April 12, 2002

Mr. William T. Cottle
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS APPROVING UPRATED CORE THERMAL POWER AND REVISING THE ASSOCIATED TECHNICAL SPECIFICATIONS (TAC NOS. MB2899 AND MB2903)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment No. 138 to Facility Operating License No. NPF-76 and Amendment No. 127 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 22, 2001, as supplemented by letters dated January 21; February 5, 14, and 27; and March 4, 2002.

The amendments revise the TSs to reflect a 1.4 percent increase in the reactor core thermal power level from 3,800 megawatts thermal (MWt) to 3,853 MWt, as requested in your application. The change applies to each Facility Operating License after replacement of the steam generators with Model Δ94 steam generators.

The request for amendments is based on a reduced core thermal power uncertainty associated with the more accurate measurement of feedwater flow by the CROSSFLOW Ultrasonic Flow Measurement (UFM) System. The CROSSFLOW UFM System was previously reviewed and approved by the Nuclear Regulatory Commission (NRC) staff in a safety evaluation report dated March 20, 2000.

In support of its request for the increase of core thermal power level the licensee performed a re-evaluation of the nuclear steam supply system parameters, safety-related systems and components, nuclear fuel, and loss-of-coolant accident (LOCA) and non-LOCA accident analyses related to operation at the increased reactor power level of 3,853 MWt. The NRC staff has reviewed the licensee's analyses. A copy of the NRC staff's related Safety Evaluation is enclosed.

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 138 to NPF-76
2. Amendment No. 127 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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DATE	4/11/02

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STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 138
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and the City of Austin, Texas (COA) (the licensees), dated August 22, 2001, as supplemented by letters dated January 21; February 5, 14, and 27; and March 4, 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, as amended, 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 license 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.

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*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio, Central Power and Light Company, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of the Facility Operating License No. NPF-76 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 138 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by T Marsh for/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: April 12, 2002

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 127
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and the City of Austin, Texas (COA) (the licensees), dated August 2, 2001, as supplemented by letters dated January 21; February 5, 14, and 27; and March 4, 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, as amended, 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 license 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.

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*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio, Central Power and Light Company, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of the Facility Operating License No. NPF-80 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 127 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date the Model △94 steam generators are installed.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by T Marsh for/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: April 12, 2002

ATTACHMENT TO LICENSE AMENDMENT NOS. 138 AND 127

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

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

REMOVE	INSERT
1-5	1-5
2-1	2-1
3/4 7-2	3/4 7-2
*6-21	*6-21
6-22	6-22
6-22a	6-22a

*Overleaf page provided to maintain document completeness. No changes on this page.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 138 AND 127 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By letter dated August 22, 2001, as supplemented by letters, dated January 21; February 5, 14, and 27; and March 4, 2002, STP Nuclear Operating Company (STPNOC or the licensee) submitted an application requesting changes to Facility Operating License Nos. NPF-76 and NPF-80 and the Appendix A, technical specifications (TSs) for South Texas Project (STP), Units 1 and 2. The supplementary letters provided clarifications and did not change the NRC staff's conclusions regarding no significant hazards consideration.

The licensee's proposed amendment revises the facility operating license to reflect a 1.4 percent increase in the reactor power level from 3,800 megawatts thermal (MWt) to 3,853 MWt. This uprated power level will be implemented for STP Unit 2 after the steam generators are replaced with Westinghouse Model Δ94 steam generators. The licensee provided the proposed changes to the existing facility operating licenses in Attachment 3, 4, and 5 of the application dated August 22, 2001 (Ref. 1). The technical analysis in support of proposed power uprate is documented in Attachment 6 of the application.

2.0 BACKGROUND

2.1 Core Thermal Power Measurement Uncertainty

The reactor core thermal power is validated by a nuclear steam supply system (NSSS) energy balance calculation. The reliability of this calculation depends primarily on the accuracy of feedwater flow, temperature, and pressure measurements. Because the measuring instruments have measurement uncertainties, margins are included to ensure the reactor core thermal power levels do not exceed safe operating levels.

At the time of the issuance of initial operating license to South Texas Project, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, required licensees to assume a 2.0 percent measurement uncertainty for the reactor thermal power and to base their transient and accident analyses on an assumed power level of at least 102 percent of the licensed thermal power level. The 2 percent power margin was intended to address uncertainties

related to heat sources and measuring instruments. Appendix K to 10 CFR Part 50 did not allow for any credit for demonstrating that the measuring instruments may be more accurate than originally assumed in the emergency core cooling system (ECCS) rulemaking.

On June 1, 2000, the Nuclear Regulatory Commission (NRC) published a final rule (65 FR 34913) that allows licensees to justify a smaller margin for power measurement uncertainty when more accurate instrumentation is used to calculate the reactor thermal power and calibrate the neutron flux instrumentation.

The licensee has performed a re-evaluation of the NSSS parameters, safety-related systems and components, nuclear fuel, and loss-of-coolant accident (LOCA) and non-LOCA accident analyses related to operation at the increased reactor power level of 3,853 MWt. The request for amendments is based on a more accurate measurement of feedwater flow due to a reduced core-thermal-power uncertainty. The improved accuracy is achieved by installation of the CROSSFLOW Ultrasonic Flow Measurement (UFM) System. The improved flow measurement instrumentation would allow the licensee to operate STP with a margin below the 2.0 percent margin previously used in the licensing basis ECCS analyses. The licensee states that as a result of the improvement in the flow measurement accuracy, the STP power measurement uncertainty has been reduced from 2.0 percent to 0.6 percent.

The CROSSFLOW UFM System was previously reviewed and approved by the NRC staff in a safety evaluation report dated March 20, 2000.

2.2 Maximum Core Thermal Power-Regulatory Guide 1.49

In the NRC Regulatory Guide (RG) 1.49, Revision 1, "Power Levels of Nuclear Power Plants," the NRC staff stated that the licensed power levels be limited to a reactor core power level of 3,800 MWt or less until January 1, 1979, at the earliest. Since the issuance of the RG 1.49 (Revision 1) in 1973, the NRC staff found other plants acceptable for operation above 3,800 MWt (e.g., Grand Gulf was approved for 3833 MWt on November 1, 1984). In 1994, the NRC staff approved the evolutionary plants General Electric (GE) Advanced Boiling-Water Reactor (ABWR) and Combustion Engineering (CE) System 80+ for power levels of 3,926 MWt and 3,914 MWt, respectively. During the review of the evolutionary plants the NRC staff informed the Commission that it planned to approve power levels above 3,800 MWt, unless the Commission disagrees with the NRC staff's decision. The Commission approved the NRC staff's proposals to exceed the RG 1.49 (Revision 1) power restriction of 3,800 MWt, without any comments. Consequently, the NRC staff concluded that it can proceed with the evaluation of the licensees request for power increase above 3,800 MWt, without prior Commission approval or notification.

3.0 EVALUATION

The NRC staff's evaluation of the licensee's application (Ref. 1) includes review of the licensee's evaluation of the NSSS parameters, safety related systems and components, nuclear fuel, and LOCA and non-LOCA accident analyses related to operation at the increased reactor power level of 3,853 MWt.

3.1 NSSS Parameters

In Reference 1, the licensee provided a summary of NSSS design parameters for 1.4 percent power uprate. The design parameters include reactor power, reactor coolant flow rate, temperatures and pressure, steam generator steam flow, and feedwater temperature. These design parameters are established for the evaluations of power uprate with the Westinghouse Model Δ94 Steam Generators. The values of these parameters are demonstrated acceptable by transient and accident analyses using these parameters. The NRC staff finds that the power uprate parameters are acceptable based on the acceptable results described in this safety evaluation.

3.2 CROSSFLOW Calculation

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 10 CFR Part 50 requires LOCA and ECCS analyses to assume that the reactor has been operating continuously at a power level at least 102.0 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. Later, 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 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 issued a final rule that allows the licensees the options of justifying a smaller margin for 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 licensed power or use the margin to relax ECCS-related TSs. 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 has a TS limit, the proposals to raise the licensed power level must be reviewed and approved under the license amendment process.

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 NSSS. This calculation is called “secondary calorimetric” for a pressurized water reactor (PWR) and “heat balance” for a boiling water reactor (BWR). The accuracy of this calculation depends primarily on the accuracy of feedwater flow and main steam and feedwater temperature and pressure measurements. Feedwater flow is the most significant contributor to the core thermal power uncertainty. An accurate measurement of these three parameters will result in an accurate determination of core thermal power, and thereby an accurate calibration of the nuclear instrumentation.

The instrumentation used for measuring feedwater flow is typically an orifice plate, a venturi meter, or a flow nozzle. These devices 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 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 that the calibration of the flow element shifts when the flow element is fouled, 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 recalibrated. Due to the high cost of recalibration and the need to improve flow instrumentation uncertainty, the industry assessed other flow measurement techniques and found the CROSSFLOW UFM to be a viable alternative. The measurement uncertainties due to venturi fouling and instrumentation drift and calibration shifts are essentially eliminated when a CROSSFLOW UFM is used. The CROSSFLOW UFM does not replace the currently installed plant venturi, but provides the licensee an in-plant capability for periodically recalibrating the feedwater venturi to adjust for the effect of fouling. A unique advantage of the CROSSFLOW UFM system is that it is installed external to the pipe in which flow is to be measured, thereby eliminating any possibility of compromising pressure boundary integrity.

The CROSSFLOW UFM consists of four ultrasonic transducers mounted on a metal support frame which is clamped on the feedwater piping. There is one upstream and one downstream transducer station, each station consisting of one transmitting and one receiving transducer. The operation of a cross-correlation UFM is based on the fact that an ultrasonic beam traveling across fluid flowing in a pipe is affected (modulated) by the turbulence (eddies) present in the flowing liquid. When this modulated signal is processed, a random signal which is a signature of the flowing eddies can be obtained. The CROSSFLOW UFM calculates the time a unique pattern of eddies take to pass between two sets of ultrasonic transducers and divides the known distance between the two sets of transducers by the calculated time to obtain the flow velocity. This measured velocity is not an average velocity (highest velocity is at the center of the pipe) and should be multiplied by the “Velocity Profile Correction Factor” (VPCF) to obtain the average velocity of the fluid flowing in the pipe.

The CROSSFLOW UFM system consists of a Mounting/Transducer Support Frame with ultrasonic transducers, a signal conditioning unit (SCU), and a data processing computer (DPC). The DPC receives a feedwater flow signal from the SCU and feedwater pressure and temperature input from the plant computer. Using a built-in signal processing algorithm, the CROSSFLOW DPC calculates fluid velocity and converts the fluid velocity to a mass flow using flow, temperature, and pressure as calculation inputs. The CROSSFLOW feedwater mass flow is periodically compared to the feedwater venturi mass flow to determine the correction factor that must be applied to the venturi mass flow to obtain the corrected mass flow. This corrected mass flow is used in calculating core thermal power and thereby calibrating nuclear instrumentation in accordance with the plant TS requirements.

The Westinghouse Topical Report CENPD-397-P-A (previously approved by the NRC staff) describes the CROSSFLOW UFM system for the measurement of feedwater flow and provides a basis for the proposed 1.4 percent uprate of the licensed reactor power. Based on calculations and tests on a typical feedwater loop (straight pipe, fully developed flow), using

CROSSFLOW UFM for feedwater flow measurement, the topical report stated that the CROSSFLOW UFM system is able to achieve an uncertainty of 0.5 percent or better with a 95 percent confidence interval. The topical report provides specific guidelines and equations for determining uncertainty values of the CROSSFLOW input parameters (VPCF, inside diameter, transducer spacing, feedwater density, and CROSSFLOW time delay). The plant-specific uncertainties are determined when the meter is installed, using the guidelines and equations provided in the topical report. The topical report stated that a trained CENP, presently the Westinghouse representative, will install the hardware and software of the CROSSFLOW UFM.

Westinghouse Topical Report WCAP-15633 provides plant-specific secondary side power calorimetric uncertainty calculation for STP Units 1 and 2 and includes proprietary information to Westinghouse Electric Company and STPNOC. Westinghouse used NRC staff approved methodology where the power measurement uncertainty components are statistically combined to determine the Secondary Side Power Calorimetric uncertainty. The secondary side power measurement uncertainties are in four principal areas: feedwater flow, feedwater enthalpy, steam enthalpy, and net pump heat addition. Feedwater flow measurement uncertainty is the largest contributor to power calorimetric measurement uncertainty. Westinghouse used a 1.0 percent instrument uncertainty for the CROSSFLOW UFM, instead of 0.5 percent or better, as stated achievable in Westinghouse Topical Report CENPD-397-P-A and confirmed achieved at the STP Units 1 and 2. The licensee indicated that, the Westinghouse Topical Report WCAP-15633 used 1.0 percent measurement uncertainty for the CROSSFLOW UFM, as a conservative margin for the power calorimetric uncertainty, based on engineering judgment for margin allocation. Subsequently, in response to the NRC staff request for additional information (RAI), the licensee stated that a new calculation limits feedwater flow uncertainty to 0.97 percent at 95 percent confidence interval which supports the assumed 1.0 percent feedwater flow uncertainty input to the thermal power calculation. The licensee stated that STP plans to implement the 1.4 percent power uprate with 0.97 percent uncertainty programmed into the CROSSFLOW UFM. This will allow STP operating flexibility and still meet the requirements for the revised rated thermal power calculation. The NRC staff review found that WCAP-15633 accounted for all components of the plant calorimetric measurement uncertainty and followed the NRC staff approved methodology for calculating power measurement uncertainty. The total power measurement uncertainty in the plant calorimetric calculation was found to be 0.6 percent to support the proposed 1.4 percent power uprate of STP, Units 1 and 2.

The NRC staff SER on Westinghouse Topical Report CENPD-397-P-A included four additional requirements to be addressed by a licensee referencing this topical report for power uprate. STPNOC's submittal addressed each of the four requirements as follows:

1. The licensee should discuss the development of maintenance and calibration procedures that will be implemented with the CROSSFLOW UFM installation. These procedures should include processes and contingencies for an inoperable UFM and the effect on thermal power measurement and plant operation.

    - The licensee stated that the CROSSFLOW UFM system has been installed in STP, Units 1 and 2 and the plant procedures have been developed for maintenance and calibration of the CROSSFLOW UFM system in accordance with the guidelines established in the referenced topical report and user's manual. The CROSSFLOW UFM system maintenance consists of both manual and automatic checks of the

system. The manual checks include verification of time-delay circuits and transducers scans for any shift and the automatic checks include automatic calculations of flow measurement uncertainty and the verification that it is enveloped by the values calculated for the power uprate.

