Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

August 22, 2001
NOC-AE-01001162
File No.: G20.02.01
G21.02.01
10CFR50.90
STI: 31336548

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Facility Operating License Nos. NPF-76 and NPF-80
Proposed Amendment to Facility Operating Licenses and
Technical Specifications Associated with a 1.4% Core Power Uprate

Pursuant to 10CFR50.90, the STP Nuclear Operating Company (STPNOC) requests an amendment to the South Texas Project Facility Operating Licenses and the Units 1 and 2 Technical Specifications. This change applies to each unit after replacement of the steam generators with the Model Δ94 steam generators. The Unit 1 steam generators were replaced with the Model Δ94 steam generators in the Spring of 2000. The South Texas Project Unit 2 steam generators are planned for replacement with the Model Δ94 steam generators in late 2002.

The proposed amendment will revise the Facility Operating Licenses and Technical Specifications to reflect a 1.4% increase in the reactor core power level from 3,800 to 3,853 megawatts thermal (Mwt). This amendment is based on a reduced core thermal power level uncertainty associated with a more accurate measurement of feedwater flow. The increased accuracy of the feedwater flow measurement is provided by the CROSSFLOW Ultrasonic Flow Measurement (UFM) System. The CROSSFLOW UFM System is discussed in Topical Report CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology." The NRC issued a Safety Evaluation dated March 20, 2000, stating the staff's acceptance for referencing the topical report by licensees who want to use the CROSSFLOW UFM System for a power uprate.

A001

The South Texas Project Plant Operations Review Committee has reviewed the proposed amendment and recommended it for approval. The South Texas Project Nuclear Safety Review Board has reviewed and approved the proposed changes.

The South Texas Project requests approval of this amendment by the Nuclear Regulatory Commission by April 1, 2002 to support the planned implementation of the power uprate modification in Unit 1 prior to the summer 2002 peak period. Once approved, a period of 60 days is requested to implement the amendment.

In accordance with 10CFR50.91(b), STPNOC is providing the State of Texas with a copy of this proposed amendment.

Attached are Westinghouse Proprietary Class 2 WCAP-15633, Revision 0, and Non-Proprietary Class 3, WCAP-15696, Revision 0, "Power Calorimetric for the 1.4% Uprating for the South Texas Project Nuclear Operating Company Units 1 and 2" and Westinghouse letter, CAW-01-1466, accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

Also attached are Westinghouse Proprietary Class 2 WCAP-15697, Revision 0, and Non-Proprietary Class 3, WCAP-15698, Revision 0, "CROSSFLOW Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2" and a Westinghouse letter, CAW-01-1474, accompanying affidavit, Proprietary Information Notice, and Copyright Notice. This information is provided to justify a reactor core power level reduction to 3,838 Mwt, if the CROSSFLOW UFM system is not available, instead of the current licensed power level of 3,800 Mwt.

Attachments 8 and 11 contain information proprietary to the Westinghouse Electric Company ("Westinghouse"). These two attachments are accompanied by affidavits signed by Westinghouse, the owner of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and addressed with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulation.

Correspondence with respect to the copyright on proprietary aspects of the items listed above or the supporting Westinghouse affidavits should reference CAW-01-1466 or CAW-01-1474, as appropriate, and should be addressed to Mr. H. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

If there are any questions associated with these Technical Specification changes, please contact either Mr. Ken Taplett at (361) 972-8416 or me at (361) 972-8757.

J. J. Sheppard
Vice President,
Engineering &Technical Services

KJT

Attachments:
1) Affidavit
2) Licensee's Evaluation
3) Proposed Replacement Pages for the Facility Operating Licenses
4) Proposed Marked-up Technical Specifications Changes
5) Technical Specifications Pages with Proposed Changes Incorporated
6) Technical Analysis Supplement
7) Westinghouse Letter, "Application for Withholding Proprietary Information from Public Disclosure" (CAW-01-1466) with Affidavit CAW-01-14667)
8) WCAP-15633, Revision 0, "Power Calorimetric for the 1.4% Uprating for the South Texas Project Nuclear Operating Company Units 1 and 2" (Proprietary)
9) WCAP-15696, Revision 0, "Power Calorimetric for the 1.4% Uprating for the South Texas Project Nuclear Operating Company Units 1 and 2" (Non-Proprietary)
10) Westinghouse Letter, "Application for Withholding Proprietary Information from Public Disclosure" (CAW-01-1474) with Affidavit CAW-01-1474
11) WCAP-15697, Revision 0, "CROSSFLOW Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2" (Proprietary)
12) WCAP-15698, Revision 0 "CROSSFLOW Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2" (Non-Proprietary)

cc: Without Attachments 7 - 12 Unless Noted By *

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Wadsworth, TX 77483

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Attachment 10

Westinghouse Letter,
"Application for Withholding Proprietary Information from Public Disclosure"
(CAW-01-1474)
with Affidavit CAW-01-1474

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

July 26, 2001

CAW-01-1474

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Crossflow Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2", WCAP-15697, Revision 0 (Proprietary), July 2001

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-01-1474 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by South Texas Project Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-01-1474 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: M. Scott/NRR/OWFN/DRPW/PDIV2 (Rockville, MD) 1L

P:DATA/DOCUMENTS/0573s

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

PATRICIA L. CROWN COMMONWEALTH OF PENNSYLVANIA NOTARY PUBLIC

H. A. Sepp

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 26th day
of July, 2001

Patricia L. Crown

Notary Public

Notarial Seal
Patricia L. Crown, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Feb 7, 2005
Member, Pennsylvania Association of Notaries

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Crossflow Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2", WCAP-15697, Revision 0 (Proprietary), July 2001 for information in support of South Texas Project Nuclear Operating Company's submittal to the Commission, transmitted via South Texas Project Nuclear Operating Company's letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention: Mr. Samuel J. Collins. The proprietary information was provided by Westinghouse Electric Company LLC.

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

(a) Provide documentation of the methods to be used to employ Westinghouse models for performing containment design basis analyses.

(b) Assist the customer in the licensing process.

Further this information has substantial commercial value as follows:

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

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

Westinghouse Electric Company, LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

June 20, 2001

CAW-01-1466

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Power Calorimetric for the 1.4% Uprating for the South Texas Project Nuclear Operating Company Units 1 and 2", WCAP-15633, Revision 0 (Proprietary), June 2001

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-01-1466 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by South Texas Project Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-01-1466 and should be addressed to the undersigned.

Very truly yours,

John S. Galembush, Acting Manager
Regulatory and Licensing Engineering

Enclosures

cc: S. Bloom/NRR/OWFN/DRPW/PDIV2 (Rockville, MD) 1L

P:DATA/DOCUMENTS/0570s

CAW-01-1466

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

John S. Galembush, Acting Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 21st day
of June, 2001

Lorraine M. Piplica
Notary Public

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

P:DATA/DOCUMENTS/0570s

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

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Samuel J. Collins. The proprietary information was provided by Westinghouse Electric Company LLC.

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

COPYRIGHT NOTICE

ATTACHMENT 1

AFFIDAVIT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of STP Nuclear Operating Company, et al., South Texas Project Units 1 and 2	)))))))	Docket Nos. 50-498 50-499

AFFIDAVIT

I, J. J. Sheppard, being duly sworn, hereby depose and say that I am Vice President, Engineering & Technical Services, of STP Nuclear Operating Company; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached Facility Operating License and Technical Specification change; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

J J Sheppard

J. J. Sheppard
Vice President,
Engineering & Technical Services

STATE OF TEXAS)
)
COUNTY OF MATAGORDA)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 22nd day of August, 2001.

NOTARY PUBLIC STATE OF TEXAS
LOIS J. MILLS
Notary Public, State of Texas
My Commission Expires
JULY 27, 2003

Lois J. Mills

Notary Public in and for the
State of Texas

ATTACHMENT 2

<u>LICENSEE'S EVALUATION</u>

SOUTH TEXAS PROJECT
UNITS 1 & 2
PROPOSED AMENDMENT TO THE FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS

LICENSEE'S EVALUATION

1.0 INTRODUCTION

1.1 The proposed amendment will revise the Facility Operating Licenses and Technical Specifications to reflect a 1.4% increase in the reactor core power level from 3,800 Megawatts thermal to 3,853 Megawatts thermal (Mwt). STP Nuclear Operating Company (STPNOC) is planning to implement the power uprate for Unit 1 in the Spring of 2002 and for Unit 2 in late 2002 following replacement of Unit 2 steam generators with Model Δ94 steam generators. Unit 1 steam generators were replaced with Model Δ94 steam generators in the Spring of 2000.

The specific changes to the Facility Operating Licenses requested by this amendment are to paragraph 2.C.(1), revising the Maximum Power Level from 3,800 to 3,853 Mwt to reflect the revised value of 100% power.

The specific changes to the Technical Specifications requested by this amendment are:

- Revise the Definition of Rated Thermal Power (1.27)
- Add the WRB-2M DNB Correlation to Technical Specification 2.1.1.1
- Revise the Maximum Allowable Power Range Neutron Flux High Setpoint values listed in Table 3.7-1
- Add Topical Report CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," to Specification 6.9.1.6.b.

1.2 Proposed change to existing Facility Operating Licenses:

See Attachment 3

1.3 Markup of existing Technical Specifications:

See Attachment 4.

1.4 Proposed Technical Specifications:

See Attachment 5.

1.5 Supplemental technical analysis information for South Texas Project Units 1 and 2 1.4% Power Uprate Project:

See Attachment 6.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The 1.4-percent core power uprate for South Texas Project Units 1 and 2 is based on eliminating unnecessary analytical margin originally required for the emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements contained in the Code of Federal Regulations (CFR) 10CFR50, Appendix K (ECCS Evaluation Models).

WCAP-15633, Revision 0, "Power Calorimetic for the 1.4% Uprating for the South Texas Project Nuclear Operating Company Units 1 and 2", (Reference 1), provides the analysis to determine the uncertainty in the daily power calorimetric for the 1.4-percent uprate. This analysis is based on the use of the CROSSFLOW UFM system to measure feedwater flow that is described in topical report CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology" (Reference 2). The CROSSFLOW UFM system provides a more accurate indication of feedwater flow, resulting in a reduced reactor core thermal power uncertainty of less than 0.6 percent, which is less than the 2 percent uncertainty assumed during the development of Appendix K requirements. The improved thermal power measurement accuracy obviates the need for the full 2-percent power margin assumed in Appendix K, thereby increasing the thermal power available. The NRC issued a Safety Evaluation dated March 20, 2000 stating the staff's acceptance of referencing the topical report by licensees who want to use the CROSSFLOW UFM system for a power uprate.

Revised Thermal Design Procedure (RTDP) Departure from Nucleate Boiling Ratio (DNBR) design limits were calculated for the 1.4-percent power uprate with the WRB-2M DNB correlation to ensure that the DNB design basis was met. The DNBR portion of the core limits and the axial offset limits were not changed resulting in no impact on the OTΔT and OPΔT setpoints. The WRB-2M DNB correlation was used to provide increased DNB margin.

The DNBR analyses for the 1.4-percent power uprate demonstrate that the DNB design basis continues to be met.

The impact of a 1.4-percent core power uprate on the potentially affected systems, components, and safety analyses has been evaluated and is discussed in Attachment 6.

The CROSSFLOW UFM system has been installed in Units 1 and 2. Plant procedures have been developed for the maintenance and calibration of the CROSSFLOW UFM system. Prior to implementation of the power uprate, plant procedures will be revised to address the unavailability of the CROSSFLOW UFM system, including actions for reducing power, and for performing the power calorimetric using the feedwater flow venturis to determine the feedwater flow rate. The core power reduction will be based on the conclusions of WCAP-15697, "CROSSFLOW Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2" (Reference 3) . The uncertainty calculation associated with this document provides the basis for a 1.0% uncertainty for the plant power calorimetric measurement if the CROSSFLOW UFM system is out of service. Thus, core power would be reduced to 3838 Mwt if the CROSSFLOW UFM system is unavailable. This satisfies Condition 1 of the NRC Safety Evaluation for CENPD-397-P-A, Revision 01.

3.0 BACKGROUND

South Texas Project Units 1 and 2 are presently licensed for a core thermal power rating of 3,800 Mwt. This licensed power level includes a 2-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, as previously required by Code of Federal Regulations (CFR) 10CFR50, Appendix K (ECCS Evaluation Models).

The Nuclear Regulatory Commission (NRC) approved a change to the requirements of 10CFR50, Appendix K (Federal Register (FR) 65 FR 34913, June 1, 2000), with an effective date of July 31, 2000. This change provides licensees with the option of maintaining the 2-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation. The proposed alternative reduced margin for ECCS evaluation has been demonstrated to account for uncertainties due to a reduction in power level instrumentation error. Based on use of the CROSSFLOW UFM system to reduce the core power level power measurement uncertainty to less than 0.6 percent, a core power uprate of 1.4% is achievable using current NRC-approved methodologies.

The use of a more accurate feedwater flow measurement instrumentation will reduce the 2-percent power margin to less than 0.6 percent, and allow an increase in core power by 1.4 percent to 3,853 Mwt. Without the CROSSFLOW UFM system available, WCAP-15697 provides justification for a reduction in core power to 3,838 Megawatts.

The core thermal-hydraulic analyses and evaluations were performed at the uprated power level of 3,853 Mwt (1.4% uprate) assuming core designs composed of Robust Fuel Assemblies and using the WRB-2M DNB correlation for the DNB analyses, in addition to the Revised Thermal Design Procedure (RTDP) DNB methodology. To address the standard and Vantage 5 Hybrid (V5H) fuel types, the WRB-1 DNB correlation was used. The WRB-2M DNB correlation is discussed in WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids" (Reference 3). The NRC approved the use of the WRB-2M DNB correlation in a Safety Evaluation dated December 1, 1998.

Since the uprate will increase the nominal NSSS power rating of the plant, the reactor trip setpoint reductions for inoperable main steam safety valves were re-calculated and are reflected in the revision to Table 3.7-1 of the Technical Specifications.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

Licensees are required to submit a safety analysis report that contains an evaluation of emergency core cooling system (ECCS) performance under loss-of-coolant accident (LOCA) conditions. 10CFR50.46 requires that ECCS performance under LOCA conditions be evaluated and that the estimated performance satisfy certain criteria. Licensees were given the option to develop a model that conforms to the requirements of Appendix K of 10CFR 50 for conducting ECCS evaluations. Before Appendix K of 10CFR50 was revised in June 2000, Appendix K specified that a power level of 102 per cent be assumed when conducting ECCS analyses. The revision to Appendix K allows licensees the option of using a value lower than 102 per cent of licensed power in their ECCS analyses, where justified. Licensees may request an increase in licensed power level where a reduced power measurement uncertainty and basis for a modified ECCS analysis can be justified.

5.0 TECHNICAL ANALYSIS

The impact of a 1.4% core power uprate on the nuclear steam supply systems (NSSS), the balance of plant (BOP) systems and components, and the safety analyses has been evaluated. This evaluation has been performed consistent with the methodology contained in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Plant" (Reference 4). The methodology contained in WCAP-10263 has been used successfully in many PWR power uprate programs. This methodology establishes the approach and criteria to be addressed in uprate programs in the following broad categories;

- NSSS performance parameters,
- design transients,
- systems, components, and interfaces between NSSS and BOP systems,
- safety analyses, and
- nuclear fuel.

Well-defined analysis input assumptions and parameter values using current NRC-approved methods with currently applicable licensing criteria and standards are used in this methodology.

A comprehensive program consistent with this methodology was performed for the South Texas Project Units 1 and 2 to justify the proposed increase in licensed core power from 3,800 Mwt to 3,853 Mwt. The following sections of Attachment 6 provide a more detailed technical analysis to support the overall conclusions in this section.

