical Specifications. This references Westinghouse setpoint methodology. PTC 19 is not referenced.**

* **The CPSES FSAR references RG 1.105 and the Westinghouse setpoint methodology not PTC 19.**

* **The CPSES Bases B 3/4 2-11 DNB parameters references the RCS total flow uncertainty as 1.8%. The uncertainty is stated to be based on Westinghouse Revised Thermal Design Procedure which includes measurements of reactor power. The methodology used to develop the associated uncertainties and includes specific treatment of feedwater flow uncertainties. PTC 19 is not referenced.**

* **FSAR Page 4.4-37 Reference 85 lists "Improved Thermal Design Procedure" as the methodology used. PTC 19 is not referenced.**

Response:

Refer to response to Question 16 (TXX-99105).

Question 3 (TXX-99115 - Attachment 7):

For Question 17 provide a calibration report from a calibration lab with accuracy traceable to NIST that indicates the accuracy of the LEFM in fully conditioned flow. Additionally provide a test report from a calibration facility that shows the LEFM accuracy is unaffected by velocity profile changes including those based on piping geometry changes (reducers, header, elbows, etc.) such that it can be confirmed the LEFM is not sensitive to plant specific piping installation effects and that the calibration facility results are directly applicable to a plant specific installation.

Response:

The Sequoyah LEFM system will be calibrated in hydraulically similar piping at Alden Research Laboratories prior to installation at Sequoyah. The results from the calibration laboratory report will be directly applicable to the plant/unit-specific installation and will be incorporated in the site-specific uncertainty analysis prepared by Caldon for the Sequoyah LEFM Check system. This analysis can be made available for NRC review.

C. TXX-99195

Question 1 (TXX-99195):

Provide a comparison of the relevant acceptance criterion to the appropriate design limit (e.g., DNBR, RCS pressure) for each of the following safety analyses:

15.4.2 Uncontrolled RCCA withdrawal from power
15.4.7 Misloaded fuel assembly
15.4.8 Rod Ejection
15.4.3 Dropped RCCA

Response:

The relevant acceptance criteria for the Uncontrolled RCCA Withdrawal at Power event are (1) peak primary and secondary pressures do not exceed 110% of design pressure and (2) minimum DNBR remains above the 95/95 DNBR limit for the correlation used. The system transient analysis, used to calculate system pressure, is unaffected by the power level upgrade and the RCS pressurizer pressure plotted in Sequoyah FSAR Figures 15.2.2-2 and 15.2.2-5 remains valid. Furthermore, the primary and secondary pressure response to an Uncontrolled RCCA Withdrawal at Power is bounded by the Loss of Electric Load event. The minimum DNBR was recalculated using the revised thermal power level and calorimetric uncertainty because the increased power level has an adverse impact on minimum DNBR. The new minimum DNBR is 1.501 and this is sufficiently greater than the BWCMV-A thermal design basis limit of 1.431. See Section 3.3.7.2 of Enclosure 6 for further discussion of the Uncontrolled RCCA Withdrawal at Power analysis for Sequoyah.

The SQN FSAR does not contain a system transient or DNB analysis of the Inadvertent Loading of a Fuel Assembly Into an Improper Position event. It was concluded that fuel misloadings are low probability events, owing to administrative controls regarding fuel pellet loading in a fuel pin, fuel pin loading in an assembly, and fuel assembly manufacture. A steady-state x-y power distribution analysis was performed that confirmed that power distribution effects resulting from misloading events will either (1) be readily detected by the in-core moveable detector system or (2) be of a sufficiently small magnitude to remain acceptable and within the design peaking limits. None of these conclusions will be affected by the power level upgrade. See Section 3.3.8.3 of Enclosure 6 for further discussion of the Inadvertent Loading of a Fuel Assembly Into an Improper Position analysis for Sequoyah.

The FSAR analysis of the Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) event considered an RCCA ejection from 0% of RTP (HZP) and an RCCA ejection from 102% of RTP (HFP). The acceptance criteria for this event are (1) average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel, (2) fuel melting limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1, and (3) peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits. The system transient analysis of this event was unaffected by the power level upgrade. The minimum DNBR was recalculated using the revised thermal power level and calorimetric uncertainty because the power uprate adversely affects the SCD DNBR analysis at hot full power. The analysis showed that the limiting case does not result in fuel damage beyond the 10% fuel melt limit. See Section 3.3.9.7 of Enclosure 6 for further discussion of the RCCA Ejection analysis for Sequoyah.

The relevant acceptance criteria for the Dropped RCCA event are (1) peak primary and secondary pressures do not exceed 110% of design pressure and (2) minimum DNBR remains above the 95/95 DNBR limit for the correlation used. The FSAR analysis of the limiting case of a dropped RCCA in automatic rod control mode assumes the initial power to be at a nominal value of 100% RTP. The nominal core power level on which the safety analysis is based will be different following the power level upgrade. However, the analysis of this event assumes that the core power increases from nominal power to the high neutron flux trip setpoint that is unchanged by the power level upgrade. The core trip on high neutron flux is unaffected by the power upgrade because the power range high neutron flux setpoint (high and low settings) will be redefined (i.e., from 118% of 3411 MWt to 116.5% of 3455 MWt) so that the reactor trips at the same value in absolute megawatts. The response to the dropped RCCA event is also dependent on the safety analysis values of reactivity insertion due to rod motion, initial axial power distribution, moderator temperature reactivity coefficient and Doppler reactivity coefficient, all of which are unaffected by the power level upgrade. With respect to system pressurization, the loss of electric load (LOEL) event is most limiting. System pressures resulting from a dropped RCCA are well bounded by the results of the LOEL. Finally, the dropped RCCA event is analyzed each fuel cycle as a part of the core design process to ensure that DNB cannot occur. Because of the above arguments, this event was not reanalyzed for the power level upgrade. See Section 3.3.7.3 of Enclosure 6 for further discussion of the Dropped RCCA analysis for Sequoyah.

Question 2 (TXX-99195):

The topical report detailing the analysis of an inadvertent boron dilution event (RXE-91-002-A) indicates that the analysis assumed a power level of 100 percent. Discuss the sensitivity of the analysis results to initial power level. Summarize the methods and results of any supporting sensitivity analysis and provide references.

Response:

The Sequoyah FSAR analysis of the Uncontrolled Boron Dilution event considers the accident to occur during reactor refueling, startup, and full power operation. This event evolves slowly and resembles a low rod worth withdrawal event. The parameters that dominate in the Sequoyah plant response to this event are the dilution flow rate, initial boron concentration, critical boron concentration, boron worth, and RCS volume. These parameters are unaffected by the 1.3% power level upgrade. Initial core power is not directly modeled in this event. Concerning system pressurization limits, the results associated with the LOEL event bound those that could be postulated for the boron dilution event. In addition, each reload fuel cycle design is evaluated for acceptability by reviewing the predicted cycle values of boron concentrations and reactivity worth.

Question 3 (TXX-99195):

Discuss the sensitivity of the analysis results to initial power level for the SG tube rupture event. Summarize the methods and results of any supporting sensitivity analysis and provide references.

Response:

The Sequoyah FSAR analysis of the SG Tube Rupture event was performed at 102% of RTP which includes a 2% calorimetric uncertainty. The current analysis can accommodate a power uprate of 1.3% and equipment changes that reduce measurement uncertainty to 0.7%. The important parameters in the system response to this event are break size, low pressurizer pressure reactor trip and safety injection setpoints, ECCS operation and capacity, and steam line safety valve setpoint and capacity. None of these parameters are affected by the power uprate.

The analysis of the environmental consequences of a postulated Steam Generator Tube Rupture is presented in FSAR Section 15.5.5. The key parameters affecting this analysis are primary-to-secondary leakage, primary coolant activity, iodine partition factor, and steam generator blowdown rate. None of the key input parameters for this event are affected by the power uprate. The current FSAR analysis of this event used an initial primary coolant activity and core thermal power based on 3582 MWt (105% of 3411 MWt). Thus, the current analysis can accommodate a power uprate of 1.3% and equipment changes that reduce measurement uncertainty to 0.7%. Section 3.3.9.4 of Enclosure 6 provides a discussion of the SG Tube Rupture event for Sequoyah.