The licensee stated that the CROSSFLOW UFM system inoperability has no immediate effect on plant operation as it does not perform any safety function and is not used to directly control any plant systems. However, prior to implementation of the power uprate, the plant procedures for the CROSSFLOW UFM maintenance and calibration will be revised to address the unavailability of the CROSSFLOW UFM system. The revised procedures will include actions for reducing power, and for performing the power calorimetric using the feedwater flow venturis to determine the feedwater flow rate. The CROSSFLOW system inoperability is automatically transmitted to the plant computer to alarm an overhead annunciator in the control room. In this condition, the plant may continue to operate at a core thermal power level of 3,853 MWt for 24 hours. In response to the NRC staff's RAI, the licensee stated that if an over-power event occurred during the time the UFM was unavailable, there would be no change to plant system or operator response to the transient. Reactor trip instrumentation and alternate reactor power indications remain calibrated and available to ensure power limits are not exceeded. If the inoperable CROSSFLOW UFM system is not restored within 24 hours, plant procedures would require reactor power to be reduced to a level less than or equal to 3,838 MWt. The core thermal power would be maintained at this reduced power level until the CROSSFLOW UFM system was restored to service and a secondary plant calorimetric was performed per the plant TS Surveillance Requirement 4.3.1.1.2.a using the CROSSFLOW UFM indication of feedwater flow.

The proposed power level reduction from 3,853 MWt to 2,838 MWt for an inoperable CROSSFLOW UFM is calculated in the Westinghouse Topical Report WCAP-15697. The topical report calculations established 1 percent uncertainty in the secondary side four loop power calorimetric measurement using venturi measurement of feedwater flow. The NRC staff reviewed WCAP-15697, Revision 0 and found that the calculations followed staff approved methodology and, therefore, are acceptable. The STP, Units 1 and 2 were licensed to operate at 3,800 MWt which included the Appendix K required 2 percent margin for instrumentation uncertainty. Under the final rule, the licensees are allowed to operate at a power level with less than 2 percent margin for plant ECCS analyses if the plant reactor thermal power measurement uncertainty is low enough to justify such an operating power level. As such, the calculated 1.0 percent uncertainty in power calorimetric measurement justifies a corresponding 1 percent increase in the licensed reactor thermal power to a power level of 3,838 MWt. The NRC staff finds the proposed increase in reactor thermal power of STP Units 1 and 2 are acceptable.

In response to the NRC staff's RAI, the licensee described its programs for the calibration of all other instrumentation, in addition to the CROSSFLOW, whose measurement uncertainties affect the plant power calorimetric uncertainties determined in Westinghouse Topical Reports WCAP-15633 and WCAP-15697. These other instruments are for feedwater pressure and temperature, steam pressure, and steam generator blowdown flow. The licensee stated that the

STP Plant Instrumentation Calibration Verification Program, Plant Surveillance Test Program, and Preventive Maintenance Program delineate controls for calibration and verification of these instruments. The STP Software Quality Assurance Procedure controls the appropriate level of validation, verification, and documentation. The STP Configuration Management Procedure establishes responsibilities and requirements to ensure conformance to the approved design and functional characteristics of the instrumentation. The licensee listed STP's procedures for performing corrective actions, reporting deficiencies to the manufacturers, and receiving and addressing manufacture deficiency reports on these instruments. The NRC staff believes that the licensee's plant procedures can sufficiently assure instrumentation capability to provide acceptable power calorimetric uncertainty for the proposed power uprate.

2. For plants that currently have CROSSFLOW UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and confirm that the instrumentation is representative of CROSSFLOW UFM and bounds the requirements set forth in Topical Report CENPD-397-P-A.

   - The licensee stated that the CROSSFLOW UFM was installed and placed in operation in December 1999 in STP, Units 1 and 2, to provide corrections for flow instrumentation, drift and venturi fouling. The installations were performed in accordance with the Westinghouse procedure for the CROSSFLOW UFM installation based on the descriptions and criteria established in the CROSSFLOW UFM Topical Report CENPD-397-P-A. Three transducer failures have been observed since the UFMs were placed in operation. The CROSSFLOW system automatic check feature identified the failure as designed, but did not introduce any bias into the UFM measurements. The licensee stated that the installed CROSSFLOW system at STP, Units 1 and 2 is representative of the CROSSFLOW UFM discussed in CENPD-397-P-A, Revision 1, and is bounded by the requirements set forth in this topical report.

3. The licensee should confirm that the methodology used to calculate the uncertainty of the CROSSFLOW UFM in comparison to the current feedwater flow 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 the UFM for comparison.

   - The licensee stated that the CROSSFLOW UFM measurement uncertainty calculations were performed by Westinghouse for STP, Units 1 and 2 and the calculation methodology is consistent with the methodology described in Topical Report CENPD-397-P-A, Revision 01 and the STP methodology for calculating instrument uncertainties.

4. The licensee of the plant at which the installed CROSSFLOW UFM was not calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should submit additional justification. This justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and plant configuration for the specific installation, including the

propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated CROSSFLOW UFM, the licensee should confirm that the piping configuration remains bounding for the original CROSSFLOW UFM installation and calibration assumptions.

- The licensee described the feedwater configuration at STP, Units 1 and 2 and stated that the CROSSFLOW meter was calibrated at the plant under actual full-power operating conditions and the plant installation followed the guidelines of Topical Report CENPD-397-P-A, Revision 01.

The NRC staff finds that STPNOC's response sufficiently addressed these requirements and adequately resolved the plant-specific concerns about CROSSFLOW UFM maintenance and calibration, hydraulic configuration, and procedures and contingency plans for an inoperable CROSSFLOW UFM. The licensee used an approved methodology to calculate the plant-specific CROSSFLOW measurement uncertainty and the plant power calorimetric measurement uncertainty.

Based on a review of the licensee's submittals and the plant-specific Westinghouse calculations of the plant power calorimetric measurement uncertainty, the NRC staff finds that the STP, Units 1 and 2 thermal power measurement uncertainty using the CROSSFLOW UFM is limited to 0.6 percent of actual reactor thermal power and can support the proposed 1.4 percent thermal power uprate. The NRC staff also found that the licensee adequately addressed the four additional requirements outlined in the NRC staff SER on the CROSSFLOW Topical Report CENPD-397-P-A.

3.3 Revised Thermal Design Procedure Uncertainties

The STP, Units 1 and 2 safety analyses of non-LOCA design-basis events were performed with the revised thermal design procedure (RTDP) methodology, described in WCAP-11397-P-A (Ref. 4). With the RTDP methodology, the uncertainties of the reactor system parameters, core power distribution, and the critical heat flux (CHF) correlation are statistically combined to establish the departure from nucleate boiling ratio (DNBR) design limit so that the nominal values of the system parameters can be used in the analysis of the design basis transients. The RTDP methodology has been accepted by the NRC for referencing in licensing applications; however, the uncertainty values of the system parameters are evaluated on a plant-specific basis. For the STP, Units 1 and 2, the RTDP methodology includes the measurement uncertainties of the core average temperature (Tavg), pressurizer pressure, RCS flow, reactor power, and fuel enthalpy rise factor. The uncertainty values of these reactor system parameters are documented in WCAP-13441, Revision 1 (Ref. 5). As described in Section 4 of Attachment 6, the power uprate has no effect on the uncertainties of Tavg, pressurizer pressure, and RCS flow. The NRC staff agrees with the licensee's assessment.

However, the power measurement uncertainty is reduced with the installation of the CROSSFLOW UFM system for feedwater measurement. The CROSSFLOW UFM system, described in CENPD-397-P-A (Ref. 6), has been approved by the NRC. In WCAP-15633 (Ref 7) and WCAP-15697 (Ref. 8), respectively, the licensee performed evaluations to determine power calorimetric measurement uncertainties of 0.6 percent using the UFM system for feedwater flow measurement, and 1.0 percent when the UFM system is unavailable. The NRC staff has reviewed these two topical reports, and concluded that the power measurement

uncertainties of 0.6 percent and 1.0 percent for the cases of the UFM availability and unavailability, respectively, are acceptable.

As discussed in Section 3.6.10 of this safety evaluation, the measurement uncertainties of T-avg, pressurizer pressure, and RCS flow, as well as the power measurement uncertainties, are used in the RTDP to determine the DNBR design limits.

3.4 Design Transients

NSSS Design Transients

To support the power uprate for STP, Units 1 and 2, the licensee evaluated the current primary and secondary side design transients to determine their continued applicability at the power uprated conditions. The licensee has concluded that operating conditions for the 1.4 percent power uprate do not change sufficiently to require a revision to any of the primary or secondary side design transients. There is sufficient conservatism in the assumptions of the original design transient development (including 2 percent power uncertainty) to accommodate the proposed power level. Therefore, the NSSS design transients do not have to be revised. The NRC staff agrees with the licensee's assessment and finds it acceptable.

Auxiliary Equipment Design Transients

The licensee has evaluated the current auxiliary equipment transients and determined that the only transients that could be potentially impacted by the 1.4 percent power uprate are those temperature transients that are impacted by the full-load T_{cold} RCS temperature. These transients are currently based on a worst case T_{cold} of 570 ^{o}F. Since the T_{cold} will range between 549.8 ^{o}F and 560.3 ^{o}F at power uprate conditions, the worst T_{cold} of 570 ^{o}F will be bounding for the current design transients and, therefore, all of the auxiliary equipment design transients are applicable to power uprate conditions.

3.5 Nuclear Steam Supply Systems

3.5.1 NSSS Fluid Systems

Reactor Coolant Systems (RCSs)

The RCS operating conditions are changed slightly at uprated power. The design operating parameters for temperature, pressure, and flow are bounded by parameters previously evaluated for the Westinghouse Model Δ94 steam generators. The maximum T_{hot} and the minimum T_{cold} values for the 1.4 percent power uprating are bounded by the Westinghouse Model Δ94 steam generator values. Therefore, the reactor coolant pump net positive suction head (NPSH) requirements, pressurizer performance, and the natural circulation cooldown capability are not affected by the proposed power uprate transient and safety analyses. The NRC staff finds that the changes of RCS operating parameters associated with power uprate are acceptable based on the acceptable results of the safety analyses described below.

NSSS Auxiliary Systems Evaluation

Chemical and Volume Control System/Boron Thermal Regeneration System

The main role of the chemical and volume control system (CVCS) is to manage RCS water inventory, boron concentration, and water chemistry. Since the system is connected to the RCS, a change in the RCS operating temperature could have some effect on its performance. The letdown line and the excess letdown line in the CVCS interface with the RCS cold leg, and cold leg temperature will be impacted by the power uprate. However, the licensee has demonstrated that the performance of the CVCS after the power uprate is bounded by the results of the analysis performed to support installation of the Model Δ94 steam generators. This analysis was based on an assumed full-load NSSS worst-case T_{cold} of 570°F which is significantly higher than the corresponding T_{cold} reached after the power uprate. The operating performance of the CVCS and the associated boron thermal regeneration system (BTRS) for 1.4 percent power uprate is, therefore, acceptable.

Boration Capability

Since the operating parameters for RCS temperature, pressure, and flow for the 1.4 percent power uprate program are bounded by the values previously determined for Westinghouse Model Δ94 steam generator values, the boration capabilities to achieve safe cold shutdown and emergency core cooling system function will not be impacted by the proposed power uprate.

Residual Heat Removal System

The licensee has performed an analysis to verify the cooldown capability of residual heat removal (RHR) system at power uprate conditions. The results indicate that for the three train RHR operation, a total time of less than 10 hours is required to reach an RCS temperature of 150 ^{o}F after reactor shutdown, which is less than the design objective of 12 hours. For the safety grade cold shutdown analysis, with one train of RHR operating, the results of this analysis show that the reactor could be brought to the cold shutdown conditions within a reasonable period time of 23 hours following the RHR initiation. The NRC staff finds this decay heat removal capability is well within the guidance of Standard Review Plan Section 5.4.7 to be capable of bringing the plant to cold shutdown within 36 hours, and is therefore acceptable.

ECCS and Containment Spray System

Also, the adequacy of the safety injection system during the injection and sump recirculating phases is verified by various safety analyses performed in support of the power uprate. There are no system modifications required to support power uprate. The NRC staff agrees with the licensee's assessment based on the acceptable results of the safety analyses as described below.

3.5.2 NSSS/Balance-Of-Plant Interface Systems

Main Steam System

Each of the STP units has 20 main steam safety valves (MSSV) with a total relief capacity of 20.65 x 10^6 lb/hr. This provides approximately 120 percent of the maximum uprated full load

steam flow of 17.2 x 10^6 lb/hr. Therefore, the capacity of MSSVs will be more than adequate to support the 1.4 percent power uprate.

The steam generator power operated relief valves (SGPORVs) provide a means for decay heat removal when the condenser is not available. In this case, the SGPORVs in conjunction with the auxiliary feedwater system, can perform plant cooldown following a reactor trip to RHR system initiation conditions. The results of the licensee’s evaluation show that the original sizing of the SGPORV is capable of performing plant cooldown over the full range of NSSS design parameters under power uprate conditions.

The main steam isolation valves (MSIVs) are designed to close within 5 seconds of the receipt of the closure signal initiated on the steam line break flow conditions. Rapid closure results in a pressure challenge to the valve components. The licensee's evaluation shows that the proposed increase of 1.4 percent in core thermal power will not increase the pressure challenge to the valve components. Therefore, the proposed core power increase of 1.4 percent has no significant impact on the performance of MSIVs. The MSIV bypass valves perform their warm up function at no load or low power conditions. The licensee states that the proposed 1.4 percent increase in core thermal power will have no significant impact on the performance of the MSIV bypass valves.

Therefore, the NRC staff concludes that the existing MSSVs, SPPORVs, and MSIVs have sufficient capacities to perform their intended functions for the 1.4 percent power uprate conditions, and are acceptable.

Steam Dump System

The licensee evaluated the steam dump system valves and piping capacity to discharge 40 percent of the rated steam flow at full load steam pressure to permit NSSS to withstand an external load of 50 percent of the plant rated electrical load, without a reactor trip. A steam dump capacity of 40 percent of rated steam flow at full load pressure prevents valves from lifting, following a reactor trip from full power.

The licensee stated that the calculated condenser steam dump valve steam flow capacity at the proposed increase of 1.4 percent core thermal power continues to meet the above sizing criterion for condenser steam dump valves.

The NRC staff concurs with the licensee's evaluation and conclusions for the steam dump valves' performance at the proposed core thermal power level, and finds that the steam dump system continues to perform adequately at 1.4 percent power uprate.

