April 30, 2002

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
ISSUANCE OF AMENDMENTS RE: 1.3-PERCENT POWER UPRATE
(TAC NOS. MB3435, AND MB3436) (TSC NO. 01-08)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 275 to Facility Operating License No. DPR-77 and Amendment No. 264 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. These amendments consist of changes to a License Condition, to the Technical Specifications (TS), and to the applicability interval of certain SQN, Unit 2 temperature versus pressure curves. The amendments are in response to your application dated November 15, 2001, as supplemented by a letter dated March 11, 2002, which provided clarifying information to the original application.

These amendments approve revisions to the TS and facility operating licenses to reflect increases in SQN, Units1 and 2, maximum steady-state core power levels from 3411 megawatts thermal (MWt) to 3455 MWt, an increase of approximately 1.3 percent. These increases are facilitated by the utilization of the Caldon Leading Edge Flowmeter for feedwater flow measurements.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Ronald W. Hernan, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 275 to License No. DPR-77
2. Amendment No. 264 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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cc w/enclosures: See next page

ADAMS Accession No. ML021220060 * See previous concurrence

OFFICE	PDII-2/PM	PDII-2/LA	OGC	PDII-2/SC(A)	PDII/D	DLPM/D
NAME	RHernan	BClayton	JHeck *	TKoshy	HBerkow	LMarsh for JZwolinski
DATE	4/29/02	4/29/02	4/25/02	4/29/02	4/29/02	4/29/02

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SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
ISSUANCE OF AMENDMENTS RE: 1.3-PERCENT POWER UPRATE
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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.275
License No. DPR-77

The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 15, 2001, as supplemented by letter dated March 11, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 275, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L. Marsh acting for/

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical Specifications

Date of Issuance: April 30, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 275

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace page 3 of Operating License DPR-77 with the attached replacement page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
1-5	1-5
3/4 7-2	3/4 7-2
B 3/4 7-1	B 3/4 7-1

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 264
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 15, 2001, as supplemented by letter dated March 11, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 264, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 120 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L. Marsh acting for/

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical Specifications

Date of Issuance: April 30, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 264

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace page 3 of Operating License DPR-79 with the attached page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
1-6	1-6
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-35	3/4 4-35
3/4 7-2	3/4 7-2
B 3/4 4-7	B 3/4 4-7
B 3/4 7-1	B 3/4 7-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 275 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 264 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated November 15, 2001 (Ref.1) (Agencywide Documents Access and Management System [ADAMS] Accession No. ML013470345), as supplemented by a letter dated March 11, 2002 (Ref. 10) (ADAMS Accession No. ML020870181), the Tennessee Valley Authority (TVA, the licensee) submitted a request for changes to the Sequoyah Nuclear Plant (SQN), Units 1 and 2, Facility Operating Licenses (FOLs) and Technical Specifications (TS). The March 11, 2002, letter provided clarifying information that did not change the scope of the original Proposed No Significant Hazards Consideration Finding published in the *Federal Register* (66 FR 64303).

The requested changes included revisions to the SQN, Units 1 and 2, FOLs and TS to reflect increases in the authorized maximum steady-state reactor core power levels from 3411 megawatts thermal (MWt) to 3455 MWt, an increase of approximately 1.3 percent. This power uprate is facilitated by using the Caldon, Inc. (Caldon) Leading Edge Flowmeter✔™ (LEFM✔™) system to measure feedwater flow. The specific changes proposed in support of the power uprate include:

- Revision of Section 2.C.(1) of the SQN, Units 1 and 2, FOLs to reflect the increased maximum steady-state reactor core power level of 3455 MWt. In addition, the wording of this paragraph for SQN Unit 2 would be revised to be identical to that in the SQN Unit 1 FOL,

- Revision of TS 1.26, “RATED THERMAL POWER (RTP),” to define total reactor core heat transfer rate to the reactor coolant of 3455 MWt,

- Revision to Table 3.7-1 to reflect a reduced setpoint for the maximum allowable power range neutron flux high setpoint with one inoperable steam line safety valve from 63 percent to 62 percent,

- Revisions to Figures 3.4-2, 3.4-3, and 3.4-4 for SQN Unit 2 only to reduce the applicable effective full-power years (EFPY) from 16 EFPY to 14.5 EFPY, as well as corresponding changes to the BASES (page B 3/4 7-1 for both units and B 3/4 4-7 for Unit 2 only).

ENCLOSURE 3

TVA's 1.3-percent power uprate request is based on a reduced uncertainty associated with measuring core thermal power using the newly installed Caldon LEFM✔™ systems to measure feedwater flow and temperature. The licensee has installed an LEFM✔™ system. The total power measurement uncertainty associated with utilization of the LEFM✔™ system is 0.7 percent, which can support a power uprate of up to 1.3 percent of rated thermal power. In its November 15, 2001, letter, TVA referenced Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✔™ System." This topical report was approved in the NRC's Safety Evaluation (SE) for the Texas Utilities' Comanche Peak, Unit 2, dated March 8, 1999 (Nuclear Documents System [NUDOCS] Accession No. 9903190065). The original topical report was supplemented by Caldon Engineering Report ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✔™ System," which was submitted to the NRC for a Watts Bar Nuclear Plant (WBN) Unit 1 power uprate request. In response to the WBN Unit 1 amendment request, the measurement uncertainties associated with the LEFM✔™ system, as described in ER-160P, were approved by the NRC, as documented in WBN Amendment No. 31 to FOL No. NPF-90, dated January 19, 2001 (ADAMS Accession No. ML010260074). Additionally, the NRC staff approved the referencing of ER-160P in Amendment Nos. 194 and 169 to FOL Nos. NPF-14 and NPF-22, dated July 6, 2001 (ADAMS Accession No. ML011760551), for Susquehanna Steam Electric Station, Units 1 and 2, respectively.

The November 15, 2001, TVA submittal also included Westinghouse Topical Report WCAP-15726, "Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program Licensing Report." This report addresses the plant-specific evaluations of the higher power level on various plant systems, reactor trip system setpoint, core safety limits, and accident analysis that could be affected by the higher power level.

The November 15, 2001, TVA submittal also included Westinghouse Topical Report WCAP-15669, "Westinghouse Power Measurement Instrument Uncertainty Methodology for Tennessee Valley Authority Sequoyah 1 and 2," which provided the basis for the instrumentation and control section of the NRC staff's evaluation.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power, and the uncertainty of the calculated values of this thermal power determines the probability of exceeding the power levels assumed in the design-basis transient and accident analyses. In this regard, Appendix K to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 requires loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses to assume that the reactor has been operating continuously at a power level at least 102 percent of the licensed thermal power to allow for uncertainties, such as instrument error. The phrase "such as" suggests that the 2-percent power margin was intended to address uncertainties related to heat sources in addition to the instrument measurement uncertainties. As documented in the *Federal Register* on June 1, 2000 (65 FR 34916), the NRC concluded that, at the time of the original ECCS rulemaking, the 2-percent power margin requirement was solely based on the considerations associated with reactor power measurement uncertainty. This development could justify a reduced margin between the licensed power level and the power level assumed in the ECCS analysis and, therefore, a power uprate.

In order to reduce an unnecessarily burdensome regulatory requirement and to avoid unnecessary exemption requests, the Commission published a final rule in the *Federal Register* on June 1, 2000. This final rule allows licensees the option of justifying a smaller margin of power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power or maintaining the current margin of 2-percent power. Licensees may apply the reduced margin to operate the plant at a level higher than the current licensed power or use the margin to relax ECCS-related TS. The final rule, by itself, does not allow licensees to increase the licensed power level without NRC staff approval. Since the licensed power level of a nuclear power plant is a specified limit in the TS and often in the FOL, the proposals to increase the licensed power level must be reviewed and approved through the license amendment process. Such license amendment requests should include a justification for the reduced power measurement uncertainty to support the proposed power uprate.

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. In support of the power uprate application, the licensee submitted safety analyses including transient and LOCA analyses, and proposed TS changes for the staff to review and approve.

The staff review is to confirm that the licensee performed safety analyses with acceptable methods, to verify that the analytical results meet the required acceptance criteria, and to ensure that the proposed TS appropriately reflect the results of acceptable safety analyses. The following evaluation is based on the staff review of the licensee's safety analyses, proposed TS changes and the responses to the staff's request for additional information. This review includes the following areas: (1) power measurement uncertainty, (2) methods used for Departure from Nucleate Boiling Ratio (DNBR) calculations, (3) non-LOCA and LOCA transients analyses, (4) station blackout analysis, (5) anticipated transients without scram (ATWS) analysis, (6) reactor vessel fluence analysis, and (7) proposed TS changes.

3.0 EVALUATION

3.1 Instrumentation and Control

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system (NSSS). The daily power is measured based on the steam generator (SG) thermal output. Assuming that the reactor coolant system (RCS) primary and secondary sides are in equilibrium, the core power is determined by summing the thermal output of the SGs and the RCS primary side heat loss, and subtracting the reactor coolant pump heat addition. The SG thermal output is determined by secondary side calorimetric measurement, which is determined by the feedwater flow times the difference in the steam and feedwater enthalpy, with the correction of SG blowdown. The feedwater flow is measured using the Caldon LEFM✔™ system placed in the feedwater header for SQN Units 1 and 2. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow and feedwater net enthalpy measurements. Thus, an accurate measurement of feedwater flow and temperature will result in an accurate calibration of the nuclear instrumentation.

The instrumentation for measuring feedwater flow typically uses an orifice plate, a venturi meter, or a flow nozzle to generate a differential pressure proportional to the feedwater velocity in the pipe. Of the three differential pressure devices, a venturi meter is most widely used for feedwater measurement in nuclear power plants. The SQN design uses a venturi in the feedwater systems of each of the four steam generators for feedwater measurement. The major advantage of a venturi meter is a relatively low head loss as the fluid passes through the device. The major disadvantage of the device is fouling, which causes the meter to indicate a higher differential pressure and hence a higher than actual flow rate. This leads the plant operator to calibrate nuclear instrumentation high. Calibrating the nuclear instrumentation high is conservative with respect to the reactor safety, but causes the electrical output to be proportionally low when the plant is operated at its thermal power rating. To eliminate the fouling effects, the flow device has to be removed, cleaned, and re-calibrated. Due to the high cost of re-calibration and the need to improve flow instrumentation uncertainty, the industry assessed other flow measurement techniques and found LEFM✓™ to be a viable alternative.