Question 4 (TXX-99195):

CPSES technical specifications contain a surveillance requirement (3.3.1.2) requiring that power levels measured by nuclear instruments and by the N-16 monitoring system be checked to within 2% of the daily calorimetric. Explain why this surveillance requirement is not being modified to require that the readings be within 1% of the calorimetric.

Response:

Sequoyah does not have the N-16 monitoring system therefore this portion of the question is not applicable.

The uncertainty associated with the accuracy of the plant calorimetric measurement is considered in the plant safety analyses. It is this uncertainty that can be reduced through the use of the improved LEFM Check system.

Technical Specification Surveillance Requirement (SR) 4.3.1.1.1, Power Range Neutron Flux Channel calibration by heat balance comparison, is a requirement for the re-normalization of the Nuclear Instrumentation System (NIS) power range channels if the allowed deviation (±2% RTP) between the power calculated by the plant calorimetric measurement and the NIS indicated power is exceeded. This deviation is considered in the uncertainty analyses of those reactor trip functions that are based on the NIS power range channels.

Question 5 (TXX-99195):

In response to a previous request for additional information the revised overpower N-16 allowable value of 113.5% of rated thermal power was defended as having been derived based on WCAP-12123 methods. Provide the detailed calculation showing how the allowable value for the N-16 overpower trip was determined.

Response:

The Sequoyah design does not employ an N-16 overpower trip. Thus, this RAI would not apply to the Sequoyah 1.3% uprate submittal.

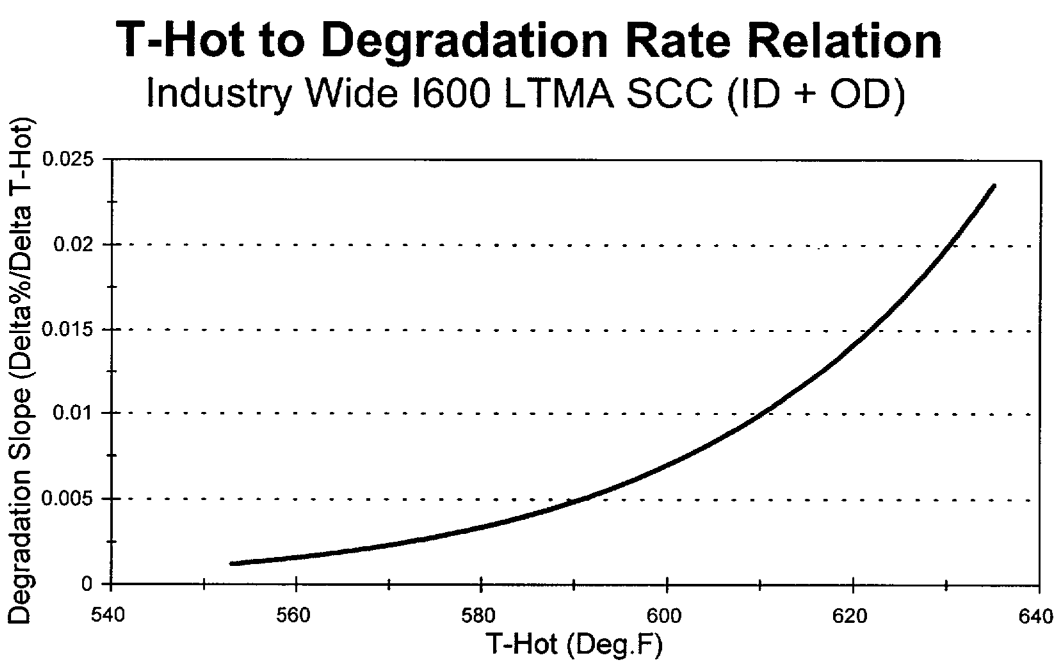
D. TXX-99164

Question 1 (TXX-99164):

The licensee needs to evaluate the effects of the power uprate on the tube degradation mechanisms (present and potential) including wear.

Response:

Of the changes proposed in the uprate, the minor change in temperature and the minor change in secondary pressure are the only changes that would affect corrosion rates. The T_{hot} change is considered to be the most sensitive to corrosion rates. The graph below (T-Hot Degradation Rate Relation) represents industry data for Inconel 600 tubing. The graph illustrates the impact of temperature to corrosion rates, however, the affect of this small increase can not be quantified.



When assessing structural integrity of indications identified during an inspection, the change in the secondary pressure would be a sensitive parameter; however, also too small to be a quantifiable impact. Secondary side pressure is an input to calculations performed during the inspection to determine tube integrity.

Question 2 (TXX-99164):

Discuss how steam generator tube inspection plan will be assessed to monitor potential tube degradation including wear. Will additional inspections be

necessary? How will TXU Electric assess their inspection plans should new degradation mechanisms be discovered?

Response:

The TVA Steam Generator Program presently contains all requirements of NEI-97-06, including assessing growth rates. The uprate would change nothing in the methodology used to perform condition monitoring and operational assessments. The primary and secondary pressure are inputs to the tube integrity calculations performed. These pressures are verified each inspection to ensure the limiting steady state delta pressure for the past cycle is used in calculations. With respect to the slightly higher temperature, the SQN Steam Generator Program presently includes consideration of growth rate analyses. Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs will be made. Degradation growth rate changes will be incorporated into the operational assessment. New degradation, which means degradation that is not expected to occur, will be entered into the SQN Corrective Action Program and a root cause analysis will be performed. Inspection plans will be expanded as necessary.

Question 3 (TXX-99164):

The licensee needs to evaluate if the Technical Specification plugging limit of 40 percent through wall degradation is still adequate.

Response:

The only mechanisms allowed to remain in service using the plugging limit of 40 percent are AVB Wear, Cold Leg Thinning, and PWSCC at drilled support plates. The 40 percent limit is conservative, and will not be affected by the small uprate. As discussed above, the affects of this small uprate are nonquantifiable.

E. TXX-99203

Question 1 (TXX-99203):

In section 6 of the Caldon Topical, reference is made to use of the LEFM to calibrate the NIs. How does CPSES plan to use the LEFM and explain the relation of the LEFM as M&TE with regards to Appendix B.

Response:

The requirement in Technical Specification Surveillance (SR) 4.3.1.1.1 (Table 4.3-1 note 2), Power Range Neutron Flux Channel calibration by heat balance comparison, is to "adjust" the power range neutron flux channels if the absolute difference between the calorimetric heat balance calculation and the NIS power range channels output is greater than 2%. Using this guideline, it is more correct to state that the NIS indication of reactor power is normalized, rather than calibrated, against the reactor power calculated with the LEFM-based secondary plant power calorimetric measurement. As such, the application of M&TE is not strictly appropriate.

The improved LEFM system is included in the non-Appendix B Quality Assurance program and designated as Quality Related as described in FSAR Section 17.2 and the Nuclear Quality Assurance Plan, TVA-NQA-PLN89-A.

Question 2 (TXX-99203):

Page 5.5 of the Caldon Topical discusses the use of the LEFM to correct the Venturi measurement. Page 8 of the TXU license amendment request also discusses the use of the LEFM for providing correction for the venturi. What are CPSES plans when the LEFM is unavailable and the venturis are used for normalizing the NIs?

Response:

The referenced page of the Caldon Topical Report discusses use of the improved LEFM for correcting the venturi-based feedwater flow indication for effects such as fouling. As detailed in response to Question 29 (TXX-98274), SQN currently uses the LEFM 8300 strap-on system measurement of feedwater flow to correct fouling effects for the venturis. This correction is used for the power calorimetric measurements only. Refer to Question 29 for additional discussion on SQN plans to continue correcting for feedwater venturi fouling based on the improved LEFM Check System.

Through the use of the improved LEFM, the power calorimetric uncertainty is shown to be less than 0.7% RTP. However, this uncertainty calculation is not applicable to the case where the power calorimetric is based on venturi-based feedwater flow indication, even if the improved LEFM is used to correct the venturi-based feedwater flow indications for effects such as fouling.

Sequoyah Units 1 & 2 will be operated in accordance with the safety analyses and the applicable power calorimetric uncertainty analysis. When the improved LEFM-based calorimetric measurement is available, the plant will be operated at a nominal core power of 3455 MWt. The reactor operators will be provided procedural guidance for those occasions when the improved LEFM is not available. For those instances a new section of the SQN Technical Requirements Manual (TRM) will specify the appropriate actions to be taken when the LEFM is unavailable. The actions required for continued power operation when the LEFM is unavailable are described in Section 4.0 of Enclosure 6.