- Section 2 discusses the revised NSSS design thermal and hydraulic parameters modified as a result of the 1.4-percent uprate and that serve as the basis for all of the NSSS analyses and evaluations.

- Section 3 describes the CROSSFLOW UFM system which provides the more accurate feedwater flow measurement.

- Section 4 discusses the RTDP uncertainties, and supports a 0.6% calorimetric uncertainty to support the 1.4% power uprate.

- Section 5 concludes that no design transient changes are required to accommodate the revised NSSS design conditions.

- Sections 6 and 7 discuss the system (e.g., safety injection, residual heat removal (RHR), and control systems) and component (e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), steam generator, and NSSS auxiliary equipment) evaluations completed for the revised design conditions.

- Section 8 provides the results of the accident analyses and evaluations performed (e.g., steam generator tube rupture, mass and energy release, loss-of-coolant-accident (LOCA), and non-LOCA). The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

- Sections 9 and 10 discuss the evaluations performed for the 1.4-percent power uprate impact on the electrical power and balance of plant systems.

- Section 11 provides the results of the 1.4-percent power uprate on the radiological analyses.

- Sections 12 and 13 discuss the evaluations performed for the 1.4-percent power uprate impact on plant operations and other licensing requirements.

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

1. Apply a 2-percent increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the 1.4% uprate conditions because the sum of increased core power level (1.4%) and the decreased power measurement uncertainty (less than 0.6%) fall within the previously analyzed conditions.

   The power calorimetric uncertainty calculation discussed in Section 4 of Attachment 6 indicates that with the CROSSFLOW UFM system installed, the power measurement uncertainty (based on a 95-percent probability at a 95-percent confidence interval) is less than 0.6-percent. Thus, these analyses only need to reflect a 0.6-percent power measurement uncertainty. Accordingly, the existing 2-percent uncertainty can be allocated such that 1.4-percent is applied to provide sufficient margin to address the uprate to 3,853 Mwt, and 0.6-percent is retained in the analysis to still account for the power measurement uncertainty. In addition, for these types of analyses, it is shown that they still employ other conservative assumptions not affected by the 1.4-percent power uprate. Taken together, the use of the calculated 95/95 power measurement uncertainty and retention of conservative assumptions indicate that the margin of safety for these analyses would not be reduced. This same conclusion can be made for the less than 1.0-percent uncertainty with a 95-percent probability at a 95-percent confidence interval if the CROSSFLOW UFM system is unavailable.

2. Employ a nominal power level. These analyses have either been evaluated or re-performed for the 1.4-percent increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.4-percent conditions. These results bound the 1-percent increased power level if the CROSSFLOW UFM system is unavailable.

3. Assume a core power level in excess of the proposed 3,853 Mwt. These analyses were previously performed at a higher power level (typically 4,100 Mwt) as part of prior plant programs. For these analyses, some of this available margin has been used to offset the 1.4-percent uprate. Consequently, the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the 1.4-percent uprate.

4. Perform analyses at zero-percent power conditions or do not actually model the core power level. Consequently, these analyses have not been re-performed since they are unaffected by the core power level.

6.0 REGULATORY ANALYSIS

The technical analysis in Section 5.0 supported by the Licensing Report in Attachment 6 demonstrates that the CROSSFLOW UFM system provides a more accurate feedwater flow measurement. The instrument uncertainty methodology presented in WCAP-15633 assuming the use of the CROSSFLOW UFM system to measure feedwater flow provides the basis for a 0.6-percent calorimetric uncertainty to support a 1.4-per cent power uprate. In addition, WCAP-15697 provides the justification for a reduction in core power if the CROSSFLOW UFM system is unavailable. This reduced power measurement uncertainty provides a basis for a modified ECCS analysis that justifies an increase in licensed power level for meeting 10CFR50.46 criteria.

Applicable acceptance limits are also addressed in Section 5.0, "Technical Analysis" and in Attachment 6, "Licensing Report for South Texas Project Units 1 and 2 1.4% Power Uprate Project".

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Pursuant to 10CFR50.91, this analysis provides a determination that the proposed change described previously does not involve any significant hazards consideration as defined in 10CFR50.92:

1. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The comprehensive analytical efforts performed to support the proposed uprate conditions include a review and evaluation of all components and systems (including interface systems and control systems) that could be affected by this change. The revised power uprate value was input to applicable safety analyses. The proposed change is not an initiator of any design-basis accident. All of the Nuclear Steam Supply System or Balance of Plant interface systems will continue to perform their intended design functions and meet all performance requirements. The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms, loop piping and supports, reactor coolant pump, steam generator, and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Therefore, there is no increase in the probability of a structural failure of these components.

   The auxiliary systems and components continue to comply with applicable structural limits and will continue to perform their intended design functions. Therefore, there is no increase in the probability of a structural failure of these components. The steam generator safety valves will provide adequate relief capacity to maintain the steam generators within design limits. The steam dump system will still relieve 40 percent of the maximum full-load steam flow.

   Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   The applicable analyses have been evaluated with respect to the increase in core power associated with this change. All applicable radiological acceptance criteria continue to be met. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   The proposed change neither causes the initiation of any accident nor creates any new limiting single failures. All of the affected systems and components continue to perform their intended design functions. The proposed change has no adverse effects on any safety-related system or component and does not challenge the performance or integrity of any safety-related system.

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

3. **The proposed change does not involve a significant reduction in a margin of safety.**

The WRB-2M DNB methodology is used to demonstrate that core thermal-hydraulic limits are maintained without any significant reduction in margin of safety for the uprated power level of 3853 Mwt (1.4-percent uprate) assuming core designs composed of Robust Fuel Assemblies. The WRB-1 DNB correlation demonstrates that there is no significant reduction in the margin of safety for core designs composed of standard or Vantage 5 Hybrid (V5H) fuel types. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that either have been reviewed and approved by the Nuclear Regulatory Commission or are in compliance with all applicable regulatory review guidance and standards.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10CFR50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

10CFR51.22(b) specifies the criteria for categorical exclusion from the requirements for a specific environmental assessment per 10 CFR 51.21. This amendment request meets the criteria specified in 10 CFR 51.22(c)(9). The specific criteria contained in this section are discussed below.

(i) the amendment involves no significant hazards consideration

As demonstrated in the No Significant Hazards Consideration Determination, the requested license amendment does not involve any significant hazards consideration.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite

The requested license amendment involves a minimal increase in rated thermal power level. The requested license amendment involves no physical change to the facility or any other change in the manner of operation of any plant systems involving the generation, collection or processing of radioactive materials or other types of effluents. Therefore, no significant increase in the amounts of effluents or new types of effluents would be created.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure

The requested license amendment involves a minimal increase in rated thermal power level. The requested license amendment involves no change to the facility and will result in minimal increase in the radiation dose resulting from the operation of any plant system. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure associated with this proposed change.

Based on the above it is concluded that there will be no impact on the environment resulting from this change. The change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to specific environmental assessment by the Commission.

9.0 PRECEDENCE

Power uprates based on a reduced core thermal power level uncertainty associated with a more accurate measurement of feedwater flow of have been previously approved by the NRC for 1.4 percent for Salem Nuclear Generating Station, Units 1 and 2 in Amendments No. 243 and 244 dated May 25, 2001, and for 1.42-percent for the San Onofre Nuclear Generating Station, Units 2 and 3 in Amendments No. 180 and 171 dated July 6, 2001.

10.0 REFERENCES

1) WCAP-15633, Revision 0, "Power Calorimetic for the 1.4% Uprating for the South Texas Project Nuclear Operating Company Units 1 and 2"

2) Topical Report CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology"

3) WCAP-15697, "CROSSFLOW Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2"

4) WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids"

5) WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Plant"

ATTACHMENT 3

PROPOSED REPLACEMENT PAGES FOR FACILITY OPERATING LICENSE

Paragraph 2.C.(1)

Revise the Maximum Power Level from 3,800 Mwt to 3,853 Mwt to reflect the revised value of 100% power in Paragraph 2.C.(1)

Note: The first page following this page is the proposed replacement page for the South Texas Project, Unit 1 Facility Operating License, NPF-76. The second page following this page is the proposed replacement page for the South Texas Project, Unit 2 Facility Operating License, NPF-80.

(1) STPNOC pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in Matagorda County, Texas, in accordance with the procedures and limitations set forth in this license;

(2) 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), pursuant to the Act and 10 CFR Part 50, to possess the facility at the designated location in Matagorda County, Texas, in accordance with the procedures and limitations set forth in this license;

(3) STPNOC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) STPNOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(6) STPNOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein.

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

(1) <u>Maximum Power Level</u>

STPNOC is authorized to operate the facility at reactor core power levels not in excess of 3,853 megawatts thermal (100% power) in accordance with the conditions specified herein.

(1) STPNOC pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in Matagorda County, Texas, in accordance with the procedures and limitations set forth in this license;

(2) 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), pursuant to the Act and 10 CFR Part 50, to possess the facility at the designated location in Matagorda County, Texas, in accordance with the procedures and limitations set forth in this license;

(3) STPNOC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) STPNOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(6) STPNOC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein.

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

(1) Maximum Power Level

STPNOC is authorized to operate the facility at reactor core power levels not in excess of 3,853 megawatts thermal (100% power) (Model Δ94 steam generators installed) or 3,800 Mwt thermal (100% power) (Model E steam generators installed) in accordance with the conditions specified herein.

ATTACHMENT 4

PROPOSED MARKED-UP TECHNICAL SPECIFICATION CHANGES

- **Revise the Definition of Rated Thermal Power (1.27) from 3,800 Mwt to 3,853 Mwt**
- **Add the WRB-2M DNB Correlation to Technical Specification 2.1.1.1**
- **Revise the Power Range Neutron Flux High Setpoint values listed in Table 3.7-1**
- **Add Topical Report CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," to Specification 6.9.1.6.b.**

Page 6-21 has no changes and provided for completeness only.

Bases pages are provided for information only

DEFINITIONS

PROCESS CONTROL PROGRAM

1.24 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

1.26 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3,8~~00~~53 Mwt (**Model Δ94 steam generators installed) or 3,800 Mwt (Model E steam generators installed).**

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in the Core Operating Limits Report.

2.1.1.1 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 $\geq$ 1.14 for the WRB-2M DNB correlation**.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58 °F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, AND 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

MODEL Δ94 STEAM GENERATORS ONLY

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	61
2	43
3	26

MODEL E STEAM GENERATORS ONLY

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	63
2	45
3	27

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle, or any part of a reload cycle for the following:

1. Safety limits for thermal power, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) for Specification 2.1,
2. Limiting Safety System Settings for Reactor Coolant Flow-Low Loop design flow, Overtemperature ΔT, and Overpower ΔT setpoint parameter values for Specification 2.2,
3. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3/4.1.1.3,
4. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
5. Control Bank Insertion Limits for Specification 3/4.1.3.6,
6. Axial Flux Difference limits and target band for Specification 3/4.2.1,
7. Heat Flux Hot Channel Factor, K(Z), Power Factor Multiplier, and (F_{xy}^{RTP}) for Specification 3/4.2.2,
8. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3, and
9. DNB related parameters for Reactor Coolant System T_{avg} Pressurizer Pressure, and the Minimum Measured Reactor Coolant System Flow for Specification 3/4.2.5.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

6.9.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP 9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July, 1985 (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 Shutdown Rod Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 - DNB Parameters.)

CORE OPERATING LIMITS REPORT (Continued)

2. WCAP12942-P-A "SAFETY EVALUATION SUPPORTING A MORE NEGATIVE EOL MODERATOR TEMPERATURE COEFFICIENT TECHNICAL SPECIFICATION FOR THE SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION UNITS 1 AND 2."

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

3. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature, ΔT Trip Functions," September 1986 (Westinghouse Proprietary Class 2)

(Methodology for Specification 2.1 - Safety Limits, and 2.2 - Limiting Safety System Settings)

4. WCAP 8385, "POWER DISTRIBUTION AND LOAD FOLLOWING PROCEDURES TOPICAL REPORT", September, 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

5. Westinghouse letter NS-TMA-2198, T.M. Anderson (Westinghouse) to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control). Approved by NRC Supplement No. 4 to NUREG-0422, January, 1981 Docket Nos. 50-369 and 50-370.)

6. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July, 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

7. WCAP-10266-P-A, Rev.2, WCAP-11524-NP-A, Rev.2, "The l98l Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", Kabadi, J.N., et al., March 1987; including Addendum 1-A, "Power Shape Sensitivity Studies," December 1987 and Addendum 2-A, "BASH Methodology Improvements and Reliability Enhancements" May 1988.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

8. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April, 1995 (W Proprietary) for Loss of Coolant Accident (LOCA) Evaluation models with ZIRLO clad fuel for rod heatup calculation.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

9. CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

(Methodology for operating at a RATED THERMAL POWER of 3,853 Mwt.)

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

FOR INFORMATION ONLY

REACTOR CORE BASES

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the WRB-1 correlation **and the WRB-2M correlation**. The WRB-1 DNB **and WRB-2M DNB** correlations hasve been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB design basis is as follows: uncertainties in the WRB-1 **and WRB-2M** correlations, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with a 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value that must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The reactor core Safety Limits are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and,
b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core Safety Limits are used to define the various Reactor Protection System (RPS) functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that, for variations in the Thermal Power, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI, the reactor core Safety Limits will be satisfied during steady state operation, normal operational transients, and AOOs.

FOR INFORMATION ONLY

TURBINE CYCLE
SAFETY VALVES BASES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1413.5 psig) of its design pressure of 1285 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

Five MSSVs, each with an orifice size of 16 in^2, are located on each main steam header, outside containment, upstream of the main steam isolation valves. The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 20.65 x 10^6 lbs/h which is 122% the total secondary steam flow of 16.94 X 10^6 lbs/h **for the Model E steam generators or 120% of the total secondary steam flow of 17.20 X 10^6 lbs/h for the Model Δ94 steam generators** at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

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

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

Where:

Hi Φ	=	Safety analysis power range high neutron flux setpoint, percent
Q	=	Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt
K	=	Conversion Factor, 947.82 (BTU/sec)/MWt
w_s	=	Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure, including tolerance and accumulation, as appropriate, in lbm/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.
h_{fg}	=	Heat of vaporization for steam at the highest MSSV operating pressure including allowances for tolerance, drift, and accumulation, as appropriate, Btu/lbm
N	=	Number of loops in the plant.

The calculated values are lowered an additional 9% full power to account for instrument and channel uncertainties.

ATTACHMENT 5

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES INCORPORATED

PROCESS CONTROL PROGRAM

1.24 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

1.26 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3,853 Mwt (Model Δ94 steam generators installed) or 3,800 MWt (Model E steam generators installed).

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in the Core Operating Limits Report.

2.1.1.1 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 $\geq$ 1.14 for the WRB-2M DNB correlation.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58 °F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, AND 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

MODEL Δ94 STEAM GENERATORS ONLY

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	61
2	43
3	26

MODEL E STEAM GENERATORS ONLY

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	63
2	45
3	27

CORE OPERATING LIMITS REPORT (Continued)

2. WCAP12942-P-A "SAFETY EVALUATION SUPPORTING A MORE NEGATIVE EOL MODERATOR TEMPERATURE COEFFICIENT TECHNICAL SPECIFICATION FOR THE SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION UNITS 1 AND 2."

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

3. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature, ΔT Trip Functions," September 1986 (Westinghouse ProprietaryClass 2)

(Methodology for Specification 2.1 - Safety Limits, and 2.2 - Limiting Safety System Settings)

4. WCAP 8385, "POWER DISTRIBUTION AND LOAD FOLLOWING PROCEDURES TOPICAL REPORT", September, 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control.)

5. Westinghouse letter NS-TMA-2198, T.M. Anderson (Westinghouse) to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control). Approved by NRC Supplement No. 4 to NUREG-0422, January, 1981 Docket Nos. 50-369 and 50-370.)

6. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July, 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

7. WCAP-10266-P-A, Rev.2, WCAP-11524-NP-A, Rev.2, "The l98l Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", Kabadi, J.N., et al., March 1987; including Addendum 1-A, "Power Shape Sensitivity Studies," December 1987 and Addendum 2-A, "BASH Methodology Improvements and Reliability Enhancements" May 1988.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

8. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April, 1995 (W Proprietary) for Loss of Coolant Accident (LOCA) Evaluation models with ZIRLO clad fuel for rod heatup calculation.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

CORE OPERATING LIMITS REPORT (Continued)

9. CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

(Methodology for operating at a RATED THERMAL POWER of 3,853 Mwt)

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

Attachment 6

TECHNICAL ANALYSIS

SUPPLEMENT

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LIST OF ACRONYMS

AFST	auxiliary feedwater storage tank
AFWS	auxiliary feedwater system
AMSAC	ATWS mitigation system actuation circuitry
ANS	American Nuclear Society
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
B&PV	Boiler and Pressure Vessel Code
BEF	best-estimate flow
BHP	brake horsepower
BOL	beginning of life
BOP	balance of plant
C&FS	condensate and feedwater system
CFR	Code of Federal Regulations
CRDM	control rod drive mechanism
CVCS	chemical and volume control system
DEPS	double-ended pump suction
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EFPY	effective full-power year
EOL	end of license
EFS	emergency feedwater system
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
FBCV	feedwater bypass control valves
FCV	feedwater control valves
FIV	feedwater isolation valve
FR	Federal Register
FTI	Framatome Technologies International
HFP	hot full power
HZP	hot zero power
IFBA	integrated fuel burnable absorber
IPCA	instrumentation port column assembly

LIST OF ACRONYMS (Cont'd)

ISA	Instrument Society of America
LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LTCC	long-term core cooling
MSIV	main steam isolation valve
MSS	main steam system
MSSV	main steam safety valve
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OBE	operating basis earthquake
OPΔT	overpower delta T
OTΔT	overtemperature delta T
PCWG	Performance Capability Working Group
PORV	power-operated relief valve
PRT	pressurizer relief tank
P-T	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized water reactor
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pump
RCS	reactor coolant system
RFA	Robust Fuel Assembly
RHR	residual heat removal
RHRS	residual heat removal system
RPS	reactor protection system
RSG	replacement steam generator
RTDP	revised thermal design procedure
RT_{NDT}	reference temperature for nil ductility transition
RTP	rated thermal power
RT_{PTS}	reference temperature, pressurized thermal shock
RTS	reactor trip switchgear
RWFS	rod withdrawal from subcritical
RWST	refueling water storage tank

LIST OF ACRONYMS (Cont'd)

SBLOCA	small-break loss-of-coolant accident
SER	Safety Evaluation Report
SG	steam generator
SGBS	steam generator blowdown system
SGTP	steam generator tube plugging
SSE	safe shutdown earthquake
STPNOC	South Texas Project Nuclear Operating Company
SRSS	square root sum of the squares
TDF	thermal design flow
UFSAR	Updated Final Safety Analysis Report
UFM	ultrasonic flow measurement
USE	upper shelf energy
V5H	Vantage 5 Hybrid
VPCF	velocity profile correction factor

1 INTRODUCTION

1.1 BACKGROUND

The South Texas Project Units 1 and 2 are presently licensed for a core power rating of 3,800 MWt. Through the use of more accurate feedwater flow measurement instrumentation, approval is sought to increase this core power by 1.4 percent, to 3,853 MWt.

The South Texas Project Nuclear Operating Company (STPNOC) 1.4-percent core power uprate for South Texas Units 1 and 2 is based on eliminating unnecessary analytical margin originally required for emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in the Code of Federal Regulations (CFR) 10 CFR 50, Appendix K (Emergency Core Cooling System Evaluation Models, ECCS). The U.S. Nuclear Regulatory Commission (NRC) approved a change to the requirements of 10 CFR 50, Appendix K (Federal Register (FR) 65 FR 34913, June 1, 2000). The change provides licensees with the option of maintaining the 2-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation. The proposed alternative reduced margin for ECCS evaluation has been demonstrated to account for uncertainties due to a reduction in power level instrumentation error. Based on the use of the CROSSFLOW instrumentation to determine core power level with a power measurement uncertainty of less than 0.6 percent, it is proposed to reduce the licensed power uncertainty required by 10 CFR 50, Appendix K, for increases of up to 1.4 percent in the license power level using current NRC-approved methodologies.

The impact of a 1.4-percent core power uprate has been evaluated on the nuclear steam supply system (NSSS) and balance of plant (BOP) systems, components, and safety analyses. This document summarizes these evaluations, analyses, and conclusions.

1.2 APPROACH FOR INCREASING THE PLANT POWER LEVEL

The South Texas Units 1 and 2 1.4-Percent Power Uprate Program has been completed consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant." Since its submittal to the NRC, the methodology has been successfully used as the basis for power uprate projects for over 20 pressurized water reactor (PWR) units.

The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects, including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents, and nuclear fuel, as well as the interfaces between the NSSS and BOP systems. The methodology includes the use of well-defined analysis input assumptions/parameter values, use of currently approved analytical techniques, and use of currently applicable licensing criteria and standards.

A comprehensive engineering review program consistent with this methodology has been performed for South Texas Units 1 and 2 to evaluate an increase in the licensed core power from 3,800 MWt to 3,853 MWt. Section 2 of this report discusses the revised NSSS design thermal and hydraulic parameters that were changed as a result of the 1.4-percent uprate, which serve as the basis for all of the NSSS analyses and evaluations. Section 3 describes the CROSSFLOW system that provides the more accurate feedwater flow measurement. Section 4 discusses the Revised Thermal Design Produce (RTDP) uncertainties that support a 0.6-percent calorimetric uncertainty to justify the 1.4-percent uprating. Section 5 concludes that no design transient modifications are required to accommodate the revised NSSS design conditions. Sections 6 and 7 present the system (e.g., safety injection, residual heat removal (RHR), and control systems) and components (e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), steam generator, and NSSS auxiliary equipment) evaluations completed for the revised design conditions. Section 8 provides the results of the accident analyses and evaluations performed for the various analyses area (e.g., steam generator tube rupture, mass and energy release, loss-of-coolant-accident (LOCA) and Non-LOCA). Section 9 discusses the impact of the uprate on the plant electrical system. Section 10 discusses the impact of the uprate on the balance of plant (BOP) systems. Section 11 provides a summary of the radiological evaluation. Section 12 discusses the impact of the uprate on plant operations. Section 13 discusses the evaluations of other licensing requirements. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

1.3 EVALUATION APPROACH FOR THE 1.4-PERCENT POWER UPRATE

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

- First, some analyses apply a 2-percent increase to the initial power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the 1.4-percent uprate conditions because the sum of increased core power level (1.4 percent) and the decreased power measurement uncertainty (less than 0.6 percent) fall within the previously analyzed conditions.

  The power calorimetric uncertainty calculation described in Section 4 indicates that, with the CROSSFLOW instrumentation installed, the power measurement uncertainty (based on a 95-percent probability at a 95-percent confidence interval) is less than 0.6 percent. Therefore, these analyses only need to reflect a 0.6-percent power measurement uncertainty. Accordingly, the existing 2-percent uncertainty can be allocated such that 1.4 percent is applied to provide sufficient margin to address the uprate to 3,853 MWt, and 0.6 percent is retained in the analysis to account for the power measurement uncertainty. In addition, for these types of analyses, it is shown that they make other conservative assumptions not affected by the 1.4-percent uprated power. In summary, the use of the calculated 95/95 power measurement uncertainty, and retention of other conservative assumptions ensure that the margin of safety for these analyses would not be reduced.

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

- Third, some analyses assume a core power level in excess of the proposed 3,853 MWt. These analyses were previously performed at a higher power level (typically 4,100 MWt) as part of prior plant programs. For these analyses, some of this available margin has been used to offset the 1.4-percent uprate. Consequently, these analyses have been evaluated to confirm that sufficient analysis margin exits to envelop the 1.4-percent uprate.

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

2 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

2.1 INTRODUCTION

The nuclear steam supply system (NSSS) design parameters are the fundamental parameters used as input in all of the NSSS analyses. They provide the reactor coolant system (RCS) and secondary system conditions (temperatures, pressures, and flow) that are used as the basis for all of the NSSS analyses and evaluations.

It was necessary to revise these parameters due to the 1.4-percent increase in licensed core power from 3,800 MWt to 3,853 MWt. The new parameters are identified in Table 2.1-1. These parameters have been incorporated, as required, into the applicable NSSS system and component evaluations, as well as safety analyses, performed in support of the 1.4-percent uprate.

2.2 INPUT PARAMETERS AND ASSUMPTIONS

The NSSS design parameters are determined based on conservative inputs, such as a conservatively low thermal design flow (TDF) and bounding steam generator tube plugging (SGTP) levels, which yield primary- and secondary-side conditions that bound plant operation.

2.3 DISCUSSION OF PARAMETER CASES

Table 2.1-1 provides the NSSS design parameter cases generated and used as the basis for the 1.4-percent uprating for the Model Δ94 steam generators (SGs).

The 1.4-percent uprating resulted in changes to some of the NSSS design parameters. The changes included a range of full-power normal operating T_{avg} from 582.7°F to 592.6°F, from the current design values of 582.3°F to 593.0°F.

2.4 CONCLUSIONS

The four cases of NSSS design parameters identified in Table 2.1-1 were used to evaluate the impact of the 1.4-percent power uprating on South Texas Units 1 and 2.

The various NSSS analyses and evaluations discussed in this report used the design parameters appropriate for the given analytical area.

Table 2.1-1 1.4% Uprate NSSS Design Parameters – South Texas Units 1 and 2

BASIC COMPONENTS			
Reactor Vessel, ID, in.	173	Isolation Valves	No
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17 RFA(3)	Model	Δ94
Rod OD, in.	0.374	Shell Design Pressure, psia	1300
Number of Grids	8R, 2L	Reactor Coolant Pump	
Active Fuel Length, in.	168	Model/Weir	100A/NA
Number of Control Rods, FL	57	Pump Motor, hp	8000
Internals Type	TGX	Frequency, Hz	60

	1.4% Uprate/RSG			
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power, %	101.4	101.4	101.4	101.4
MWt	3874	3874	3874	3874
10^6 Btu/hr	13,219	13,219	13,219	13,219
Reactor Power, MWt	3853	3853	3853	3853
10^6 Btu/hr	13,147	13,147	13,147	13,147
Thermal Design Flow, Loop gpm	98,000	98,000	98,000	98,000
Reactor 10^6 lb/hr	145.21	145.21	147.39	147.39
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	8.5(1)	8.5(1)	8.5(1)	8.5(1)
Reactor Coolant Temperature, °F				
Core Outlet	630.0	630.0	620.9	620.9
Vessel Outlet	624.8	624.8	615.5	615.5
Core Average	597.4	597.4	587.4	587.4
Vessel Average	592.6	592.6	582.7	582.7
Vessel/Core Inlet	560.3	560.3	549.8	549.8
Steam Generator Outlet	559.9	559.9	549.4	549.4
Steam Generator				
Steam Temperature, °F	551.4	549.9	540.7	539.2
Steam Pressure, psia	1057(2)	1045(2)	968(2)	956(2,5)
Steam Flow, 10^6 lb/hr total	17.20/16.06	17.19/16.06	17.12/16.00	17.11/15.99
Feed Temperature, °F	441.8/391.8(6)	441.8/391.8(6)	441.8/391.8(6)	441.8/391.8(6)
Moisture, % max.	0.10	0.10	0.10	0.10
Tube Plugging, %	0	10	0	10
Zero-Load Temperature, °F	567	567	567	567

HYDRAULIC DESIGN PARAMETERS	
Mechanical Design Flow, gpm	110,000
Minimum Measured Flow, gpm total	403,000(4)

FOOTNOTES:
(1) Includes 2% for upper head T_{cold} conversion and 2% for thimble plug removal.
(2) 15 psi internal steam generator pressure drop incorporated.
(3) Parameters apply to both Robust Fuel Assembly (RFA) and V5H fuel.
(4) Includes a 2.8% flow measurement uncertainty.
(5) Minimum steam pressure limited to 957 psia to be consistent with existing analysis values.
(6) Minimum feedwater temperature reduced to 390°F to be consistent with existing analyses values.

3 CROSSFLOW CALCULATION

3.1 CROSSFLOW ULTRASONIC FLOW MEASUREMENT

The CROSSFLOW system is used to measure feedwater flow. Feedwater flow is an input for determining the plant secondary calorimetric power to verify the core thermal power output. The CROSSFLOW system uses a cross-correlation technique to determine the velocity of the fluid. This is done by measuring the time a unique pattern of eddies takes to pass between two sets of ultrasonic transducers, each transducer is set at a known distance apart, injecting ultrasonic signals perpendicular to the pipe axis.

This flow measurement method yields highly accurate flow readings and has been approved by the U. S. Nuclear Regulatory Commission (NRC) for power uprate applications as documented in topical report CENPD-397-P-A, Revision 01.

3.2 USE OF CROSSFLOW TO DETERMINE CALORIMETRIC POWER

The CROSSFLOW system receives feedwater pressure, feedwater temperature, and feedwater flow inputs that are transmitted via a datalink from the plant computer. The CROSSFLOW computer then determines the fluid velocity in the common header and converts the fluid velocity to a mass flow by using the feedwater temperature, pressure, and known pipe dimensions 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 then used to determine the core thermal power. This core thermal power determination will be used directly to calibrate the nuclear instrumentation in accordance with the Technical Specifications.

3.3 CROSSFLOW FAILURE

CROSSFLOW system failures are detected and transmitted to the plant computer. This causes an overhead annunciator point to alarm when CROSSFLOW abnormal conditions exist to make the operators aware of the CROSSFLOW system status. The CROSSFLOW system does not perform any safety function and is not used to directly control any plant systems. Therefore, system inoperability has no immediate effect on plant operation.

If the CROSSFLOW system becomes unavailable, plant operation at a core thermal power level of 3,853 MWt may continue for 24 hours. The 24-hour period is based upon the use of previous feedwater CROSSFLOW system corrections to the venturi mass flow having good quality and being representative of actual steady-state plant conditions. If the feedwater CROSSFLOW 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. This power level is based upon the methodology and instrumentation configuration assumed in WCAP-15697, "CROSSFLOW Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2". Core power would be maintained at a level less than or equal to 3,838 MWt, until the CROSSFLOW system was returned to service and a secondary plant calorimetric required to satisfy Technical Specification Surveillance Requirement 4.3.1.1.2.a was performed using the CROSSFLOW system indication of feedwater flow.

3.4 MAINTENANCE AND CALIBRATION

Maintenance of the CROSSFLOW meter is performed in accordance with the guidelines established in the referenced topical report (CENPD-397-P-A, Revision 01) and users manual. Proper maintenance is assured through both automatic and manual checks of the system. Manual checks are performed using site-specific procedures developed from (CENPD-397-P-A, Revision 01) and the user manual.

Manual checks include verification of the time delay circuits using an internal time delay circuit. A check is also run on the transducers for the observance of any shift. It should be noted that shifts in the transducer scans do not affect the accuracy of the meter's output. The effect of such a shift is to reduce the number of flow measurements per unit time, but each of the measurements would still be normally distributed about the true flow.

The final check is a planned calibration of the signal conditioning unit time-delay circuits using a National Institute of Standards and Technology traceable standard. Automatic checks are also performed by the CROSSFLOW meter to assure that the flow measurement accuracy is maintained. This check includes the automatic calculation of the uncertainty of the flow measurement, and the verification that it is enveloped by the uncertainty calculation, that formed the bases for the magnitude of the Appendix K power uprate.