The Caldon LEFM✓™ is an ultrasonic flow meter, using acoustic energy pulses to determine the feedwater mass flow rate. The meter is based on time-of-flight (transit time or counter-propagation) technology. The transit time technology sends an ultrasonic signal diagonally through the fluid and then measures the time it takes to travel upstream and downstream. The sound travels faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The difference in these times is proportional to the velocity of the fluid in the pipe. The LEFM✓™ uses these transient times and the time differences between pulses to determine the fluid velocity and temperature. There are two designs of LEFMs. One is intrusive, using multiple chordal paths with transducers mounted on a spool and the other is a clamp-on type, which straps on the feedwater pipe. SQN Units 1 and 2 currently use the LEFM 8300 strap-on system as a basis for determining the correction factor for feedwater venturi fouling. However, the accuracy and repeatability of measurements with this LEFM are not high enough to justify power uprate.

TVA will install an improved Caldon LEFM✓™ system for feedwater flow measurement at SQN Units 1 and 2. This system consists of an electronic cabinet and a measurement section, or a spool piece. The spool piece will be permanently installed on the feedwater header. The improved LEFM✓™ is a digital system controlled by software using the ultrasonic transit time method to measure four line integral velocities at precise locations with respect to the pipe centerline. The system numerically integrates the four measured velocities to determine the mass flow rate and the fluid temperature. These measurements are used by the plant computer to determine the reactor thermal output. TVA stated that, although the system's function is not nuclear safety-related (providing flow and temperature inputs only to the calorimetric calculation), the system's software has been developed and will be maintained under a verification and validation (V&V) program. The V&V program has been applied to all system software and hardware, and includes a detailed code review. The LEFM✓™ will significantly improve measurement accuracy and measurement reliability, and will allow on-line verification of the accuracy of the feedwater flow and temperature measurements. TVA stated that it will continue to use venturi-based feedwater flow measurement for feedwater control and other functions that it is currently used for. The venturi-based indication may be periodically adjusted on the basis of the LEFM✓™ indication as a backup to determine calorimetric power when the LEFM✓™ is not available.

Caldon Topical Report ER-80P and its supplement ER-160P (both previously approved by the staff) describe the improved LEFM✓™ system for the measurement of feedwater flow and temperature to determine reactor thermal power and provide a basis for up to a 1.4 percent uprate of the licensed reactor power. The topical report stated that the LEFM✓™ is superior to the venturi-based instrumentation currently in use on the basis of the following:

1. The elements of LEFM✓™ accuracy can be verified on-line,

2. The LEFM✓™ measurement accuracy results in an uncertainty ±0.6 percent of thermal power, with 95-percent confidence limit, whereas the measurement uncertainty of the current venturi flow element instrumentation is ±1.4 percent.

In approving Caldon Topical Report ER-80P, the staff included four additional requirements to be addressed by a licensee requesting a power uprate. These four requirements, and the licensee responses, are as follows:

1. 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.

- In Enclosure 8 to the November 15, 2001, submittal, TVA stated that the LEFM✓™ installation will include development of procedures for maintenance and calibration of the LEFM✓™ system, and will be included in the SQN preventive maintenance program. Software and hardware configuration control of the LEFM✓™ system is maintained by TVA Standard Programs and Processes (SPP)-9.Q, “Engineering.” TVA software control for LEFM✓™ is in accordance with SPP-2.6, “Computer Software Control.” Corrective actions and necessary maintenance will be procedurally controlled, and training on the operation and maintenance of the LEFM✓™ system is provided by Caldon. Operator training will include changes associated with the power uprate. Maintenance will be performed by SQN plant personnel per Caldon recommendations contained in Caldon supplied instructions. All adverse conditions that are identified will be documented in accordance with the SQN corrective action program, and TVA has required Caldon to notify SQN of any deficiencies that could affect the design basis accuracy.

  TVA has stated that, if the LEFM✓™ becomes unavailable during the interval between daily performances of the calorimetric heat balance comparison with the nuclear instrumentation system per TS Surveillance Requirement (SR) 4.3.1.1.1, and cannot be returned to service prior to the performance of SR 4.3.1.1.1, the Technical Requirements Manual will 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 LEFM✓™ software will be maintained under Caldon’s V&V program with a requirement that Caldon will notify SQN of any deficiency that could affect the design basis accuracy of the LEFM✓™.

2. For plants that currently have an LEFM✓™ installed, the licensee should provide an evaluation of the operational and maintenance history of the 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.

- SQN Units 1 and 2 currently use the LEFM 8300 strap-on system 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✓™ system and, therefore, will not be used as a basis for the 1.3-percent uprate. The new LEFM✓™ system that will be installed at SQN is bounded by the analysis and assumptions set forth in the Caldon Topical Report ER-80P. The new LEFM✓™ 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. The testing associated with the commissioning will document that the new system is bounded by the Topical Report.

3. 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 alternate methodology is used, the application should be justified and applied to both the venturi and ultrasonic flow measurement instrumentation installation for comparison.

- In Enclosure 8 to the November 15, 2001, submittal, TVA states that 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. Westinghouse states in WCAP-15669 that they are following staff approved setpoint methodology to calculate the plant-specific total power measurement uncertainty at SQN Units 1 and 2 using the LEFM✓™. 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. The LEFM✓™ flow uncertainty of 0.482 percent, as provided by Caldon, was used by Westinghouse in its calculations. The calculated total power uncertainty is ±0.65 percent, based on a 95-percent probability that all errors should fall within the tolerance limits of the uncertainty analysis. Since the calculated uncertainty is slightly higher than the generic value of 0.6 percent, the licensee is unable to take advantage of the full 1.4-percent generic power uprate and chose to limit the power uprate to 1.3 percent. Since the combination of the 0.65 percent uncertainty value and the 1.3-percent power uprate (1.95 percent) is less than the current margin of 2-percent power, the 1.3-percent power uprate value is acceptable.

4. Licensees for plant installations where the ultrasonic meter (including the 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 either that the meter installation is 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 the plant configuration 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.

- TVA has stated that the LEFM✓™ systems to be installed at SQN Units 1 and 2 were calibrated to a plant/unit specific piping configuration at Alden Research Laboratories. This testing was documented in Caldon Report No. ER-223.

The staff finds that the TVA's response to these criteria has sufficiently resolved the plant-specific concerns about LEFM✓™ maintenance and calibration, hydraulic configuration, processes and contingencies for an inoperable LEFM✓™, and the methodology for the plant-specific calculations of the LEFM✓™ power measurement uncertainty.

The staff evaluation finds the calculation of the power calorimetric measurement uncertainty for the SQN power uprate to be acceptable. Based on the staff's review of the licensee plant-specific LEFM✓™ error band calculation, the staff finds that the SQN LEFM✓™ thermal power measurement uncertainty of 0.65 percent of actual reactor thermal power can support the proposed 1.3-percent uprate of the SQN licensed thermal power. The staff also finds that the licensee sufficiently addressed the four additional requirements outlined in the staff SE of Caldon Topical Report ER-80P. The staff, therefore, finds the licensee request for a 1.3 percent thermal power uprate to be acceptable with regard to instrumentation and control.

3.2 Reactor Systems and Accident Analysis Evaluation

3.2.1 Methods used for DNBR Calculations

TVA used the statistical core design (SCD) methodology for performing statistical core thermal-hydraulic analysis. Unlike the deterministic method, in which the uncertainties of various plant and operating parameters are assumed simultaneously at their worst uncertainty limits in the safety analysis, the SCD methodology statistically accounts for the uncertainties of key thermal-hydraulic parameters (such as reactor core power, core power distribution, reactor coolant temperature and flow rate) that affect DNBRs. The SCD methodology establishes an SCD DNBR limit (SDL) that accounts for the effects on DNB of the key parameters. TVA calculated DNBRs using the BWCMV-A critical heat flux (CHF) correlation for the Mark-BW fuel design. The SDL has been established by TVA to assure that there is at least a 95 percent probability at a 95 percent confidence that the hot rod in the core does not experience a DNB during transients. The staff finds the SCD methodology and CHF correlation (Refs. 5 through 8) have been previously approved by the NRC for calculating DNBRs applicable to the Mark-BW fuel. In response to the staff's request, TVA also has confirmed that its use of the referenced methods in the power uprate application fully satisfies (Ref. 10) the restrictions specified in the staff's SE for the referenced approved methods. Therefore, the staff concludes that TVA's methods used for DNBR calculations are acceptable.

3.2.2 Non-LOCA and LOCA Transients Analyses

TVA discussed the SQN Updated Final Safety Analysis Report (UFSAR) Chapter 15 non-LOCA and LOCA transients in Reference 2 for the power uprate conditions. TVA identified the limiting cases for each event category discussed in FSAR Chapter 15 and evaluated the effects of power increase on plant transients. For those cases that were bounded by the corresponding cases in UFSAR Chapter 15, TVA provided supporting rationales. For those cases with values of plant parameters outside the applicable range of the corresponding UFSAR cases, TVA provided results of reanalyses to show the compliance with applicable acceptance criteria used in the corresponding UFSAR analysis. The staff reviewed TVA's discussion of safety analyses (Ref. 2) and the responses to the requests for additional information (Refs. 9 and 10), and our SE is provided below.

3.2.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

The UFSAR analysis for this event was performed at zero power conditions. Therefore, the uprated power level has no effect on the core power assumption. Also, the effect of the power uprate on the input parameters (such as reactivity insertion due to rod motion, initial axial power distribution, moderator temperature and Doppler reactivity coefficients) that affect the transient is negligibly small. Therefore, the UFSAR results of the transient analysis remain valid for the power uprate application.

3.2.2.2 Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power

For the system response calculation, the UFSAR analysis assumed an initial power of 102 percent of the rated power. Therefore, the 2-percent measurement uncertainty assumed in the current analysis can accommodate a power uprate of 1.3 percent and equipment changes that reduce power measurement uncertainty to 0.7 percent. The reactor trip on high neutron flux remains unchanged following the power uprate because the high neutron flux setpoints (high and low settings) specify that the reactor trip occurs at same neutron flux level in term of absolute megawatts before and after power uprating. The reactor trip on OT delta-T is also unaffected by the power uprate because the delta-T_0 used in modeling the OT delta-T setpoint is the delta-T_0 at 102-percent power. Therefore, the current analysis for the system response remains valid for the power uprating conditions.

Since the current DNBR calculations were based on the full rated power, TVA reanalyzed the limiting DNBR cases using the revised thermal power level and power measurement uncertainty. TVA's analysis shows that the calculated minimum DNBR meets the acceptable DNBR safety limit and confirms that the UFSAR conclusions remain valid. Therefore, the staff concludes that UFSAR analysis is acceptable.