F. TXX-98274, TR-ER80P

Question 3 (TXX-98274, TR-ER80P):

Describe how the LEFM is used in calorimetric power determinations.

Response:

See a general description for the use of the LEFM in Enclosures 1 and 4 and responses to Questions 4 (TXX-99195), 1 and 2 (TXX-99203).

Question 5 (TXX-98274, TR-ER80P):

Who is responsible and how are Calibration, Maintenance, and Training performed and achieved?

Response:

The Verification Test of the LEFM spoolpiece is contracted by Caldon and performed at Alden laboratory before the installation in the main feedwater header at SQN. The requirements for installation are within SQN Design Change Notices (DCN's) 20637(unit 1) and 20638(unit 2). The LEFM software has provisions for on-line monitoring and diagnostics and will alert the operator if the system has failed or the performance of the system indicates a maintenance/alert condition. In that event, it may become necessary for maintenance to be performed. This necessary maintenance will be procedurally controlled.

Training on the operation and maintenance of the LEFM system is contractually provided by Caldon. Maintenance will be performed by SQN plant personnel per vendor recommendations contained in vendor supplied instructions and does not require any special skills that would be beyond that encompassed in the SQN I&C technician training program.

Question 6 (TXX-98274, TR-ER80P):

How will monitoring, verification, and error reporting be handled? Provide clarification (list) of Quality Control standards used by Caldon in the design and manufacturing of the LEFM. Provide clarification (list) as to the standards followed under Caldon's verification and validation program.

Response:

TVA believes this question was intended to mainly provide clarification to Caldon's Quality Control Program which TVA will not attempt to address in this response.

SQNP will include the LEFM in the calibration and maintenance program including the preventative maintenance program. The system will be monitored by the System Engineer for reliability. As a plant instrument, all equipment problems fall under the site work control process. All adverse conditions that are identified will be documented on a Problem Evaluation Report (PER) in accordance with the TVAN Corrective Action Program. SQNP has required Caldon to maintain the LEFM software under their V & V Program with requirements that Caldon notify SQNP of any deficiencies that could affect the design basis accuracy.

Question 10 (TXX-98274, TR-ER80P):

How does the LEFM uncertainty compare to the venturi uncertainty at Comanche Peak, in measuring reactor thermal power?

Response:

Refer to Enclosure 1 and Question 16 (TXX-99105) for response.

Question 29 (TXX-98274, TR-ER80P):

How is the LEFM used currently to provide correction factors to the venturis? Is the correction determined on the basis of the absolute accuracy or the repeatability of the LEFM?

Response:

The LEFM 8300 strap-on system that is currently installed at SQN Units 1 & 2 is only used as a basis for determining the correction factor for feedwater venturi fouling. The correction is based on the absolute accuracy of the LEFM but a high degree of repeatability is also required. SQN plans to continue correcting for feedwater venturi fouling based on the improved LEFM Check System. This correction factor will be based on the improved accuracy of feedwater mass flow measurement.

Question 30 (TXX-98274, TR-ER80P):

What action is taken when the LEFM fails?

Response:

The actions required for continued power operation when the LEFM is unavailable are described in Section 4.0 of Enclosure 6. Also see response to Question 2 (TXX-99203).

Question 34 (TXX-98274, TR-ER80P):

Provide a figure analogous to figure 5-2 in the topical using the Comanche Peak site-specific uncertainty values for the venturi and LEFM instruments.

Response:

This question is addressed in Figure 3 and accompanying text in Caldon's engineering report ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓ System", submitted as Enclosure 2 of LAR TVA-WBN-TS-00-06, Docket No. 50-390. This report has been confirmed to be applicable to SQN and is bounding for the 1.3% power uprate.

G. WBN Follow-up submittal dated August 24, 2000

MECHANICAL ENGINEERING BRANCH

Question 1 (WBN - 08/24/00)::

In Section III.5.1.1 of Enclosure 1 and Page E6-16 of Enclosure 6, you stated that in most cases (but not all), revised fatigue usage and stress intensities of the reactor vessel components did not need to be calculated for the power uprate. Please identify components that are impacted by the power uprate and require further calculation. For these components evaluated for the uprated conditions, provide the maximum calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of these components. Also, provide the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification.

Response:

Refer to section 2.4.1.1 of Enclosure 6 for an evaluation of the effects that the Sequoyah 1.3% uprating conditions have on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress reports and addenda. The evaluation concluded that the maximum ranges of stress intensities and fatigue usage factors reported in the current stress reports for the Sequoyah reactor vessel continue to remain bounding with the 1.3% uprating conditions and satisfy the applicable limits of Section III of the ASME Code. The applicable values and limits are presented in Table 2 below. The 1968 ASME code edition (no addenda) was used in the evaluation; this is the same edition as the code of record.

Table 2: Stress Intensities and Fatigue Usage Factors for the Reactor Vessel

Location	$P_L + P_b + Q$ Range	U_c
Main Closure Flange Region 1. Closure Head Flange 2. Vessel Flange 3. Closure Studs	 [ ]a,c [ ]a,c [ ]a,c	 [ ]a,c [ ]a,c [ ]a,c
Bottom Head to Shell Juncture	[ ]a,c	[ ]a,c
Vessel Shell (Transition Taper)	[ ]a,c	[ ]a,c
Inlet Nozzles and Supports	[ ]a,c	[ ]a,c
Outlet Nozzles and Supports	[ ]a,c	[ ]a,c

CRDM Housings	[ ]a,c	[ ]a,c
Bottom Head Instrumentation Tubes	[ ]a,c	[ ]a,c
Core Support Pads	[ ]a,c	[ ]a,c
Auxiliary Head Adapter Stub	[ ]a,c	[ ]a,c
Auxiliary Head Adapter Pipe Cap (References 10 and 11)	[ ]a,c	[ ]a,c

Question 2 (WBN - 08/24/00):

In regard to Section III.5.2.3 of Enclosure 1, provide the maximum calculated stress and CUF at the critical locations of the reactor internal components (such as lower and upper core plates, core barrel, baffle/barrel, and fuel assembly) for the power uprate condition. If codes are used in the evaluation for the power uprate, provide the allowable Code limits, and the Code and Code edition. Confirm that methodology, assumptions and allowable limits used for the power uprate evaluation are the same as those in the current licensing basis of record.

Response:

The reactor internals are not licensed to a Code version and were designed based on sound engineering practice as summarized in the FSAR. Section 2.4.1.4 of Enclosure 6 describes the evaluation performed for the reactor internals components. Following is a summary of the results relative to maximum calculated stress and CUF.

Baffle-Barrel Region Components:

No new CUF calculations were performed. The existing design transients remained valid for the 1.3% power uprate. The heat generation rates seen by the baffle-barrel region for the 1.3% power uprate were bounded by the existing analysis. The existing analysis remains applicable for the 1.3% power uprate conditions.

Upper Core Plate:

No new CUF calculations were performed. The existing design transients remained valid for the 1.3% power uprate. The temperature effects due to the fluid and heat generation on the upper core plate remained essentially the same compared to the previous fuel cycles. The analysis of record remains applicable for the 1.3% power uprate conditions.

Lower Core Plate:

New CUF calculations were performed for the 1.3% power uprate conditions. This was required due to the increase in heat generation seen by the lower core plate. The existing design transients remained valid for the 1.3% power uprate. The results were:

Stress Intensity (SI) = [ ]+a,c KSI
$3S_m$ = [ ]+a,c KSI
Margin = $(3S_m/SI)-1$ = [ ]+a,c

CUF = [ ]+a,c (Allowable Limit = 1.0)

The Sequoyah reactor internals were designed prior to the introduction of Subsection NG of the ASME B&PV Code Section III. The ASME B&PV Code, Section III, Division I, 1989 Edition was used as the acceptance criteria for allowable stresses.

Question 3 (WBN - 08/24/00):

In regard to Section III.5.2.2 of Enclosure 1, provide an assessment of flow-induced vibration of the reactor internal components due to the changes of T_{hot} and T_{cold} for the power uprate.