3.5 OPERATIONS AND MAINTENANCE HISTORY OF THE INSTALLED CROSSFLOW INSTRUMENTION

The CROSSFLOW meter was installed in Units 1 and 2 in December of 1999. The installations were performed in accordance with Westinghouse's installation procedures (formerly ABB's procedures at the time of installation). These procedures, for meter installation, were produced in accordance with the descriptions and criteria established by the referenced topical report (CENPD-397-P-A, Revision 01).

A total of three transducer failures have been observed between the two units since the systems were placed in operation to correct for flow instrumentation drift and venturi fouling. It should be noted that the failures of these transducers manifested itself as an increase in the standard deviation of the flow data, which in turn caused the accuracy criteria to be intermittently exceeded. However, at no time did the mean flow become biased in either the high or low direction. Therefore, as described in the topical report, the system of automatic checks identified a failure, but did not introduce a bias into the meter readings.

The CROSSFLOW system installed at South Texas Project Units 1 and 2 is representative of the CROSSFLOW ultrasonic flow measurement (UFM) discussed in the topical report CENPD-397-P-A, Revision 01, and is bounded by the requirements set forth in this topical report.

3.6 UNCERTAINTY DETERMINATION METHODOLOGY

The methodology used to calculate the CROSSFLOW UFM uncertainties is consistent with American National Standards Institute/Instrument Society of America (ANSI/ISA)-S67.04 Part I and ISA-RP67.04-Part II, which uses an industry-accepted square root sum of the squares (SRSS) methodology. The South Texas Project (STP) currently uses this methodology in the development of the majority of its calculated instrument uncertainties.

With respect to the CROSSFLOW UFM uncertainties, uncertainty calculations have been performed and determined a mass flow accuracy of better than 0.5 percent of rated flow for the South Texas Units 1 and 2. These calculations are consistent with the methodology described in topical report CENPD-397-P-A, Revision 01.

Additionally, the CROSSFLOW UFM uncertainty calculations are performed to achieve a 95-percent confidence interval flow measurement with the primary terms of:

- Inside pipe diameter measurement and associated uncertainty
- Transducer spacing measurement and associated uncertainty
- Velocity profile correction factor (VPCF) and associated uncertainty
- Flow density and associated uncertainty
- CROSSFLOW time delay calibration data and associated uncertainty

These terms are developed and appropriately combined consistent with the current STP instrument uncertainty methodology.

STP maintenance procedures and CROSSFLOW system operating instructions will ensure that the assumptions and requirements of the uncertainty calculation remain valid.

3.7 SITE-SPECIFIC PIPING CONFIGURATION

The plant-specific installation follows the guidelines of topical report CENPD-397-P-A, Revision 01. The CROSSFLOW meter was calibrated at the plant under actual full-power operating conditions. The feedwater piping configuration at South Texas includes a 36-inch common header, which feeds four 18-inch individual feedwater loops. All 4 loops included a straight run of pipe from the common header to the venturis, followed by another 7 pipe diameters of straight run from the venturi discharge to a thermal well. Beyond the thermal well, 3 of the 4 loops turn abruptly, requiring the meter to be located just upstream of the thermal well in a location for which there was no calibration data. The fourth loop included a long run of straight pipe downstream of the thermal well, where the flow would be fully developed.

To calibrate the meters just upstream of the thermal wells, a second meter was installed on the fourth loop in the fully developed flow section, with the intent of using this meter to establish a calibration factor for the meters that were located upstream of the thermal wells. However, after collecting the flow data from both meters on the fourth loop, it was observed that the difference in the readings, of 0.1 percent, was well within the uncertainty of the meter. Based on these results, it was concluded that the flow would also be fully developed for all 4 meters located upstream of the thermal well, eliminating the need to include a calibration factor for each of the meters. This conclusion is supported by plant operating data.

4 REVISED THERMAL DESIGN PROCEDURE UNCERTAINTIES

Westinghouse WCAP-13441 provides the basis for the revised thermal design procedure (RTDP) uncertainties that are used in the South Texas Units 1 and 2 safety analyses. These include T_{avg} (rod) control, pressurizer pressure control, reactor coolant system (RCS) flow measurement (calorimetric) and indication, and power measurement (calorimetric). The effect of the power uprating on these uncertainties is discussed in the following subsections.

4.1 POWER CALORIMETRIC

While not covered by WCAP-13441, typical plant safety analysis evaluations for Condition II non-departure from nucleate boiling (DNB), Condition III, and Condition IV events assume a power calorimetric uncertainty of 2.0-percent of rated thermal power (RTP). The power uprate is based on a reduction in the power calorimetric uncertainties, such that the calculated uncertainties, plus the magnitude of the power uprate, remains within the 2.0-percent RTP assumption of these evaluations. Therefore, the final calculated uncertainties determine the magnitude of the power uprate. The primary means of reducing the power calorimetric uncertainties is a reduction in the uncertainties associated with the measurement of secondary-side feedwater flow. New calculations were performed to determine the uncertainties for the daily power calorimetric assuming the use of the CROSSFLOW measurement system to determine total feedwater flow. The uncertainty allowance for feedwater system flow is ±0.5 percent of the flow per loop. The flow error, in combination with the remaining uncertainty components, results in a total 95/95 power measurement uncertainty of ±0.6-percent RTP. A power measurement uncertainty of ±0.6 percent allows a power uprate of 1.4-percent RTP. The methodology used to determine the power calorimetric uncertainties is documented in WCAP-15633. With the CROSSFLOW UFM system unavailable, the components used in the power calorimetric calculation results in a total 95/95 power measurement uncertainty of ±1.0-percent RTP. The methodology used to determine the power calorimetric uncertainties with the CROSSFLOW UFM system out of service is documented in WCAP-15697.

4.2 T_{AVG} (ROD) CONTROL AND PRESSURIZER PRESSURE CONTROL

The uncertainties associated with the T_{avg} and pressurizer pressure control systems are not affected by changes in plant parameters for the 1.4-percent power uprate conditions. Therefore, the uprating does not require changes to the uncertainties documented in WCAP-13441 for these controllers.

4.3 REACTOR COOLANT SYSTEM FLOW CALORIMETRIC

The RCS flow calorimetric calculation uses nominal plant conditions for feedwater temperature and steam pressure as part of the input assumptions for the uncertainty calculations. The small changes in these plant parameters due to the power uprating conditions do not change the final calculated RCS flow uncertainties. Therefore, the uprating does not require changes to these uncertainties documented in WCAP-13441. Additionally, the 1.4-percent uprate does not affect the results and conclusions of the RCS flow measurement using the elbow tap methodology.

4.4 RTS/ESFAS UNCERTAINTIES

The reactor trip switchgear (RTS)/engineered safety feature actuation system (ESFAS) uncertainties that are used in the South Texas Units 1 and 2 safety analyses were evaluated for impacts associated with the 1.4-percent uprate. These include the loss of flow and steam generator water level functions. The effect of the power uprating on the loss of flow and steam generator water level uncertainties is discussed in the following subsections. All other RTS/ESFAS functions are unaffected by the uprating.

4.4.1 RCS Loss of Flow

The small changes in the plant parameters due to the power uprating conditions do not change the final calculated RCS flow calorimetric uncertainties. Therefore, the uprating does not require changes to the uncertainties for the RCS loss of flow trip function.

4.4.2 Steam Generator Water Level

The small change in nominal steam pressure due to the power uprating conditions does not change the final calculated steam generator water level channel uncertainties. Therefore, the uprating does not require changes to the uncertainties for the steam generator water level trip functions.

5 DESIGN TRANSIENTS

5.1 NUCLEAR STEAM SUPPLY SYSTEM DESIGN TRANSIENTS

The basis for the nuclear steam supply system (NSSS) design transient definitions is the work performed for the Model Δ94 steam generators (SGs).

Based upon a comparison to the work performed for the Model Δ94 steam generators, the operating conditions for the 1.4-percent uprating do not change sufficiently to require a revision to any of the primary-side or pressurizer design transients. Considering the conservatisms generally included in the design transient development (such as including minimum rod worths or minimum fuel reactivities, initial power level accounting for 2-percent power uncertainty, and conservative assumptions made in the transient definition to maximize the parameter changes), the primary-side design transients do not have to be revised.

This conclusion is also true for the secondary-side design transients. The current design transient time-history variations are still valid, based on the conservatisms discussed above.

5.2 AUXILIARY EQUIPMENT DESIGN TRANSIENTS

The review of the NSSS auxiliary equipment design transients was based on a comparison between the NSSS design parameters for the 1.4-percent power uprating described in Section 2 and the NSSS design parameters that make up the current auxiliary equipment design transients.

A review of the current auxiliary equipment transients determined that the only transients that could be potentially impacted by the 1.4-percent power uprating are those temperature transients that are impacted by the full-load T_{cold} NSSS design temperature. These transients are currently based on an assumed full-load NSSS worst-case T_{cold} of 570°F. This NSSS temperature was originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS design temperatures.

A comparison of the 1.4-percent power uprating NSSS T_{cold} design temperature range (549.8° – 560.3°F) with the T_{cold} value used to develop the current design transients indicates that the 1.4-percent power uprating design temperature range for T_{cold} is less than the value assumed to develop the design transients. Therefore, the actual temperature transients are less limiting than the current design temperature transients.

As the temperature transients dictated by the 1.4-percent power uprating conditions are less limiting than those that established the current auxiliary equipment design transients, it can be concluded that all of the applicable auxiliary equipment design transients for South Texas Units 1 and 2 still apply for the 1.4-percent power uprating conditions.

6 NUCLEAR STEAM SUPPLY SYSTEMS

This section discusses the evaluations performed on the nuclear steam supply systems (NSSSs) in support of the revised design parameters discussed in Section 2. The systems that could potentially be affected by the South Texas Units 1 and 2 1.4-Percent Power Uprate Program that are discussed in this section are the NSSS fluid systems, the NSSS/balance-of-plant (BOP) interface systems, and NSSS control systems.

6.1 NSSS FLUID SYSTEMS

6.1.1 Reactor Coolant System

The required net positive suction head (NPSH) for the reactor coolant pumps (RCPs) was determined for the Model Δ94 steam generators. Since the T_{hot} and T_{cold} values for the 1.4-percent power uprating (as discussed in Section 2) are bounded by the Model Δ94 values, the RCP NPSH requirements are not affected by the 1.4-percent power uprate.

The RCP brake horsepower (BHP) was determined for the Model Δ94 steam generators. Since the T_{hot} and T_{cold} values for the 1.4-percent uprate (as discussed in Section 2) are bounded by the Model Δ94 values, the RCP BHP requirements are not affected by the 1.4-percent uprate.

The design operating parameters for RCS temperature, pressure, and flow (as discussed in Section 2) are bounded by the Model Δ94 steam generator parameters previously evaluated. Therefore, the pressurizer bypass spray flow, pressurizer surge line performance, overpressure protection system, and chemical and volume control system/residual heat removal system (CVCS/RHRS) to reactor coolant system (RCS) interface pressure is unaffected by the 1.4-percent power uprate.

The design-basis pressurizer relief tank (PRT) performance is dependent on pressurizer level (i.e., pressurizer steam volume and mass). Since the T_{hot} and T_{cold} values for the 1.4-percent power uprate (as discussed in Section 2) are bounded by the Model Δ94 steam generator values, the PRT performance is not affected by the 1.4-percent power uprate.

The 1.4-percent power uprate does not affect the RCS heat capacity, since no mass change occurs.

The natural circulation cooldown capability is not affected because the T_{hot} and T_{cold} values for the 1.4-percent power uprate (as discussed in Section 2) are bounded by the Model Δ94 steam generator values. The Loss of External Load event that takes credit for natural circulation was analyzed with 2-percent power measurement uncertainty as discussed in Section 8.3. The use of the 2-percent power 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.

6.1.2 NSSS Auxiliary Systems Evaluation

Chemical and Volume Control System/Boron Thermal Regeneration System

The letdown line and excess letdown line in the CVCS interface with the RCS at the cold legs. Since the T_{hot} and T_{cold} values for the 1.4-percent power uprating (as discussed in Section 2) are bounded by the Model Δ94 steam generator values, the CVCS (including the boron thermal regeneration system) operating temperatures associated with the 1.4-percent power uprating are acceptable.

Boration Capability

Since the T_{hot} and T_{cold} values for the 1.4-percent uprate program (as discussed in Section 2) are bounded by the Model Δ94 steam generator values, the boration capabilities for safety-grade cold shutdown will not be impacted by the 1.4-percent uprate.

Residual Heat Removal System

The 1.4-Percent Power Uprate Program affects the RHR cooldown times. Therefore, new RHR cooldown cases to account for the 1.4-percent uprate conditions were analyzed. The 3-train system alignment case was considered. In addition, single-train RHR cooldown analyses were performed to support the worst-case scenarios for the safety-grade cold shutdown analysis and fire hazards analysis.

1. For the three-train RHR operation, the nominal design basis for RHR cooldown was modeled. The analysis determined that a total time of 9.8 hours is required to reach an RCS temperature of 150°F after reactor shutdown, which is less than the design objective of 12 hours.

2. Two modes of single-train RHR operation were considered in support of the safety-grade cold shutdown analysis.

   Case A (in support of the safety-grade cold shutdown requirements):

   For the safety-grade cold shutdown analysis, with 1 train of RHR and 1 train of CCW operating, the RCS is cooled down from 365° to 200°F. The RHR cooldown was initiated at 22 hours after reactor trip. The analysis determined that a total time of 45.2 hours is required to reach an RCS temperature of 200°F after shutdown. This result is acceptable since the requirement is to be at safety-grade cold shutdown within a reasonable period of time.

   Case B (in support of the fire hazard requirements):

   The fire hazards analysis, assumed that the RHR cooldown was initiated at 55 hours after reactor trip, with 1 train of RHR and 1 train of CCW operating, and the RCS is cooled_down from 365° to 200°F. The analysis determined that a total time of 67.8

hours is required to reach an RCS temperature of 200°F after shutdown. This result is acceptable since the requirement is to be at cold shutdown (less than 200°F) is within 72 hours.

Emergency Core Cooling System and Containment Spray System

The 1.4-percent power uprate design operating parameters for RCS temperature, pressure, and flow (as discussed in Section 2) are bounded by the Model Δ94 steam generator values. Therefore, there is no impact on the emergency core cooling system (ECCS) and containment spray system (CSS) flows previously calculated.

6.2 NSSS/BALANCE-OF-PLANT INTERFACE SYSTEMS

Five BOP fluid systems were reviewed based on the NSSS design parameters (as discussed in Section 2) to assess compliance with the NSSS/BOP interface requirements. The BOP systems evaluated are the following:

- Main steam system
- Steam dump system
- Condensate and feedwater system (C&FS)
- Auxiliary feedwater system (AFWS)
- Steam generator blowdown system (SGBS)

6.2.1 Main Steam System

The following subsections summarize the evaluation of the major steam system components relative to the proposed NSSS operating parameters. The major components of the main steam system include the steam generator safety valves (MSSVs), steam generator power-operated relief valves (PORVs), main steam isolation valves (MSIVs), and main steam isolation bypass valves.

Steam Generator Safety Valves (Main Steam Safety Valves - MSSVs)

The steam generator safety valves must have sufficient capacity to ensure that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code) for the worst-case loss-of-heat-sink event.

Each South Texas unit has 20 safety valves with a total 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, based on the NSSS design parameters (contained in Table 2.1-1), the capacity of the MSSVs will be more than adequate to satisfy the sizing criterion of the ASME Code, and is acceptable for the 1.4-percent uprate.