3.2.2.3 RCCA Misalignment

The RCCA misalignment events include (1) a dropped full-length assembly, (2) a dropped full-length assembly bank, and (3) a statically misaligned full-length assembly. Previous reload analyses have demonstrated that the dropped RCCA peaking margins bound that of the statically misaligned single RCCA event. In the UFSAR analysis, the dropped RCCA event was analyzed to assure no fuel failure occurred due to the low DNBR. Since the current analysis assumed the initial power at full-power level, TVA reanalyzed the dropped RCCA transient based on the revised power level and power measurement uncertainty. The DNBRs for the uprated conditions were calculated for all the possible dropped RCCA configurations including dropped banks. TVA's analysis confirms the validity of the UFSAR analysis showing no DNB to occur. Therefore, the UFSAR remains acceptable. With respect to the system pressurization, a dropped RCCA is bounded by the loss of external electrical load event, which is the limiting pressurization event for the faults of moderate frequency and remains valid for the uprated power conditions (as discussed later in this section.) Therefore, the UFSAR analysis for the RCCA misalignment event is acceptable.

3.2.2.4 Uncontrolled Boron Dilution

This analysis ensures that there is sufficient time for operator actions before loss of shutdown margin occurs. The primary parameters in the determination of the time available to terminate the dilution include the dilution flow rate, initial and final boron concentrations, boron worth and the overall RCS active volume. The increase of power does not affect any of these parameters. The existing analysis remains conservative and bounding. Therefore, it is acceptable for the power uprate applications.

3.2.2.5 Partial and Complete Loss of Forced Reactor Coolant Flow

The partial loss of reactor coolant flow event is assumed to occur from the mechanical or electrical failure in a reactor coolant pump (RCP) or a fault in an RCP bus. The complete loss of reactor coolant flow event initiates from simultaneous loss of electrical supplies to all RCPs. At initiation of the events, the RCP begins to coast down. Since a loss of all RCPs event results in a larger amount of RCS flow reduction and a greater rate of flow decrease (thus, lower DNBRs), the complete loss of RCS flow event results in a more severe transient than the partial loss of RCS flow event. Even though the complete loss of flow event was categorized as a Condition III event, it was analyzed to satisfy Condition II acceptance criteria, which do not allow DNB to occur during the transient. Therefore, the analysis of complete loss of the flow event bounds the partial loss of flow event, a Condition II event.

The current system response analysis for the total loss of RCS flow event was performed based on an initial power of 102 percent of the rated thermal power, which included 2-percent measurement uncertainty. Therefore, the current analysis for the system response remains valid for the power uprate conditions. Although the system response for this event was not reanalyzed, the minimum DNBR was recalculated using the uprated thermal power level and reduced power measurement uncertainty. The reanalysis shows that the safety DNBR limit continues to be met and confirms that the conclusions of the UFSAR analysis remain valid.

3.2.2.6 Startup of an Inactive Reactor Coolant Loop

Startup of an idle reactor coolant pump without bringing the inactive loop hot-leg temperature close to the core inlet temperature results in the injection of cold water into the core. In the presence of a negative moderator reactivity coefficient, the transient causes a rapid reactivity increase and subsequent power increase. The UFSAR analysis of this event was performed based on 72-percent power level with a 2-percent power measurement uncertainty. The primary input parameter that affects the analysis of this event is initial reactor coolant average temperature (578.2°F) which remains unchanged for the power uprate condition. Therefore, the UFSAR analysis remains valid.

3.2.2.7 Loss of External Electrical Load and/or Turbine Trip

The UFSAR analysis for system response includes two loss-of-load cases: one at 102 percent and one at 52-percent power. Both of these cases include a 2-percent power measurement uncertainty in the initial power assumption. The current analysis for system response is, therefore, applicable to the power uprate conditions. The minimum DNBR for this event is bounded by the calculated value for the complete loss of reactor coolant flow event, which confirms that the safety DNBR limit is not exceeded for uprating power conditions. Therefore, the conclusions of the UFSAR analysis for this event remain valid.

3.2.2.8 Loss of Normal Feedwater and Loss of Offsite Power to the Station Auxiliaries

The UFSAR analysis of system response for both events was performed based on an initial power of 102 percent of the rated thermal power, which included 2-percent measurement uncertainty. Therefore, the UFSAR analysis for the system response remains valid for the power uprating conditions.

The minimum DNBR for the loss of normal feedwater event is bounded by the calculated value for the complete loss of reactor coolant flow event, because the complete loss of reactor coolant flow event assumes immediate RCP trip from at-power conditions while the loss of normal feedwater event assumes continuous operation of the RCPs. An earlier RCP trip results in a higher power-to-flow ratio and lower DNBRs.

During a loss of offsite power (LOOP) event, both the reactor and RCPs trip coincidentally. In comparison, the pumps trip before reactor trip in a complete loss of reactor coolant flow event, resulting in a greater power to flow ratio. As a result, DNBR margins associated with a LOOP are bounded by those predicted for a complete loss of rector coolant flow event, which confirms that the safety limit DNBR is met for power uprate conditions. Therefore, the UFSAR conclusions remain valid for both events.

3.2.2.9 Excessive Heat Removal due to Feedwater System Malfunctions

The UFSAR analysis of this event was performed at both full-power and zero-power levels. The zero-power analysis is unaffected by the power uprate. The full-power analysis assumed an initial power level of 102 percent of the rated thermal power, including a 2-percent power measurement uncertainty. Therefore, the UFSAR analysis can accommodate a power increase of 1.3 percent and equipment changes that reduce power measurement uncertainty to 0.7 percent.

3.2.2.10 Excessive Load Increase Event

The UFSAR analysis includes four cases following a 10-percent step load from full-power conditions: (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 analysis assumes an initial core power of 102 percent of rated thermal power with inclusion of 2-percent power measurement uncertainty. Therefore, the UFSAR analysis remains valid for the power uprate application.

3.2.2.11 Accidental Depressurization of the Reactor Coolant System

This event is initiated by the inadvertent opening of a pressurizer safety valve or by the failure of a valve to close following an overpressurization event. The event causes a reduction in RCS inventory and subsequent reduction in RCS pressure. The UFSAR analysis assumed an initial power of 102 percent of rated thermal power. The primary parameters that affect the results of the analysis for this event are: the RCS pressure and temperature, moderator density, Doppler reactivity feedback, and pressurizer safety valve capacity. None of these parameters are affected by the power uprate. The UFSAR analysis is, therefore, applicable to the power uprate conditions.

3.2.2.12 Accidental Depressurization of the Main Steam System and Major Rupture of a Main Steamline

Accidental depressurization of the main steam system and main steamline breaks (SLB) could produce RCS overcooling. In the presence of negative moderator reactivity feedback, the events result in a power excursion. The UFSAR analysis assumed an initial zero-power for both events. Thus, the analysis is unaffected by the power uprate. The same operational conditions were assumed in the UFSAR analysis for both events. Modeling considerations, such as heat removal capabilities and reactivity feedback, were identical. Since an SLB presents a much greater overcooling event and resulting power excursion, the SLB event results in a lower minimum DNBR than the accidental depressurization of the main steam system event. Even though the SLB event was categorized as a Condition IV event, it was analyzed to satisfy Condition II acceptance criteria, which do not allow DNB to occur during the transient. Therefore, the analysis of the SLB event bounds the accidental depressurization of the main steam system event. The primary parameters that affect the system and core responses to the SLB are the heat transfer area of the steam generator tubes and break size. None of these parameters is affected by the power uprate. Therefore, the UFSAR analysis remains valid following the power uprate.

3.2.2.13 Spurious Operation of the Safety Injection System at Power

This analysis assumed that the safety injection was inadvertently actuated. The UFSAR analysis considered two cases: (1) the reactor trip occurred at the same time as spurious injection started and (2) the reactor protection system produced a reactor trip later in the transient. The cases were analyzed for pressurizer overfilling due to continued ECCS injection and reactor coolant expansion resulting from residual heat generation. Since the analysis was performed at 102-percent power, it bounded the proposed 1.3-percent uprate conditions in which power measurement uncertainty is limited to 0.7 percent. The UFSAR analysis showed that the DNBR increased throughout the transient and the minimum DNBR remained to be greater than the initial value. The analysis confirms the validity of the UFSAR results. Therefore, the UFSAR analysis remains acceptable following the power uprate.

3.2.2.14 Inadvertent Loading of a Fuel Assembly into an Improper Position

The UFSAR analysis concluded that power distribution effects resulting from misloading events were readily detected either by the in-core movable detector system, or of a sufficiently small magnitude to remain within the design peaking limits. Since the UFSAR conclusions are unaffected by the power uprate, the UFSAR analysis remains valid for the power uprate applications.

3.2.2.15 Single RCCA Withdrawal at Full Power

The UFSAR analysis of this event considered two cases: (1) a continuous withdrawal of a single RCCA with the reactor in manual control mode and (2) a withdrawal of a single RCCA resulting in the immobility of the other RCCAs in the controlling bank with the reactor in automatic control mode. Since the UFSAR analysis for the RCCA bank withdrawal at power event considered a spectrum of withdrawal rates that encompassed that expected from a single RCCA, the core power and system response resulting from an RCCA withdrawal event were bounded by the RCCA bank withdrawal event. The UFSAR analysis calculating the system response for the RCCA bank withdrawal event was assumed to initiate from 102 percent of the rated power.

For the fuel performance analysis, the UFSAR case assumed the initial power level at the nominal full-power level. TVA reanalyzed the fuel performance calculations at uprated power conditions. The results show that the 1.3-percent power increase has a negligible effect on the number of fuel pins failed, and the calculated total number of failed pins remains less than the limiting fuel failure of 5-percent in the UFSAR. Therefore, UFSAR conclusions remain valid.

3.2.2.16 Steam Line Break (SLB) Coincident with Rod Withdrawal at Power

The UFSAR analysis of system response for this event considered a spectrum of SLB sizes coincident with withdrawal of control bank D at 102 percent of the rated power. The UFSAR analysis, therefore, remains applicable for the power uprate applications. For the minimum DNBR calculations, the UFSAR case assumed the initial power level at the nominal full-power level. TVA recalculated the minimum DNBR using the revised thermal power level and power measurement uncertainty. The reanalysis shows that calculated DNBRs meet the DNBR safety limit and confirms the validity of the UFSAR conclusions. Therefore, the UFSAR analysis remains acceptable.

3.2.2.17 Feedwater Line Break

The feedwater line break event reduces the ability to remove heat generated by the core from the RCS because fluid in the steam generator is discharged through the break, and the break is large enough to prevent the addition of the main feedwater after the trip. During the event, the reactor trip is actuated on SG low-low level trip signal and the auxiliary feedwater provides water to SGs to prevent overpressurization of the RCS and maintain core coolability. The UFSAR analysis was performed at 102 percent of the rated thermal power. Therefore, the feedwater line break analysis in the UFSAR remains valid for the power uprate.