Response:

As discussed in section 2.4.1.4 of Enclosure 6, the T_{HOT} and T_{COLD} fluid densities change by less than 0.1% for the 1.3% uprate conditions. These changes are judged to be negligible when compared to the design basis temperatures. The mechanical design flow is also unaffected by the uprate. Therefore, the effect on the flow-induced vibration of the reactor internals is negligible or essentially nonexistent.

Question 4 (WBN - 08/24/00):

In reference to Section III.5.3 of Enclosure 1, provide an evaluation of the control rod drive mechanism with regard to the stress and fatigue usage as a result of the power uprate. Also, provide the allowable Code limits for the critical components evaluated, and the Code and Code edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

Response:

Section 2.4.2 of Enclosure 6 contains the evaluation of the CRDMs for the Sequoyah 1.3% uprate. The CRDMs are affected by the cold leg temperature. The 1.3% uprate results in a decrease in T_{cold} of 0.4°F. The current stress and fatigue evaluations are based on operating temperatures which bound the expected change in T_{cold}. As such, the allowable stress and fatigue usage limits continue to be satisfied.

As noted in section 2.4.2 of Enclosure 6, the Code version is the same as the Code of record. Depending on which CRDM design is evaluated, the Code of record is either the ASME B&PV Code Section III 1968 Edition or the 1971 Addenda.

Question 5 (WBN - 08/24/00):

In reference to Section III.6.4 of Enclosure 1, you stated that the 2-percent increase in forces (loop forces increase due to a reduction of T_{cold}) was offset by a more representative characterization of the loop at the break location. Explain more about the approach using "the more representative characterization of the loop," which was claimed to result in 17-percent reduction in loop force at the break location. Is this approach currently used by WBN for a licensing basis documented in the UFSAR?

Response:

Refer to section 2.5.3 of Enclosure 6. The evaluation performed for the effect of the 1.3% uprate on LOCA Hydraulic Forces indicates that the forces will increase with the uprated power conditions. However, the dynamic effects of a primary coolant pipe break have been eliminated from the SQN design basis based on the application of a leak-before-break analysis methodology. Dynamic pipe break forces for Sequoyah are currently based on an accumulator line break on the cold leg and a pressurizer line break on the hot leg. The hydraulic force increase in these pipe breaks associated with power uprate will remain bounded by the current analysis of record. The current analysis of record evaluates a reactor coolant pipe break for the original power conditions.

Question 6 (WBN - 08/24/00):

Provide evaluation of the potential of flow induced vibration for the steam generator U-Bend tubes quantitatively based on the increase in feedwater flow and the increase in pressure difference between the primary system pressure

(unchanged at 2250 psi) and the decreased steam pressure for the proposed power uprate.

Response:

Section 2.4.5.3 of Enclosure 6 contains the evaluation of the effect of the 1.3% uprate on SG U-bend fatigue due to flow-induced vibration. It concludes that a small number of tubes are susceptible to fatigue if the plant operates below a certain steam pressure. The pressure at which the tubes become susceptible is much less than the normal operating pressure such that no immediate actions are required. Tubes will be evaluated on a cycle-to-cycle basis in accordance with the TVA Steam Generator inspection program.

Question 7 (WBN - 08/24/00):

In Section III.7, "Balance of Plant," you stated that as part of design change process for the power uprate, additional heat balance studies will be performed at higher ambient conditions to assess potential impact on individual BOP components. Please provide such an evaluation and identify systems and components that will be affected by the higher ambient conditions for the power uprate.

Response:

This evaluation has been included in our response to Question 1 (TXX-99115 - Attachment 3).

Question 8 (WBN - 08/24/00):

On Page E6-22 of the reference, you indicated that the licensing basis conditions for the motor-operated valves (MOVs) program by TVA bound the uprated conditions and therefore, the safety-related MOVs at WBN will be capable of performing their intended function(s) following the power uprate. Please discuss effects of the proposed power uprate on the pressure locking and thermal binding of the safety-related power-operated gate valves for Generic Letter (GL) 95-07 and on the evaluation of overpressurization of isolated sections of piping segment for GL 96-06. Identify mechanical components for which functionality at the uprated conditions could not be confirmed.

GL 96-06 Response:

Generic Letter GL 96-06 addresses the overpressurization of isolated piping segments. Two other issues are addressed in the GL but were not pertinent to the RAI.

The isolated segments of pipe have been previously evaluated and have been upgraded, where required, to meet the criteria of the GL. Isolated segments of piping that are susceptible to overpressurization due to thermal stresses imposed by the environment or due to internal heat sources have been provided with thermal safety-relief valves or have been determined to be structurally adequate to withstand the stresses imposed by the thermal loading. LOCA analyses that affect the containment side of the isolated segment of piping have been performed at 102% of 3411 MWt and remain bounding for this power uprate.

The environmental conditions, imposed by LOCAs or secondary side line breaks, have not been affected in an adverse manner that would compromise the piping that is capable of being isolated by segments. The main steam temperature has decreased and this evaluation bounds the small increase in the feedwater temperature in the environmentally qualified rooms.

There is no increase in the possibility of overpressurization of isolated segments of piping.

Response to GL 95-07:

A review of the documentation and evaluations of GL 95-07 was performed to determine if the proposed 1.3% power increase would adversely affect any conclusions or qualifications that were approved by the NRC upon closure of the subject Generic Letter. Of particular interest was the TVA Pressure Locking and Thermal Binding (PLTB) Evaluation Matrix Notes and the NRC's Safety Evaluation (TAC Nos. M93519 and M93520) and the comments and conclusion therein. The conditions that were in the evaluation remain bounding for the 1.3% power uprate conditions and the conclusions, and the NRCs understanding of the basis for those conclusions, remain valid.

The support systems that are in close proximity to the NSSS loop were reviewed for impact. The second isolation valves for the support systems were assumed to be unaffected by the small NSSS hot leg temperature increase. The valves that were listed in the GL 95-07 as being modified to eliminate the potential for PLTB remain unaffected. (Refer to Table 1, Item A, attached). Valves that were listed in the GL 95-07 response as being evaluated to ensure their ability to open under pressure locking conditions were reevaluated for the hot leg temperature increase of 0.4 degrees F. (Refer to Table 1, Item B, attached). In accordance with the guidance of Information Notice 95-14 (Susceptibility of Containment Sump Recirculation Gate Valves to Pressure Locking), a 1 degree F temperature rise may result in a 33 psi pressure increase. The 0.4 degrees F temperature rise results in a pressure increase of approximately 13 psi. The increase in thrust per the Commonwealth Edison Company thrust-prediction methodology due to the 0.4 degrees F is insignificant and is bounded by the current calculations. Additionally, the evaluation assessed the impacts of the 1.3% uprate on the GL 89-10 and Limitorque Technical Bulletin 98-01 update programs and found these to be acceptable. No impacts were identified on the secondary side due to the lower Main Steam system operating pressure. The feedwater temperature increase is bounded by the existing evaluations.

This proposed power uprate does not introduce any increased challenge for thermal binding and/or pressure locking and the responses and conclusions of GL 95-07 and GL 96-06, as well as GL 89-10, remain valid.