Condensate and Feedwater System (C&FWS)

C&FWS automatically maintains steam generator water levels during operations. The proposed 1.4 percent increase in core thermal power will increase the volumetric flow and the system pressure drop. The licensee evaluated the impact of the increase in core thermal power on the feedwater isolation valves (FIVs), the backup feedwater control valves (FCVs), and the feedwater system pumps. The FIVs and the backup FCVs are required to be capable of closure within 10 seconds after receipt of the closure signal. The rapid closure requirements imposed on the FIVs and FCVs impose potentially large dynamic pressure changes. Since the

worst load conditions occur after a steam line break from no-load conditions, with conservative assumption of the feedwater pumps in service, the increase in maximum core thermal power will have no impact on the maximum dynamic pressure changes. The NRC staff concurs with the licensee's conclusion, and finds that C&FWS continues to perform adequately at 1.4 percent power uprate.

Auxiliary Feedwater System (AFWS)

AFWS is the backup system to the normal feedwater system. It also functions as the emergency feedwater system (EFS) to mitigate the core damage and overpressurization following transients and accidents, such as a loss of normal feedwater system or a secondary-system pipe break. The licensee states that its evaluation confirms that the current AFWS design basis performance remains acceptable.

The AFWS takes its suction from auxiliary feedwater storage tank (AFST). Sufficient AFWS capacity is required to satisfy the requirements of postulated transient and accident conditions. The licensee states that AFWS minimum volume of 485,000 gallons is adequate for 102 percent of the rated power, or 3876 MWt. Allowing for the 0.6 percent uncertainty, the capacity is adequate for the proposed 1.4 percent core thermal power increase. The NRC staff agrees with the licensee's conclusion, and finds that the AFWS continues to perform adeqately at 1.4 percent power uprate.

Steam Generator Blowdown System (SGBD)

The SGBD system is used to control chemical composition and buildup of solids in the steam generator shell-side (secondary side) water. The SGBD system flow rates are determined by water chemistry and by the tube-sheet sweep required for controlling buildup of solids. The rate at which dissolved solids are introduced into the secondary water of the steam generator depends on condenser leakage, quality of secondary makeup water, and the amount of corrosion products generated by flow accelerated corrosion (FAC). However, only the generation of corrosion products by FAC could be affected by the power uprate because it increases with increasing feedwater flow which is expected to occur after the power uprate. The licensee has determined that because the increase in feedwater flow will be insignificant there will be no impact of the power uprate on the generation of corrosion products and the chemistry environment in the shell-side of the steam generator will remain unchanged. The NRC staff concurs with the licensee's conclusion.

The inlet pressure to the SGBD system varies with the steam generator operating pressure. As the pressure decreases, the SGBD system control valves have to open to maintain the desired SGBD system flow rate into the system's flash tank. The current range of plant design parameters allows a maximum decrease in steam pressure from no-load to full-load. The licensee has determined that these pressures remain unchanged when they are based on the revised range of NSSS design parameters associated with the power uprate. Therefore, the range of design parameters associated with the power uprate will not impact SGBD system flow control. The NRC staff reviewed the licensee's analyses and concur with the licensee's conclusion that SGBD continues to control the chemical composition and buildup of solids in the steam generators at 1.4 percent power increase. The NRC staff finds the analysis acceptable.

3.5.3 NSSS Control Systems

The licensee has evaluated the capabilities of NSSS control systems to respond to Condition I transients to confirm that the plant can respond to these transients without reactor trip or actuation of ESF. The results of the licensee's evaluation are discussed below.

Condition I Transient Evaluations

The capabilities of NSSS control system to respond to a 10 percent step load decrease, 50 percent load rejection and turbine trip below 50 percent power level without reactor trip, and ESF actuation are analyzed by the licensee. The analyses are performed for the Model Δ94 steam generators based on a power level of 3,821 MWt with a power uncertainty of 2 percent or 3,897 MWt. Since 3,897 MWt is greater than 3,876 MWt, which is the 1.4 percent power uprate condition of 3,853 MWt plus 0.6 percent power measurement uncertainty, the NRC staff agrees that these analyses are also valid and bounding for the 1.4 percent power uprate.

Other Considerations

The licensee indicated that the existing cold overpressure setpoints are unaffected by the 1.4 percent uprating, since cold overpressure events can only occur during reactor shutdown, which is not impacted by the 1.4 percent uprate. Since the cold overpressure events are not affected by the power uprate, the NRC staff agrees that the cold overpressure protection system setpoints remains acceptable for the 1.4 percent power uprate.

3.6 NSSS Components

3.6.1 Reactor Vessel Structural Evaluation

The licensee stated that the normal operating vessel outlet temperature (T_{hot}) and normal operating vessel inlet temperature (T_{cold}) remain within the bounds of the previous reactor vessel structural evaluations performed for the Model △94 steam generators. There are no changes to any of the primary-side design transients that were considered for the Model △94 steam generators. The reactor vessel LOCA loads for the Model △94 steam generators still apply to the 1.4 percent power uprating. The licensee further stated that the previous reactor pressure vessel system seismic analysis is not changed due to the 1.4 percent uprating, since neither the seismic response spectra nor the mass inputs for the equipment are changed. Therefore, the faulted condition blowdown (LOCA), plus safe shutdown earthquake (SSE) seismic loads previously considered in the reactor vessel structural analysis are not impacted. As a result, there are no changes to the maximum stress intensities, the maximum ranges of stress intensity, or the maximum cumulative fatigue usage factors that were previously reported in the STP, Units 1 and 2 reactor vessel stress reports. The licensee concluded that the STP, Units 1 and 2 reactor vessels continue to satisfy the applicable requirements of Section III (Nuclear Power Plant Components) of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code, 1971 Edition through the summer 1973 Addenda, in accordance with the reactor vessel design requirements. The NRC staff agrees with this conclusion.

3.6.2 Reactor Vessel Integrity-Neutron Irradiation

The licensee has reviewed the effect of 1.4 percent power uprate on STP, Units 1 and 2 reactor vessel (RV) integrity. RV integrity is impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The RV integrity evaluation for the 1.4 percent uprating included the following evaluations:

1.04 Review of the reactor vessel surveillance capsule removal schedules to determine if changes are required as a result of changes in the vessel fluence due to the 1.4 percent power uprate.

1.05 Review of the existing pressure-temperature (P-T) limit curves to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.

1.06 Review of the existing pressurized thermal shock reference temperature (RT_{PTS}) values to determine if the effects of the uprated fluence projections result in an increase in RT_{PTS} for the beltline materials in the STP, Units 1 and 2 reactor vessels at the end of license (EOL) (32 EFPY).

1.07 Review the upper shelf energy (USE) values at EOL for all reactor vessel beltline materials in the STP, Units 1 and 2 reactor vessels to assess the impact of the uprated fluence projections.

Surveillance Capsule Withdrawal Schedule

The licensee stated that the revised fluence projections considering the 1.4 percent power uprate exceed the fluence projections used in the development of the current withdrawal schedules for STP, Units 1 and 2. The licensee performed a calculation of the adjusted reference temperature (ART) value at 32 EFPY for each RV material to determine if the increased fluences alter the number of capsules to be withdrawn for both units. The calculation determined that the maximum shift in ART value using the uprated fluences for STP, Units 1 and 2 at 32 EFPY is less than 100 ^{0}F. Per Appendix H, 10 CFR Part 50 and the American Society for Testing and Materials (ASTM) Standard E185-82, Table 1, if the predicted shift in ART values is less than 100 ^{0}F, three capsules are required to be withdrawn from each unit. This is consistent with the current withdrawal schedule. Therefore, no change is required to the current withdrawal schedules as a result of the proposed power uprate, and the withdrawal schedule is in compliance with Appendix H, 10 CFR Part 50.

Heatup and Cooldown P-T Limit Curves

The licensee stated that STP, Units 1 and 2 are currently operating using 32 EFPY P-T limit curves, which are specified in the TSs (TS). The licensee completed a review of the current heat up and cooldown curve to determine the applicability dates for both units. The review indicated that the revised maximum ART after the 1.4 percent power uprate will be lower than the ART value used in developing the current P-T limit curves for STP, Units 1 and 2 at 32 EFPY. Therefore, no changes in applicability dates are required and the 32 EFPY P-T curves for both units remain valid.

Emergency Response Guideline Limits

The licensee stated that the current peak inside surface reference temperature (RT_{NDT}) values at EOL that were calculated are 82 ^{0}F (Unit 1), and 61 ^{0}F (Unit 2). The limiting material for STP Unit 1 was the Intermediate Shell Plate R1606-3, while the limiting material at STP Unit 2 was the Intermediate Shell Plate R2507-2. These peak values would currently put STP, Units 1 and 2 in emergency response guideline (ERG) Category 1. The licensee calculated the EOL fluence projections considering the power uprate and concluded that Units 1 and 2 will still remain in ERG Category 1 after the power uprate, and this is consistent with the guidance outlined in the Westinghouse Owner's Group guidelines.

Pressurized Thermal Shock (PTS)

The licensee calculated the RT_{PTS} values for STP, Units 1 and 2 using the guidance in RG 1.190 for the estimation of the EOL projected vessel fluence. After the power uprate, the projected EOL fluence values for each RV will be smaller than the current projected EOL fluence values. The smaller fluence values are due to low leakage core loadings for future cycles. Therefore, to evaluate the effects of the 1.4 percent uprate, the RT_{PTS} values for the most limiting material from each unit were re-evaluated using the uprated fluences. Based on this evaluation, all RT_{PTS} values remain below the NRC screening criteria of 10 CFR 50.61(b) using the projected uprated fluence values through 32 EFPY for both units.

Upper Shelf Energy (USE)

The licensee stated that all beltline materials are expected to have a USE greater than 50 ft-lb through EOL as required by 10 CFR Part 50, Appendix G. The EOL USE for each RV material was predicted using the EOL 1/4T fluence projection (i.e., the fluence at a depth 1/4 of the way from the inside surface to the outside surface of the RV). The revised fluence projections associated with the 1.4 percent power uprating have exceeded the fluence projections used in developing the current predicted EOL USE values. However, it has only affected the 1/4T fluence by less than 1 percent. This small amount has no measurable effect on the percent decrease in USE. Therefore, the current predicted USE values for STP, Units 1 and 2 remain valid, and is in compliance with 10 CFR Part 50, Appendix G.

The licensee concluded that the fluence projections associated with the 1.4 percent uprated condition will exceed the current fluence projections used in the evaluation of withdrawal schedules, ERG category, PTS, and USE. The effect of the higher fluence values is minimal for PTS and the capsule withdrawal schedule. The effect of the higher fluences is negligible for the ERG limits and predicted EOL USE. With respect to the P-T limit curves, the existing STP Units 1 and 2 P-T limit curves are based on original, conservative fluence projections. These fluence projections have not been updated based on surveillance capsule data and knowledge of current fuel loading patterns which tend to reduce fluence. Therefore, even considering the power uprate, the RV EOL fluences will remain below those used to generate the existing P-T limit curves. Thus, the existing P-T limit curves remain valid for 32 EFPY.

The NRC staff finds the licensee's evaluation and reasoning to be acceptable and, therefore, the NRC staff concluded that the proposed 1.4 percent uprate for STP, Units 1 and 2 will not have a significant impact on the reactor vessel integrity.

3.6.3 Reactor Internals

The licensee indicated that it has performed analyses to demonstrate that the reactor internals can perform their intended design functions at the 1.4 percent uprated conditions.

An evaluation was performed to determine the hydraulic lift forces on the various reactor internal components to ensure that the reactor internals assembly would remain seated and stable for all conditions. The licensee stated that the evaluation results indicate that the downward force remains essentially unchanged, indicating that the reactor internals would remain seated and stable for the 1.4 percent uprate conditions.

The licensee stated that, with regard to flow- and pump-induced vibration, the current analysis uses a mechanical design flow and T_{cold} that did not change for the revised design conditions. Flow-induced vibration stress levels on the core barrel assembly and upper internals remain low and well below the material high-cycle fatigue endurance limit. Therefore, the 1.4 percent uprate conditions do not affect the mechanical loads.

The reactor internals components subjected to heat generation effects (either directly or indirectly) are the upper core plate, the lower core support, the core baffle plates, the former plates, the core barrel, the neutron panel, the baffle-former bolts, and the barrel-former bolts. The licensee indicated that, for all of the reactor internal components, except the lower core support plate and the upper core plate, the stresses and cumulative fatigue usage factors for STP, Units 1 and 2 were unaffected by the 1.4 percent uprate conditions, because the previous analyses remain bounding. The licensee stated that, due to the lower support plate's proximity to the core and thermal expansion of fuel rods at power, the heat generation rates in the lower support plate due to gamma heating can cause a significant temperature increase in this component. Structural evaluations were performed to demonstrate that the structural integrity of the lower support plate is not adversely affected by the revised design conditions. The cumulative fatigue usage factor of the lower support plate due to the increase in the heat generation rates remains very small, and the lower support plate is structurally adequate for the 1.4 percent uprate conditions. The licensee further stated that the maximum stress contributor in the upper core plate is the membrane stress resulting from the average temperature difference between the center portion of the upper core plate and the rim. The increased stress was determined as a function of the gamma heat generation rate increment. The fluid temperature effect due to the 1.4 percent uprating was insignificant. The results show that the structural integrity of the upper core plate is maintained for the 1.4 percent uprate conditions, since the upper core plate analysis was performed based on conservative assumptions. The cumulative fatigue usage factor of the upper core plate caused by the increase in the heat generation rates remains less than unity. The NRC staff agrees with the licensee's conditions because the results are within the allowables of the original analysis of records.

In its submittal and the response to the NRC staff's questions, the licensee provided an evaluation of the effects of the 1.4 percent power uprate on the reactor coolant flow behavior and the reactor internal system. Since the core inlet temperature remains unchanged, the 1.4 percent power uprate will increase slightly the core coolant temperature (T_{avg} increases 0.4 ^{o}F), which has insignificant effects on the reactor coolant flow, the core bypass flow, and the reactor internals. The rod control cluster assembly drop time remains within 2.8 seconds specified in the TSs. The maximum calculated core bypass flow is less than the 8.5 percent

assumed in power uprate analysis. The change in the bypass flow velocities through the pressure relief holes on the baffle plates is insignificant. The hydraulic lift force on the various reactor internal components remain essentially unchanged. Therefore, the power uprate will have essentially no effect on the reactor internals structural integrity and functionality.

As a result of these evaluations, the licensee concluded that the reactor internal components at STP, Units 1 & 2 will be structurally adequate for the proposed power uprate conditions. The NRC staff concludes that the resulting stresses and fatigue factors of the components are within the allowable of the original analysis of records.

3.6.4 Piping and Supports

The licensee evaluated the impact of the 1.4 percent power uprating on the STP, Units 1 & 2 existing analyses for the reactor coolant loop (RCL), pressurizer surge line, the Class 1 auxiliary lines, and their supports.