3.2.2.18 Steam Generator Tube Rupture

The UFSAR analysis of the steam generator tube rupture (SGTR) event was performed to determine steam releases for calculating offsite radiation doses. The UFSAR analysis was performed at 102 percent of rated thermal power. The primary input parameters that affect the results of the analysis for the SGTR event are: break size; low pressurizer pressure reactor trip and safety injection setpoints; emergency core cooling system operation and capacity; and steam line safety valve setpoint and capacity. None of these parameters are affected by the power uprate. Therefore, the SGTR analysis in the UFSAR remains valid for the power uprate applications.

3.2.2.19 Single Reactor Coolant Pump Locked Rotor

There are two issues associated with the single reactor coolant pump locked rotor event:

(1) the peak primary and secondary pressure transient with respect to RCS and main steam system (MSS) pressure limits, and (2) the percentage of fuel rods expected to experience DNB. For the RCS and MSS pressure calculations, the UFSAR case assumed the initial power level at 102 percent of the rated thermal power. Thus, the current analysis remains bounding for the 1.3-percent power uprating and its associated 0.7-percent power measurement uncertainty. For the fuel performance analysis, the UFSAR case assumed the initial power level at the nominal full-power level. TVA reanalyzed the fuel performance calculations at uprated power conditions. The results show a slight increase in the number of fuel pins failed. However, the calculated total number of failed pins remains less than the limiting fuel failure of 10-percent in the UFSAR. Therefore, the UFSAR conclusions remain valid.

3.2.2.20 RCCA Ejection

The UFSAR analyzed two cases for the RCCA event: the full-power case and zero-power case. Since the full-power case assumes an initial power level at 102 percent of the rated thermal power and the zero-power case is unaffected by the power uprate, the UFSAR analysis for the core and system responses remains applicable for the power uprate applications. For the minimum DNBR calculations, the UFSAR case assumed the initial power level at the nominal full-power rating. TVA recalculated the minimum DNBR using the revised thermal power level and measurement uncertainty. The reanalysis shows that the limiting case does not result in fuel damage greater than 10-percent fuel failure in the core for the UFSAR limiting case. Therefore, the UFSAR conclusions remain valid.

3.2.2.21 Large- and Small-Break LOCAs

The UFSAR analyses were performed for LOCA events with various break sizes. The analyses for the limiting cases of both large- and small-break LOCA events show that the results satisfy the acceptance criteria of 10 CFR 50.46: the calculated peaked cladding temperature is less than 2200°F; the maximum localized oxidation is less than 17 percent of total cladding thickness; the maximum hydrogen generation is within 1.0 percent of the total amount of zircaloy in the core; the core geometry remains coolable and the decay heat can be removed for the extended period of time. The analyses used an initial power of 102 percent. Since the power increase of 1.3 percent with a corresponding 0.7-percent measurement uncertainty is within the assumed initial power limit of the current LOCA analyses, the UFSAR analyses remain valid.

3.2.3 Anticipated Transients Without Scram (ATWS) Analysis

An ATWS event is defined as an anticipated operational occurrence (such as loss of normal feedwater, loss of load or loss of offsite power) combined with an assumed failure of the reactor trip to shut down the reactor. For the pressurized water reactors (PWRs) manufactured by Westinghouse, the basic requirements of the ATWS rule are specified in paragraph (c)(1) of 10 CFR 50.62. TVA satisfies the ATWS rule with an NRC-approved ATWS Mitigating System Actuation Circuitry (AMSAC). During the course of the power uprate review, the staff requested TVA to provide a discussion of an ATWS analysis demonstrating that the plant at the power uprate condition is within bounds considered by the staff during TVA's documentation of compliance with the ATWS rule. In response, TVA indicated (Ref. 10) that SQN currently relies upon the generic ATWS analyses to demonstrate the acceptance of the analytical results. The generic analyses were performed by Westinghouse and documented in NS-TM-2182 (Ref. 11).

These analyses include 2-, 3-, and 4-loop PWRs with various SG models. The base case of the generic ATWS analyses for a 4-loop PWR with Westinghouse Model 51 SGs and a power level of 3411 MWt adequately represents the current plant configurations and licensed power level and, therefore, is applicable to the SQN units. The sensitivity study (Ref. 11) of the generic ATWS analyses was performed to determine the changes of the calculated peak RCS pressure resulting from the changes in various initial plant conditions such as initial power level, PORVs relief capacity and auxiliary feedwater capacity. The results of the sensitivity study show that the calculated RCS peak pressure for 4-loop plant with a power increase of 2-percent from the base case remains below the acceptable pressure of 3200 psig. TVA also confirmed (Ref. 10) that the design of the auxiliary feedwater (AFW) system and pressurizer safety valves (PSV) at SQN is consistent with the total AFW flow and PSV relief flow assumed in the ATWS analysis base case for Westinghouse 4-loop PWRs. In addition, TVA stated that the original SQN pressurizer power operated valves (PORVs) supplied by Westinghouse were replaced in 1983 with valves manufactured by Target Rock. TVA confirmed that the capacities of current PORVs are consistent with those modeled in the generic ATWS analyses. For the SQN units, the current TS requirement on moderator temperature coefficient (MTC) is limited to < 0 pcm/°F at all power levels. TVA also stated (Ref. 10) that the reactivity feedback for the SQN units remains sufficiently negative to be comparable to the generic Westinghouse analyses presented in Reference 11. Since the power level, PSV, PORV relief capacities, AFW flow and MTC are within the applicable range of the ATWS base case with a 2-percent increase in power for Westinghouse 4-loop PWRs, the staff concludes that the ATWS analysis for the SQN units with power uprate conditions is bounded by the generic analyses for 4-loop PWRs in Reference 11 and, therefore, is acceptable.

3.2.4 TS Changes

TVA submitted to NRC the proposed TS changes (Ref. 3) in its support of safe operation of the SQN plant at a maximum power level of 3455 MWt. The following is the staff review of the TS changes.

3.2.4.1 TS 1.26 - Rated Thermal Power

The TS defines 3455 MWt (increased from 3411 MWt) as the rated thermal power. The TS change is acceptable since the power level of 3455 MWt is considered as the rated power in the acceptable transient and accident analysis for the power uprate application.

3.2.4.2 TS Figures 3.4-2and 3.4-3- RCS Heatup and Cooldown Limitations

As stated in Section 2.6, the current pressure-temperature (PT) limit for both units is 16 EFPYs. Both SQN Units have a PT Limits Report (PTLR) that includes the 32 EFPY PT values. Due to the power uprate both sets of limits have been reconsidered. The result is as follows:

Unit 1: At 16 EFPY, no TS change is needed because the original value is higher than the revised value. At 32 EFPY, the revised value is 31.3 EFPY. However, this change does not need to be made until the current limits expire.

Unit 2: At 16 EFPY, the revised applicability range is 14.5 EFPY. TS Figures 3.4-2 and 3.4-3 would be changed to reflect this shorter applicability. The curves do not change. At 32 EFPY, the range will be limited to 31.8 EFPY. However, the change does not need to be made until after the end of the current PT limits.

The TS changes correctly reflect the Unit 2 change of the PT limits and, therefore, are acceptable.

3.2.4.3 TS Figure 3.4-4 - PORV Nominal Lift Setting (Applicable to Unit 2)

Figure 3.4-4 specifies the PORV setpoints for the low-temperature-overpressure-protection (LTOP) systems. This figure for Unit 2 would be revised to change the EFPY applicability from 16 EFPY to 14.5 EFPY. This change is consistent with the change in Figures 3.4-2 and 3.4-3, reactor coolant system heatup and cooldown limitations, which revise the EFPY applicability from 16 EFPY to 14.5 EFPY for Unit 2. It does not change any assumptions previously made in determination of PORV setpoints for the LTOP. Therefore, the change is acceptable.

3.2.4.4 TS Table 3.7-1 and Associated Bases - Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves

Operability of all the main steam line safety valves ensures that the secondary system pressure is limited to less than 110 percent of its design pressure during the most severe transient. With less than full main steam safety valves capacity available, operation may be allowed at reduced power levels. TS Table 3.7-1 lists the maximum allowable setpoints for high power range neutron flux trip for conditions with the main steam safety valves inoperable. The revised table lowers the trip setpoint from 63 percent of the original rated thermal power of 3411 MWt to 62 percent of the uprated power of 3455 MWt for operating conditions with one inoperable safety valve on any operating SGs. The proposed trip setpoint (in term of MWt) is lower than the setpoint of the current TS, and is more restrictive. Therefore, the TS change is acceptable.

In the associated Bases section, the total relieving capacity for the steamline safety valves is changed from $1.9X10^7$ lbs/hr to $1.6X10^7$ lbs/hr. TVA states that the change is a correction to the safety valve relieving capacity and is not a result of the power uprate effect. To reflect the power uprate, the secondary steam flow is changed from $1.493X10^7$ lbs/hr to $1.514X10^7$ lbs/hr. As a result of changes in the safety valve relieving capacity and the total steam flow, the percentage of the total relieving capacity for all steamline safety valves to the total secondary steam flow is changed from 127 percent to 106.4 percent. The changes in the Bases sections adequately reflect the power uprate and the correction in the valve capacity. Therefore, the TS Bases changes are acceptable.

3.2.5 Reactor Systems and Accident Analysis Conclusion

The NRC staff has reviewed TVA's safety analyses and the associated TS changes in support of operation of the SQN Nuclear Plant at the maximum core power level of 3455 MWt. The staff finds that the supporting safety analyses use acceptable methods, and results show that the uprated power conditions are either bounded by the current UFSAR analyses or the reanalyses meet the applicable acceptance criteria used in the corresponding UFSAR analysis. Therefore, the staff concludes that the supporting safety analyses discussed in this evaluation are acceptable. The staff also finds that the proposed TS discussed in Section 3.2.4 of this evaluation adequately reflect the results of the acceptable supporting analyses, and therefore, concludes that the proposed TS are acceptable for the SQN power uprate applications.

3.3 Nuclear Steam Supply System (NSSS) Design Parameters

The NSSS design parameters provide the reactor coolant system (RCS) and secondary system conditions for use in the NSSS analyses and evaluations. TVA presented the new parameters for the power uprate and incorporated them into its evaluations. The modified input assumptions include an increased NSSS power level of 3467 MWt. The NSSS power is the summation of the reactor core power of 3455 MWt and the thermal power generated by the reactor coolant pumps (RCPs) of 12 MWt. Other parameters that change as a result of the 1.3-percent power uprate include the RCS temperatures (RCS hot leg temperature (T_{hot}) would increase by 0.4 ^{o}F and RCS cold leg temperature (T_{cold}) would decrease by 0.4 ^{o}F) and small changes in secondary-side parameters such as steam temperature (T_{steam}), steam pressure (P_{steam}), and steam mass flow rate (M_{steam}). The largest of these secondary-side parameter changes would be a drop in steam pressure from 802 psig to 795 psig (0.85 percent). The NRC staff has evaluated these changes to the plant conditions and finds them to adequately represent the uprated plant behavior; therefore, the NSSS design parameters are acceptable.