TABLE 1

LISTING OF VALVES EVALUATED UNDER GL 95-07

A. Valves that have been previously modified to eliminate PLTB:

1,2-FCV-63-008	RHR to SI
1,2-FCV-63-011	RHR Pump To SI Pump Suction
1,2-FCV-63-022	SI Pump To Cold Leg Injection
1,2-FCV-63-025	Boron Injection Tank Outlet Isolation
1,2-FCV-63-026	Boron Injection Tank Outlet Isolation
1,2-FCV-63-039	Charging Pump Injection
1,2-FCV-63-040	Charging Pump Injection
1,2-FCV-63-072	Containment Sump to RHR Pump Suction
1,2-FCV-63-073	Containment Sump to RHR Pump Suction
1,2-FCV-63-172	RHR Hot Leg Injection
1,2-FCV-72-002	Containment Spray Pump Discharge
1,2-FCV-72-039	Containment Spray Pump Discharge
1,2-FCV-72-040	Containment Spray Header Isolation
1,2-FCV-72-041	Containment Spray Header Isolation
1,2-FCV-74-001	RHR Pump Suction from Hot Legs
1,2-FCV-74-002	RHR Pump Suction from Hot Legs
1,2-FCV-74-003	RHR Pump A Suction
1,2-FCV-74-021	RHR Pump B Suction
1,2-FCV-74-033	RHR Crosstie
1,2-FCV-74-034	RHR Crosstie

B. Valves re-evaluated to ensure operability with the uprate:

1,2-FCV-01-016	Steam Supply to Turbine Driven AFW Pump

1,2-FCV-62-138	Emergency Boration Isolation
1,2-FCV-63-001	RHR Pump Suction From Refueling Water Storage Tank
1,2-FCV-63-006	SI Pump Suction from RHR
1,2-FCV-63-007	SI Pump Suction from RHR
1,2-FCV-63-156	SI Pump A to Hot Legs
1,2-FCV-63-157	SI Pump A to Hot Legs
1,2-FCV-68-332	Pressurizer Power Operated Relief Valves Block
1,2-FCV-68-333	Pressurizer PORV Block

Question 9 (WBN - 08/24/00):

Describe superscripts "a" and "c" which are not defined in Tables 1 and 2 on Pages E6-20 and E6-21

Response:

This question has no applicability for Sequoyah.

Question 10 (WBN - 08/24/00):

Do you project modifications to piping or equipment supports for the proposed power uprate? If any, provide examples of pipe supports requiring modification and discuss the nature of these modifications.

Response:

Section 2.4.3 of Enclosure 6 concludes that there are no modifications required to any of the primary loop or auxiliary system piping or supports as a result of the 1.3% uprating since the existing design basis loads and displacements remain valid.

REACTOR SYSTEMS ENGINEERING BRANCH

Question 1 (WBN - 08/24/00):

The SG Atmospheric Relief Valves (ARVs) are discussed on Page E1-16 of TVA's application. Provide additional information to justify the adequacy of the ARVs' design relief capacity for the 1.4% uprate.

Response:

The ARV sizing criterion in the report (Section 2.3.2.2.2 of Enclosure 6) indicates that the ARVs are adequately sized for the 1.3% uprate conditions.

Question 2 (WBN - 08/24/00):

With respect to the discussion of BELBLOCA, Page E1-35 of TVA's application, discuss the relationship between the MONTEC computer Code and WCOBRA/TRAC and whether it may be used separately from WCOBRA/TRAC.

Response:

This question has no applicability for Sequoyah.

Question 3 (WBN - 08/24/00):

Section 6.5.1, beginning on page E1-37 of TVA's application provides a discussion of the affects on the Non-LOCA/Transient Analyses for the 1.4% power uprate. Please provide additional information to justify the conclusion that DNBR margins remain acceptable.

Response:

DNBR margins were evaluated for each non-LOCA transient as part of the Sequoyah FSAR Chapter 15 safety evaluation. The results of these evaluations are documented in Sections 3.3.7, 3.3.8, and 3.3.9 of Enclosure 6.

Question 4 (WBN - 08/24/00):

TVA's application discusses the Rod Ejection Event, on Page E1-44. Please discuss the acceptance criteria for the fuel pellets with respect to 10 CFR 50, Appendix-A, General Design Criteria 28.

Response:

The FSAR analysis of this event considered an RCCA ejection from 0% of RTP (HZP) and an RCCA ejection from 102% of RTP (HFP). The acceptance criteria for this event are (1) average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel, (2) fuel melting limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1, and (3) peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits. The HZP case is unaffected by the power uprate and the HFP case included a 2% calorimetric uncertainty. The current analysis, therefore, can accommodate a power uprate of 1.3% and equipment changes that reduce measurement uncertainty to 0.7%. Additional parameters that affect the fuel pellet enthalpy, RCS pressure, and centerline fuel melting are time in core life, ejected rod worth and Doppler temperature/power coefficient. These parameters are unaffected by the power uprate. In addition, cycle specific checks will be performed to verify that the RCCA ejection analysis remains applicable for the power uprate.

Although the system response for this event was not reanalyzed, the MDNBR was recalculated using the revised thermal power level and calorimetric uncertainty because the power uprate adversely affects the SCD DNBR analysis at hot full power. The analysis shows that the limiting case does not result in fuel damage beyond the 10% fuel melt limit for the limiting case.

The analysis of the environmental consequences of a postulated Rod Ejection Accident is presented in FSAR Section 15.5.7. The environmental consequences of a rod ejection accident are bounded by the environmental consequences of a loss of coolant accident. Since the loss of coolant accident environmental consequence analysis was found to be unaffected by the power level upgrade, the rod ejection accident environmental consequence analysis is also unaffected.

See Sections 3.3.9.7 and 3.3.10.7 of Enclosure 6 for further discussion of the RCCA Ejection analysis for Sequoyah.

Question 5 (WBN - 08/24/00):

Please provide additional information to justify TVA's conclusion on Page E1-45, that Reactor Trip and ESFAS Setpoints remain acceptable for the 1.4% Power Uprate.

Response:

Uncertainty calculations that are affected by the uprating are those that account for plant operating conditions as part of the assumptions that form the bases for the uncertainty calculations, or those that may have a change to the safety analysis limits. The uncertainty calculations that typically use plant operating conditions to support the input assumptions are:

OTΔT
OPΔT
Steam Generator Level (low and high)
RCS Low Flow Trip

Westinghouse anticipates that the changes to the uncertainties due to the uprated conditions alone will have a negligible effect on the above trip setpoints. It is also anticipated that none of the safety analysis limits input will change, due to the uprating conditions. TVA has incorporated changes to its instrument uncertainty calculations based on the uprated conditions. Framatome has evaluated the safety analysis limits relative to the uprated conditions and TVA has incorporated any changes in the applicable design basis documents.

Reactor trip and ESFAS setpoints were originally defined by the plant vendor. These setpoints, including relevant uncertainties and measurement errors, are used in safety analyses. Safety models incorporate calorimetric error sufficient to accommodate the proposed combined power uprate and calorimetric error reduction.

The power uprate results in a slight increase in vessel ΔT and has the potential to affect overpower and overtemperature ΔT (OPΔT and OTΔT) reactor trips. Coefficients for these trips (K1 term in the OTΔT and K4 term in the OPΔT) have been validated for use at power uprate conditions by assuring that the trip lines remain within existing core safety limits. In the plant safety analyses, the OT ΔT and OPΔT reactor trips are initialized in a manner that represents conditions characteristic of rated thermal power plus uncertainty, the combination of which does not change with the power uprate. Therefore, both the OTΔT and OPΔT trips continue to be applicable before and after the 1.3% power uprate without change.

The "analysis" values of the high neutron flux trips were re-defined to represent identical power limits in MWt, both before and after the power uprate. "Technical Specification" values of these trips were left as-is in terms of percent of thermal power but are actually increased, in terms of MWt. Calorimetric error is included explicitly in the calculation of channel statistical allowance for the high neutron flux trip. This allowance was re-calculated utilizing the reduction in calorimetric error associated with the proposed 1.3% power uprate and shown to be sufficiently small to allow for the above changes.

The only other parameters having the potential to affect reactor and/or ESFAS trips are secondary system parameters - steam pressure and steam flow. The changes in these parameters are small and are judged to be insufficient to affect errors in the existing steam generator level trips (lo-lo and hi-hi narrow range level trips). Existing steam generator level trips, therefore, continue to be applicable subsequent to a 1.3% power level increase.

MATERIALS AND CHEMICAL ENGINEERING BRANCH

STEAM GENERATOR RELATED QUESTIONS

The following questions relate to steam generator tube degradation as discussed in Sections 5.6.5, 5.6.6, and 5.6.7 in TVA's submittal dated June 7, 2000.

Question 1 (WBN - 08/24/00):

Section 5.6.5 - TVA stated that ". . . T_{hot} is expected to increase by 0.4 degree F for the 1.4% uprate and is considered to be the most sensitive operating parameter with respect to corrosion . . ." TVA also stated that ". . . these changes are expected to have an insignificant effect on the tube corrosion mechanisms since they are relatively minor and are comparable to the range of uncertainties used in assessing corrosion . . . " (1) TVA should expand on why the increase in T_{hot} is the most sensitive operating parameter with respect to corrosion. (2) If the increase in T_{hot} is within the range of uncertainties used in assessing corrosion and is relatively minor, TVA needs to describe the uncertainties in terms of quantitative or qualitative analysis to support the above statement.