Since the uprate will increase the nominal NSSS power rating of the plant, the reactor trip setpoint reductions for inoperable main steam safety valve had to be re-calculated. The method used for re-calculation of these setpoint reductions is described in the Technical Specification Bases and can be found in Attachment 4 to this licensing amendment request.

Steam Generator Power-Operated Relief Valves

The primary function of the steam generator PORVs is to provide a means for decay heat removal and plant cooldown. This is accomplished by discharging steam to the atmosphere when the condenser, the circulating water pumps, or steam dump to the condenser is not available. Under these circumstances, the PORVs, in conjunction with the auxiliary feedwater system, can cool the plant down from the pressure setpoint of the lowest-set MSSV to the point where the RHRS can be placed in service.

The steam generator PORVs are sized to permit a plant cooldown to RHRS operating conditions in approximately 10 hours assuming 4 hours at hot standby with the four Δ94 steam generators available. This capacity is compatible with the capacity of the normally aligned auxiliary feedwater system water supply. An evaluation of the installed capacity (68,000 lb/hr at 100 psia) indicates that the original design basis, in terms of cooldown capability, can still be achieved over the full range of NSSS design parameters associated with the 1.4-percent power uprate. Therefore, the steam generator PORVs are adequately sized for the 1.4-percent uprate.

Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside of containment and downstream of the steam generator safety and relief valves. The MSIVs function to prevent the uncontrolled blowdown of more than one steam generator, and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the MSIVs must be capable of closure within 5 seconds of receipt of a closure signal against steam line break flow conditions in either the forward or reverse direction.

Rapid closure of the MSIVs following postulated steam line breaks causes a significant differential pressure across the valve seats, and a thrust load on the main steam system piping, and piping supports in the area of the MSIVs. The worst cases for pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), the throat area of the steam generator flow restrictors, the valve seat bore, and the no-load operating pressure. Because these variables are not affected by the 1.4-percent power uprating, the design loads and associated stresses resulting from rapid closure of the MSIVs do not change. Therefore the 1.4-percent power uprate has no significant impact on the interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steam lines and equalize the pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low-power conditions, where the 1.4-percent power uprate has no significant impact on main steam conditions (e.g., steam flow and steam pressure). Therefore, the 1.4-percent power uprating has no significant impact on the interface requirements for the MSIV bypass valves.

6.2.2 Steam Dump System

The steam dump system creates an artificial steam load by dumping steam from ahead of the turbine throttle and governor valves to the main condenser. The sizing criterion recommends that the steam dump system (valves and piping) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of the plant-rated electrical load, without a reactor trip. A steam dump capacity of 40 percent of rated steam flow at full-load steam pressure prevents the steam generator safety valves from lifting following a reactor trip from full power.

Condenser Steam Dump Valves

Each South Texas unit has 12 condenser steam dump valves. Each valve has a flow capacity of 708,576 lb/hr at a valve inlet pressure of 1,000 psia. The total valve capacity provides a steam dump capability of approximately 52.8 percent of the original maximum steam flow (16.96×10^6 lb/hr), or 8.95×10^6 lb/hr at a full-load steam generator pressure of 1,100 psia.

Operation of the NSSS within the proposed range of operating parameters at lower steam generator pressures and increased steam flows will result in a small decrease in steam dump capacity. Based on the range of NSSS operating parameters associated with the 1.4-percent power uprate, an evaluation was performed and the results confirmed that total steam dump capacity continues to meet the sizing criterion. The sizing criterion specifies that the steam dump system (valves and pumps) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of the plant-rated electrical load without a reactor trip. Therefore, the condenser steam dump capacity is adequate for 1.4-percent power uprating.

Conclusions

The capacity of the steam dump system is more than adequate for the proposed 1.4-percent power uprating.

6.2.3 Condensate and Feedwater System

The C&FS must automatically maintain steam generator water levels during steady-state and transient operations. The range of NSSS design parameters for the 1.4-percent power uprating will impact both the feedwater volumetric flow and the system pressure drop. These impacts have been evaluated, and are discussed below.

Feedwater Isolation Valves

The feedwater isolation valves (FIVs) are located downstream of the feedwater control valves (FCVs) and the feedwater bypass control valves (FBCVs). The FIVs function, in conjunction with the FCVs or the FBCVs, and the backup trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive cooldowns of the RCS. To accomplish this function, the FIVs and the backup FCVs must be capable of closure within 10 seconds after receipt of a closure signal under all operating and accident conditions. This includes a maximum flow condition with all main feedwater pumps delivering to one steam generator.

The quick closure requirements imposed on the FIVs and the backup FCVs cause potentially large dynamic pressure changes. These changes must be considered in the design of the valves and associated piping. The worst loads occur following a steam break from no-load conditions, with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these assumptions are not affected by the 1.4-percent power uprating, the design loads and associated stresses resulting from rapid closure of these valves are not impacted.

Feedwater Control Valves, Condensate, and Feedwater System Pumps

The C&FS available head, in conjunction with the FCV characteristics, must provide sufficient margin for feedwater control to ensure adequate flow to the steam generators during steady-state and transient operations. A continuous, steady feedwater flow should be maintained at all loads. To ensure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the FCVs at rated flow (100-percent power) should be approximately equal to the dynamic losses from the feedwater pump discharge through the steam generator.

For the range of NSSS design parameters associated with the 1.4-percent power uprating, the current speed control program results in a small change in the FCV pressure drop , and a corresponding small change in valve lift at 100-percent power. This small change in system hydraulics results in an absolute valve lift at full load that falls well within the acceptance range for good feedwater control. Therefore, based on the NSSS design parameters associated with the 1.4-percent power uprate, operation of the FCVs is acceptable for both steady-state and transient operations.

Conclusions

The hydraulics of the C&FS, in conjunction with the current feedwater pump speed control program, will provide acceptable feedwater control valve operation over the entire range of full-power NSSS design conditions associated with the 1.4-percent power uprating.

6.2.4 Auxiliary Feedwater System

The AFWS serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink provided by the steam generators. The system provides an alternate to the main feedwater system and/or the startup steam generator feed pump during startup, hot standby, and cooldown conditions. It also functions as an emergency feedwater system (EFS). In its EFS function, the AFWS is relied upon to prevent core damage and system overpressurization in the event of transients and accidents, such as a loss of normal feedwater or a secondary-system pipe break. The minimum flow requirements of the AFWS are determined by the safety analyses. Evaluations of the limiting transients and accidents have confirmed that the current AFWS design basis performance remains acceptable for the 1.4-percent power uprate.

Auxiliary Feedwater Storage Requirements

The AFWS pumps are normally aligned to take suction from the auxiliary feedwater storage tank (AFST). Sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

In accordance with Nuclear Regulatory Commission (NRC) Branch Technical Position RSB 5-1, the inventory in the AFST shall be sufficient to permit plant operation at hot standby for at least 4 hours, followed by a cooldown to the conditions permitting RHR operation. The inventory required for cooldown shall be based on the longest cooldown time needed with either only onsite, or only offsite power available, with an assumed single failure. In light of these design bases requirements, the South Texas Units 1 and 2 minimum AFWST contained volume of 485,000 gallons is adequate for all design-bases transient and accident conditions.

The AFST minimum contained volume of 485,000 gallons assumes that the reactor power is equal to 102 percent of rated reactor power, or 3,876 MWt. Since the power uprating of 1.4 percent is based on a reduction in calorimetric error from 2 percent to 0.6 percent, no change is required to the minimum AFST volume.

6.2.5 Steam Generator Blowdown System

The SGBS is used to control the chemical composition of the steam generator secondary-side water within specified limits. The blowdown system also controls the buildup of solids in the steam generator secondary.

The blowdown flow rates required during plant operation are based on chemistry control and tubesheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is a function of condenser in-leakage, total dissolved solids in the plant service water, allowable primary-to-secondary leakage, and the performance of the condensate polishers. Since these variables are not impacted by the 1.4-percent power uprating, the blowdown will not be impacted by the 1.4-percent power uprating.

The inlet pressure to the SGBS varies with the steam generator operating pressure. Therefore, as the steam generator full-load operating pressure decreases, the inlet pressure to the SGBS control valves decreases, and the valves must open to maintain the required blowdown flow rate into the system flash tank. The current range of NSSS design parameters allows a maximum decrease in steam pressure from no load to full load. Based on the revised range of NSSS design parameters associated with the 1.4-percent power uprating, the no-load steam pressure and the minimum full-load steam pressure remain the same. Therefore, the range of design parameters associated with the power uprating will not impact the blowdown flow control.

6.3 NSSS CONTROL SYSTEMS

Condition I transients are evaluated to confirm that the plant can respond to these transients without generating a reactor trip or engineered safety features actuation system (ESFAS) actuation.

The transients evaluated included the following:

- 10-percent step load increase
- 10-percent step load decrease
- 50-percent load rejection
- Turbine Trip without Reactor Trip below P-9

The analyses performed for the Model Δ94 steam generators were reviewed for continued acceptability for the 1.4-percent uprating.

6.3.1 Condition I Transient Evaluations

The analyses performed for the Model Δ94 steam generators were based on a nominal power level of 3,821 MWt with a power uncertainty of 2 percent. Therefore, they are also valid and bounding for the 1.4-percent uprating. Analyses were done for the limiting beginning-of-life (BOL) fuel condition and the minimum full-power steam pressure. The analyses demonstrated that there is acceptable margin to the low steam line pressure ESFAS actuation setpoint for the 1.4-percent uprating condition.

10-Percent Step Load Decrease

The analyses performed for the Model Δ94 steam generators were based on a nominal power level of 3,821 MWt with a power uncertainty of 2 percent. Therefore, they are also valid and bounding for the 1.4-percent uprating. Analyses were performed for the limiting BOL fuel condition, and for both the high and low T_{avg} and feedwater temperature conditions. The analyses demonstrated that there is acceptable margin to the pressurizer PORV actuation setpoint for a 10-percent step load decrease transient actuated from the 1.4-percent uprating condition.

50-Percent Load Rejection

The 50-percent load rejection was analyzed for the 1.4-percent uprating at a power level 2-percent higher than the nominal power level of 3,821 MWt used for the Model Δ94 steam generators , which bounds the 1.4-percent uprating. The analyses demonstrated that the 50-percent load rejection can be accommodated for the 1.4-percent uprating without challenging any of the reactor trip setpoints.

Turbine Trip without Reactor Trip below P-9

The turbine trip without reactor trip below P-9 analysis was performed to demonstrate that the control system can sustain a turbine trip from 50-percent power without actuating the pressurizer PORVs. The results of the analysis confirmed that the PORVs would not be actuated during this transient at the 1.4-percent uprate conditions.

Based on these analyses, all of the above transients can be accommodated for the 1.4-percent uprate conditions.

6.3.2 Other Considerations

The existing pressurizer pressure control component sizing is acceptable for the 1.4-percent uprated conditions. The 5-percent/minute loading/unloading, 10-percent step, and 50-percent large-load rejection transients can be accommodated and meet the design basis requirements. 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.

7 NSSS COMPONENTS

7.1 REACTOR VESSEL STRUCTURAL EVALUATION

The South Texas Units 1 and 2 reactor vessels have been evaluated for impact due to the 1.4-percent power uprate. The 1.4-percent uprating has no effect on the results in the South Texas Units 1 and 2 reactor vessel analytical reports, since there is no change to any of the design inputs that were previously considered in the reactor vessel evaluations for the Model Δ94 steam generators.

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 Δ94 steam generators. There are no changes to any of the primary-side design transients that were considered for the Model Δ94 steam generators. The reactor vessel loss-of-coolant-accident (LOCA) loads for the Model Δ94 steam generators still apply to the 1.4-percent power uprating. 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 South Texas Units 1 and 2 reactor vessel stress reports. The South Texas Units 1 and 2 reactor vessels continue to satisfy the applicable requirements of Section III (Nuclear Power Plant Components) of the ASME B&PV Code, 1971 Edition through the Summer 1973 Addenda, in accordance with the reactor vessel design requirements.

7.2 REACTOR VESSEL INTEGRITY-NEUTRON IRRADIATION

Reactor vessel integrity is impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The neutron fluence increases resulting from the South Texas Units 1 and 2 1.4-percent power uprating have been evaluated to determine the impact on reactor vessel integrity.

The reactor vessel integrity evaluation for the 1.4-percent uprating included the following evaluations:

1. 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 uprating.

2. 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.

3. Review of the existing RT_{PTS} values to determine if the effects of the uprated fluence projections results in an increase in RT_{PTS} for the beltline materials in the South Texas Units 1 and 2 reactor vessels at the end of license (EOL) (32 EFPY).

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

Surveillance Capsule Withdrawal Schedule

The revised fluence projections considering the 1.4-percent power uprating have exceeded the fluence projections used in the development of the current withdrawal schedules for South Texas Units 1 and 2. A calculation of ΔRT_{NDT} at 32 EFPY was performed to determine if the increased fluences alter the number of capsules to be withdrawn for South Texas Units 1 and 2. This calculation determined that the maximum ΔRT_{NDT} using the uprated fluences for South Texas Units 1 and 2 at 32 EFPY is less than 100°F. Per American Society for Testing and Materials (ASTM) E185-82, these ΔRT_{NDT} values would require three capsules to be withdrawn from each unit. This is unchanged from the current withdrawal schedule. Therefore, no change is required to the current withdrawal schedules.

Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves

South Texas Units 1 and 2 are currently operating to 32 EFPY P-T limit curves, which are contained in the Technical Specifications. A review was completed of the current heatup and cooldown curve applicability dates for South Texas Units 1 and 2. This review indicates that the revised adjusted reference temperature (ART) after the 1.4-percent power uprating will be lower than that used in developing the current ART values for South Texas Units 1 and 2 at 32 EFPY. Therefore, no changes in applicability dates are required and the 32 EFPY P-T curves for the South Texas Units 1 and 2 remain valid.

Emergency Response Guideline Limits

The current peak inside surface RT_{NDT} values at EOL that were calculated are 82°F (Unit 1), and 61°F (Unit 2). The limiting material for South Texas Unit 1 was the Intermediate Shell Plate R1606-3, while the limiting material at South Texas Unit 2 was the Intermediate Shell Plate R2507-2. These values would currently (pre uprating) put South Texas Units 1 and 2 in Category I. Even though the revised fluence projections after the 1.4-percent power uprating have exceeded the fluence projections used in development of the current peak inside surface RT_{NDT} values at EOL, South Texas Units 1 and 2 will still remain in the same emergency response guideline (ERG) category.

Pressurized Thermal Shock

The pressurized thermal shock (PTS) calculations were performed for South Texas Units 1 and 2 using the latest procedures required by the Nuclear Regulatory Commission (NRC) in the PTS Rule. The calculated neutron fluence values for the 1.4-percent uprated condition for South Texas Units 1 and 2 have exceeded the current fluences. Therefore, to evaluate the effects of the 1.4-percent uprating, the 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 values using the projected uprated fluence values through 32 EFPY for South Texas Units 1 and 2.

Upper Shelf Energy (USE)

All beltline materials are expected to have a USE greater than 50 ft-lb through the end of license (EOL, 32 EFPY) as required by 10CFR50, Appendix G. The EOL (32 EFPY) USE was predicted using the EOL 1/4T fluence projection.

The revised fluence projections associated with the 1.4-percent power uprating have exceeded the fluence projections used in developing the predicted EOL USE values. However, it has only affected the 1/4T fluence by less than 1 percent. This small amount has no measurable effect on the percent decrease in USE. Therefore, the current predicted USE values for South Texas Units 1 and 2 remain valid.