The licensee stated that, since there is no significant impact on the RCL analyses, there are no changes in any of the steam generator or reactor coolant loop displacements, the primary equipment nozzle qualification, or the magnitude of the support loads. Therefore, the piping and the support load evaluations for the Model Δ94 steam generators are still applicable for the 1.4 percent power uprating. The maximum primary and secondary stresses and the maximum fatigue usage factors from the existing analysis remain applicable for the 1.4 percent power uprating. The RCS supports were shown to meet the allowable stresses for all loading combinations for the Unit 1 Model Δ94 steam generator loads.

The licensee indicated that the steam generator, reactor coolant pump (RCP), reactor vessel, and pressurizer supports have been qualified for piping and component loads resulting from the Unit 1 Model Δ94 steam generators. Since the 1.4 percent power uprating does not significantly change the loads exerted on the piping and the support structures, the licensee concluded that the piping and the supports will remain qualified for the 1.4 percent power uprate condition. The NRC staff agrees with the licensee's conclusion.

3.6.5 Control Rod Drive Mechanisms (CRDMs)

The CRDM parameters are established by the vessel inlet data. The licensee performed the evaluation for the uprated NSSS power of 3,874 MWt (3,853 MWt core power). The licensee indicated that the upper bound vessel inlet temperature is shown to decrease from the current value of 567°F to 560.3°F. The higher temperature previously evaluated for the Model Δ94 steam generators remains bounding for the 1.4 percent uprating. The NRC staff agrees with the licensee's statement because the original analysis of record still applies.

3.6.6 Reactor Coolant Pumps and Motors (RCPMs)

The RCPs are located between the steam generator outlet and the RV inlet in the RCL. The maximum vessel inlet (RCP outlet) temperature is 560.3° F for the 1.4 percent uprate conditions. The licensee stated that this temperature is lower than the design full power qualification temperature of 561.2°F, and, therefore, represents a less limiting condition. The revised pressure changes (ΔPs), temperature changes (ΔTs), and the maximum pressure and

temperature of the governing transients are less than those previously evaluated and remain bounded for the 1.4 percent uprating.

The licensee indicated that the limiting design parameter of the RCP motor is the horsepower loading at continuous hot and cold operation. Loads on the RCP motors, based on the Model △94 steam generator outlet temperature of 549.4°F and best-estimate flow (BEF) of 105,400 gpm, were calculated to be less than the nameplate rating of the motors. Therefore, no analysis was necessary for operation at the 1.4 percent uprate conditions. Since the 1.4 percent uprated BEFs are bounded by the previously evaluated Model △94 steam generator BEFs, with no change in the steam generator outlet temperature, the RCP motor loads for the Model △94 steam generator project remain bounding and applicable to the 1.4 percent uprate RCP motor loads.

Based on the above evaluation, the licensee concluded that the current RCP and RCP motor evaluations are bounding for the 1.4 percent uprate condition. Therefore, the STP, Units 1 and 2 RCPs and RCP motors are acceptable for the 1.4 percent uprate condition. The NRC staff concurs with the licensee's conclusions because the original analysis of records remains valid.

3.6.7 Steam Generators (SGs)

The licensee's structural evaluation focused on the critical SG components as determined by the stress ratios and fatigue usage. Comparisons of the primary-side transients and RCS parameters were performed to determine the scale factors that would be applied to the baseline analyses for the maximum stress range and fatigue usage factors. The baseline analysis results for various components were updated for the 1.4 percent uprate conditions.

For the primary-side components (particularly the divider plate, the tubesheet and shell junctions, the tube-to-tubesheet weld, and tubes), the applicable scale factors were the ratios of the primary-to-secondary-side differential pressure for the baseline and uprated conditions. For the secondary-side components, such as the feedwater nozzle and secondary man way studs, the decrease in secondary-side pressure was the basis for determining the applicable scale factor. The scale factor was then used for the lower bound stresses, which in turn conservatively increased the stress ranges involving transients that originate from, or lead to, full power. The increased stress ranges were addressed in the evaluation of the secondary-side components and factored into the calculation of fatigue usage.

The licensee concluded that the resultant primary stresses due to design, faulted, emergency, and test conditions were unchanged from the baseline analysis values. Since the primary-to-secondary △P that results from the original design parameters bounds that of the 1.4 percent uprate conditions, and the design transients remain unchanged, the previous △P evaluation remains valid and the 1600 psi design pressure differential is not exceeded. The licensee replaced the original SGs at STP Unit 1 with Westinghouse Model Δ94 replacement SGs in 2000. The tubes in the replacement SGs are more resistant to degradation than the tubes in the original SGs because the thermally treated alloy 690 material used in the replacement SG tubing has shown to be more resistant to degradation than the mill-annealed alloy 600 material used in the original SG tubing. In addition, the tube support configuration and installation of the tube in the tubesheet have been modified in the replacement SGs to mitigate potential degradation in those regions.

The existing SGs at STP Unit 2 are scheduled to be replaced with the Westinghouse Delta Δ94 SGs during the unit's winter 2002 refueling outage. The proposed power uprate will be implemented at STP Unit 2 after the SG replacement. The following evaluation is applicable to the Westinghouse Model Δ94 SGs in STP Unit 1 (not applicable to the existing SGs at STP Unit 2), and will be applicable to STP Unit 2 when its steam generators are replaced by Model Δ 94 steam generators in the winter of 2002.

The licensee reviewed the structural integrity of the following Model Δ94 SG components: The divider plate, the tubesheet and shell junctions, the tube-to-tubesheet weld, the tubes, the feedwater nozzle, and the secondary man way studs. The licensee reviewed the primary-side transients and RCS parameters used in the baseline safety analyses with respect to the changes under the power uprated conditions. On the basis of the power uprated conditions, the licensee applied scale factors to the baseline structural analyses. The original design parameters of the primary-to-secondary differential pressure bound that of the power uprated conditions. Therefore, the design transients remain unchanged, the original differential pressure evaluation remains valid, and the 1600 psi design pressure differential is not exceeded. The licensee concluded that under the power uprated conditions, the resultant primary stresses due to design, faulted, emergency, and test conditions were unchanged from the baseline analysis. The NRC staff finds that the licensee's assessment of the above components is acceptable, because the original design parameters bound the power uprated conditions and the result and stresses or components are unchanged.

With regard to the structural analysis of tubing, the licensee evaluated the minimum wall thickness of Model Δ94 SG tubing in accordance with RG 1.121. The evaluation is documented in the Westinghouse report, WCAP-15095, Revision 1. The licensee calculated the minimum acceptable wall thickness for several different tube locations and considered two different tube plugging levels, 0 percent and 10 percent. The licensee verified that the transient loading conditions analyzed in accordance with RG 1.121 bound the power uprated conditions. The bounding structural limit for the tube wall thickness is 0.015 inch, which is about 62.5 percent of the actual tube wall thickness. This minimum tube wall thickness is conservatively calculated assuming uniform thinning of a tube wall and represents an acceptable wall thickness that meets relevant ASME Code criteria using minimum ASME Code material properties. In addition to the structural considerations, RG 1.121 recommends that the tube repair limit (the wall thickness that requires tube repair) should consider measurement uncertainty and potential flaw growth. The licensee has implemented a tube repair limit of 40 percent through wall in the plant TSs. The NRC staff finds that the tube repair limit in the plant TSs will not be affected under the power uprated conditions because there is a sufficient margin between the tube repair limit of 40 percent through wall and the structural limit of 62.5 percent through wall, considering measurement uncertainty and potential flaw growth. The NRC staff finds that the licensee's evaluation of the minimum tube wall thickness under the power uprated conditions is acceptable because it follows RG 1.121.

With regard to the leakage integrity of the tubes, the licensee stated that the steam line break differential pressure is controlled by the pressurizer power-operated relief valve setpoint which does not change with power uprate. The licensee stated that the Electric Power Research Institute (EPRI) Report, "PWR Primary to Secondary Leak Guidelines-Revision 2," TR-104788-R2, will ensure leakage integrity of SG tubes. Application of the EPRI guidelines implies that there is a 94.4 percent probability that the burst pressure of a single indication leaking at 75 gallons per day will be greater than the steam line break pressure differential. At

the TSs leakage limit of 150 gallons per day, there is a 79.4 percent probability that the burst pressure of a single leaking indication will be greater than the steam line break pressure differential.

The licensee also evaluated the leakage integrity of the SG tubes using an outside diameter stress corrosion cracking (ODSCC) model. The ODSCC model will predict a longer 100 percent through wall flaw length for leakage than for primary water stress corrosion cracking. The licensee stated that in a recent pulled tube examination, the estimated 100 percent through wall flaw length based on laboratory leak test results provided an excellent match with the 100 percent through wall flaw length measured by destructive examination. Using the calculated 100 percent through wall ODSCC flaw length that provides a normal operating condition leak rate of 150 gallons per day, the nominal predicted burst pressure is above the steam line break pressure differential. If partial through wall flaw depths are included such that the overall flaw length is twice the 100 percent through wall flaw length, the predicted burst pressure using the lower tolerance limit flow stress is still greater than the steam line break pressure differential. The licensee concluded that the leakage integrity is maintained for the SG tubing. The NRC staff finds that the licensee's evaluation on leakage integrity under the power uprated conditions is acceptable, because the licensee used a conservative approach to analyze the tube leakage and the burst capability of the tube is maintained.

With regards to tube plugs, the licensee evaluated the tapered welded plugs used in the STP Unit 1 SGs under the power uprated conditions. The licensee found that the stress vs. allowable ratios are less than unity, indicating that all primary stress limits are satisfied for the plug shell wall between the top land and the plug end cap. The licensee also found that there is adequate friction to prevent dislodging of the plug and adequate leakage integrity for the limiting steady state and transient loadings. The plug shell satisfies the fatigue requirements of NB-3222.4 of the 1989 edition of the ASME Code. The licensee indicated that alloy 690 thermally treated welded plugs have high chromium content and high annealing temperature that would result in improved carbide precipitation on the grain boundaries and would preclude the occurrence of primary water stress corrosion cracking. The NRC staff finds that the licensee's evaluation of tube plugs under the power uprated conditions is acceptable because it follows the ASME Code.

With regard to potential tube degradation under the power uprated conditions, the licensee stated that the operating parameters that would affect tube degradation are temperature and pressure differential. The temperature and pressure differential at the uprated conditions are basically the same as the reference operating conditions. For the zero-plugging condition, the maximum steam temperature at the uprated condition is 1°F higher than the pre-uprate design condition. This slightly increased temperature has negligible effect on the corrosion potential of the thermally treated Alloy 690 tubing. There is no change in the pressure differential between the uprated conditions compared to the reference conditions. The licensee concluded that there is no significant increase in the corrosion degradation potential of the tubes. There is no change to the tube inspection intervals in the plant TSs under the uprated conditions because there is no significant degradation on the SG tubing. The NRC staff agrees with the licensee that the power uprated conditions will not affect tube degradation and the tube inspection intervals significantly.

The licensee stated that it will comply with the industry guidance as specified in the Nuclear Energy Institute (NEI) Document, NEI 97-06, with regard to condition monitoring and

operational assessments after each SG inspection. The licensee stated that because there is no significant change in corrosion potential and the inspection intervals in the TS is conservative, the power uprated conditions will have insignificant impact on the conditional monitoring and operational assessments. The NRC staff has not approved NEI 97-06; however, the NRC staff agrees with the licensee that the power uprated conditions will have insignificant impact on the future conditional monitoring and operational assessments.

On the basis of the NRC staff review of the licensee's submittal, the staff concludes that the licensee has followed RG 1.121 and the ASME Code to demonstrate that the structural and leakage integrity of the replacement SGs under the power uprated conditions will satisfy the General Design Criterion (GDC) 14 of Appendix A to 10 CFR Part 50. GDC-14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. However, the NRC staff's conclusion applies only to the Model Δ94 replacement SGs at STP Unit 1. The NRC staff's conclusion will be applicable to STP Unit 2 SGs when the existing SGs are replaced with Westinghouse Model Δ94 generators in the winter of 2002.

Thermal-Hydraulic Evaluation

The licensee performed a thermal-hydraulic evaluation of the replacement Model Δ94 SG to confirm the acceptability of the secondary side operating parameters at the 1.4 percent uprate conditions. The steam flow will increase with the power uprate with no change on the mixture flow in the tube bundle. The steam pressure is reduced but remains within the current design operating values. The heat flux through the SG tubes will increase, with only a minimal impact on the margin for the departure from nucleate boiling (DNB) transition based on small changes in the peak void fraction. The moisture carryover, based on the Model Δ94 test data, is predicted to be less than 0.005 percent, which is well below the 0.1 percent moisture carryover limit. Therefore, the secondary side operating characteristics are adequate for the 1.4 percent power uprate.

3.6.8 Pressurizer

The licensee indicated that any changes in T_{hot} and T_{cold} are very small, and are bounded by the existing pressurizer stress analysis. No changes were made to the design transients that are applicable to the pressurizer. Additionally, there are no changes to the pressurizer nozzle loads as a result of the 1.4 percent power uprating. Therefore, the current design transients are still applicable.

The licensee concluded that the pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III (Rules for Construction of Nuclear Vessels), 1974 Edition, for plant operation at the 1.4 percent power uprate conditions. The NRC staff concurs with the licensee's conclusions because the original analysis of records is still valid for the 1.4 percent power uprate conditions.

3.6.9 NSSS Auxiliary Equipment

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks. The licensee performed an evaluation to determine the impact that the revised design conditions will have on these equipment. Only the safety injection accumulators and boron injection tanks

have transients associated with them. None of the transients associated with these tanks are impacted by the 1.4 percent power uprate; therefore, these tanks are not affected by the 1.4 percent power uprate. Additionally, the 1.4 percent power updating has no affect on the pressurize relief tank or the volume control tank.

The licensee has evaluated the revised design conditions with respect to the impact on the auxiliary heat exchangers, valves, pumps, and tanks. The licensee concluded that the auxiliary equipment continues to meet the design P-T requirements, as well as the fatigue usage factors and allowable limits, which the equipment is designed for. The NRC staff concurs with the licensee's conclusion because the original analysis of record still apply.

3.6.10 Fuel Evaluation

The current loadings utilize robust fuel assemblies (RFA) and V5H 14 ft fuel assemblies. These assemblies are hydraulically compatible, therefore, no mixed-fuel core penalty is assessed. From the values of $F_{\Delta H}$ and DNBR it is clear that there is sufficient thermal margin for the proposed power uprate. The NRC staff finds that the Nuclear Design for the proposed uprate is acceptable.