Response:

Of the changes proposed in the uprate, the minor change in temperature and the minor change in secondary pressure are the only changes that would affect corrosion rates. The T_{hot} change is considered to be the most sensitive to corrosion rates. When assessing structural integrity of indications identified during an inspection, the change in the secondary pressure would be a sensitive parameter; however, also too small to be a quantifiable impact. (2) The uncertainties mentioned in section 5.6.5 are burst equation uncertainties and material property uncertainties.

Question 2 (WBN - 08/24/00):

Section 5.6.5 - TVA stated that "...With regard to pre-heater wear, the 1.4% uprate conditions result in a slight increase in flow through the main feedwater nozzle which can impact the rate of wear. This slight increase in flow is not expected to result in a significant increase in the wear rate, and the resultant flow is within the pre-heater design flow..." (1) What is the flow rate through the main feedwater nozzle after the uprate? (2) What is the design flow rate for the pre-heater? (3) Does increase in T_{hot} affect the pre-heater wear?

Response:

The Sequoyah Nuclear Plant utilizes Westinghouse Model 51 Steam Generators which contained a feedring design with bottom flow holes for distribution of feedwater to the steam generator. This model does not utilize a pre-heater. Therefore, this question is not applicable for Sequoyah.

Question 3 (WBN - 08/24/00):

Section 5.6.5 - TVA stated that "...For anti-vibration bar (AVB) wear, the slightly increased steam flow and reduced steam pressure can impact the flow induced vibration and wear. The revised design conditions will have a negligible impact on the projected AVB wear rate..." These two statements seem to be incongruent. The first statement indicates that the increase in steam flow and pressure reduction will affect the AVB wear. The second statement indicates that these changes will have negligible impact on the AVB wear rate. TVA needs to clarify the ambiguity.

Response:

For anti-vibration bar (AVB) wear, evaluations have shown that a significant increase in steam flow (> 5 %) and a significant decrease in steam pressure (> 100 psi) can impact the flow induced tube vibration and wear. However, 1.4% uprating, will only slightly increase the steam flow rate (1.4% only) and decrease slightly its steam pressure. These changes will have negligible impact on the projected AVB wear rate. Thus, the 1.4% uprate will not significantly impact future tube wear at the AVB sites.

Question 4 (WBN - 08/24/00):

Section 5.6.5 - TVA needs to address (1) whether the steam generator tubes would satisfy Regulatory Guide 1.121 under the power uprate condition. (2) the impact of the power uprate on the tube inspection during future outages.

Response:

Tubes in service at SQN presently are not affected by the uprate condition. Tube inspections are driven by the degradation assessment. This uprate has not affected the degradation assessment; therefore, no changes will be made to the inspection plan for the upcoming outage. Future inspections will be determined by active degradation, potential degradation, industry experience, and plant-specific operating experience. If the temperature change affects degradation growth rates, the repair limit will be assessed during the operational assessment.

Question 5 (WBN - 08/24/00):

Section 5.6.6 - TVA performed a preliminary assessment to confirm that the existing 40% through wall plugging criteria will remain adequate for the power uprate condition. Provide the final assessments for staff review.

Response:

This question was prompted for Watts Bar since both preliminary and final assessments were performed for that plant. For Sequoyah, a final assessment has already been completed. As stated previously (i.e., , the only mechanisms allowed to remain in service using the plugging limit of 40 percent are AVB Wear, Cold Leg Thinning, and PWSCC at drilled support plates. The 40 percent limit is conservative, and will not be affected by the small uprate. As discussed above, the affects of this small uprate are nonquantifiable. Therefore, the 40 percent limit remains bounding and no changes are required.

Question 6 (WBN - 08/24/00):

Section 5.6.7 - Discuss whether the increase in T_{hot} would affect the proposed outside diameter stress corrosion cracking (ODSCC) voltage-based alternate repair criteria (ARC).

Response:

As discussed in question 1, the affect of the 0.4 degree F Thot increase is non quantifiable. However, if the T_{hot} increase affects growth rates of the ODSCC ARC tubes, this will be evaluated as part of the operational assessment.

Question 7 (WBN - 08/24/00):

Section 5.6.7 - TVA stated that ". . . The ODSCC ARC was developed to replace the application of the generic 40% depth plugging criterion for tube cracking at elevations corresponding to tube support plate intersections. . . " It should be noted that the ODSCC ARC are applicable only to predominate axial tube cracking at tube support plates. The ARC are not applicable to circumferential cracking. Clarify if that is the intent of the above statement.

Response:

ODSCC ARC are only applicable to predominate axial tube cracking at tube support plates.

Question 8 (WBN - 08/24/00):

Section 5.6.7 - TVA stated that "...The loading conditions compared to applicable criteria are only operative during faulted conditions, since the tube degradation is confined to the tube/tube support plate intersection crevice during normal operation...." (1) Clarify the above statement. Specifically, what is meant by ". . .the loading conditions compared to applicable criteria are only operative during faulted conditions . . .?" (2) Do the temperature and primary-to-secondary pressure differential change for the faulted condition under power uprate?

Response:

During normal operation, the presence of tube support plates (TSPs) prevents the burst of ODSCC indications at the TSP intersections even if the tubes are overpressurized. During faulted conditions, however, TSPs may be displaced exposing ODSCC indications to free span conditions. Therefore, the limiting condition to be considered for demonstrating acceptability of the voltage-based repair criteria stipulated in Generic Letter 95-05 is a postulated main steam line break (MSLB) event. The peak pressure differential across the steam generator tube wall during the design-basis MSLB event, which is not affected by the proposed uprate. Therefore, the uprating has no impact on the applicability of voltage-based repair criteria.

As noted above, the limiting condition for evaluating the applicability of voltage-based repair criteria is the design-basis MSLB event, and the peak primary-to-secondary pressure differential for this event is not affected by the proposed uprate. The primary temperature during the event is expected to change to the same extent as the T_{hot} increase (0.4°F), and its impact is insignificant.

Question 9 (WBN - 08/24/00):

Section 5.6.7 - TVA stated that "…the structural and leakage criteria do apply during the application of faulted loading conditions; however, these are unaffected by the 1.4% uprate….
(1) Discuss how the conclusion was reached. (2) Was there any calculations or assessments performed?"

Response:

The structural and leakage criteria in question are those applicable to ODSCC indications in TSP crevices. As noted in the response to Question 8, these criteria are based on the peak primary-to-secondary pressure differential occurring during the design-basis MSLB event, which is not affected by the proposed 1.4% uprate. Therefore, the applicability of the Generic Letter 95-05 voltage-based repair criteria is unaffected by the 1.4% uprate.

No analysis was needed to evaluate the impact of 1.4% uprate on the applicability of the Generic Letter 95-05 voltage-based repair criteria.

Question 10 (WBN - 08/24/00):

Section 5.6.7 - TVA needs to address (1) the impact of the power uprate on tube degradation itself, i.e., would the power uprate affect the ODSCC degradation mechanism? (2) The impact of power uprate on the methodology (the assumptions and parameters used) for condition monitoring and operational assessments.

Response:

Refer to the response to question 1. The affect of the 0.4 degree F temperature change is non quantifiable. (2) The uprate would change nothing in the methodology used to perform condition monitoring and operational assessments. The primary and secondary pressure are inputs to calculations performed. These pressures are verified each inspection to ensure the limiting steady state delta pressure for the past cycle is used in calculations. With respect to the 1.4% uprate, the SQN Steam Generator Inspection Program will include consideration of the higher temperatures in growth rate analyses. Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs will be made. Degradation growth rate changes will be incorporated into the operational assessment.

Question 11 (WBN - 08/24/00):

TVA needs to make an overall conclusion as to the structural and leakage integrity of steam generator tubes under power uprate conditions.

Response:

The responses to the above questions illustrate that the minor change in temperature and secondary side pressure will have non-quantifiable affects on degradation rates, structural integrity, and/or leakage integrity. Steam generator inspections are driven by degradation assessments and repair decisions are driven by operational assessments. The current methodology of performing condition monitoring and operational assessments will not change due to this minor uprate.