Conclusions

The fluence projections associated with the 1.4-percent uprated condition, while considering actual power distributions incorporated to date, will exceed the current fluence projections used in the evaluations of withdrawal schedules, ERG category, PTS, and USE. The effect of the higher fluence values is minimal for PTS and the withdrawal schedule. As for the ERG limits and predicted EOL USE, the effect of the higher fluences is negligible. With respect to the P-T curves, the current Technical Specification curves are the original curves that were developed, and were never updated at the last capsule withdrawal. These P-T curves used a more restricted fluence that was developed without the benefit of current operating conditions (i.e., loading patterns) which tend to reduce fluences. Therefore, the uprated fluences remain lower than those used for the P-T curves in the South Texas Units 1 and 2 Technical Specifications. These curves remain valid for 32 EFPY.

It is concluded that the 1.4-percent uprating program for South Texas Units 1 and 2 will not have significant impact on the reactor vessel integrity.

7.3 REACTOR INTERNALS

The reactor internals support the fuel and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support in-core instrumentation. The changes in the reactor coolant system (RCS) temperatures produce changes in the boundary conditions experienced by the reactor internals components. Also, increases in core power may increase nuclear heating rates in the lower core plate, upper core plate, and baffle-barrel former region. This section describes the analyses performed to demonstrate that the reactor internals can perform their intended design functions at the 1.4-percent uprated conditions.

7.3.1 Thermal-Hydraulic Systems Evaluations

A key area in evaluation of core performance is the determination of the hydraulic behavior of the coolant flow and its effect within the reactor internals system. The core bypass flows are required to ensure reactor performance and adequate vessel head cooling. The rod cluster control assembly

(RCCA) scram time is affected by the flow conditions. The hydraulic lift forces are critical in the assessment of the structural integrity of the reactor internals and hold-down spring functionality. Baffle plate pressure-relief hole velocities are affected by pressure differences between the core and baffle former region.

The results of these evaluations are discussed below.

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region. The principal core bypass flows are the barrel-baffle region, vessel head cooling spray nozzles, vessel outlet nozzle gap, baffle plate cavity gap, and the thimble tubes.

The maximum calculated bypass flow is less than the bypass flow assumed for the revised design conditions associated with the 1.4-percent uprate. Therefore, the calculated bypass flow is bounded by the bypass flow assumed for the 1.4-percent uprate conditions.

Rod Control Cluster Assembly (RCCA)

An analysis was performed to demonstrate that the RCCA drop time is still within the current value of 2.8 seconds (required by the Technical Specifications) for the revised design conditions. The analysis indicated that the revised design conditions (primarily T_{cold}) will have a negligible effect on the drop time, and the time will still be less than 2.8 seconds.

Hydraulic Lift Forces and Pressure Losses

The reactor internals hold-down spring maintains a net clamping force between the reactor vessel head flange and the upper internals flange and the reactor vessel shell flange and the core barrel flange of the internals. 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 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.

Pressure-Relief Hole Velocities of the Baffle Plate

Pressure-relief hole velocities result from the radial pressure difference between the core and baffle-barrel annulus. Pressure relief holes sized for LOCA blowdown have been drilled through the baffle plates at various axial and circumferential locations. Some of this bypass flow impacts the surrounding fuel assemblies. The resulting bypass flow velocities exiting or entering the pressure relief holes did not significantly change and still meet the fuel interface requirements for the 1.4-percent uprate conditions.

7.3.2 Mechanical Evaluations

The 1.4-percent uprate conditions do not affect the current design bases for seismic and LOCA loads. Therefore, it was not necessary to re-evaluate the structural affects from the seismic operating basis earthquake (OBE) and SSE loads and the LOCA hydraulic and dynamic loads. 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.

7.3.3 Structural Evaluations

Evaluations were performed to demonstrate that the structural integrity of the reactor components is not adversely affected by the 1.4-percent uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of various components.

The core support structure components affected by the 1.4-percent uprating are discussed below. The primary inputs to the evaluations are the revised RCS temperatures (as discussed in Section 2) and the gamma heating rates. The gamma heating rates were revised, as required, to account for the 1.4-percent increase in core power.

The instrumentation port column assembly (IPCA) pressure boundary is unaffected by the 1.4-percent uprated operating conditions because the highest stresses result from reactor coolant pressure and seismic loads, which remain unchanged.

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. 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 South Texas Units 1 and 2 were unaffected by the 1.4-percent uprate conditions, because the previous analyses remain bounding.

Lower Support Plate Structural Analysis

The lower support plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid flow through the plate. The fluid flow is provided to each fuel assembly and the baffle-barrel region.

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.

Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel into which baffle plates are installed. They are supported by bolting interconnecting former plates to the baffle and core barrel.

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

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

A comparison between the current and revised design conditions shows that the original design cycle bounds T_{hot}, the core average ΔT, the peripheral power distribution, and the highest average fluid temperature along the plate. Therefore, the baffle-barrel region thermal and structural analysis results are still bounding for the revised design conditions associated with the 1.4-percent uprate.

Upper Core Plate Structural Analysis

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

The 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 from the increased gamma heating was determined as a function of the 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.

7.4 PIPING AND SUPPORTS

7.4.1 Nuclear Steam Supply System Piping

The impact of the 1.4-percent power uprating on the South Texas Units 1 and 2 existing reactor coolant loop (RCL) and pressurizer surge line analyses was evaluated. The parameters associated with the 1.4-percent power uprating were reviewed for impact on the existing analyses for the reactor coolant loop piping and the Class 1 auxiliary lines evaluation.

Since there is no significant impact on the reactor coolant loop 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 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.

7.4.2 Reactor Coolant Loop Support System

Reactor coolant loop supports are designed to support the reactor coolant equipment and piping for normal operating, seismic, and postulated accident conditions. The support structures were recently re-evaluated for the revised loading associated with the Model Δ94 steam generators for Unit 1, and are in the process of reconciliation for Unit 2 and will be completed prior to implementation of the uprate in Unit 2.

The 1.4-percent power uprating does not significantly affect any of the loads applied to the equipment supports by the primary equipment and piping. Therefore, the design basis of the supports as reconciled for the Model Δ94 steam generators does not change 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 steam generator, reactor coolant pump, 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 upon the support structures, the supports will continue to be qualified for the 1.4-percent power uprate condition.

7.4.3 Leak-Before-Break Analysis

The current leak-before-break (LBB) evaluation was performed for the primary loop piping, pressurizer surge lines, and the accumulator lines to provide technical justification for eliminating pipe rupture as the structural design basis for South Texas Units 1 and 2.

To demonstrate the elimination of RCS primary loop, pressurizer surge line, and accumulator line pipe breaks, the following objectives were met:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability
- Demonstrate margin on the applied load
- Demonstrate that fatigue crack growth is negligible

There is an insignificant change in loads due to the 1.4-percent uprating parameters. The effect of material properties due to the changes in temperature, shown in Table 2.1-1, will have a negligible impact on the LBB margins. Also, there is no significant impact on loads in the pressurizer surge line and the accumulator line LBB due to the South Texas Units 1 and 2 1.4-percent uprating.

Therefore, the existing LBB analyses conclusions remain applicable for the 1.4-percent uprating condition for South Texas Units 1 and 2.

7.5 CONTROL ROD DRIVE MECHANISMS

This evaluation determined the impact of the 1.4-percent uprate parameters relative to the current parameters evaluated, to determine if the current parameters remain bounding and applicable.

The control rod drive mechanism (CRDM) parameters are given by the cold leg data, which is the vessel inlet data, presented in Table 2.1-1. The evaluation was performed for the uprated NSSS power of 3,874 MWt (3,853 MWt core power). 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.

7.6 REACTOR COOLANT PUMPS AND MOTORS

The RCPs and RCP motors were evaluated to determine the impact of the revised RCS conditions to demonstrate that the RCP structural integrity is not adversely impacted.

7.6.1 Reactor Coolant Pump

The RCPs are located between the steam generator outlet and reactor vessel inlet in the RCL. The maximum vessel inlet (RCP outlet) temperature is 560.3°F for the 1.4-percent uprate conditions, as shown in Table 2.1-1. 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) and temperature changes (ΔTs) and the maximum pressure and temperature of a transient are less than those previously evaluated and remain bounded for the 1.4-percent uprating.

7.6.2 Reactor Coolant Pump Motor

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 Δ94 steam generator outlet temperature of 549.4°F and best-estimate flow (BEF) of 105,400, were calculated. The results show a hot-loop motor load of 7,700 HP and a cold-loop motor load of 9,870 HP. The South Texas RCP motors have a nameplate rating of 8,000 HP hot and 10,000 HP cold. Since the loads are less than the nameplate rating of the motors, no analysis was necessary for operation at the 1.4-percent uprate conditions.

The previous Model Δ94 steam generator motor evaluation was based on BEFs ranging from 105,400 gpm to 107,800 gpm. The 1.4-percent uprate BEFs range from 105,600 gpm to 107,800 gpm and are, therefore, bounded by the current BEFs. In addition, the steam generator outlet temperature of 549.4°F associated with the Model Δ94 steam generator motor evaluation has remained unchanged for the 1.4-percent uprating. Since the 1.4-percent uprated BEFs are bounded by the previously evaluated Model Δ94 steam generator BEFs, with no change in the steam generator outlet temperature, the Model Δ94 steam generator motor loads remain bounding and applicable to the 1.4-percent uprate motor loads.

Based upon the above evaluation, it is concluded that the current RCP motor evaluation is bounding for the 1.4-percent uprate condition. Therefore, the South Texas Units 1 and 2 RCP motors are acceptable for the 1.4-percent uprate condition.

7.7 STEAM GENERATORS

Evaluations of the thermal-hydraulic performance, structural integrity, and mechanical hardware have been performed to address operation at a 1.4-percent power uprate.

7.7.1 Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluation of the Model Δ94 steam generator focused on the changes to secondary-side operating characteristics at the 1.4-percent uprate conditions. The following evaluations were performed to confirm the acceptability of the steam generator secondary-side parameters.

Bundle Mixture Flow Rate

The steam flow rate increases proportionally with the 1.4-percent uprate when operating with the same T_{hot} and feedwater temperature . The steam flow decreases with a reduction in the feedwater temperature, and decreases further with reductions in both T_{hot} and feedwater temperatures due to the increased enthalpy difference. Circulation ratios increase proportionally with both the T_{hot} and feedwater temperature reductions. Since the tube bundle mixture flow rate is the product of the circulation ratio and the steam flow rate, the resulting bundle flow rate remains almost the same in all cases. The 1.4-percent uprate and the changes in these temperatures have no effect on the mixture flow in the tube bundle.

Steam Pressure

The steam pressure is affected by tube plugging, but not by a reduction in feedwater temperature. Operating at the original design value of T_{hot}, of 624.8°F, the steam pressure decreases to the level of 1048 psia with the 1.4-percent increase in power and 10-percent tube plugging. The steam pressure is still above the current valves-wide-open value of approximately 1040 psia .

Heat Flux

Average heat fluxes are inversely proportional to the heat transfer area in service, and increase proportionally with the 1.4-percent power uprate. A measure of the margin for departure from nucleate boiling (DNB) transition in the bundle is a check of the ratio of the local quality, to the estimated quality at DNB transition, or (X/XDNB).

The analysis results show that changes in the peak void fraction from the base design value case are small, indicating a minimal impact due to the 1.4-percent uprating.

Moisture Carryover

The moisture separator system design of the Model Δ94 steam generator was extensively analyzed and tested in the laboratory with a full-size unit cell test model The test was conducted per WNEP-9829, " Moisture Separator Design Report - Delta 94 Replacement Steam Generator South Texas Project Nuclear Operating Company South Texas Nuclear Plant Unit 1," Revision 0, December 1998. The moisture carryover performance of the Unit 1 Model Δ94 steam generator A, B, C, and D at 100-percent power was also measured on June 1, 2000. Data from these tests and measurements were correlated to arrive at a predicted moisture carryover of less than 0.005 percent for the 1.4-percent power uprate. The moisture carryover would be well below the 0.1-percent limit at the 1.4-percent uprate condition. This demonstrates that the 1.4-percent uprating will have no effect on the moisture separator of the steam generators.

7.7.2 Structural Integrity Evaluation

The structural evaluation focused on the critical steam generator components as determined by the stress ratios and fatigue usage.

The following discussions address the evaluations of the primary and secondary components. The mechanical repair hardware evaluations are discussed later in this section.

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 manway 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 resultant primary stresses due to design, faulted, emergency, and test conditions were unchanged from the baseline analysis values.

Evaluation of Primary-to-Secondary-Side Pressure Differential

Since the primary-to-secondary ΔP that results from the original design parameters bounds that of the 1.4-percent uprate conditions, and the design transients remain unchanged, the previous ΔP evaluation remains valid and the 1600 psi design pressure differential is not exceeded.

Evaluation of Tube Minimum Wall Thickness

Calculations were performed to establish a minimum wall thickness for the steam generator tubes. This value of minimum wall is used to develop the steam generator tube inspection and repair criteria for the units.

The calculations used to determine the required tube minimum wall thickness are unaffected by changes in the feedwater temperature. Therefore, it can be concluded that the 1.4-percent uprating will not affect the required tube minimum wall thickness.

Evaluation of Mechanical Repair Hardware

Both the "long" and "short" 11/16-inch ribbed mechanical plugs and straight-leg cable stabilizers were re-evaluated for the operating conditions and transients at the 1.4-percent uprate conditions.

Mechanical Plugs

All of the stress/allowable ratios are less than unity, indicating that all primary stress limits are satisfied for the plug shell wall between the top land and the plug end cap. The plug shell meets the Class 1 fatigue exemption requirements per NB-3222.4 of the 1989 Edition of the American Society of

Mechanical Engineers (ASME) Code, no Addenda. There is adequate friction to prevent dislodging of the plug, and there is adequate leakage resistance for the limiting steady-state and transient loadings.

Results of the analyses performed for the mechanical plug for the Model Δ94 steam generators show that both the long- and short-mechanical plug designs satisfy all applicable stress and retention acceptance criteria at the 1.4-percent power uprate conditions.

Straight-Leg Cable Stabilizer

The qualification of the stabilizer is based on the relative wear coefficients and the cross-sectional areas of the tube and stabilizer components and is independent of the primary-fluid and secondary-fluid conditions. Therefore, changing the fluid conditions in the steam generators for the 1.4-percent uprate will have no effect on the stabilizer performance. The stabilizer performance will continue to be acceptable for the Model Δ94 steam generators at the 1.4-percent uprate conditions.

Structural Evaluation Conclusions

Results of the analyses performed on the Model Δ94 steam generators show that all steam generator components continue to meet ASME Code Section III limits for the 1.4-percent uprate conditions. The primary-to-secondary pressure differential remains below the design value of 1600 psi. The tube structural minimum wall thickness limits remain acceptable. In addition, the mechanical plugs and straight-leg stabilizers remain qualified for use in the Model Δ94 steam generators.

7.8 PRESSURIZER

A review of the revised temperature parameters presented in Table 2.1-1 showed 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. Therefore, the current design transients are still applicable. Additionally, there are no changes to the pressurizer nozzle loads as a result of the 1.4-percent uprating. Therefore, it is concluded that the revised parameters would not have any impact on the pressurizer stress and fatigue analysis and that the current evaluations remain valid.

It is 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.

7.9 NSSS AUXILIARY EQUIPMENT

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks. An evaluation was performed to determine the impact that the revised design conditions will have on the 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 uprate, therefore these tanks are not affected by the 1.4-percent uprate. Additionally, the 1.4-percent uprating has no affect on the pressurizer relief tank or the volume control tank.

The revised design conditions have been evaluated with respect to the impact on the auxiliary heat exchangers, valves, pumps, and tanks. The results of this review concluded that the auxiliary equipment continues to meet the design pressure and temperature requirements, as well as the fatigue usage factors and allowable limits, which the equipment is designed for.