Fuel Rod Design

In 1998, South Texas started fuel design transition from V5H to RFA using 10 CFR 50.59 evaluation. Because of the possible different type of the fuel design, the NRC staff requested a review of the licensee's 50.59 evaluation. The licensee's 50.59 evaluation was based on the approved Westinghouse's Fuel Criteria Evaluation Process (FCEP). This process allows a licensee to make minor fuel design changes without the NRC formal review. The NRC staff reviewed the licensee's 50.59 evaluation and determined that the V5H and RFA fuel designs were very similar albeit different names. In addition, the NRC staff determined that the licensee's analyses were consistent with the FCEP, and satisfied with all the requirements set forth in the FCEP. Thus the NRC staff concludes that the licensee's 50.59 evaluation of fuel transition from V5H to RFA is acceptable.

The licensee analyzed the future core of power uprate for the standard, V5H, and RFA fuel types. The analyses showed that all fuel designs would be meet licensing design limits with adequate conservatism. The NRC staff reviewed the analyses and agreed with the licensee's assessment. Based on the licensee's reload core analysis, the NRC staff concludes that the fuel designs are acceptable for the power uprated core of STP, Units 1 and 2.

Core Thermal Hydraulic Design

The core thermal-hydraulic analyses and evaluations were performed at the uprated core power level of 3,853 MWt. The analyses assumed that the uprated core designs are composed of the 17x17 RFA.

For the RFA fuel design, the DNBR analysis was performed using the WRB-2M CHF correlation and the VIPER-01 subchannel code. The NRC staff has approved the WRB-2M correlation for the modified V5H fuel assemblies with a DNBR limit of 1.14 when used with the subchannel analysis codes THINC-IV or VIPER-01 as described in WCAP-15025-P-A (Ref. 9) and WCAP-14565-P-A (Ref. 10). WCAP-15025-P-A describes the WRB-2M CHF correlation test data base

and derivation of correlation DNBR limit. The licensee in its response to a NRC staff question stated that the RFA design is similar to the fuel design with the modified V5H mixing vane grids described in WCAP-15025-P-A. Therefore, the use of WRB-2M/VIPER-01 for the RFA DNBR calculation is acceptable. Other fuel designs such as V5H and STD fuel that may be reloaded in the power uprate core, will continue to be analyzed with the WRB-1 CHF correlation with corresponding design DNBR limit. However, the presently loaded STD or V5H fuel assemblies will not be limiting due to the significant amount of burnup of the reload assemblies.

In response to a NRC staff question regarding the effects of a mixed core with both the RFA and V5H fuel on the DNBR analysis, the licensee stated that the RFA fuel design without the intermediate flow mixers is hydraulically fully compatible with the STD and V5H fuel, and, therefore, there is no mixed core penalty. The NRC staff concludes that the small amount of uncertainty, if any, would be compensated by the large thermal margin preserved by the use of higher safety analysis DNBR limit, as discussed below.

The non-LOCA safety analysis are performed using the RTDP methodology. New RTDP design DNBR limits for the RFA fuel assemblies are determined to account for the reduction in the reactor power measurement uncertainty, as described in Section 4.0 of this report, and the use of the WRB-2M CHF correlation for the RFA fuel assemblies. In its response to a NRC staff question, the licensee provided the new RTDP design DNBR limits with the WRB-2M correlation, i.e., 1.24 and 1.23 for the typical cells and thimble cells, respectively. Since the licensee did not provide the actual derivation of the design DNBR limits, the NRC staff performed an audit calculation to confirm the DNBR design limits. The audit calculation used the methodology described in WCAP-11397, the WRB-2M correlation data base statistics described WCAP-15025-P-A, the reactor system parameter uncertainty values provided in WCAP-13441, the revised power measurement uncertainties of 0.6 and 1.0 percent, the enthalpy rise factor uncertainty of 4 percent, and the sensitivity factors provided in the licensee's response to an NRC staff request for additional information. Based on its calculation, the staff confirmed that the RTDP design DNBR limits of 1.24 and 1.23 for typical and thimble cells, respectively, are acceptable. In addition, the licensee has determined to use the RTDP safety analysis DNBR limit of 1.52 for both typical and thimble cells. This safety analysis DNBR limit provides a large thermal margin, which can be used to compensate for the rod bow penalty and other uncertainties, such as the effects of mixed core.

In response to a NRC staff question, the licensee also provided a revised core thermal limits based on the DNBR safety analysis limit of 1.52. The licensee has evaluated the over temperature ΔT (OTΔT) and overpower ΔT (OPΔT) reactor trip function using the approved method described in WCAP-8745-P-A (Ref. 11) and the revised core thermal limits. Based on this evaluation, the licensee confirmed that the existing OTΔT and OPΔT set points remain valid. Since the OTΔT and OPΔT set points are evaluated with the NRC-approved methodology and the core thermal limits based on an acceptable DNBR limit, the NRC staff concludes they are acceptable.

Fuel Structural Evaluation

The fuel structure is designed to remain intact during seismic and LOCA loading conditions. The licensee analyzed the fuel structural response for power uprate and concluded that the power uprate had insignificant impact on the fuel assembly functional requirements. The NRC staff reviewed the analysis and agrees with the licensee's assessment. Based on the

conservatism of this assessment, the NRC staff concludes that the fuel structure designs are acceptable for the power uprated core of STP, Units 1 and 2.

3.7.0 NSSS Accident Evaluation

3.7.1 LOCA & LOCA Related Evaluations

Large Break Loss-of-Coolant Accident (LBLOCA) and Small Break Loss-of-Coolant Accident (SBLOCA)

The licensee performed the current STP LBLOCA and SBLOCA analyses assuming 102 percent (3,876 MWt) of the licensed power (3,800 MWt) for the plants. The licensee proposed to continue to use the present LBLOCA and SBLOCA analyses at 3,853 MWt as the licensing basis LOCA analyses for the STP plants assuming 101.4 percent of the proposed licensed power (2,689 MWt). The licensee's proposal is based on its use of the Westinghouse/CE CROSSFLOW UFM system technology to reduce the STP power measurement uncertainty to 0.6 percent, from the previously assumed 2.0 percent. The remaining 1.4 percent is available for increasing the licensed power without changing the LOCA analysis initial power assumption. This is now permitted by 10 CFR Part 50, Appendix K, Section I.A, in its most recent revision dated June 1, 2000.

The licensee showed that the LBLOCA and SBLOCA analysis methodologies presently approved for STP continue to apply specifically to the STP plant(s) in a letter dated February 5, 2002, as clarified in a letter dated February 13, 2002, by providing a statement that STP and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as operated plant values for those parameters.

Based on the consistency of the licensee's proposed uprated LOCA analysis input initial power assumption with 10 CFR Part 50, Appendix K, Section I.A, and the licensee's demonstration that the LBLOCA and SBLOCA analysis methodologies presently approved for STP continue to apply specifically to the STP plant(s), the NRC staff finds that the LOCA analysis methodologies presently approved for STP continue to apply to the STP plants and are suitable for inclusion in plant licensing documentation, including TSs and core operating limits reports.

Post-LOCA Long Term Core Cooling

When a sufficient amount of water has been injected by the ECCS to satisfy various design requirements, including providing enough water in the ECCS sump to meet ECCS pump net positive suction head requirements, but prior to depletion of the Refueling Water Storage Tank (RWST) inventory, the ECCS suction is switched from the RWST to the ECCS containment sump. The increased residual heat due to the higher (uprated) power could impact the timeliness of actions necessary to effect the switch over the injection source. In a letter dated February 25, 1995, the licensee stated that the ECCS flow to the reactor core is not significantly reduced during this switch over. The NRC staff also noted that in the NRC SER for STP (NUREG-0178, April 1986), the only operator action is to secure certain valve positions to assure continued flow to the core. Therefore, the core cooling demonstrated by the LBLOCA and SBLOCA analyses is maintained through the switch over. The NRC staff, therefore, finds this acceptable.

Hot Leg Switch over

The licensee stated that it was reallocating the existing 2 percent uncertainty margin in its hot leg switch over analysis to 1.4 percent for the increase in licensed power level and 0.6 percent for power measurement uncertainty, consistent with the recent change in Section I.A of 10 CFR Part 50 Appendix K. The existing licensed power level is 3,800 MWt and the new licensed power level is 3,853 MWt. The assumed power level for the analysis is 3,876 MW, consistent with (3,800)(1.02) = 3,876 MWt and (3,853)(1.006) = 3,876 MWt.

The existing analyses of record are acceptable for the requested thermal power level because the assumed power used in the licensee's previously approved calculations is not affected by the relocation of the 2 percent uncertainty margin between the actual power and uncertainty in the power.

3.7.2 Non-LOCA Analysis

Section 8.3 of Attachment 6 addresses the impact of 1.4 percent power uprate on the non-LOCA analyses.

The STP analyses of the non-LOCA design-basis events were performed using (1) a deterministic method for events such as a steam line break, in which the power measurement uncertainty of 2 percent was explicitly assumed for the core power, and (2) the RTDP methodology for other non-LOCA events, in which the nominal core power are used in the analysis with the power measurement uncertainty statistically included in the design DNBR limit. For those events previously analyzed with the deterministic method with an initial power level of 102 percent, which explicitly accounted for a measurement uncertainty of 2 percent, no re-analysis is required because the 1.4 percent power uprate is covered by the reduction in the power measurement uncertainty from 2 percent to 0.6 percent. For those events that were analyzed with the RTDP methods, an evaluation is necessary because of the increase in the nominal power level of 1.4 percent and the reduction of power measurement uncertainty. Of the Chapter 15 non-LOCA design-basis events, four events are re-analyzed, with the remaining events determined to be bounded by the existing analyses.

For those non-LOCA events that need to be re-analyzed to support the 1.4 percent power uprate, the initial conditions are based on the design operating parameters given in Table 2.1-1 of Attachment 6 to the power uprate application. Table 2.1-1 provides the thermal design parameters for four cases. These four cases are described in licensee response Cases 1 and 2 vary from 0 percent and 10 percent SGTP, respectively, while maintaining a T_{avg} of 592.6 ^{o}F, and Cases 3 and 4 vary from 0 percent to 10 percent SGTP while maintaining T_{avg} of 582.7 ^{o}F. As the RTDP methodology with the safety analysis DNBR limit of 1.52 for the RFA fuel is used for the DNBR analysis of these non-LOCA events, the nominal values of reactor power, flow, temperature and pressure are used.

As described in this report, the current OTΔT and OPΔT set points remain unchanged based on the revised set of core thermal limits to accommodate the increased core power.

Non-LOCA Events Evaluated

Excessive Load Increase

An excessive load increase incident is caused when a rapid increase in steam flow causes a power mismatch between the reactor core power and the SG load demand. The RCS accommodates a 10 percent step load increase or a 5 percent per minute ramp load increase between 15 and 100 percent power.

The licensee evaluated the load increase at beginning-of-cycle and end-of-cycle conditions, with and without rod control. From this evaluation, they determined that the plant conditions reached during the transient at the uprated power will not meet those required to exceed their core thermal limits. Since the core thermal limits are not challenged, the licensee states that the minimum DNBR values for the accident will remain above their limit values.

Because the DNBR values for this accident remain below their limit values, the transient will continue to meet the applicable acceptance criteria for the uprated conditions. Therefore, the NRC staff finds this transient acceptable for the proposed power uprate.

Loss of Non-Emergency Alternating Current (AC) Power and Loss of Normal Feedwater

A loss of normal feedwater reduces the capability of the secondary system to remove the heat generated in the reactor core. The most limiting loss of normal feedwater event is caused by a loss of offsite power (LOOP), which is also called a loss of non-emergency AC power transient. The current loss of normal feedwater analysis model uses a 2 percent power uncertainty, which bounds the uprated conditions. Since the current analysis bounds the power uprate, the NRC staff finds it acceptable.

The loss of non-emergency AC power transient results in a loss of all power to the plant auxiliaries, including the RCPs, condensate pumps, etc. On the loss of power, core cooling and removal of residual heat is accomplished by natural circulation in the reactor coolant loops, aided by auxiliary feedwater on the secondary side. This analysis also models a 2 percent power uncertainty, which bounds the uprated conditions. Since the current analysis bounds the power uprate, the NRC staff finds it acceptable.

A variation of the loss of AC power/loss of normal feedwater analysis is performed in support of the auxiliary feedwater system reliability evaluation. This analysis is based directly on the loss of AC power and loss of normal feedwater analyses, which model a 2 percent power uncertainty. Since the current analysis bounds the uprated conditions, the NRC staff finds its use acceptable.

Feedwater (FW) System Pipe Break

FW system pipe break accidents are breaks in FW lines large enough to prevent the addition of water to the SGs to maintain shell-side fluid inventory. However, if the break is postulated in a FW line between the check valve and the SG, the fluid from the SG will discharge through the break. Depending on the size of the break and the plant operating conditions at the time of the break, it could cause either an RCS cooldown (by excessive discharge through the break) or an RCS heat up (because of loss of flow).

The licensee's analysis for the FW system pipe break models a 2 percent power uncertainty. This 2 percent uncertainty level bounds the 1.4 percent power uprate because of the reduced uncertainty achieved through the implementation of the Westinghouse/CE UFM (0.6 percent uncertainty). Since this accident is still bounding for the uprated conditions, the NRC staff finds this analysis acceptable to support the power uprate.

Partial and Complete Loss of Forced Reactor Coolant Flow

The partial or complete loss of forced reactor coolant flow events may result from either a mechanical or electrical failure for the RCPs. If the reactor is at power during the accident, the loss of coolant flow will cause a rapid increase in the coolant temperature. This increase will result in a decrease in the DNBR.

The licensee performed an evaluation of accident for the uprated conditions and found that the existing state points for the limiting complete loss of flow event remain valid, except for the nominal core heat flux. The nominal core heat flux increases because of the 1.4 percent uprate. Even with the increase to core heat flux, the licensee's analysis showed that the DNB design basis continues to be met. Because the design basis continues to be met, the NRC finds the licensee's assessment to be acceptable.

Single RCP Locked Rotor and RCP Shaft Break

A RCP locked rotor accident is an instantaneous seizure of a RCP rotor. The flow through the affected reactor coolant loop rapidly reduces, and the reactor trips on a low reactor coolant flow signal. The RCP shaft break accident is similar to the locked rotor accident, except the impeller remains free to spin. The RCP locked rotor accident bounds the RCP shaft break accident.

For the uprated conditions, the licensee's analysis shows that the assumptions in their original safety analysis continue to be met. Specifically, the number of rods in DNB remains below 10 percent. Also, the RCS pressure criteria continues to be met since the current model includes a 2 percent power uncertainty, which bounds the power uprate. Because the original assumptions of the safety analysis remain applicable, the NRC staff finds the licensee's assessment to be acceptable.

Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical Condition

An uncontrolled RCCA Bank Withdrawal accident can be caused by a malfunction of the reactor control or rod control systems. This withdrawal will cause an uncontrolled reactivity addition to the reactor core, which results in a power excursion. The accident is terminated by the power range neutron flux - low set point reactor trip. For the current analysis, this trip is assumed to take place at 35 percent power (1,330 MWt). However, for the uprate, the trip will take place at 35 percent of the uprated power, which is about 1,349 MWt. Because the power level increases rapidly in the accident, this difference in power levels will be reached on the order of milliseconds during the accident. This small change in trip timing has a negligible effect on the results of the current Final Safety Analysis Report (FSAR) accident analysis.

However, the nominal core heat flux increases because of the 1.4 percent uprate. The licensee analyzed the effects of the increased nominal core heat flux and the flow asymmetry penalty

and found that the DNBR design basis continues to be met for the uprated conditions. Since the DNBR design basis continues to be met for this accident, the NRC staff finds the licensee's assessment to be acceptable.

RCCA Misalignment

The RCCA misalignment accidents include the cases of one or more dropped RCCAs within the same group, a dropped RCCA bank, a statically misaligned RCCA, and withdrawal of a single RCCA.

The single or multiple dropped RCCAs within the same group transients result in a negative reactivity insertion. However, the core is not adversely affected since the power is decreasing rapidly. With the plant in manual rod control following the dropped RCCAs, the plant establishes a new equilibrium condition without control system interaction. In automatic rod control mode, on the other hand, the rod control system will detect the drop in power and initiate a control bank withdrawal to compensate. A power overshoot may occur for this case, on which, the control system will insert the control bank in order to restore nominal power. Because of the power overshoot, the automatic rod control mode is most limiting for this transient.

Similar to the dropped RCCA accidents, during a dropped RCCA bank transient, the core is not adversely affected because the power decreases rapidly. The accident continues to progress like the dropped RCCA accidents, but the return to power effects will be less because of the greater reactivity worth of the entire dropped bank.

The licensee evaluated the adequacy of the DNB design basis for the dropped RCCA transients using the approved methodologies of WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event." They found that the approved methodology uses generic state-points in its analysis and that these state-points were still applicable for use at the 1.4 percent uprated conditions. The licensee evaluated the DNB design basis with respect to these generic state-points and the increased nominal heat flux associated with the uprate and confirmed that the DNB design basis continues to be met. Since the DNB design basis continues to be met for these transients, the NRC staff finds the analyses acceptable for the uprated conditions.

For the statically misaligned RCCA analysis, the most severe misalignment situations with respect to DNBR occur when one RCCA is fully inserted or when bank D is fully inserted with one RCCA fully withdrawn. The results of the full power insertion limit calculations for control bank D vary from cycle to cycle, depending on fuel arrangement and other limiting criteria. These insertion limits allow for bank D to be fully inserted with any one assembly fully withdrawn without the DNBR falling below the limit value. Since the DNBR limit values continue to be met, the NRC staff finds this transient acceptable.

The analysis of the single RCCA withdrawal transient, on the other hand, shows that the fuel rods do reach DNB conditions. The current Updated Final Safety Analysis Report (UFSAR) Section 15.4.3.2.3 analysis uses cycle-specific power distributions to determine the number of rods in a DNB condition. To meet the current UFSAR limits, the transient must have fewer than 5 percent of the rods in DNB at uprated conditions. The licensee states that for the power uprate with the current fuel cycle, less than 0.5 percent of the rods will be in DNB at Unit 1, and less than 1 percent of the rods will be in DNB at Unit 2. Since these results are sensitive to the

cycle-specific power distributions, the licensee states that they will continue to ensure that the number of rods in DNB remain acceptable in future cycles by analyzing this transient using the approved Westinghouse reload safety evaluation methodology. Since the UFSAR design basis for this transient continues to be met, and since the licensee will continue to perform the calculations per the approved reload methodology, the NRC staff finds this transient acceptable at the uprated plant conditions.

Startup of Inactive Loop at Incorrect Temperature

The startup of an inactive loop at the incorrect temperature accident occurs when one reactor coolant pump is out of service. The hot leg temperature of the inactive loop is lower than the reactor core inlet temperature. This startup results in the injection of cold water into the core, which causes a reactivity insertion and subsequent power increase.

This event is analyzed at a power level of 2,748 MWt, which includes a 2 percent model of power uncertainty. The assumed uncertainty bounds the uprated uncertainty. Because the analysis of record remains bounding, the NRC staff finds the analysis to be acceptable for the uprated conditions.

CVCS Malfunction

The licensee evaluated the CVCS malfunction (boron dilution) event for the uprated power conditions over the spectrum of plant operations, from power operation to refueling. However, the power uprate only affects the results of the analysis for power operation. The analyses for all other plant conditions remain unchanged.

Currently, during power operation, with the reactor in automatic control, the power and temperature increase from a boron dilution event would result in insertion of the RCCAs and a decrease in the shutdown margin. The rod insertion limit alarms (low and low-low settings) provide the operator with adequate time (on the order of 56 minutes) to determine the cause of dilution, isolate the unborated water source, and initiate reboration before the total shutdown margin is lost due to dilution. With the reactor in manual control, and if no operator action is taken, the power and temperature rise will cause the reactor to reach the over temperature delta-T trip set point. Prior to the OTDT trip, an over temperature delta-T alarm will actuate. After the reactor trip, there is on the order of 35 minutes available for the operator to determine the cause of dilution, isolate the unborated water source, and initiate reboration before the reactor can return to criticality.

For the uprated conditions, the licensee evaluated the CVCS malfunction and determined that the 1.4 percent power increase has an insignificant impact on the automatic reactor trip times assumed in the FSAR analysis. Since the trip times remain valid, the operator action times shown in the current analysis remain applicable. These times far exceed the SRP required operator action times of 30 minutes to terminate dilution during refueling and 15 minutes at all other plant conditions. Since the SRP acceptance criteria continue to be met for this accident at the uprated conditions, the NRC staff finds the licensee’s assessment to be acceptable.

Rupture of CRDM Housing

The consequences of a CRDM pressure housing rupture results in the ejection of a RCCA and drive shaft to their fully withdrawn position. The consequences of this failure include a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The CRDM housing rupture and RCCA ejection event is analyzed at the beginning and end of the core life for both hot full power (HFP) and hot zero power (HZP) conditions, in order to bound the fuel cycle and expected operating conditions. The HFP analysis was performed at 102 percent of the currently licensed power level. The 102 percent power level bounds the proposed uprated level. The HZP analysis, on the other hand, is performed at 0 percent power. It remains unaffected by the change in licensed power levels.

The licensee also considered the effects of the power increase on the reactor trip time in the accident. The trip set points modeled in the analyses were based on 35 percent for the HZP and 118 percent for the HFP cases, which would equate to power levels of 1,330 MWt and 4,484 MWt. For the uprated power, the 35 percent and 118 percent reactor trip set points would be close to 1,348 MWt and 4,546 MWt, respectively. Since the CRDM housing rupture accident is a fast transient, these small differences in power level equate to milliseconds of difference in the reactor trip times. Since the timing change is small, and the initial heat flux used in the analysis is based on 102 percent power, the effects of the uprate on this accident is also small.

The licensee's evaluation also shows that the conclusions in their UFSAR safety analysis continue to be met. Specifically, the fuel pellet enthalpies remain below 225 cal/gm for unirradiated fuel, the enthalpies remain below 200 cal/gm for the irradiated fuel, and the fuel melt in the hot area remains below 10 percent. Since the UFSAR safety analysis remains valid, the NRC staff finds its use for the uprated power to be acceptable.

Inadvertent Operation of ECCS and CVCS Malfunction that Increases RCS Inventory

The shutoff head for the high-head safety injection pumps is about 1,600 psi and the low-head safety injection pump shutoff head is 300 psi. At power, the RCS is pressurized to 2,250 psi, which means that the pumps provide no flow to the RCS on inadvertent operation in this condition. Therefore, a CVCS malfunction that increases the RCS inventory bounds the inadvertent ECCS operation. A CVCS malfunction resulting in the inadvertent injection of borated water could lead to filling the pressurizer to a water-solid condition. The most limiting malfunction would result if charging were in automatic control and the pressurize level channel being used for charging control failed in a low direction. This failure would cause the maximum charging flow to be delivered to the RCS and letdown flow to be isolated. The analysis of this failure is performed to ensure that the RCS pressure boundary is not breached, and the fuel design limits are not exceeded.

Currently, a power level of 3,853 MWt (2 percent power uncertainty) is assumed for the licensing-basis CVCS malfunction analysis. Since this power level is equivalent to the uprated assumptions (3,853 MWt with a 0.6 percent uncertainty), the analysis of record remains bounding for the uprated conditions. Since the analysis of record remains bounding, the NRC staff finds it acceptable for use at the uprated conditions.

Accidental Depressurization of RCS

The accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. A safety valve is sized to relieve approximately twice the steam flow rate of a relief valve. Therefore, the inadvertent opening of a safety valve will cause a much more rapid depressurization of the RCS. The faster depressurization, in turn, will cause a lower DNBR for the accident. For the power uprate, the licensee evaluated the event and determined that the minimum DNBR for this accident remains above the applicable limit value and the conclusions in the UFSAR remain valid. Because the applicable requirements are met for this accident, the NRC staff finds that the accident analysis remains valid for the uprated conditions.

Anticipated Transients Without Scram (ATWS)

The licensee analyzed the ATWS event for the requested power uprate as required by 10 CFR 50.62, " Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants" (known as the ATWS Rule). The requirements of 10 CFR 50.62 apply to all commercial light-water-cooled nuclear power plants. For Westinghouse designed plants, the implementation of ATWS mitigation system action circuitry (AMSAC) is a requirement of the Final ATWS rule, 10 CFR 50.62(b). STP, Units 1 and 2 have installed AMSAC and, therefore, meet the requirements of 10 CFR 50.62(b). In addition, the NRC staff requested additional analyses be performed to confirm that the proposed power uprate will not result in a transient peak RCS vessel pressure above ASME stress level C limit of 3,200 psig.

The licensee analyzed two ATWS cases: (1) Loss of Load (LOL) without reactor trip and (2) Loss of Flow (LOF) without reactor trip, to demonstrate that there are more than adequate safety relief valve (SRV) capacity in the design of STP, Units 1 and 2 to compensate for any increasing in the peak RCS pressure following an ATWS due to the increased power level. The licensee did a simple sensitivity study to compare the STP units to the reference plant. The STP pressurize SRV capacity is 501,000 lbm/hr. Compared to the pressurizer SRV capacity of 420,000 lbm/hr assumed in the referenced plant used in the Westinghouse ATWS analysis, this is an increase in relief capacity of almost 20 percent, which would provide additional margin not included in the sensitivity study. The results of the licensee's analysis showed that LOL ATWS case is more limiting. The results of this analysis show that the peak RCS pressure during this ATWS case is only slightly above the code value of 3,200 psig. Since the amount of increased peak RCS pressure is relatively small, the NRC staff concluded that this will not significantly impact the safe operation of the plant. Therefore, the NRC staff finds the licensee's ATWS analysis acceptable to support its proposed power uprate (Ref.12).

Non-LOCA Events Analyzed

During the review of the transient and accident analyses postulated for STP, the licensee identified four events which required reanalysis as part of their proposed power uprate. These events are sensitive to the effects of higher initial power on the ability to comply with appropriate safety limits. The NRC staff reviewed the results of the analyses submitted by the licensee to verify that safety margin was retained and regulatory requirements were met. The NRC staff used NUREG-800, "Standard Review Plan (SRP)," Rev. 3, to determine the

acceptance criteria for each event. The licensee identified several events to which reanalyses are required for the STP power uprate include: (1) Excessive Heat Removal Due to Feedwater System Malfunction, (2) Spectrum of SG System Piping Failures Inside and Outside Containment, (3) Turbine Trip, and (4) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power.

Excessive Heat Removal Due to Feedwater System Malfunction

The licensee analyzed this event at power uprated conditions using methods that the NRC staff has previously approved. The acceptance criteria for this event are as follows: (1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values, and (2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit specified in the TS safety limits.

The licensee provided the results of its analysis to demonstrate that the acceptance criteria for this event were met. The licensee assumed the failure of the feedwater control valves to the fully open position as an initiating event to ensure that the maximum amount of feedwater reached one SG. The licensee performed this analysis from the proposed uprated power condition to determine if the TS safety limits were satisfied. Since the power uprate has no effect on the HZP condition analysis of this event, the licensee did not perform the reanalysis for that condition. The results of analysis demonstrated that the minimum DNBR calculated remained above the safety limit to prevent damaging the fuel cladding. Since this is a cooldown transient, the peak pressures of the primary and secondary systems for this event will not exceed the initial pressures of these systems and over pressure criteria were met. The NRC staff have reviewed the assumptions and the results of the licensee's analysis and find that the assumptions used in this analysis are conservative and the results of the analyses meet all acceptance criteria for this event. Therefore, the NRC staff concludes that the licensee's analyses are acceptable to support the proposed power uprate.

Spectrum of SG System Piping Failures Inside and Outside Containment

The licensee analyzed this event at power uprated conditions using methods that the NRC staff has previously approved. The acceptance criteria for this event are as follows: (1) Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, and (2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit specified in the TS.

The licensee provided the results of its analysis to demonstrate that the acceptance criteria for this event were met. The licensee analyzed this event over a spectrum of break sizes to identify the limiting rupture size resulting in the highest power peaking. The licensee performed this analysis from the proposed uprated power condition to determine if the TS safety limits were satisfied. Since the power uprate has no effect on the HZP condition analysis of this event, the licensee did not amend the analysis for that condition. The results of the licensee's analysis demonstrated that the minimum DNBR calculated was above the safety limit to prevent damaging the fuel cladding. Since this is a cooldown transient, the peak pressures of the primary and secondary systems for this event will not exceed the initial pressures of these systems and overpressure criteria were met for this event. The NRC staff have reviewed the assumptions and the results of the licensee's analysis and find that the assumptions used in this analysis are conservative and the results of the analysis meet all acceptance criteria for this

event. Therefore, the NRC staff concluded that the licensee's analyses are acceptable to support the proposed power uprate.

Loss of External Electrical Load /Turbine Trip

The licensee analyzed this event at power uprated conditions using methods that the NRC staff has previously approved. The acceptance criteria for this event are as follows: (1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values, and (2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit specified by the TS safety limits.