3.4 Design Transients

3.4.1 NSSS Design Transients

To support the power increase for SQN, Units 1 and 2, TVA reviewed the current primary and secondary side design transients to determine their continued applicability at the uprated design conditions. TVA compared the current analysis parameters to those expected at the uprated conditions. When the existing parameters remained bounding, TVA concluded that the design transient analyses remain valid.

The NRC staff has reviewed TVA's methodology for determining and reevaluating the initial design transient parameters. The NRC staff finds that the methodology is conservative and, therefore, is acceptable.

3.4.2 Auxiliary Equipment Design Transients

TVA reviewed the NSSS auxiliary equipment design transients by comparing the revised operating conditions to the current transient conditions. TVA determined that the only transients affected by the uprate were the temperature transients (i.e., those impacted by the changes to the full-load NSSS operating temperatures, T_{hot} and T_{cold}). The temperature transients for SQN were previously analyzed for a worst-case T_{hot} of 630^{o}F and T_{cold} of 560^{o}F. The uprated T_{hot} and T_{cold} are 611.6 ^{o}F and 544.8 $^{\circ}$F, respectively. These temperatures remain within the limits set in TVA's transient analyses. Therefore, the NRC staff finds that the current auxiliary equipment design transient analyses remain acceptable for the power uprate.

3.5 NSSS

3.5.1 RCS

The RCS consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop contains one RCP, which circulates water through the loops and reactor vessel and a steam generator, where heat is transferred to the main steam system which drives the main electrical generator. The RCS contains a pressurizer that controls system pressure using electrical heaters and water sprays. Overpressure protection is provided by the PORVs connected to the pressurizer. The PORVs discharge to the pressurizer relief tank (PRT).

TVA performed various assessments and demonstrated that the RCS design-basis functions would be met at the revised design conditions. Based on these assessments, the NRC staff provides the following conclusions.

The major components of the main steam system support the increased heat transfer requirements. The residual heat removal system adequately removes the increased decay heat. The RCS control and protection functions are not significantly affected. The RCS thermal design flow does not change as a result of the uprate. The pressurizer spray flow rate of 800 gallons-per-minute (gpm) is still achievable. The RCS and pressurizer design temperature and pressure parameters continue to remain bounding. The PRT sizing and setpoint, pressurizer relief valve sizing and discharge piping pressure drop, pressurizer relief valve inlet pressure drop, and pressurizer surge line pressure drop parameters are not affected by the power uprate.

On the basis of these conclusions, the NRC staff finds that the RCS remains acceptable for the power uprate.

3.5.2 Chemical and Volume Control System (CVCS)

The SQN Units 1 and 2 CVCSs provide 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.

In TVA's assessment of CVCS operation at revised RCS operating temperatures, the maximum expected RCS vessel/core outlet temperature (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 regard to the CVCS thermal performance, the T_{COLD} of 544.8°F is still lower than the design system inlet temperature of 560^{0}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. The staff reviewed the licensee's analysis and concurred with the licensee's conclusion.

3.5.3 Residual Heat Removal (RHR) System

The RHR system 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 RHR system is used to transfer refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations.

TVA performed an evaluation of the effects of the power uprate on the design-basis operation of the RHR system and concluded that the existing design-basis analysis is acceptable for the power uprate. TVA's analysis showed that, during a normal cooldown at the higher power level, the plant could be cooled down to less than 140°F in less than 33 hours. The NRC guidelines (Appendix R to 10 CFR 50) specify cooldown to less than 140°F in less than 72 hours.

Based on the information provided by TVA, the NRC staff finds that TVA's evaluation is conservative and acceptable. Therefore, the NRC staff finds the RHR system acceptable for the power uprate.

3.5.4 Spent Fuel Pool Cooling System (SFPCS)

The SQN SFPCS removes the decay heat generated by stored fuel assemblies in the spent fuel pit. TVA analyzed the thermal-hydraulic effects on the SFPCS due to the power uprate and determined that the proposed power uprate will have little or no impact on the performance of this system.

Based on the information provided by TVA and the experience gained from prior NRC staff reviews of other power uprate applications for similar PWRs, the NRC staff concurs with TVA that operations at the proposed 1.3-percent uprate power level will have little impact on the operation of this system. Therefore, the NRC staff finds the SFPCS acceptable for the power uprate.

3.5.5 NSSS Control Systems

TVA evaluated the following transients to ensure that the plants would respond without generating a reactor trip system or an engineered safety features actuation system actuation. These transients include a 10-percent step load increase and decrease, large load rejection (100 percent to 50 percent), and 5-percent ramp load increase and decrease. The current analyses for these transients include a 2-percent power calorimetric uncertainty. Since the Caldon LEFM✔™ systems reduce the calorimetric uncertainty to no more than 0.6 percent, the initial uncertainty assumption bounds the 1.3-percent uprate, and the initial analyses remain valid. The results showed a very smooth transient response with no oscillatory or diverging parameter responses. The overtemperature setpoint was the most limiting reactor trip setpoint with margin to the setpoint of approximately 10-percent. Therefore, the NRC staff finds the analyses acceptable for the power uprate conditions.

3.6 NSSS Components

The NRC staff reviewed the SQN power uprate amendment as it relates to the effects of the power uprate on the structural and pressure boundary integrity of the NSSS and balance-of-plant (BOP) systems. Affected components in these systems included piping, in-line equipment and pipe supports, the reactor pressure vessel (RPV), core support structures (CSS), reactor vessel internals (RVI), SGs, control rod drive mechanisms (CRDM), RCPs, and pressurizer. The staff's safety evaluation (SE) concerning the effects of the power uprate on the pertinent components is provided below.

3.6.1 Reactor Vessel Structural Evaluation

The proposed power uprate represents an increase of approximately 1.3 percent over the currently licensed level of 3411 MWt in core power. The licensee reported that the power increase will result in changing the design parameters provided in Enclosure 6, Table 2.1-1 (Ref. 2). Table 2.1-1 provides a comparison of the current design parameters and the corresponding revised parameters for use in the power uprate analysis at SQN Units 1 and 2.

The licensee evaluated the reactor vessel for the effects of the revised design conditions in Table 2.1-1 on the most limiting vessel locations with regard to ranges of stress intensity and fatigue usage factors in each of the components, as identified in the reactor vessel stress reports. The evaluations considered the operating parameters, which were identified for the uprated power condition. The existing NSSS design transients were not affected by the 1.3-percent power uprate. The components of the reactor vessel affected by the power uprate include outlet and inlet nozzles, the RPV (main closure head flange, studs, and vessel flange), CRDM housing, bottom head to shell juncture, core support pads and the instrumentation tubes. The licensee evaluated the maximum stresses and cumulative fatigue usage factors (CUFs) for the critical components at the uprated core power conditions. The evaluation was performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1968 Edition, which is the Code of record.

The calculated maximum stresses and the maximum CUFs for the reactor vessel critical locations are provided in Enclosure 7 of Reference 1. The results indicate that the maximum stresses are within the allowable limits, and the CUFs remain below the allowable ASME Code limit of 1.0. The licensee concluded that the current design of the reactor vessel continues to be in compliance with licensing basis codes and standards for the power uprate condition. Based on its review, the staff agrees with this conclusion.

3.6.2 Reactor Vessel Integrity

TVA evaluated changes in neutron fluence resulting from the proposed SQN Units 1 and 2 power uprate to determine the effect on reactor vessel integrity. The assessment included a review of the current material surveillance capsule withdrawal schedules, applicability of the plant heatup and cooldown PT limit curves, applicability of the Emergency Response Guideline (ERG) limits, and the effect on the pressurized thermal shock (PTS) values (10 CFR 50.61, known as the PTS Rule).

3.6.2.1 Reactor Vessel Fluence Analysis

Reactor vessel fluence analyses (Ref. 12) were performed at the end of Cycle 9 for both units at the rated power level of 3411 MWt. These calculations have been revised to reflect the 1.3 percent power uprate to 3455 MWt. The neutron source assumed for all cycles beyond Cycle 9 is the average of Cycles 5 to 9, which have low leakage loadings. Because both units will operate at the higher rated power, TVA normalized the EFPY-affected curves to the new power level (i.e., divided the EFPY by 1.013). For example, for Unit 1 the 10.03 EFPY at 3411 MWt at the end of Cycle 9 becomes 9.90 EFPY (10.03/1.013) at 3455 MWt.

The revised peak inside surface fluence values for 32 EFPY at 3455 MWt are: 1.86×10^{19} and 1.85×10^{19} neutrons/square centimeter (n/cm^2) for Units 1 and 2, respectively. These values compare very closely to 1.84×10^{19} and 1.82×10^{19} n/cm^2 estimates for 32 EFPY at the present power level for units 1 and 2, respectively.

3.6.2.2 Heatup and Cooldown PT Curves

Both SQN units are currently operating to PT curves for 16 EFPY, per Westinghouse Topical Reports WCAP-12970 (Unit 1) and WCAP-12971 (Unit 2). SQN has a PTLR which includes the 32 EFPY curves. However, the increased power level and corresponding vessel fluence increase requires adjustment of the period of applicability for both the 16 and 32 EFPY PT limits. TVA states that, for Unit 1, the 16 EFPY curves are applicable for the uprated power because the original fluence value was higher than the revised value. However, the 32 EFPY curves are limited to 31.3 EFPY. Likewise, for Unit 2, the 16 EFPY reduces to 14.5 EFPY and the 32 EFPY is reduced to 31.8 EFPY.

TVA completed a review of the current heatup and cooldown curve applicability dates for SQN Units 1 and 2. The review indicated that the revised fluence projections after the power uprating exceed the fluence projections used in developing the current adjusted reference temperature (ART) values for SQN Unit 1 at 32 EFPY and SQN Unit 2 at 16 and 32 EFPY. Therefore, TVA calculated new applicability dates which are documented in Table 2.4.1.3-5 of its submittal. The calculation indicated that only the 16 EFPY curves for SQN Unit 2 are affected by reducing the applicability period to 14.5 EFPY. This change is reflected in the proposed change to SQN Unit 2 TS.

3.6.2.3 Pressurized Thermal Shock (PTS)

As noted above, the revised fluence values for 32 EFPY differ only slightly from the corresponding current values; therefore, the estimated reference temperature for pressurized thermal shock (RT_{PTS}) values are practically identical to the original values. The critical element for Unit 1 is the lower shell forging 4 and for Unit 2 the intermediate shell forging 5. Therefore, the screening criterion required by 10 CFR 50.61 is 270°F. Both units satisfy this requirement by a wide margin.