7.10 FUEL EVALUATION

This section summarizes the evaluations performed to determine the effect of the 1.4-percent uprating on the nuclear fuel. The core design for South Texas Units 1 and 2 is performed for each specific fuel cycle and varies according to the needs and specifications for each cycle. However, some fuel-related analyses are not cycle-specific. The nuclear fuel review for the 1.4-percent uprate evaluated the nuclear design, fuel rod design, core thermal-hydraulic design, and fuel structural integrity.

The 1.4-percent uprate fuel evaluation is applicable to the standard, V5H, and RFA fuel types. Reload-specific evaluations that confirm the loading patterns and associated fuel types utilized in future reload designs will be performed.

7.10.1 Nuclear Design

The core design criteria that are evaluated for a standard reload design have been evaluated for the 1.4-percent uprate. Adequate margin to the limits associated with all reload safety analysis parameters that are evaluated for each cycle have been confirmed by a review of recent cycles. This provides assurance that these limits will not be challenged by the 1.4-percent uprate.

Cycle-specific core designs are performed for each reload cycle to ensure that all core design and reload safety analysis parameters will be satisfied for the specific operating conditions associated with that cycle.

7.10.2 Fuel Rod Design

The fuel rod design criteria evaluated for a standard reload design have been evaluated for both a transition core and equilibrium cycle conditions for South Texas Units 1 and 2. A 1.4-percent uprated power level (3,853 MWt) was analyzed, and conservative thermal-hydraulic conditions were assumed. The current feed product (similar to Unit 1 Region 12 and Unit 2 Region 11) was assumed for all fuel in the equilibrium cycle uprated core and for feed fuel in the transition cores. Integrated fuel burnable absorber (IFBA) loadings in the range of 1X to 1.6X were evaluated. The results of these evaluations demonstrated that the fuel would be expected to meet all fuel rod design limits with margin.

7.10.3 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 robust fuel assemblies (RFAs).

The WRB-2M DNB correlation was used for the 1.4-percent uprate DNB analyses, in addition to the continued use of the Revised Thermal Design Procedure (RTDP) DNB methodology. The WRB-2M DNB correlation was used to provide DNB margin over the WRB-1 DNB correlation. The WRB-1 DNB correlation was used where the WRB-2M DNB correlation is not applicable.

To support operation at the 1.4-percent uprate conditions with the use of the WRB-2M DNB correlation, revised RTDP DNBR design limits were calculated. The DNBR design limits are calculated to confirm that the DNB design basis is met. The DNBR safety analysis limits were revised to create increased DNB margin. The DNBR portion of the core limits and the axial offset limits were unchanged, to minimize the impact on the OTΔT and OPΔT protection setpoints.

In conclusion, the DNBR analyses at the 1.4-percent uprate conditions showed that the DNB design basis continues to be met.

7.10.4 Fuel Structural Evaluation

The 17x17 XL RFA and the 17x17 XL V5H fuel assembly designs were evaluated to determine the impact of the 1.4-percent uprate on the fuel assembly structural integrity. The original core plate motions remain applicable for the 1.4-percent power uprate. Therefore, there is no impact on the fuel assembly seismic/LOCA structural evaluation. The 1.4-percent uprate has an insignificant impact on the operating and transient loads, such that there is no adverse affect on the fuel assembly functional requirements. Therefore, the fuel assembly structural integrity is not affected, and the seismic and LOCA evaluations for the 17x17 XL RFA and the 17x17 XL V5H fuel assembly designs remain applicable. This evaluation was done specifically for the 17x17 XL RFA and the 17x17 V5H fuel assembly designs, however, other fuel designs can also be used, if justified by cycle-specific evaluations.

8 NSSS ACCIDENT EVALUATIONS

8.1 LOCA HYDRAULIC FORCES

The purpose of a loss-of-coolant-accident (LOCA) hydraulic forces analysis is to generate the reactor coolant system (RCS) component hydraulic forcing functions and hydraulic loads resulting from a postulated LOCA. These forcing functions and loads are considered in the structural design of the nuclear steam supply system (NSSS) components. In general, LOCA hydraulic forces increase with an increase in RCS coolant density associated with lower RCS temperatures.

A LOCA hydraulic forces evaluation was performed for the 1.4-percent uprate design conditions, as shown in Table 2.1-1. A review of the robust fuel assembly (RFA) and the Model Δ94 steam generator LOCA forces analyses determined that these analyses assumed RCS conditions that bound the 1.4-percent uprate design. Therefore, there are no changes to methodology, results, or margin of safety with respect to LOCA hydraulic forces as a result of the 1.4-percent uprate conditions.

The 1.4-percent uprate LOCA hydraulic forces evaluation concluded that the recent RFA and Model Δ94 steam generator vessel/internals, loop, and steam generator LOCA hydraulic forcing functions remain valid for the 1.4-percent uprate design conditions.

8.2 LOCA AND LOCA-RELATED EVALUATIONS

8.2.1 LBLOCA and SBLOCA

The current licensing basis large-break LOCA (LBLOCA) and small-break LOCA (SBLOCA) analyses employ a nominal core power of 3,800 MWt. However, the licensing basis methodology includes a 2-percent calorimetric power measurement uncertainty (yielding an assumed core power 3,876 MWt) in accordance with the original requirements of the Code of Federal Regulations (CFR) 10 CFR 50, Appendix K. Consistent with the recent change to the 10 CFR 50 Appendix K Rule, South Texas Project Nuclear Operating Company (STPNOC) proposes to reduce the power measurement uncertainty to 0.6-percent, based on the use of the CROSSFLOW UFM system. The existing 2-percent uncertainty margin in the LBLOCA and SBLOCA analyses is reallocated, with 1.4 percent applied to the increase in the licensed core power level and 0.6 percent retained to account for the power measurement uncertainty. Therefore, the total core power (including uncertainties) assumed in the analyses remains at 3,876 MWt.

8.2.2 Post-LOCA Long-Term Core Cooling

The requirements of 10 CFR 50.46, Paragraph (b), Item (5), "Long-term cooling," is satisfied by concluding that the reactor will remain shut down by borated emergency core cooling system (ECCS) water contained in the RCS/sump following a LOCA. Credit for the control rods is not taken for an LBLOCA. Therefore, the borated ECCS water provided by the refueling water storage tank (RWST) and accumulators must have a boron concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady-state conditions at the initiation of a LOCA, and considers both borated and unborated fluid in the post-LOCA containment sump. The water volumes and associated boric acid concentrations are not directly affected by the 1.4-percent power uprate. The core reload licensing process will confirm that there are no required changes to these volumes and concentrations. Therefore, there is no impact on the long-term core cooling (LTCC) analysis.

8.2.3 Hot Leg Switchover

For a post-LOCA cold leg break, some of the ECCS injection into the cold leg will circulate around the top of the full downcomer and out the broken cold leg. Flow stagnation in the core and the boiling off of nearly pure water will increase the boron concentration of the remaining water. As the boron concentration increases, the boron will eventually precipitate and potentially inhibit core cooling. Thus, at a designated time after a LOCA, the ECCS configuration is switched to hot leg injection to flush the core with water and keep the boron concentration below the precipitation point. The licensing basis analysis methodology employs a 2-percent calorimetric power uncertainty, in accordance with the original requirements of 10 CFR 50, Appendix K. Consistent with the recent change to Appendix K, STPNOC proposes to reduce the power measurement uncertainty to 0.6-percent, based on the use of the CROSSFLOW UFM system. The existing 2-percent uncertainty margin in the hot leg switchover (HLSO) analysis is reallocated with 1.4 percent applied to the increase in the licensed power level and 0.6 percent retained to account for the power measurement uncertainty. The total core power (including uncertainties) assumed in the analysis is 3,876 MWt.

8.3 NON-LOCA ANALYSIS

This section addresses the impact of the South Texas Units 1 and 2 1.4-percent uprate on the non-loss-of-coolant-accident (non-LOCA) analyses presented in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR).

The non-LOCA design-basis events are documented in the South Texas UFSAR in Sections 15.1 through 15.6. Of those events, four non-LOCA events were re-analyzed. The remaining non-LOCA design basis events were evaluated, and the existing analyses were determined to be bounded by the changes. The analytical evaluations are contained in Section 8.3.1 of this report. Discussions of the analyses performed are contained in Section 8.3.2. The non-LOCA methodology used for the evaluations and analyses performed is discussed in this section. Table 8.3-1 indicates which of the events were evaluated, and which were analyzed in support of the South Texas Units 1 and 2 1.4-percent power uprate.

Table 8.3-1	Non-LOCA Design Basis Events	
UFSAR 15.1.1	Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature	Evaluated
UFSAR 15.1.2	Feedwater System Malfunctions Causing an Increase in Feedwater Flow	Analyzed
UFSAR 15.1.3	Excessive Increase in Secondary Steam Flow	Evaluated
UFSAR 15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing a Depressurization of the Main Steam System	Evaluated
UFSAR 15.1.5	Spectrum of Steam System Piping Failures Inside and Outside Containment	Hot-Full-Power (HFP) Case Analyzed
UFSAR 15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	Evaluated
UFSAR 15.2.2	Loss of External Electrical Load	Evaluated
UFSAR 15.2.3	Turbine Trip	Analyzed
UFSAR 15.2.4	Inadvertent Closure of Main Steam Isolation Valves	Evaluated
UFSAR 15.2.5	Loss of Condenser Vacuum and Other Events Causing a Turbine Trip	Evaluated
UFSAR 15.2.6	Loss of Non-Emergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)	Evaluated
UFSAR 15.2.7	Loss of Normal Feedwater Flow	Evaluated
UFSAR 15.2.8	Feedwater System Pipe Break	Evaluated
UFSAR 15.3.1	Partial Loss of Forced Reactor Coolant Flow	Evaluated
UFSAR 15.3.2	Complete Loss of Forced Reactor Coolant Flow	Evaluated
UFSAR 15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Evaluated
UFSAR 15.3.4	Reactor Coolant Pump Shaft Break	Evaluated
UFSAR 15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition	Evaluated
UFSAR 15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Analyzed

Table 8.3-1	Non-LOCA Design Basis Events (Cont'd)	
UFSAR 15.4.3	Rod Cluster Control Assembly Misoperation	Evaluated
UFSAR 15.4.4	Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	Evaluated
UFSAR 15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Evaluated
UFSAR 15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Evaluated
UFSAR 15.5.1	Inadvertent Operation of the Emergency Core Cooling System (ECCS) During Power Operation	Evaluated
UFSAR 15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Evaluated
UFSAR 15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Evaluated

The following discussions describe which events need to be re-analyzed and which events could be addressed by evaluation.

The following non-LOCA analyses are currently analyzed with an explicit 2-percent power measurement uncertainty. The use of a 2-percent power 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.

- Loss of External Electrical Load and/or Turbine Trip – overpressure analysis (UFSAR Sections 15.2.2 and 15.2.3)
- Loss of AC Power and Loss of Normal Feedwater (UFSAR Sections 15.2.6 and 15.2.7)
- Major Rupture of a Main Feedwater Pipe – full-power case (UFSAR Section 15.2.8)
- Single Reactor Coolant Pump Locked Rotor – overpressure, maximum cladding temperature, and maximum zirconium-water reaction analysis (UFSAR Section 15.3.3)
- Startup of an Inactive Reactor Coolant Loop (UFSAR Section 15.4.4)
- Rupture of a Control Rod Drive Mechanism Housing (UFSAR Section 15.4.8) – full-power cases
- Chemical and Volume Control System Malfunction – that increases reactor coolant inventory UFSAR Section 15.5.2)

The improved thermal power measurement accuracy eliminates the need for the full 2-percent power uncertainty assumed in the analysis. The small changes in the plant initial operating conditions resulting from the 1.4-percent uprating were evaluated, and it was determined that these analyses remain valid. As such, the results and conclusions associated with these analyses remain valid at the 1.4-percent uprated power conditions.

Analyses that do not explicitly consider a 2-percent power uncertainty, such as those that use the Revised Thermal Design Procedure (RTDP) methodology, must be evaluated or analyzed to determine the effect of the 1.4-percent power increase. An evaluation was sufficient to determine the effect that the 1.4-percent increase in nominal core power has on the following events:

- Excessive Load Increase Incident (UFSAR Section 15.1.3)
- Partial and Complete Loss of Forced Reactor Coolant Flow (UFSAR Sections 15.3.1 and 15.3.2)
- Single Reactor Coolant Pump Locked Rotor – departure from nucleate boiling (DNB) case (UFSAR Section 15.3.3)
- Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical Condition (UFSAR Section 15.4.1)
- Rod Cluster Control Assembly Misalignment (UFSAR Section 15.4.3)
- Chemical and Volume Control System Malfunction (UFSAR Section 15.4.6)
- Rupture of a Control Rod Drive Mechanism Housing – zero-power cases (UFSAR Section 15.4.8)
- Accidental Depressurization of the Reactor Coolant System (UFSAR Section 15.6.1)
- Anticipated Transients Without Scram (UFSAR Section 15.8)

The following events required an analysis to support the 1.4-percent power uprating:

- Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR Section 15.1.2)
- Spectrum of Steam Generator System Piping Failures Inside and Outside Containment (UFSAR Section 15.1.5, HFP Case)
- Turbine Trip (UFSAR Section 15.2.3)

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR Section 15.4.2)

The following events continue to be bounded by related events or are otherwise not affected by the uprating:

- Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature (UFSAR Section 15.1.1). Bounded by UFSAR Section 15.1.2.
- Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing a Depressurization of the Main Steam System (UFSAR Section 15.1.4). Bounded by UFSAR Section 15.1.5.
- Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow (UFSAR Section 15.2.1). Continues to be a non-credible event for South Texas Units 1 and 2.
- Loss of External Electrical Load (UFSAR Section 15.2.2). Bounded by UFSAR Section 15.2.3.
- Inadvertent Closure of Main Steam Isolation Valves (UFSAR Section 15.2.4). Bounded by UFSAR Section 15.2.3.
- Loss of Condenser Vacuum and Other Events Causing a Turbine Trip (UFSAR Section 15.2.5). Bounded by UFSAR Section 15.2.3.
- Reactor Coolant Pump Shaft Break (UFSAR Section 15.3.4). Bounded by UFSAR Section 15.3.3.
- Inadvertent Operation of the Emergency Core Cooling System (ECCS) During Power Operation (UFSAR Section 15.5.1). Bounded by UFSAR Section 15.5.2.

Design Operating Parameters and Initial Conditions

Design operating parameters that were used as a basis for the evaluations and analyses performed to support the 1.4-percent power uprate are given in Table 2.1-1.

For accident analyses that are performed to demonstrate that the DNB acceptance criteria are met, nominal values of initial conditions are assumed. In accordance with the RTDP methodology, uncertainty allowances on power, temperature, and pressure are considered in the convolution of uncertainties to statistically establish the DNB ratio (DNBR) limit.

For accidents that are not DNB limited, or in which RTDP is not utilized, the initial conditions assumed in the analysis include the maximum steady-state errors applied in the direction that yields the more limiting analysis results.

The only uncertainty that changed as a result of the 1.4-percent power uprate is the power measurement uncertainty, which now is ±0.6 percent. All of the other uncertainties (i.e., average RCS temperature, pressurizer pressure, and RCS flow) did not need to be revised.

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

Core Limits and Overtemperature and Overpower ΔT Setpoints

Two essential inputs to the non-LOCA safety analyses are the core thermal limits and the resulting overtemperature ΔT and overpower ΔT (OTΔT/OPΔT) setpoints.

A revised set of core thermal limits was developed due to the 1.4-percent increase in core power. It was determined that the OTΔT and OPΔT setpoints did not need to be revised based on the revised set of core thermal limits to accommodate the increased core power. The effect of the change in the core thermal limits on the non-LOCA analyses was addressed as part of the evaluations and analyses described in Sections 8.3.1 and 8.3.2.