Question on Section 4.2.5 - SGBD System (WBN - 08/24/00):

In the submittal you have indicated that the required flow rates in the steam generator blowdown system are not expected to be significantly affected by the 1.4% power uprate. The reason you gave was that the power uprate will not significantly impact addition of dissolved solids and particulates into the steam generators. Please, provide technical basis justifying that the power uprate will not significantly change dissolved solids and particulates introduced into the steam generators and there will be no need, therefore, for changing the flow rates in the blowdown system.

Response:

The rate of addition of dissolved solids to the secondary systems is a function of condenser leakage and the quality of secondary makeup water. The rate of generation of particulates is a function of erosion-corrosion (E/C) within the secondary systems. Since neither condenser leakage nor the quality of secondary makeup water are impacted by power uprate, the rate of blowdown required to address dissolved solids should not be impacted by power uprate. Theoretically, the potential for E/C increases with any increase in secondary system flowrates that may result from the slightly increased flows from the uprate. However, the overall effect of the minor increases in secondary system velocities is not expected to alter the E/C rates appreciably. Therefore, the evaluation of the SGBD System concluded that the required blowdown to control secondary chemistry and particulates will not be significantly impacted by power uprate.

Sequoyah currently maintains SG secondary chemistry within EPRI guidelines. Site goals are based upon achieving and maintaining INPO Top Decile Chemistry Performance Indicator (1.01). These limits are maintained using ETA, hydrazine, boric acid and ammonium chloride secondary chemical injection to maintain Steam Generator chemistry

Sequoyah operates with a SGBD flow of about 60 gpm to Cooling Tower blowdown for secondary chemistry control. Condensate Polishers are maintained in standby to support transient response to a Main Condenser tube leak, a Steam Generator tube leak, or other corrosion product or contaminant transients induced by large load changes or removing balance of plant equipment from service then returning it to service. ETA chemistry has pacified the secondary system components to minimize corrosion products and corrosion product fouling of stainless steel surfaces such as feedwater venturis. An initial temporary increase in corrosion products associated with increased secondary flow rates due to power uprate of 1.3% is anticipated. Secondary chemistry should remain within site goals and return to normal equilibrium within a week to ten days of this load increase. Steam Generator Blowdown flow may be increased to control this transient, but flow would be returned to normal values when chemistry values returned to normal equilibrium values. Placing Condensate Polishers in service is not anticipated to maintain site chemistry goals.

Reactor Vessel Fluence Question (WBN - 08/24/00):

In Section 5.1.2, TVA indicates that existing neutron fluence projections bound the corresponding projections for the 1.4% uprated conditions. What are the existing values and the uprated values?

Response:

Section 2.4.1.2 of Enclosure 6 provides a discussion of the effect of the 1.3% uprate on the Sequoyah Units 1 and 2 neutron fluence projections. For the Sequoyah uprate, these fluence projections increased from the previously existing values. The updated neutron fluences were subsequently utilized in the reactor vessel integrity assessment (see section 2.4.1.3) to determine the impact on the reactor vessel heatup and cooldown curves.

(A) Questions on Caldon Topical Report ER-160P, Enclosure 2

Question (A)1 (WBN - 10/06/00):

Table 1 on page A-3 is proprietary while the same information are non-proprietary on page 3 and in TU-ELECTRIC response to staff question 31, dated December 17, 1998. Clarify the inconsistency.

Response:

Table 1 on page A-3 can be considered non-proprietary. There is other information on this page that is considered proprietary.

Question (A)2 (WBN - 10/06/00):

Figure 3 shows equal probability of exceeding 102% of the current power level for using the current instrument or the LEFM with the proposed power uprate. Explain figure 3 indications and differences, between this figure and figure 5-2 of ER-80P.

Response:

The differences between Figure 3 in ER-160P and Figure 5-2 in ER-80P are two: the content of the data are different, and the data are presented in a different format.

The content difference has to do with the power uprate percentage. Figure 5-2 in ER-80P presents a 1% power uprate case and Figure 3 in ER-160P presents a 1.4% power uprate case. In both figures the LEFM thermal power uncertainty is presented as 0.6%, a bounding or limit value. Figure 5-2 illustrates that in the 1% uprate case the probability of exceeding 102% power with the LEFM is less than with the original instrumentation and no uprate. Figure 3 of ER-160P illustrates that in the 1.4% uprate case the probability of exceeding 102% power with the LEFM is the same as with the original instrumentation and no uprate.

The second difference is in the format of the data presented. Figure 5-2 in ER-80P presents a probability density while Figure 3 of ER-160P presents a probability. As described in detail in TU-ELECTRIC response to staff question 31, plotting the probability instead of the probability density permits one to read the vertical axis directly in probability percentage units. (On a probability density curve, the vertical axis is presented in statistical units established so that the area under the curve will integrate to equal 1.) This change was made to improve the comprehension of the curve, but does not change the content conclusion described in the above paragraph.

Both ER-80P and ER-160P state that there are two assumptions that must be met to use this probability argument. The first is that the instrument uncertainty is based on elemental errors that are normally or uniformly distributed. The second is that there is assurance the instrument is operating within this uncertainty bounds at all times. The LEFM systems meet both of these criteria, as described in ER-80P and in ER-160P.

For SQNP the uprate request of 1.3% falls between the case presented in ER-80P of a 1% uprate and that presented in ER-160P of 1.4%, therefore, the ER-160P is applicable to and is a bounding analysis for the 1.3% uprate at SQN.

Question (A)3 (WBN - 10/06/00):

An effective value of 0.62% uncertainty is calculated in the appendix. Explain why 0.6% total power uncertainty is used for the uprate as shown in Table 1.

Response:

This question is not applicable to Sequoyah since the requested uprate of 1.3% allows a total power uncertainty of 0.7%. See WCAP-15669 located in Enclosure 4 for the SQN specific instrument uncertainty calculation results.

(B) Questions on Westinghouse Plant Specific Uncertainty Calculation, Enclosure 4

Question (B)1 (WBN - 10/06/00):

Section 3 of the calculation states that the combined accuracy of feedwater flow and temperature measurement (density and enthalpy components), as provided by Caldon, is 0.483%. In Section 1 of the report this statement is listed as an assumption. Table 11a of the calculation lists the density and enthalpy effects in addition to 0.483%. Is this uncertainty value included in Table 1 of the Caldon Topical Report ER-160P or was it provided by Caldon as a plant specific LEFM flow measurement uncertainty? Also explain the relationship between the instrument error and power uncertainties listed in table 11a.

Response:

The value of 0.482% flow is for LEFM flow and temperature measurement, including LEFM contributors to feedwater density and enthalpy. There is a separate small contributor to feedwater density and enthalpy from the feedwater pressure uncertainty, based on pressure transmitters outside the LEFM. Table 1 of WCAP-15669, Rev.0 includes an estimated bounding value for the pressure error. WCAP-15669, Rev.0 was completed prior to the hydraulic testing for the Sequoyah LEFM.

The sensitivities listed in Table 2 of WCAP-15669, Rev.0 define the relationship between the instrument uncertainties and the power uncertainties listed in Table 3. For example, multiplying the feedwater pressure measurement uncertainty from Table 3 times the sensitivity of the power uncertainty to the feedwater pressure impact on density from Table 2, results in the power uncertainty contribution shown in Table 3.

Question (B)2 (WBN - 10/06/00):

Table 1 of ER-160P lists total power uncertainty as 0.6%, Tables 3-1 and E-3 of ER-80P list it as 0.57%, and Table 11a of the Westinghouse calculation lists it as 0.58%. Compare these values and explain.

Response:

As discussed in response to Question (B)1, these are all values estimated prior to the completion of hydraulic testing and calculated to ensure they would be bounding of the actual result. The uncertainty analysis for the installed LEFM at Sequoyah will not be final until the installation and commissioning is complete. However, the best current estimate is 0.5% total power uncertainty based on the results obtained for the Watts Bar system, which is similar to the Sequoyah system, and testing of the Sequoyah spool piece at Alden Labs documented in Caldon Engineering Report ER-223.

Specifically, Table 1 of ER-160P estimates a bounding value for a generic single header measurement at 0.6%, rounded to one significant figure. Table 3-1 of ER-80P estimates a bounding value for a generic single header measurement of 0.61% and for a two loop installation (not applicable to Sequoyah) of 0.57%. Table E-3 estimates the same 0.61% and 0.57% as Table 3-1 in ER-80P.