TVA performed PTS calculations for SQN 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 its PTS Rule. The calculated neutron fluence values for the uprated condition for SQN Units 1 and 2 have exceeded the current fluences. Therefore, TVA evaluated the effects of the uprate on the PTS values for the most limiting material from each unit using the uprated fluences. This evaluation is presented in Table 2.4.1.3-7 of TVA's submittal. 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 SQN Units 1 and 2, respectively.

In summary, we note that the SQN Units each have a PTLR, which employs NRC approved methods in the calculation of the vessel fluence. The effect on the 32 EFPY fluence of the 1.3-percent power increase in combination with low leakage loadings is minimal. The current period of applicability of the PT curves is shortened slightly but the pressurized thermal shock RT_{PTS} is unchanged and well within the required limits. Therefore, the existing PT curves are acceptable and applicable for the time periods noted.

3.6.2.4 Surveillance Capsule Withdrawal Schedule

Each NRC licensee is required to develop a surveillance capsule withdrawal schedule 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 prepared in terms of EFPY of plant operation, with a projected design life of 32 EFPY.

TVA stated that the capsules removed from the SQN Units 1 and 2 vessels to date meet the intent of ASTM E185-82. However, the revised fluence projections after the power uprate exceed the fluence projections used in development of the current withdrawal schedules for SQN Units 1 and 2. Therefore, TVA performed calculation of ΔRT_{NDT} at 32 EFPY using the projected increased fluences for SQN Units 1 and 2. The calculation is documented in Tables 2.4.1.3-1 and 2.4.1.3-2 of TVA's submittal. The calculation shows that the maximum ΔRT_{NDT} values using the uprated fluences at 32 EFPY are 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 reference the future fluence values. The updated withdrawal schedules are documented in Tables 2.4.1.3-3 and 2.4.1.3-4 of TVA's submittal.

3.6.2.5 Emergency Response Guideline (ERG) Limits

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

3.6.2.6 Upper Shelf Energy (USE)

TVA stated that based on WCAP-15224 (Unit 1) and WCAP-15320 (Unit 2), all beltline materials are expected to have a USEs 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 exceed the fluence projections used in developing the predicted EOL USE values. This small amount in fluence increase has no measurable effect on percent decrease in USE. Therefore, the current predicted USE values for SQN Units 1 and 2 remain valid.

The staff reviewed TVA's analysis and reasoning and concurred with its conclusions.

3.6.3 Reactor Core Support Structures and Vessel Internals

TVA indicated that a 1.3-percent power uprate does not affect the current design basis seismic and LOCA loads and the design basis NSSS transients. The evaluation of the reactor vessel core support and internal structures was performed for the power uprate condition. The limiting reactor internal components evaluated include the lower core plate, core barrel, baffle plate, baffle/barrel region bolts, and the upper core plate. TVA indicated that the reactor internal components were not licensed to the ASME B&PV Code, but the evaluation is in compliance with the design criteria as documented in the SQN FSAR. However, the lower core structural integrity was evaluated in Enclosure 7 using the 1989 Edition of the ASME Section III Code allowable stress for acceptance.

TVA evaluated these critical reactor internal components considering the revised design conditions provided in Enclosure 6, Table 2.1-1 (Ref. 2). TVA indicated that for the baffle-barrel region components and the upper core plate, the current structural and thermal analyses of record for SQN remain bounding for the power uprate condition. Enclosure 7 of Reference 1 identifies the maximum calculated stress intensity and CUF for the lower core plates. The calculated stresses are less than the Code-allowable limits based on the 1989 Edition of ASME Section III Code and the CUF is less than the limit of 1.0. The remaining reactor internal components are less limiting. In addition, the potential for the flow-induced vibration does not increase for the power uprate condition. As a result of these evaluations, TVA concluded that the reactor internal components at SQN will be structurally adequate for the proposed power uprate conditions. The staff concurs with TVA's assessment.

3.6.4 Control Rod Drive Mechanisms

The pressure boundary portions of the CRDMs are exposed to the vessel/core inlet fluid. TVA evaluated the adequacy of the CRDMs by reviewing the SQN current CRDM design specifications and stress report to compare the design-basis input parameters against the revised design conditions for the power uprate in Enclosure 6, Table 2.1-1 (Ref. 2). The comparison shows that the current design analyses are based on a range of operating temperatures which bound the expected change in the vessel/core inlet temperature for the 1.3-percent power uprate. Therefore, the existing stresses and fatigue usage are not affected by the power uprate and continue to satisfy the allowable stress and fatigue usage limits.

On the basis of its review, the staff concurs with TVA's conclusion that the current design of CRDMs continues to be in compliance with the licensing basis codes and standards for the uprated power conditions.

3.6.5 Reactor Coolant Pumps

TVA reviewed the existing design basis analyses of the SQN RCPs to determine the impact of the revised design conditions in Table 2.1-1. TVA indicated that the ASME Code editions used in the evaluation are those identified as the Code of record.

After the core power uprate, the RCS pressure remains unchanged. The most limiting design parameter is the SG outlet temperature, as provided in Table 2.1-1 of Reference 1, was decreased slightly by 0.4°F for the power uprate condition. There are no significant changes to the design thermal transients. Typically, a higher SG outlet temperature results in a greater actual stress. As a result of the evaluation, TVA indicated that the current stress and CUFs in the stress reports for the SQN RCPs remain bounding for the 1.3-percent power uprate.

On the basis of its review, the staff concurs with TVA's conclusion that the current RCPs, when operating at the proposed conditions with 1.3-percent power increase from the current rated power, will remain in compliance with the requirements of the codes and standards under which the SQN units were originally licensed.

3.6.6 Steam Generators

SQN Units 1 and 2 each have four Westinghouse model 51 steam generators. TVA reviewed the existing structural and fatigue analyses of the SGs at SQN, and compared the power uprate conditions with the design parameters of Model 51 SGs stress reports. The comparison of key parameters is shown in Table 2.1-1 of Reference 1. Based on the comparison of these input parameters, TVA developed scaling factors which were used to scale up the original stresses and fatigue usage factors for the power uprate conditions. The evaluation was performed in accordance with the requirements of ASME Code, Section III, 1971 Edition through the Summer 1972 Addendum, which is the Code of record for SGs at SQN Units 1 and 2.

The calculated maximum stresses and cumulative fatigue usage factors for the critical SG components are provided in Enclosure 7, Table 1 (Reference 1). The results indicate that the maximum calculated stresses are below the Code-allowable limits. The results provided in Table 1 also show that the calculated CUFs are within the allowable limit of unity for the 40-year service life. TVA also evaluated the flow-induced vibration of the U-Bend tubes. The evaluation showed that the tubes are acceptable for the 1.3-percent power uprate condition.

On the basis of its review, the staff concludes that TVA has demonstrated the maximum stresses and CUFs for the critical SG components to be within the Code-allowable limits and, therefore, acceptable for the proposed 1.3-percent power uprate.

TVA stated that the uprating of SQN to 3455 MWt will incorporate steam generator tube plugging in the range from 0 percent - 15 percent maximum in any steam generator. To assess TVA's evaluation of SG structural and leakage integrity, the staff reviewed the effect, if any, that the power uprate would have on the SG tube degradation and the SG inspection program including condition monitoring and operational assessments. The staff also reviewed the plugging criteria and the alternate repair criteria applicable to Units 1 and 2. This review is contained in the following sub-sections.

3.6.6.1 Steam Generator Degradation Mechanisms and Anti-vibration Bar (AVB) Wear Rates

According to TVA, changes in the reactor vessel outlet temperature (T_{HOT}) and the secondary-side pressure are key parameters in determining corrosion effects. These parameters are inputs into calculations used to determine tube integrity. The changes in T_{HOT} and the secondary-side pressure are approximately 0.4°F and 7 psia, respectively, and will not produce a quantifiable impact on degradation rates or structural and leakage integrity. In addition, TVA will disposition new degradation (unanticipated degradation) through the SQN corrective action program and the performance of a root cause analysis.

Experience with power uprates at other plants has shown that a significant increase in steam flow (>5 percent) and a significant decrease in steam pressure (>100 psi) may affect flow-induced tube vibration and result in increased AVB wear. However, the 1.3-percent power uprating only slightly increases the steam flow rate (1.4-percent increase) and slightly decreases the steam pressure. TVA concluded that the 1.3-percent power uprate will have a negligible impact on the projected AVB wear rate and will not significantly impact future tube wear at the AVBs. On the basis of the information TVA provided, the staff agrees with TVA's conclusion that the power uprate will not have a significant impact on AVB wear rates and the existing SG degradation mechanisms.

3.6.6.2 Steam Generator Inspection Program

SQN Units 1 and 2 follow the TVA Steam Generator Program to assess the structural and leakage integrity of the SQN steam generators. This program contains the requirements of Nuclear Energy Institute document NEI 97-06, including the assessment of degradation growth rates. TVA stated that the methodology used to perform condition monitoring and operational assessments will not change as a result of this uprate. TVA will make repairs and may expand the inspection plan based on condition monitoring and the operational assessment of inspection results. Active degradation, potential degradation, industry experience, and plant-specific operating experience will determine TVA's scope of future inspections. On the basis of the information TVA provided, the staff concludes that TVA will include the impact of the power uprate rate in the inspection program through degradation assessments, inspection plans, condition monitoring and operational assessments.

3.6.6.3 Plugging Limit

TVA performed an assessment to confirm that the existing 40-percent through-wall plugging criterion will remain adequate under uprate conditions. This plugging limit applies to AVB wear, cold leg thinning, and primary water stress corrosion cracking (PWSCC) at drilled support plates. TVA stated that the current plugging limit is conservative and bounds the uprate conditions. TVA further asserted that if the uprate does affect degradation rates, TVA will assess the plugging limit during the operational assessment. On the basis of the information provided, the staff finds that the 40-percent plugging limit continues to be appropriate under the proposed power uprate conditions.

3.6.6.4 Steam Generator Loose Parts and Alternate Repair Criteria

3.6.6.4.1 Secondary-Side Loose Parts

TVA performed calculations to determine the change in the wear rate on tubes affected by a loose part on the secondary side of SG 2 in Unit 1 under uprate conditions. According to TVA, this loose part has not been removed because it is inaccessible, and this steam generator is scheduled to be replaced during the next Unit refueling outage. Furthermore, the orientation of this loose part has not changed since 1997. The staff recognizes that TVA's projected wear rate calculations attempt to characterize the uprate-related impact of this loose part on the steam generator. However, the staff believes it is difficult to accurately project loose part-related wear rates in general based on the behavior of one loose part. As presented in the discussion regarding AVB wear, a 1.3-percent uprate only slightly changes the parameters that would affect wear rates (steam flow rate, steam pressure, T_{HOT}, etc.). Therefore, the staff concludes that the steam generators would experience only minor changes, if any, in wear rates related to loose parts.