The licensee provided the results of its analysis (for minimum DNBR case) to demonstrate that the acceptance criteria with respect to minimum DNBR for this event were met. The licensee analyzed this event considering cases with and without the availability of pressurize pressure control. To provide a degree of conservatism, the licensee does not credit the turbine trip signal for causing a reactor trip, but relies on the reactor protection signals. The licensee performed this analysis from the proposed uprated power condition to determine that the TS safety limits were satisfied. The licensee's analysis demonstrated that the minimum DNBR calculated was above the safety limit to prevent damaging the fuel cladding. The overpressure concern of this event has been previously analyzed with an explicit 2 percent power measurement uncertainty combined with the current power level, is equivalent to modeling the plant at the 1.4 percent uprated power level with the reduced uncertainty of 0.6 percent. The results of that analysis is documented in the current UFSAR which confirmed that the peak primary and secondary pressures for this event are within acceptable limits. Therefore, the over pressure criteria were met for this event. The NRC staff have reviewed the assumptions and the results of the licensee's analysis and find that the assumptions used in this analysis are conservative and the results of the analysis meet all acceptance criteria for this event. Therefore, the NRC staff concluded that the licensee's analysis are acceptable to support the proposed power uprate.

Uncontrolled RCCA Bank Withdrawal at Power

The licensee analyzed this event at power uprated conditions using methods that the NRC staff has previously approved. The acceptance criteria for this event are as follows: (1) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit specified by the TS safety limits, and (2) Fuel centerline temperatures (FCTs) do not exceed the melting point.

The licensee provided the results of the analysis to demonstrate that the acceptance criteria for this event were met. The licensee performed this analysis from the proposed uprated power condition to demonstrate that the TS safety limits were satisfied. The analysis showed that the minimum DNBR calculated was above the safety limit to prevent damaging the fuel cladding. Additionally, the results of the analysis showed that the peak core average heat flux is below its limit of 118 percent which confirms that there is no fuel melt in this event. To approach using peak core average heat flux as an indicator of potential fuel melt is consistent with the current licensing basis at STP. The licensee also showed that the reactor coolant and main steam systems over pressure criteria were met for this event. The NRC staff have reviewed the assumptions and the results of the licensee's analysis and find that the assumptions used in this analysis are conservative and the results of the analysis meet all acceptance criteria for this

event. Therefore, the NRC staff concluded that the licensee’s analysis are acceptable to support the proposed power uprate.

3.7.3 Steam Line Break Evaluation

The licensee evaluated the effects of the proposed power uprate on steam line breaks inside and outside containment for both short and long terms. These evaluations demonstrated that the existing analyses of record bound the results at uprated power levels.

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

The licensee performed the existing analysis of record for STP with a 2 percent uncertainty added to the NSSS power. The power uprate considers a 0.6 percent uncertainty combined with the 1.4 percent uprated power. The effect of combining the new uncertainty and uprated power level results in the same initial conditions for the analysis as those currently used. Therefore, the current analysis of record provides bounding results.

Short-Term Steam Line Break Mass and Energy Releases Outside Containment

The STP existing analysis of record assumed no load conditions which resulted in the highest SG pressures and inventories as well as critical mass flow rates. The power uprate will not affect the no-load thermodynamic conditions. Additionally, the current analysis for the main steam line break forcing functions uses sufficiently conservative initial conditions to bound the uprated power conditions. The licensee’s selection of these bounding values provides margin to ensure that the results of this event will be less severe than those predicted by the design calculations.

Short-Term SG Blowdown Line Break Mass and Energy Releases Outside Containment

The STP existing analysis of record assumed no load conditions which resulted in the highest SG pressures and inventories as well as critical mass flow rates. The power uprate will not affect the no-load thermodynamic conditions. The results of this event will be unaffected by the power uprate.

3.7.4 Post-LOCA Containment Hydrogen Generation

The licensee evaluated the hydrogen generation in containment following a LOCA at the proposed 1.4-percent increase in core thermal power. The analysis assumed operation of a single hydrogen recombiner at 90-percent recombination efficiency. Recombiner operation starts 24 hours into the LOCA. The licensee's analysis showed that 1.4-percent increase in core thermal power will have no impact on the peak calculated concentration of hydrogen remaining below 4.00 v/o limit specified in RG 1.7. The NRC staff concurs with licensee's conclusions.

3.7.5 LOCA Mass and Energy Releases

Long-Term LOCA Mass and Energy Releases Analysis

The licensee's analysis demonstrates the adequacy of the containment systems to mitigate the consequences of the postulated LBLOCA. The licensee states that the analysis of record was based on core thermal power of 3,876 MWt. This value reflects 2 percent increase due to power measurement uncertainty above the current licensed value of 3,800 MWt. The improved thermal power measurement accuracy reduces the power measurement uncertainty to 0.6 percent, or provides a core thermal power margin of 1.4 percent. The containment systems for mitigating consequences of postulated LBLOCA will be adequate for proposed increase in core thermal power. The NRC staff concurs with the licensee's conclusions.

Short-Term LOCA Mass Energy Releases

The NRC staff previously approved the licensee's Leak Before Break Methodology. The licensee states that the only break locations that need to be considered are pressurizer spray line and RHR line from the hot leg to the first isolation valve. The proposed power uprate may potentially change the initial reactor coolant temperature. Since the short term release due to a LOCA event lasts less than 3 seconds, the effect of the reactor power is not significant.

The licensee states that the critical flow correlation used in the mass and energy release calculations will yield an increase in mass and energy release at lower coolant temperature. For the proposed power uprate, the RCS cold leg temperature remains the same as in the analysis of record. However the hot leg temperature increases approximately 0.7° F. This will result in a lower mass and energy release from hot-leg breaks, and will result in lower compartment pressure. Therefore, the current licensing basis remains bounding for a short-term LOCA sub-compartment pressurization analysis. The NRC staff concurs with the licensee's conclusions.

3.7.6 SG Tube Rupture (SGTR)

The analysis of the SGTR event is used to demonstrate that there will be no overfill to the ruptured SG, and radiological consequences remains below the 10 CFR Part 100 limits. The licensee performed the SGTR analysis assuming an initial power level of 102 percent. This power level bounds the power uprate of 1.4 percent with a 0.6 percent uncertainty. Therefore, the current analysis for the SGTR event will bound the proposed power uprate of 1.4 percent.

3.8 Electrical Power

The offsite power system includes the grid, eight 345-kV transmission lines, transformers, and associated control systems and features provided to supply electric power to safety-related and other equipment. Power from the grid is supplied to the onsite electrical system through their respective unit auxiliary transformers and two plant standby transformers and 138 kV emergency transformer. The onsite standby power supply consists of three independent standby diesel generators for each unit. The ESF ac and dc power systems are designed with redundancy and independence of onsite power sources, distribution systems, and control in order to provide a reliable supply of electrical power to the ESF electrical loads necessary to

achieve safe plant shutdown, or to mitigate the consequences of postulated accidents. Steady-state and transient stability studies demonstrate that the loss of both units do not impair the ability of the system to supply power to the ESF electrical system. The electrical distribution system has been previously evaluated to conform to 10 CFR Part 50 Appendix A GDC 17.

3.8.1 Electrical Distribution System

The electrical distribution system connects the ESF buses to the UAT and standby auxiliary transformers. Additionally, 13.8 kV emergency transformer supply power to the ESF buses. There is no change to the safety-related loads at uprate conditions and, therefore, the ESF buses will not be affected by 1.4 percent power uprate and can perform their safety-related functions during a LOOP/LOCA.

The NRC staff concludes that since no new loads are added to the ESF buses, the electrical distribution system can be operated safely at the uprated power condition.

3.8.2 Turbine Generator

The licensee evaluated the capability of turbine generator to perform adequately at the proposed 1.4 percent increase in core thermal power. The licensee concluded that the throttles, high and low pressure turbines, the generators and exciters, and the supporting equipment have adequate margins to support operations at the proposed uprated power level.

The licensee concluded that the steam conditions will cause the low pressure turbine disc temperature to increase above the original temperature, and the turbine missile probability will increase slightly. The licensee states that the maintenance program will ensure that turbine missile generation requirement of the missile generation probability of 1E-04 continues to be met.

The licensee concluded that based on a revised heat balance, the proposed increase of core thermal power raises generator gross thermal output to 1,344 megawatts electric (MWe). The increased power is within the turbine generators rating of 1,405 MVa@0.9 power factor. The NRC staff concurs with the licensee's conclusions.

3.8.3 Main Generator

The main generator is rated at 1,505 MVA (1,354 MWe) at 0.9 power factor (pf). The 1.4 percent updating will result in a generator gross output power of 1,344 MWe for each unit. The capability of main generator, exciter as well as the associated equipment to perform at the 1.4 percent uprated condition (3,853 MWt) was evaluated. All generator components have sufficient margin to support operation at the 1.4 percent uprated power condition. No changes to the equipment protection relay settings for the generator are required for the 1.4 percent power uprate.

The NRC staff concludes that the 1.4 percent uprate is within the generator nameplate rating and; therefore, operating the main generator at the uprated power condition is acceptable.

3.8.4 Main Power Transformer

The size and features of the transformers in the step-up bank of STP Unit 1 (700/784 MVA and 650/728 MVA in parallel) and STP Unit 2 (two 700/784 MVA transformers in parallel), are adequate to deliver the electric power output supplied by the main generator set of STP, Units 1 and 2, after the 1.4 percent power uprate.

The NRC staff concludes that the main transformer banks for STP, Units 1 and 2 are sized to deliver the power output supplied by the main generator at the uprated power condition. Therefore, the main power transformers will operate within applicable limits at the power uprate condition.

3.8.5 Isolated Phase Bus

The isolated phase bus main section is rated at 25 kV, 36,600 amperes. The maximum current in the main generator terminals is 36,581 amperes which is less than the isolated phase bus rated current carrying capability.

The NRC staff concludes that the isolated phase buses will support the 1.4 percent updating and, therefore, the design is acceptable.

3.8.6 Impact on Balance of Plant (BOP)

The STP, Units 1 and 2 BOP systems were reviewed for impact due to the 1.4 percent power updating to 3,853 MWt reactor core power. New electrical loads were generated based on the uprated brake horsepower requirements for the pumps. The evaluation determined that the change in electrical loads is minor and is bounded by the design capacity of the distribution system.

The NRC staff concludes that the station BOP loads will remain within the existing design loads, the design is, therefore, acceptable for the power uprated condition.

3.8.7 Grid Stability

The steady state and transient stability study have been performed to demonstrate that the transmission system remains stable under the loss of both STP, Units 1 and 2, or the loss of external transmission circuits, and does not endanger the ability of the system to supply power to the engineered safety feature (ESF) electrical system. It was determined that there is sufficient margin to accommodate the 1.4 percent rating without impacting the grid stability.

The NRC staff concludes, that there is a minimal effect on the grid stability due to power uprate. Therefore, the NRC staff has reasonable assurance that safety functions will ensure that (1) fuel design limits and design conditions of the reactor pressure boundary are not exceeded as a result of the anticipated operational occurrences, and (2) the core and containment integrity and other vital functions are maintained in the event of a postulated accident, as required by GDC-17.

3.8.8 Emergency Diesel Generators

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 NRC staff concludes that the emergency diesel generators systems are not impacted by the 1.4 percent power uprate and, therefore, can operate safely at the power uprate condition.

3.9 Balance of Plant (BOP)

The licensee stated that the STP, Units 1 and 2 BOP systems that could potentially be impacted due to the 1.4 percent power uprating are the turbine and main generator, main steam and reheat steam, steam dump, SG blowdown, extraction steam, main feedwater, condensate, heater drips, drains, and vents, secondary sampling, auxiliary feedwater, condensate polishing, circulating water, open-and closed-loop auxiliary cooling, component cooling water, and essential cooling water. Each of these systems was evaluated for conformance with the design basis calculations considering thermal-hydraulic performance, equipment design adequacy, piping integrity, and instrumentation and control (I&C) system review. These systems were reviewed by the licensee for impact due to the 1.4 percent power uprating to 3,853 MWt reactor core power.

The licensee concluded that the existing BOP plant system components are adequate for the 1.4 percent power updating and continue to comply with all their original design requirements. The NRC staff agrees that the BOP systems were previously evaluated for power level above the proposed 3,853 MWt. Therefore, the proposed power uprate of 1.4 percent was bounded by previous analysis.

3.10 Radiological Consequences

The licensee stated in its submittal dated August 22, 2001, that all radiological consequence assessments for the design-basis accidents except the small line failure analysis have been performed at a reactor core thermal power level of 4,100 MWt, which is 107.9 percent of the current power rating of 3,800 MWt, bounding 1.4 percent power uprate. In its response dated January 21, 2002, to a NRC staff's RAI, the licensee restated that the small line failure analysis was also performed at 4,100 MWt. Therefore, the licensee stated that the current radiological consequence assessments in the STP, Units Nos 1 and 2 UFSAR, Revision 7, remain valid for increased core thermal power operation at 4,100 MWt. The licensee also submitted as Attachment 2 to its letter dated January 21, 2002, updated UFSAR Table 15.6-2, "Parameters Used in Sample Line Failure Radiological Analysis," and Table 15.6-3, "Parameters Used in SG Tube Rupture Analyses," stating that these tables will be included in the next UFSAR revision. These updated tables reflected that the radiological consequence analyses for the sample line break and SG tube rupture accidents were performed at a reactor power level of 4,100 MWt.

The NRC staff reviewed the radiological consequences analyzed for the design-basis accidents in Chapter 15 of the STP, Units 1 and 2 UFSAR along with the licensee's submittals. The NRC staff finds that the radiological consequences analyzed for the design-basis accidents at power level of 4,100 MWt bound the requested power level of 3,853 MWt, and that the radiological consequences calculated at a reactor core thermal power level of 4,100 MWt met the relevant dose acceptance criteria specified in 10 CFR Part 100 for the site boundaries and GDC 19 for

the control room operator. The licensee proposed no new parameters or methods for analyzing the radiological consequences for the postulated design-basis accidents.

The NRC staff concludes that the current design basis dose analyses, as documented in the STP, Units 1 and 2 UFSAR, Revision 7, remain acceptable in that reasonable assurance exists that the radiological consequences, with proposed 1.4 percent reactor core thermal power uprate, will remain the same or bounded by the current values. Therefore, the NRC staff further concludes that the proposed power uprate is acceptable with respect to the radiological consequences resulting from the postulated design-basis accidents.

3.11 Human Factors

The NRC staff reviewed the following operator performance topics discussed in the licensee's application (Ref. 1).