WCAP-15669 Revision 0, included in Enclosure 4, provides the power uncertainty analysis results for the uprated conditions using site-specific values at Sequoyah. Since the discussion above is based on utilizing the entire 1.4% margin for the uprate at Watts Bar this question does not specifically apply to

SQNP since the requested uprate of 1.3% allows a total power uncertainty allowance of 0.7%.

(C) Questions on Description and Evaluation of the Proposed Change, Enclosure 1

Question (C)1 (WBN - 10/06/00):

Section 6.6 states that WCAP-12096, revision 8, provides the basis for the RTS and ESF actuation setpoints and WCAP-14738, revision 0, provides the basis for the RCS control system uncertainties that are used in the plant safety analyses. Are these topical reports applicable to Watts Bar Nuclear Plant and were they reviewed by the staff?

Response:

WCAP-11239, "Westinghouse Setpoint Methodology for Protection Systems, Sequoyah 1 & 2, Eagle-21 Version" is applicable to Sequoyah Nuclear Plant and has been previously reviewed by the NRC Staff. There is no topical report that provides the basis for the Sequoyah RCS control system uncertainties.

The current version of the reactor protection system setpoint methodology document is maintained by TVA and has not been submitted to the staff for review.

Question (C)2 (WBN - 10/06/00):

Section 6.6 states that based on evaluations performed for other plant uprates, it is judged that the 1.4% uprate would have a negligible effect on the steam generator narrow range water level instrument tap and thus, there would be no impact on the existing instrumentation setpoints and allowable values. Provide a sample of those plant uprate evaluations with a comparison to justify the judgment.

Response:

See response to Question 5 from Reactor Systems Engineering Branch (WBN - 08/24/00).

ENCLOSURE 9

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

WESTINGHOUSE APPLICATION FOR WITHHOLDING
PROPRIETARY INFORMATION

CAW-01-1486 FOR WCAP-15669
(ENCLOSURE 4)
CAW-01-1489 FOR RAI RESPONSE
(ENCLOSURE 7)

LICENSING TRANSMITTAL TO NRC
SUMMARY AND CONCURRENCE SHEET

THE PURPOSE OF THIS CONCURRENCE SHEET IS TO ASSURE THE ACCURACY AND COMPLETENESS OF TVA SUBMITTALS TO THE NRC.

DATE ____________ DATE DUE NRC ____________

SUBMITTAL PREPARED BY Keith Weller

SUBJECT: Technical Specification (TS) Change No. 01-08, Increase Maximum Allowed Reactor Power Level To 3455 Mega-Watt Thermal (MWt)

PURPOSE/SUMMARY To request the necessary Technical Specification and Operating License changes to increase reactor power by 1.3% to a maximum power output of 3455 MWt.

RESPONDS TO ______________________ (RIMS NO.)

NEW COMMITMENTS ____ YES ____ NO

INDEPENDENT TECHNICAL REVIEW (ITR) (2)____________ DATE: ____________
(Engineering)
(it affects design basis)

LICENSING BASIS CHANGE - If this submittal requires a change to the licensing basis, a change has been initiated in accordance with NADP-7. ____________ DATE ____________

A concurrence signature reflects that the signatory has assured that the submittal is appropriate and consistent with TVA Policy, applicable commitments are approved for implementation and supporting documentation for submittal completeness and accuracy has been prepared.

CONCURRENCE (3)

NAME	ORGANIZATION	SIGNATURE	DATE
D. L. Koehl	SQN Plant Manager		
M. J. Lorek	SQN Asst. Plant Mgr		
K. C. Weller	SQN Licensing Engr		
J. D. Smith	SQN Licensing Mgr.		
10 CFR 50.54(f) oath or affirmation required [X] Yes [ ] No [ ] N/A			
PORC	PORC Chairman		
J. A. Bailey	NSRB Chairman		
	SQN OPS Shift Mgr		
J. K. Wilkes	SQN OPS Supt.		
E. E. Freeman	SQN OPS Manager		
D. L. Lundy	SQN Engineering Mgr.		
D. H. Morris	Corp. Engr - Mech		
S. A. Cutts	Corp. Engr -Elec		
G. P. Cooper	Corp. Engr. Mgr.		

I:license/TS Submittal/TSC 01-08
KCW:PMB

ENCLOSURE 9

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

WESTINGHOUSE APPLICATION FOR WITHHOLDING
PROPRIETARY INFORMATION

CAW-01-1486 FOR WCAP-15669
(ENCLOSURE 4)
CAW-01-1489 FOR RAI RESPONSE
(ENCLOSURE 7)

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

September 27, 2001

CAW-01-1486

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Westinghouse Power Measurement Instrument Uncertainty Methodology for Tennessee Valley Authority Sequoyah Units 1 and 2 (1.3% Uprate to 3467 MWt - NSSS Power)", WCAP-15669, Revision 0 (Proprietary) July 2001

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-01-1486 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. 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-01-1486 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures
cc: M. Scott/NRR/OWFN/DRPW/PDIV2 (Rockville, MD) 1L

P:DATA/DOCUMENTS/0579s

CAW-01-1486

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, Manager, 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 Company LLC ("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 28th day
of September, 2001

Lorraine M. Piplica

Notary Public

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

P:DATA/DOCUMENTS/0579s

(1) I am Manager, Regulatory and Licensing Engineering, in Nuclear Services of the Westinghouse Electric Company LLC ("Westinghouse"), 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.

(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 Electric Company LLC 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 WCAP-15669 "Westinghouse Power Measurement Instrument Uncertainty Methodology For Tennessee Valley Authority Sequoyah Units 1 and 2 (1.3% Uprate to 3467 MWt – NSSS Power)", Revision 0, (Proprietary) July 2001. This information is being transmitted by Tennessee Valley Authority letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention: Mr. Samuel. J. Collins. The proprietary information as submitted for use by the Tennessee Valley Authority for the Sequoyah Units 1 and 2 is expected to be applicable in other licensee submittals in response to certain NRC requirements for licensing of a 1.3% power uprate to 3467 MWt.

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

(a) Provide documentation supporting the determination of power measurement uncertainty associated with 1.3% uprate.

(b) Provide the applicable engineering evaluations which establish the technical basis for the 1.3% power uprate.

(c) Provide licensing information to support license amendments.

Further this information has substantial commercial value as follows:

(a) Westinghouse's 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 this information 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 licensing support documentation 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 design 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.

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

October 17, 2001

CAW-01-1489

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Applicability of Comanche Peak and Watts Bar Power Uprate RAI's to Sequoyah 1 & 2 Uprate", (Proprietary) October 2001

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-01-1489 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. 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-01-1489 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures
cc: M. Scott/NRR/OWFN/DRPW/PDIV2 (Rockville, MD) 1L

P:DATA/DOCUMENTS/0580s

CAW-01-1489

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, Manager, 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 Company LLC ("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, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 19th day
of October, 2001

Lorraine M. Piplica

Notary Public

LORRAINE M. PIPLICA COMMONWEALTH OF PENNSYLVANIA NOTARY PUBLIC

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

P:DATA/DOCUMENTS/0580s

(1) I am Manager, Regulatory and Licensing Engineering, in Nuclear Services of the Westinghouse Electric Company LLC ("Westinghouse"), 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.

(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 Electric Company LLC 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 "Applicability of Comanche Peak and Watts Bar Power Uprate RAIs to Sequoyah 1 & 2 Uprate", (Proprietary) October 2001. This information is being transmitted by Tennessee Valley Authority letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention: Mr. Samuel. J. Collins. The proprietary information as submitted for use by the Tennessee Valley Authority for the Sequoyah Units 1 and 2 is expected to be applicable in other licensee submittals in response to certain NRC requirements for licensing of a 1.3% power uprate to 3467 MWt.

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

(a) Provide documentation supporting the acceptability of the 1.3% power uprate.

(b) Provide the applicable engineering evaluations that establish the technical basis for the 1.3% power uprate.

(c) Provide licensing information to support license amendments.

Further this information has substantial commercial value as follows:

(a) Westinghouse's 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 this information 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 licensing support documentation 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 design 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.

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