TVA's current steam generator program includes the identification and disposition of loose parts either by removal or monitoring. In addition, the steam generator program provides for the evaluation of the impact of loose parts through condition monitoring and operational assessments. Therefore, the staff concludes that TVA has provided reasonable assurance that SQN steam generator integrity can be protected from secondary-side loose parts because TVA manages loose parts via the current SQN steam generator program.

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

Generic Letter (GL) 95-05 provides guidance on the implementation of the voltage-based ARC for outer-diameter stress corrosion cracking (ODSCC) at tube support plate intersections. The NRC staff has approved the ODSCC ARC for SQN Units 1 and 2 only for predominately axial tube cracking. TVA states that during normal operation, ODSCC indications at the TSP intersections will not cause a tube to burst because the TSP will restrain the tube. However, under faulted conditions, such as a main steam line break (MSLB), ODSCC indications at the TSP intersections may become exposed and possibly burst. Therefore, to demonstrate the acceptability of the ARC, the limiting condition is a postulated MSLB, as stipulated by GL 95-05. The design basis MSLB event generates a peak differential pressure across the SG tube wall. This peak pressure does not change under power uprate conditions. TVA concluded that the power uprate will have no impact on the applicability of the ODSCC ARC. On the basis of the information provided, the staff finds that the power uprate will not affect the applicability of the ODSCC ARC.

3.6.7 Pressurizer

TVA evaluated the structural adequacy of the pressurizer and components at limiting locations in the pressurizer spray nozzle, the surge nozzle, and upper shell for operation at the uprated conditions. The evaluation was performed by comparing the key parameters in the current SQN pressurizer stress report against the revised design conditions in Table 2.1-1 for the proposed power uprate. Table 2.4.6.1, Enclosure 6 of Reference 1 provides the comparison of the current and uprated pressurizer design parameters. For components affected by hot leg temperature (e.g., surge line), the temperature difference for the uprate parameters is bounded by the current design conditions since the decrease in temperature difference reduces the associated thermal stress in the components. The limiting component affected by the cold leg temperature is the spray nozzle, for which the temperature difference increases slightly from 107.8°F to 108.2°F. However, it is bounded by the design basis temperature difference of 125°F. Therefore, the existing design basis analyses remain valid for the proposed power uprate conditions. TVA concluded that the existing pressurizer components will remain adequate for plant operation with the proposed 1.3-percent power increase while the RCS pressure remains unchanged. Based on its review of the relevant section of the pressurizer evaluation, the staff agrees with TVA's conclusion.

3.6.8 NSSS System Piping and Pipe Supports

The proposed power uprate of SQN Units 1 and 2 involves an increase of temperature difference across the RCS. TVA evaluated the NSSS piping and supports by reviewing the existing design basis analysis against the uprated power condition, with regard to the design system parameters, transients, and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. The methods, criteria and requirements used in the existing design basis analysis for SQN were used for the power uprate evaluation.

The RCS pressure remains unchanged for the proposed core power uprate. The actual hot leg temperature for the power uprate is projected to be slightly greater than the hot leg temperature at the current rated power level. The cold leg temperature for the power uprate condition will be less than that for the current power level. TVA indicated that there is sufficient margin in the existing analysis for stresses associated with the temperature changes defined in Table 2.1-1 of Reference 1.

TVA also indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for SQN power uprate. The potential of a slight increase in loop hydraulic forces due to the decrease in the cold leg temperature and the increase in water density at the power uprate condition was offset by the existing margin in the current design basis analysis. TVA determined that the existing design-basis RCL LOCA hydraulic forcing functions and the reactor pressure vessel LOCA displacement remain bounding for the uprated power condition. The RCL analysis is based on criteria in the United States of America Standard (USAS) B31.1, "Power Piping" Code and does not require a fatigue analysis. TVA concluded that the existing stresses and loads remain bounding for the power uprate for the NSSS components including the reactor cooling loop piping, the primary equipment nozzles, the primary equipment supports, pipe supports and the auxiliary equipment (i.e., heat exchanger, pumps, valves and tanks). Therefore, these components will continue to be in compliance with the Code of record.

On the basis of its review of TVA's submittal, the staff concurs with TVA's conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the design bases criteria, as defined in the SQN UFSAR, and are, therefore, acceptable for the power uprate.

3.6.9 BOP Systems and Motor-Operated-Valves (MOVs)

TVA evaluated the adequacy of the BOP systems based on comparing the existing design basis parameters with the uprated input parameters in Table 2.1-1 for the core power uprate conditions. The BOP piping systems evaluated for the power uprate are main steam, condensate and feedwater, auxiliary feedwater, and steam generator blowdown. As a result, TVA concluded that for the BOP piping, pipe supports, and components, the existing design basis analyses, using maximum differential temperatures and pressures for normal operation and worst case conditions, remain bounding for the uprated power level of 3455 MWt.

TVA also reviewed the programs, components, structures, and generic letter (GL) issues as they are affected by the power uprate. In Enclosure 8 of Reference 1, TVA confirmed that there are no changes to the TVA MOV program as a result of the 1.3-percent power uprate. Safety-related valves were not found to be impacted by the 1.3-percent power uprate and this is acceptable. This determination was confirmed by verifying that changes in system operating temperature, pressure and flow rate were bounded by the design basis requirements. Additionally, in Enclosure 8 of the amendment request, TVA assessed the impacts of the 1.3-percent power uprate on the GL 89-10 MOVs and the Limitorgue Technical Bulletin 98-01 update programs and found them to be acceptable.

TVA assessed the current evaluation of GL 95-07 associated with pressure locking and thermal binding (PLTB) for valves that were listed in the GL 95-07 as being modified to eliminate the potential for PLTB, and found them to remain bounding for the 1.3-percent power uprate. The existing evaluation for GL 96-06 was performed at 102 percent of the current power and is, therefore, bounding for the proposed power uprate of 1.3 percent. On the basis of the above review, the staff concurs with TVA's conclusions that the power uprate will have no adverse effects on the safety-related valves, and that the conclusions of the TVA responses to GL 95-07, and GL 96-06, as well as GL 89-10 programs, remain valid for the power uprate condition.

As a result of the above evaluation, the staff concludes that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and continue to satisfy the design-basis requirements for the proposed 1.3-percent power uprate.

3.6.10 Steam Generator Blowdown System

The SQN Units 1 and 2 steam generator blowdown system 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 SG 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 SGs 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 affected by the 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. The staff reviewed TVA's analysis and concurred with TVA's conclusion.

3.6.11 Flow Accelerated Corrosion (FAC)

TVA's submittal did not discuss effects of FAC on plant systems and components resulting from the proposed power uprate. As a result the staff issued a Request for Additional Information (RAI) to TVA on this subject. In response to the staff's RAI, TVA stated that the FAC program at SQN complies with the requirements of NRC GL 89-08 and that the 1.3-percent power uprate is expected to have negligible effect on FAC. TVA also stated that CHECWORKS 1.0G is the current predictive model in use and includes parameters such as flow velocity, temperature, and pressure. In addition, this program ranks components according to susceptibility and provides wear rates based on parameters entered into the model. Based on TVA's preliminary assessment, system parameter changes vary by system. Some systems will have a negligible change in wear rates while other systems will have a potential for a slight increase or slight decrease in FAC wear rates. Updates to the model will be in place prior to implementation of the power uprate. In addition, the inspection scope and frequency will be adjusted as needed based on results from the model. TVA also stated that incorporation of the 1.3-percent power uprate into the FAC program will not circumvent the basic elements or implementation of the FAC program.

Based on the information provided by TVA, the staff concludes that the proposed power uprate results in negligible effects on FAC and TVA's FAC program.

3.6.12 Conclusion

On the basis of its review, the NRC staff concurs with the evaluations performed by the licensee for the NSSS and BOP piping, components, and supports, the reactor vessel and internal components, the CRDMs, SGs, RCPs and the pressurizer. The staff finds TVA's evaluation to be bounded by the licensing code of record and the original design basis, and therefore, concludes the foregoing components to be acceptable for SQN Units 1 and 2 uprate operations at the proposed core power level of 3455 MWt.

3.7 Plant Electrical Systems

The SQN units receive shutdown power from the TVA 161 kilovolt (kV) system through two physically 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 the 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 megavolt-ampere (MVA), 500-161 kV inter-tie transformer bank.

3.7.1 Switchyard

All 500 kV switches and breakers that interface with SQN 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 161 kV switches and breakers that interface with 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.

Based on its review, the staff concludes that the switchyard will accept the additional load without the need for any hardware modifications due to power uprate. Therefore, the staff has reasonable assurance that GDC-17 will be met at the power uprate condition.

3.7.2 Grid Stability

The current grid stability study has analyzed the safe shutdown of the plant following a loss of offsite power (LOOP) with Unit 1 at 1198.7 MWe and Unit 2 at 1198.5 Mwe. This analysis considered one power line out at the time of the LOOP and a subsequent simultaneous LOCA of the unit and a fault and trip of another power line. TVA analyzed the grid stability at the new power level and the staff reviewed it. The power uprate has a negligible impact on grid stability and is within the accuracy range of the existing stability calculation.

Based on its review, the staff concludes that there is a minimal effect on the grid stability due to power uprate. Therefore, the staff has reasonable assurance that GDC-17 will be met at the power uprate condition.

3.7.3 Main Generator

The main generator is rated at 1356 MVA (1221 MWe) at 0.9 power factor (pf). This rating is based upon 75 psig hydrogen pressure within the generator, which is supplemented with water cooling for the stator and rotor.

At the current thermal rating of SQN Units 1 and 2 of 3411 MWt, the electrical output of the main generators is typically about 1186 MWe with an approximate 4 MWe increase periodically observed during the colder winter months (i.e., 1190.3 MWe in the winter months). The net increase of 12 MWe lies well within the nameplate rating of the generator of 1221 MWe at 0.9 pf. No changes to the equipment protection relay settings for the main generator are required for the 1.3-percent power uprate; although some alarm setpoints for the generator and the exciter may require adjustment.

Based on its review, the NRC staff concludes that the net increase of 12 MWe (i.e., 1198.3 MWe) due to power uprate lies well within the nameplate rating of the generator of 1221 MWe at 0.9 pf. Therefore, the staff has reasonable assurance the main generator can operate safely at the uprated power condition.

3.7.4 Main Power Transformer

The main bank transformers for Unit 1 have a manufacturer's nameplate output rating of 415 MVA per phase (1245 MVA for the bank) with a 55°C winding temperature rise above ambient or 465 MVA per phase (1395 MVA for the bank) with a 65°C winding temperature rise above ambient. The main bank transformer for Unit 2 has a manufacturer's nameplate output rating of 420 MVA per phase (1260 MVA for the bank) with a 55°C winding temperature rise above ambient or 470 MVA per phase (1410 MVA for the bank) with a 65°C winding temperature rise above ambient. The Unit 1 main bank transformer, being the more limiting of the two units with a nameplate rating of 1395 MVA, will remain above the anticipated maximum net output after the uprate of 1336.3 MVA. The main bank transformers, at the 65°C winding temperature rise above the ambient, are also rated above the main generator nameplate rating of 1356 MVA.