3.11.1 Changes in Emergency and Abnormal Operating Procedures (EOPs and AOPs)

The licensee stated that the power uprate has no significant effect on the type and scope of the EOPs and AOPs. Procedure limitations on power operations due to BOP equipment unavailability will be revised as necessary to account for the increase in NSSS power to 3,853 MWt. Changes associated with power uprate will be treated in a manner consistent with any other plant modification. Procedures required for the operation and maintenance of CROSSFLOW system have been implemented. Specific actions to be taken when the CROSSFLOW system is inoperable will be addressed in the procedural guidance.

The NRC staff finds that the licensee's response is satisfactory because the procedures have been revised to incorporate the CROSSFLOW system prior to implementation of the power uprate. The licensee will treat plant procedure changes due to power uprate in a manner consistent with any other procedure changes.

3.11.2 Changes to Risk-Important Operator Actions Sensitive to Power Uprate

The licensee stated that ESF system design and set points, and procedural requirements already bound the proposed power uprating. The responses of the reactor operators to any event will be essentially unaffected by a change in rated thermal power (RTP).

There will be minimal impact on alarms, and procedural requirements for 1.4-percent uprating. The CROSSFLOW system will have alarms in control rooms to alert operators of conditions that impair its availability and accuracy. No other alarm impacts are expected. It is anticipated that any existing alarms will be modified or deleted [as appropriate]. Alarms will be recalibrated as necessary to reflect small set point changes. However, no significant or fundamental set point changes are anticipated. Also, the operator response to existing alarms is anticipated to remain as before.

When the power uprating is put in place, the nuclear instrumentation system will be adjusted to indicate the new 100-percent RTP in accordance with TS requirements and plant administrative controls. Since the power uprate is predicated on the availability of the CROSSFLOW system, procedural guidance will be implemented to facilitate operation when the CROSSFLOW system

is unavailable. The reactor operators will be trained on the changes in a manner consistent with any other design modification.

The power uprating will be reflected in the plant simulator. These changes should be virtually transparent to the reactor operators.

The NRC staff finds that the licensee's response is satisfactory because the licensee has adequately addressed the question of operator actions sensitive to the power uprate, and shown that the responses of the reactor operators to any event will be essentially unaffected by 1.4 percent increase in the rated thermal power. The licensee will implement procedures and guidance as required for operator actions when CROSSFLOW is unavailable.

3.11.3 Changes to Control Room Controls, Displays, and Alarms

The licensee has addressed the change in controls, displays, and alarms, as discussed above in Section 13.11.2.

The NRC staff finds that the licensee's response is satisfactory because the licensee has adequately identified and described the changes that will occur to alarms, displays, and controls as a result of the power uprate.

3.11.4 Changes to the Safety Parameter Display System (SPDS)

The licensee stated that only process parameter scaling changes will be made, as required, to Qualified Display Parameter System (QDPS), and that there are no other impacts to the QDPS due to 1.4-percent uprate.

The NRC staff finds that the licensee's response is satisfactory because the licensee will identify and make the necessary scaling changes to the QDPS as a result of the power uprate.

3.11.5 Changes to the Operator Training Program and the Control Room Simulator

The licensee's response to this question is folded in response under Section 3.11.2.

The NRC staff finds the licensee's response satisfactory, because the licensee has adequately described how the changes to operator actions will be addressed by training and how the simulator will accommodate the changes.

3.11.6 Human Factors Evaluation Summary

The NRC staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The NRC staff further concludes that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability.

3.12 Other Evaluations

3.12.1 10 CFR Part 50, Appendix R

The licensee states that fire hazards/cold shutdown analyses were performed at 102 percent of the original core power level (3,876 MWt). Therefore, there is sufficient margin to bound the 1.4 percent uprated power (3,876 MWt). The allowance of 0.6 percent will be available to address any uncertainty. Since the RCS temperatures are approximately the same as those previously analyzes in the fire hazards analyses, and since decay heat is bounded, there is no adverse impact, and the fire hazards/cold shutdown analyses of record remain valid.

3.12.2 Electrical Equipment Qualification

As stated in the FSAR, equipment relied on to perform a necessary safety function can maintain function operability under all service conditions (including exposure to radiation) postulated to occur for the duration it is required to operate, during its installed life. The licensee evaluated the effect of the uprate on equipment qualification for the 1.4 percent power uprate. The equipment doses determined on the basis of post-accident area dose rates would increase by 1.4 percent, however, this increase is bounded by the analyses. Therefore, the proposed power uprating would have negligible impact on the doses determined.

The component-specific analyses were examined and it was determined that a 1.4 percent increases in the resultant doses would not change the conclusions of the analyses (e.g., the revised dose would remain below the applicable limit in the calculation). All calculations important to the equipment qualification design criteria are bounding. The peak values for accidents also include design margins. The temperatures, pressures and radiation levels established in the design criteria will remain the same. Therefore, analyses remain valid for the power uprate.

The NRC staff has concluded that operating at the uprated power condition is acceptable, because, in conformance with 10 CFR 50.49, the licensee's program for qualifying the electric equipment important to safety is adequate to ensure (a) the integrity of the reactor coolant boundary, (b) the capability to shutdown the reactor and maintain it in safe shutdown condition, and (c) the capability to prevent or mitigate the consequences of accidents.

3.12.3 Station Blackout (SBO)

SBO is defined in 10 CFR 50.2 as the complete loss of preferred off site and Class 1E onsite emergency ac power system. No changes are required for SBO coping and mitigation due to the 1.4 percent power uprate. The condensate inventory requirements are bounded by the current SBO analysis. All other associated parameters are either independent of the core power level, or have margin available to accommodate the 1.4 percent power uprate.

The NRC staff has concluded that the uprate does not adversely affect the ability of the plant to mitigate a postulated SBO event for the uprate condition.

3.12.4 Pipe Break Evaluation

The licensee indicated that it has reviewed its evaluations of the consequences of a pipe break or cracks for all fluid systems that experienced a change in energy (pressures, temperatures, or flow rates) due to the 1.4 percent power updating. These systems include the main steam system, the FW system, and the SG blowdown and cleanup system. In addition, the impact of the blowdown from the RCS through the piping attached to the RCS was reviewed to determine the impact of revised NSSS temperatures on these systems.

The licensee stated that the 1.4 percent (3,853 MWt) power uprate will have negligible impact on the pipe stress and, therefore, the power uprate does not result in any new or revised pipe break locations. The existing mass and energy releases due to a main steam line break and LOCA incorporate a 2 percent margin to account for the power calorimetric measurement uncertainty. Therefore, the current 102 percent power, or 3,876 MWt analyses continue to bound the 1.4 percent (3,853 MWt) power uprate case, and the associated 0.6 percent calorimetric uncertainty. The NRC staff concludes that the licensee's design basis analyis of record continues to bound the proposed 1.4 percent power increase, and there is no change in the consequences of a pipe break accident.

3.12.5 Flow-Accelerated Corrosion (FAC)

FAC is a corrosion mechanism causing wall thinning of high energy pipes in the power conversion system which may lead to their failure. Since failure of these pipes may result in undesirable challenges to the plant's safety systems, the licensee has a program for predicting, inspecting, and repairing or replacing of the components whose wall thinning exceeds the values required for their safe operation. The program uses the EPRI developed CHECWORKS computer code for predicting thinning of the walls in the components subjected to FAC. The licensee revised the CHECWORKS models to incorporate flow and process system conditions that are determined for the 1.4 percent power uprate. Although the revised model predicted negligible change in wear rates (<0.01 percent), the licensee will factor the upgraded model results into the future surveillance plans. The NRC staff considers this licensee's action adequate for ensuring integrity of the high energy pipes.

3.12.6 Safety-Related Motor Operated Valves (MOVs)

The licensee stated that it has reviewed the input parameters discussed in NRC GLs 89-10 and 96-05 regarding MOV thrust and torque requirements calculations and discussed in NRC Generic Letter 95-07 regarding MOV pressure locking and thermal binding requirement calculations. The licensee concluded that the 1.4 percent uprate will not require any changes to the parameters discussed in those GLs. The P-T calculations for various accident scenarios are not affected by the 1.4 percent power uprate since these calculations used conservative input that bound the input for the power uprate. Therefore, the licensee concluded that the 1.4 percent power uprate will not impact the MOV calculations discussed in the NRC GLs 89-10, 95-07, or 96-05. The NRC staff concurs with the licensee's conclusions.

3.12.7 Impact of Probabilistic Safety Assessment Results

The licensee evaluated the impact of the proposed power uprate on the licensee's PSA results. The licensee has performed the evaluation of the following PSA issues, and proposed that approriate changes will be made to reflect the power uprate.

Initiating Events

Since the power uprate is bounded by the uncertainty reduction, the licensee does not expect any significant impact on the frequency of initiating events.

System Success Criteria

The licensee will reevaluate the calculated system success criteria to ensure that the criteria used in the current PRA analyses continue to be valid at uprated power.

Operator Actions

The licensee's PRA includes credit for operator actions to recover from core damaging sequences of events. The licensee will reevaluate the event sequences to determine if uprated power has any influence on the operator actions assumptions in the PRA. However, since the current power uprate is bounded by the analyses of record, the licensee does not expect any changes in operator action assumptions.

Fission Product Inventory

The licensee's PRA includes a Level II (containment response) evaluation in its PRA. As a result of the proposed power uprate, the containment release categories will be slightly modified. Therefore, the Large Early Release Fractions (LERF) will be recalculated. The licensee does not expect a significant change in LERF, because the proposed 1.4 percent increase in power is small, and it is bounded by the uncertainty reduction.

The NRC staff concludes that the licensee has properly identified the impacts of the power uprate, and has plans in place to take the appropriate actions to upgrade its PRA methodology, if the evaluations indicate a need for changes. The NRC staff concurs with the licensee's conclusions.

3.12.8 Spent Fuel Pool (SFP) Cooling System

The SFP maintains the spent fuel and containment storage area at prescribed temperature limits by removing the decay heat generated in the pool. The licensee states that its calculation for water temperature and gamma heating of the concrete walls were performed with initial condition of operation at 102 percent power, 40,000 effective full-power hours of burn up (60,000 MWD/MTU), and a 105° F component cooling water temperature. The fuel pool cooling calculations bound the proposed 1.4 percent increase due to the reduction of flow measurement uncertainty. The licensee evaluated the spent fuel pool dose consequences due to the proposed power uprate of 1.4 percent. The licensee concluded that 1.4 percent power

uprate conditions with reduced flow uncertainty are bounded by the assumption in the analysis of record. The NRC staff agrees with the licensee's conclusions.

3.13 Technical Specification Changes

The TSs changes for STP, Units 1 and 2 due to power uprate are revised as follows:

TS 1.2.7 - The definition of RTP is revised to reflect the increase from 3,800 MWt to 3,853 MWt for the plants. Because this TS change reflects the actual proposed change in the plant and because it is supported by the results of the safety analysis, the NRC staff finds the change acceptable.

TS 2.1.1.1 - The licensee proposed to add "and > 1.14 for the WRB-2M DNB correlation" to the TS Safety Limits 2.1.1.1 so that it will read as follows:

"In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$1.17 for the WRB-1 DNB correlation and > 1.14 for the WRB-2M DNB correlation."

The proposed TS change is acceptable. The NRC staff has approved the WRB-2M DNB correlation with a DNBR limit of 1.14 for application to the modified Vantage 5H fuel as described in WCAP-15025-P-A and WCAP-14565-P-A. Since the Robust Fuel Assembly design is similar to the modified V5H fuel, the addition of "and >1.14 for the WRB-2M DNB correlation" for application to the RFA fuel is acceptable.

TS Table 3.7-1 - The maximum power levels permissible with inoperable main steam safety valves (MSSVs) for the Westinghouse Model Δ94 SGs are changed to be consistent with the uprated power. The bases are also changed to be consistent with the TS revision.

In TS Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Set points with Inoperable Steam Line Safety Valves During 4 Loop Operation," the licensee inserted new values for the maximum allowable power range neutron flux high setpoints with inoperable MSSVs for the Westinghouse Model) 94 SGs, to be consistent with the power uprate. For Units 1 and 2 (Westinghouse Model) 94 SGs, with 1, 2, and 3 MSSVs inoperable, the licensee inserted the maximum allowable setpoints of 61 percent, 43 percent, and 26 percent, respectively. For the original Model E SGs at the pre-uprated power level, the set points were 63 percent, 45 percent, and 27 percent for 1, 2, and 3 inoperable MSSVs, respectively. The licensee calculated these values for the Westinghouse Model Δ94 SGs using the method described in their TS Bases, which is a conservative heat balance calculation with an appropriate allowance for instrument and channel uncertainties. Therefore, the NRC staff finds the use of the conservative heat balance calculation described in the TS Bases section, with the appropriate allowance for instrument and channel uncertainties, for the determination of the maximum allowable power range neutron flux high setpoints with inoperable MSSVs to be acceptable. Furthermore, the NRC staff reviewed the licensee's proposed calculated values and found them acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component, located 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 66472 dated December 26, 2001). Accordingly, the amendment meets 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 amendment.

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 amendment 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 J.J. Sheppard, STP Nuclear Operating Company to US NRC, "Proposed Amendment to Facility Operating Licenses and TSs Associated with a 1.4 percent Core Power Uprate," dated August 22, 2001.

2. Letter from J.J. Sheppard, STP Nuclear Operating Company to US NRC, "Additional Information to Support the Request for Approval of Power Uprate and a Revision to the TSs," dated January 21, 2002.

3. Letter from J.J. Sheppard, STP Nuclear Operating Company to US NRC, "Additional Information to Support the Request for Approval of Power Uprate and a Revision to the TSs," dated February 5, 2002.

4. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

5. WCAP-13441, Revision 1, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for South Texas Units 1 and 2 Project," July 1999.

6. CENPD-397-P-A, Revision 01, “Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology,” May 2000.

7. WCAP-15633 (Proprietary), WCAP-15696 (Non-Proprietary), Revision “Power Colorimetric for the 1.4 Percent Updating for the STP Nuclear Operating Company Units 1 and 2,” June 2001.

8. WCAP-15697 (Proprietary), WCAP-15698 (Non-proprietary), CROSSFLOW Out of Service Power Calorimetric Uncertainties for STP Nuclear Operating Company Units 1 and 2,” July 2001.

9. WCAP-15025-P-A, “Modified W.B.-2 Correlation, W.B.-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,” April 1999.

10. WCAP-14565-P-A, “VIPER-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

11. WCAP-8745-P-A, “Design Bases for the Thermal Overpower ΔT and Thermal Over temperature ΔT Trip Functions,” September 1986.

12. Letter from J.J. Sheppard, Vice President, Engineering & Technical Services, to the U.S. Nuclear Regulatory Commission, “Additional Information to Support the Request for Power Uprate,” dated March 4, 2002.

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