Based on its review, the staff concludes that the main bank transformers will operate within the nameplate rating at the 1.3-percent power uprate conditions. Therefore, the staff has reasonable assurance the main bank transformers can operate safely at the power uprate condition.

3.7.5 Isolated Phase Bus

The isolated phase bus main section is designed to American National Standards Institute/Institute of Electrical and Electronics Engineers Std. ANSI/IEEE C37.23, "Metal Enclosed Bus and Calculating Losses in isolated Phase Bus," and is rated at 34,300 amperes. The maximum current in the main generator terminals is 32,625 amperes, which is less than the isolated phase bus rated current carrying capability.

Based on its review, the staff concludes that the isolated phase buses will support the 1.3-percent uprating and, therefore, the design is acceptable.

3.7.6 Station Auxiliary Electrical Distribution

The dc loads are not expected to change as a result of a 1.3-percent reactor power uprate. The ac pump motor loads such as No. 3 and No. 7 heater drain tank pumps, and the condensate booster pumps are expected to increase a small amount. The increase is less than 0.5 percent of their current load and remains bounded by the current analysis. The plant ac and dc auxiliary electrical loads will not increase above their ratings. The voltage controls and grid source impedance at the 500 kV and 161 kV grids will not be affected by this uprate; therefore, the evaluated voltages and short-circuit values at different levels of station electrical distribution system will not change because of this uprate.

Based on its review, the staff concludes that the station electrical distribution system loads will remain within the design capacity. The design is, therefore, acceptable for the power uprated condition.

3.7.7 Emergency Diesel Generators

The power required to perform safety-related functions (pump and valve loads) is not increased with the power uprate, and the current emergency power system remains adequate.

The staff reviewed the emergency power system loads and concluded that the 1.3 percent power uprate does not impact the emergency diesel generators and, therefore, they can operate safely at the power uprate condition.

3.7.8 Station Blackout (SBO)

SBO is defined in 10 CFR 50.2 as the complete loss of the preferred offsite and Class 1E onsite emergency ac power system. The existing calculations used to demonstrate the capability to withstand an SBO event of 4-hour duration without uncovering the core were reviewed for the 1.3-percent uprate conditions. The later stages of the existing analysis credit operator action to maintain the RCS temperature and pressure below specified limits; the steam generator PORVs are used to accomplish this action. The capacity of the steam generator PORVs is adequate to accommodate the 1.3-percent uprated condition. The current SBO analysis bounds the condensate inventory requirements. The calculations remain valid for the power uprate condition.

Based on its review, the staff has concluded that the uprate does not adversely affect the ability of the plant to mitigate a postulated SBO event.

3.7.9 Electrical Equipment Qualification

TVA assessed the normal environments for the plant buildings. The uprate has an insignificant effect on process fluid temperatures in the auxiliary, safeguards, electrical and control buildings. TVA used the computer models for the post-accident thermal environmental parameters for mass and energy releases during postulated pipe breaks in the building structures. The evaluations concluded that 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 power uprate has no impact on the non-radiological equipment qualification program. TVA also evaluated the effects of post-accident radiological consequences on equipment qualification and concluded that all doses used for equipment qualification remain within existing design basis values. Therefore, operating at the uprated power condition is acceptable and in conformance with 10 CFR 50.49.

3.7.10 Conclusion

The staff has evaluated the effect of extended power uprate on the necessary plant electrical power systems, grid stability, the SBO coping capability, and the environmental qualification of electrical equipment. Results of these evaluations show that the increase in core thermal power would have negligible impact on the grid stability, SBO, or the environmental qualification of electrical components. This is consistent with GDC-17, 10 CFR 50.63, and 10 CFR 50.49 and the proposed change is, therefore, acceptable.

3.8 <u>Radiological Consequences</u>

TVA performed an evaluation of the impact of the proposed changes on the inputs and assumptions used to perform the current design-basis accident (DBA) analyses of record. Many of the DBA analyses were originally performed assuming an initial RTP of 102 percent (3479 MWt) or greater. The measurement uncertainty for the LEFM✔™ is ±0.7 percent. The sum of the proposed power increase of 1.3 percent and the LEFM✔™ measurement uncertainty is 2 percent. As such, analyses originally performed using a power level equivalent to 102 percent of RTP continue to be bounding and reanalysis is not necessary. DBA analyses which assume hot zero power as the initial condition are not affected by the power uprate. The remaining UFSAR Chapter 15 analyses are either not dependent on the RTP or are adequately bounded by the other events. TVA determined that the key inputs used in the analysis of the DBAs continue to be applicable or bounding for a 1.3-percent power uprate coincident with a 1.3-percent decrease in flow measurement uncertainty. TVA concluded that there would be no increase in the postulated radiological consequences of DBAs previously analyzed.

3.8.1 Staff Evaluation

The NRC staff reviewed the amendment request with regard to the DBA radiological consequences of the proposed changes. In addition to the material included in the submittal, the staff also considered relevant information in the SQN UFSAR. Only docketed information was relied upon in making this safety finding.

The potential radiological consequences of DBAs are proportional to the quantity of radioactive material released to the environment. This release is a product of the radioactive material released from the core or from the RCS, the transport of the released material to the effluent release point, and the transport (e.g., atmospheric dispersion) in the environment. The transport in the environment is not considered further as it is not affected by the power uprate. In general, the inventory of fission products in the core and the quantity of radioactive material in the RCS are directly proportional to the RTP. An increase in the RTP, as proposed, can be expected to increase the inventory of radioactive material available for release. The transport of the released material is dependent on plant process parameters, such as process stream flows, temperatures, and pressures. An increase in the RTP and any associated plant modifications could affect the assumptions made in previous consequence analyses.

The staff reviewed TVA's submittal and the SQN UFSAR. The decrease in reactor power measurement uncertainty due to the installation of the LEFM✔™ effectively offsets the proposed increase of 1.3 percent in RTP. Staff review of the dose analyses in SQN UFSAR Chapter 15 indicated that the current analyzed power level bounds the requested uprate power level when the reduced power uncertainty associated with the LEFM✔™ is considered.

SQN UFSAR Section 15.5 addresses the environmental consequences of postulated DBAs at SQN. For each analyzed accident there is a discussion providing the methodology, assumptions, and inputs used in performing the current radiological consequence assessments. These discussions and the associated tables indicate that the analyses were performed using a power level equivalent to 105 percent of RTP (104.5 percent for fuel-handling accident). The core inventory and the concentration of radioactive materials in the RCS and in the steam generators used in those analyses were determined using a power level equivalent to 105 percent of RTP. Since the current mass and energy releases were determined using a power level equivalent to 102 percent of RTP, the results of those analyses continue to bound the proposed 1.3-percent uprate plus 0.7-percent measurement uncertainty.

In the SQN licensing basis, the radiological consequences of the rod cluster control assembly ejection accident (REA) and single RCP locked rotor accident (LRA) are considered to be bounded by the loss-of-coolant accident and no specific radiological analyses were performed.

The validity of this conclusion is not affected by the power uprate since the fuel safety analyses for the REA and LRA used a power level equivalent to 102 percent of RTP, which bounds the proposed 1.3-percent uprate plus 0.7-percent measurement uncertainty.

3.8.2 Staff Conclusion

The staff reviewed TVA's amendment request and has determined that the decrease in reactor power measurement uncertainty due to the installation of the LEFM✔™ effectively offsets the proposed increase of 1.3 percent in RTP. The staff has concluded that there is reasonable assurance that operation at the increased RTP of 3455 MWt will not result in postulated DBA consequences that exceed the analysis results currently documented in the SQN UFSAR. Since these DBA analysis results were previously found to meet the acceptance criteria of 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the 1.3-percent power uprate coincident with a 1.3-percent decrease in flow measurement uncertainty and the proposed conforming technical specification changes are acceptable with regard to the DBA radiological consequences.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 64303). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from P. Salas (TVA) to NRC, 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)," November 15, 2001.

2. Enclosure 6 of Reference 1, WCAP-15726, "Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program Licensing Report."

3. Enclosures 2 and 3 of Reference 1, Tennessee Valley Authority Sequoyah Plant (SQN) Units 1 and 2 - Proposed Technical Specification (TS) Change 01-08, Description and Evaluation of the Proposed Pages and Revised Pages, Respectively.

4. Enclosures 4 and 5 of Reference 1, WCAP-15669 (Proprietary Version) and WCAP-15670 (Non-Proprietary Version), "Westinghouse Power Measurement Instrument Uncertainty Methodology for Tennessee Valley Authority, Sequoyah Units 1 and 2 (1.3% Uprate to 3467 MWt - NSSS Power.)"

5. Letter from A. C. Thadani (NRC) to J. H. Taylor (B&W), Acceptance for Referencing of Augmented Topical Report BAW-10159P, "BWCMV Correlation for Critical Heat Flux in Mixed Vane Grid Fuel Assembly," May 22, 1989.

6. Letter from G. M. Holahan (NRC) to J. H. Taylor (B&W), Acceptance for Referencing of Babcock & Wilcox Fuel Company Topical Report BAW-10189(P), "CHF Testing and Analysis of the Mark-BW Fuel Assembly Design," January 3, 1995.

7. Letter from NRC to J. H. Taylor (B&W), Acceptance for Referencing of Topical Report BAW-10170P, "Statistical Core Design for Mixing Vane Cores (TAC No. 66318)" September 1987.

8. BAW-10220P-A, "Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2," Framatome Cogema Fuels, March 1996 (Reference 25 in Enclosure 2 of letter from R. W. Hernan (NRC) to O. D. Kingsley, Jr. (TVA), "Issuance of Technical Specification Amendments for the Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. M95144 and M95145)," April 27, 1997.

9. Page E8-4 in Enclosure 8 of Reference 1, "Applicability of Comanche Peak and Watts Bar Power Uprate RAIs to Sequoyah 1&2 Uprate (non-proprietary)."

10. Letter from P. Salas (TVA) to NRC, Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Technical Specification (TS) Change No. 01-08, "Response to Request for Additional Information (RAI) (TAC Nos. MB3435 and MB3436)," March 11, 2002.

11. NS-TMA-2182, Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC) dated December 30, 1979, "ATWS Submittal."

12. Section 2.4.1.2 in Enclosure 6 of Reference 1, WCAP-15726, "Sequoyah Units 1 and 2 1.3-Percent Power Uprate Program Licensing Report."

13. Westinghouse WCAP-10263, “A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor,” January 1983

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