8.3.1 Non-LOCA Events Evaluated

As shown in Table 8.3.1-1, the majority of the non-LOCA events applicable to South Texas Units 1 and 2 have been evaluated in support of the 1.4-percent power uprate. The evaluations are discussed by individual event in this section. The changes addressed were previously discussed in this section.

The following subsections provide the details of the evaluations completed for the individual events.

8.3.1.1 Excessive Load Increase Incident (UFSAR Section 15.1.3)

This transient is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. Cases are evaluated at beginning-of-life and end-of-life conditions, with and without rod control, to demonstrate that the DNB design basis is met. The transient response to this accident is relatively mild, such that the reactor stabilizes at a new equilibrium condition corresponding to conditions well above that which would challenge the DNBR limit, without generating a reactor trip.

This transient was evaluated by comparing plant conditions, conservatively bounding deviations in core power, average coolant temperature, and RCS pressure, to conditions corresponding to those required to exceed the core thermal limits. The evaluation concluded that there is sufficient margin to the core thermal operating limits in each case considered. Therefore, since the core thermal limits are not challenged, the minimum DNBR remains above the limit value for all cases. Therefore, the conclusions documented in the UFSAR remain valid.

8.3.1.2 Loss of Non-Emergency AC Power and Loss of Normal Feedwater (UFSAR Sections 15.2.6 and 15.2.7)

Both the loss of AC power and loss of normal feedwater analyses model a 2-percent power uncertainty. Since the power level assumed in the current analyses is equivalent to that based upon the uprated power of 3,853 MWt, combined with the lower uncertainty of 0.6 percent, the results of these analyses are still applicable. Therefore, these analyses support operation at the 1.4-percent uprated power conditions.

A variation of the loss of AC power/loss of normal feedwater analysis is performed in support of the auxiliary feedwater system reliability evaluation presented in Appendix 10A of Chapter 10 of the UFSAR. This analysis is based directly on the loss of AC power and loss of normal feedwater analyses, which model a 2-percent power uncertainty. As such, it is also unaffected by the 1.4-percent power uprating.

8.3.1.3 Feedwater System Pipe Break (UFSAR Section 15.2.8)

The Feedwater System Pipe Break analysis models a 2-percent uncertainty on power. Since the power level assumed in the current analysis is equivalent to that based upon the uprated power of 3853 MWt combined with the lower uncertainty of 0.6-percent, the results of this analysis are still applicable. Therefore, this analysis supports operation of South Texas Units 1 and 2 at uprated power conditions.

8.3.1.4 Partial and Complete Loss of Forced Reactor Coolant Flow (UFSAR Sections 15.3.1 and 15.3.2)

The partial/complete loss of forced reactor coolant flow events may result from mechanical or electrical failure(s) in the reactor coolant pumps (RCPs). These faults may occur from an undervoltage condition in the electrical supply to the RCPs or from a reduction in motor supply frequency to the RCPs due to a frequency disturbance of the power grid. These analyses demonstrate that the minimum DNBR remains above the limit value. The limiting results are obtained at full-power conditions and occur very quickly following initiation of the event.

Since the 1.4-percent increase in core power may have an adverse effect on the minimum DNBR, an evaluation was completed for this event. The evaluation concluded that the existing statepoints for the limiting complete loss of flow event remain valid, with the exception of the nominal core heat flux. The nominal core heat flux increases due to the 1.4-percent uprating. The power statepoints, which are fractions of the nominal value, must therefore be applied to a higher nominal heat flux.

Revised statepoints that include the increased nominal heat flux were evaluated with respect to the DNBR. The analysis showed that the DNB design basis is satisfied. Therefore, the conclusions presented in the UFSAR remain valid.

8.3.1.5 Single Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break (UFSAR Sections 15.3.3 and 15.3.4)

A single RCP locked rotor event is based on the sudden seizure of an RCP impeller, or failure of the RCP shaft. A reactor trip via the low RCS flow protection function terminates this event very quickly. Since the 1.4-percent increase in core power may have an adverse effect on the minimum DNBR, an evaluation was completed to confirm that the number of rods that exceed the DNBR limit is less than that assumed in the dose analysis. The evaluation concluded that the existing statepoints for this event remain valid with the following exceptions:

- The nominal core heat flux increases due to the uprating.
- The effects of RCS loop flow asymmetry.

Revised statepoints that include the increased nominal heat flux and flow asymmetry penalty were evaluated with respect to the rods-in-DNB limit. It was found that the amount of rods-in-DNB will not exceed 10 percent that is assumed in the dose analysis.

The case completed to confirm that the RCS pressure criterion is met was not re-analyzed, since it currently models a 2-percent power uncertainty, which is equivalent to modeling the reduced uncertainty of 0.6 percent, combined with the 1.4-percent uprated power level. As such, the RCS pressure criterion continues to be met for the locked rotor event.

The analysis of the locked rotor event conservatively bounds the reactor coolant pump shaft break event presented in UFSAR Section 15.3.4.

8.3.1.6 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (UFSAR Section 15.4.1)

This event is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of one or more RCCA banks, resulting in a rapid power excursion. This transient is promptly terminated by the power range neutron flux - low setpoint reactor trip. Due to the inherent thermal lag in the fuel pellet, heat transfer to the RCS is relatively slow. The purpose of the analysis is to demonstrate that the minimum DNBR remains above the limit value.

The rod withdrawal from subcritical (RWFS) event occurs from a subcritical core condition with the RCS at no-load temperature conditions. The limiting case occurs when the reactivity insertion is terminated by the power range low reactor trip. The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10 percent increase is assumed for the power range low flux trip setpoint raising it from the nominal value of 25 percent to 35 percent. However, since the power range neutron flux low setpoint of 35 percent is not changing, the power level (in MWt) at which the plant trips during the event will be slightly higher than in the current analysis. The power level increases at a very rapid rate in the RWFS analysis, such that the delay in reaching 35 percent of the uprated power versus 35 percent of the current power would be on the order of milliseconds. This magnitude of delay (i.e., a small timing change) would have an insignificant effect on the results of the analysis. Therefore, the existing statepoints for the RWFS event remain valid, with the exception of the nominal core heat flux. The nominal core heat flux increases due to the 1.4-percent uprating. The power statepoints, which are fractions of the initial value, must therefore be applied to a higher nominal core heat flux.

The effects of RCS loop flow asymmetry were also considered for this event.

Revised statepoints that include the increased nominal core heat flux and flow asymmetry penalty were evaluated with respect to DNBR. The analysis showed that the DNB design basis is satisfied. Therefore, the conclusions presented in the UFSAR remain valid.

8.3.1.7 Rod Cluster Control Assembly Misalignment (UFSAR Section 15.4.3)

The RCCA misalignment analysis includes the following events:

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

The dropped RCCA transients (including the dropped RCCA bank) were previously analyzed using the methodology described in WCAP-11394, "Methodology for the Analysis of the Dropped Rod Event," and were reviewed to demonstrate that the DNB design basis is met.

The methodology described in WCAP-11394 involves the use of generic statepoints for the dropped rod event. Sensitivity studies on the effect of a power increase on the generic statepoints were previously performed for a 4-loop plant. The studies quantified the effect of an ~5-percent increase in power on the 4-loop generic statepoints, and found that the statepoints were still applicable for use at the uprated conditions. Since the uprating is much smaller (1.4 percent) than the uprate (~5 percent) used in the sensitivity studies, the generic statepoints also continue to be applicable. Although the statepoints are unaffected, the increase in nominal heat flux must be addressed with respect to the calculated DNBR. An evaluation of the DNB design basis using the generic statepoints and increased nominal heat flux confirmed that the DNB design basis continues to be met. Therefore, the conclusions presented in the UFSAR remain valid.

8.3.1.8 Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature (UFSAR Section 15.4.4)

The Startup of an Inactive Reactor Coolant Loop analysis models a 2-percent uncertainty on power. Since the power level assumed in the current analysis is equivalent to that based upon the uprated power of 3853 MWt combined with the lower uncertainty of 0.6-percent, the results of this analysis are still applicable. Therefore, this analysis supports operation of South Texas Units 1 and 2 at uprated power conditions.

8.3.1.9 Chemical and Volume Control System Malfunction (UFSAR Section 15.4.6)

The chemical and volume control system (CVCS) malfunction (resulting in a boron dilution) event was analyzed to demonstrate that the operator has at least 15 minutes to terminate an RCS dilution before a complete loss of shutdown margin occurs. The critical parameters in the determination of the time available to terminate the dilution include the overall RCS active volume, the dilution flow rate, and the initial and critical boron concentrations. The analysis did not explicitly model or consider the initial power level.

An evaluation of the Mode 1 analysis was performed and showed that the 1.4-percent power increase has an insignificant impact on the automatic reactor trip time used in the analysis. Since the reactor trip time assumed in the analysis is still valid, the results of the Mode 1 analysis also remain valid. With respect to the Modes 2 through 6 analyses, the increase in power does not affect the results of these analyses, since the reactor is not at full power. Therefore, the conclusions documented in the UFSAR remain valid.

8.3.1.10 Rupture of a Control Rod Drive Mechanism Housing (UFSAR Section 15.4.8)

The rupture of a control rod drive mechanism housing event models the power range neutron flux setpoints, which have not been changed for the 1.4-percent uprate conditions. Therefore, it was necessary to confirm that the event acceptance criteria continue to be met. The event is the result of the assumed mechanical failure of a control rod mechanism pressure housing, such that the RCS would eject the control rod and drive shaft to the fully withdrawn position. The transient responses for the hypothetical RCCA ejection event are analyzed at beginning and end-of-life, for both full-(HFP) and zero-(hot zero power (HZP)) power operation, in order to bound the entire fuel cycle and expected operating conditions. The analyses were performed to show that the fuel and cladding limits are not exceeded. Since this study is not performed to evaluate the minimum DNBR, the RTDP methodology is not utilized (the limiting fuel rod is conservatively assumed to undergo DNB very early in the transient, thus maximizing fuel temperature response).

The HFP analysis is performed at 102 percent of licensed core power. As such, the increase in core power, combined with the reduction in the power uncertainty, is bounded by the current assumption in the analysis.

The HZP analysis is unaffected, since it is performed at 0-percent power. A change in the 100-percent power value does not change the results.

The effect of the power increase on the reactor trip time was also considered. The trip setpoint modeled in these analyses is 35 percent and 118 percent for the HZP and HFP cases, respectively. The power level increases at a very rapid rate in this analysis, such that the delay in reaching 35 percent of the 1.4-percent uprated power, versus 35 percent of the current power, would be on the order of milliseconds. This magnitude of delay (i.e., a small timing change) would have an insignificant effect on the results of the analysis. The initial heat flux utilized in the analysis is based upon 102 percent of core power, and the normalized curve is applied to an initial heat flux that is equivalent to the 1.4-percent uprated power level.

Based upon the results of this evaluation, the fuel pellet enthalpies remain below 225 cal/gm for unirradiated fuel, and 200 cal/gm for irradiated fuel. In addition, the maximum amount of fuel melted at the hot spot remains less than 10 percent. Therefore, the conclusions presented in the UFSAR remain valid.

8.3.1.11 Inadvertent Operation of Emergency Core Cooling System and CVCS Malfunction that Increases RCS Inventory (UFSAR Sections 15.5.1 and 15.5.2)

As identified in the UFSAR, the CVCS malfunction event that results in an increase in the RCS inventory, bounds the inadvertent operation of the ECCS. A malfunction of the CVCS results in the inadvertent injection of borated water, which could lead to filling the pressurizer water-solid. The CVCS malfunction analysis is performed to ensure that the RCS pressure boundary is not breached, and that the fuel design limits are not exceeded. These criteria are met by demonstrating that the pressurizer does not become water-solid, and that the DNBR limit is not exceeded. The analysis takes credit for the pressurizer level high trip and operator action within ten minutes to secure the malfunction. Neither the pressurizer level high trip nor the operator action time is changing as a result of the 1.4-percent power uprate.

A 2-percent power uncertainty is assumed in the current licensing-basis CVCS malfunction analysis. Since the initial power level assumed in the current analysis is equivalent to that based on the uprated power of 3,853 MWt, combined with the lower uncertainty of 0.6 percent, the results of this analysis support the 1.4-percent uprated power conditions.

8.3.1.12 Accidental Depressurization of the Reactor Coolant System (UFSAR Section 15.6.1)

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. The purpose of the analysis is to demonstrate that the minimum DNBR remains above the limit value.

An evaluation was performed on the limiting case to assess the impact of the 1.4-percent increase in core power. Sensitivity analyses were completed to confirm that the DNBR limit continues to be met for this event. The results of the evaluation show that the minimum DNBR remains above the applicable limit value, and that the conclusions currently presented in the UFSAR remain valid.

8.3.1.13 Anticipated Transients Without Scram (UFSAR Section 15.8)

For Westinghouse designed PWRs, the implementation of anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) is a requirement of the Final ATWS rule, 10 CFR 50.62(b). South Texas Units 1 and 2 have installed AMSAC and, therefore, meet the requirements of 10 CFR 50.62(b). The AMSAC will continue to be operable at South Texas Units 1 and 2 at the 1.4-percent uprated power conditions in compliance with requirements of the Final ATWS

Rule. The current AMSAC design for South Texas Units 1 and 2 is based on the Logic 1 generic AMSAC design for Westinghouse pressurized water reactors (PWRs) as described in WCAP-10858P-A, Revision 1, and was approved for implementation at South Texas Units 1 and 2 via U.S. Nuclear Regulatory Commission (NRC) safety evaluation report (SER), NUREG-0781, Supplements 6 and 7. The South Texas Units 1 and 2 will maintain and operate AMSAC consistent with the AMSAC design as specified in WCAP-10858-A, Revision 1 and approved by the NRC for the 1.4-percent power uprating. Therefore, no specific evaluation of AMSAC or plant-specific ATWS-related analyses are considered necessary to support operation at the 1.4-percent uprated power conditions.

8.3.2 Non-LOCA Events Analyzed

As shown in Table 8.3.1-1, four of the non-LOCA events applicable to South Texas Units 1 and 2 have been re-analyzed in support of the 1.4-percent power uprate. Each of the re-analyses specifically models the increased power level. The events that were analyzed and the subsections that contain the details of the analysis are as follows:

- Section 8.3.2.1 — Feedwater System Malfunctions Causing an Increase in Feedwater Flow (UFSAR Section 15.1.2)
- Section 8.3.2.2 — Spectrum of Steam System Piping Failures Inside and Outside Containment (UFSAR Section 15.1.5)
- Section 8.3.2.3 — Turbine Trip (UFSAR Section 15.2.3)
- Section 8.3.2.4 — Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR Section 15.4.2)

8.3.2.1 Feedwater System Malfunctions Causing an Increase in Feedwater Flow

This event results from an increase in primary-to-secondary heat transfer caused by an increase in feedwater flow, that can result in the primary-side temperature and pressure decreasing significantly. The negative moderator and fuel temperature reactivity coefficients, and the actions initiated by the reactor rod control system can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of a reactor protection system (RPS) reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits.

An increase in feedwater flow can be caused by a failure in the feedwater control system, that leads to the simultaneous full opening of the feedwater control valves. At power, this excess flow causes a greater load demand on the primary side due to increased subcooling in the steam generator. With the plant at zero-power conditions, the addition of relatively cold feedwater may cause a decrease in primary-side temperature, and, therefore, a reactivity insertion due to the effects of the negative moderator temperature coefficient.

Transients initiated by increases in feedwater flow are attenuated by the thermal capacity of the primary and secondary sides. If the increase in reactor power is large enough, the primary RPS trip functions (e.g., high neutron flux, OTΔT, OPΔT) will prevent any power increase that can lead to a DNBR less than the safety analysis limit value. The RPS trip functions may not actuate, if the increase in power is not large enough.

The analysis presented herein is for a feedwater system malfunction that causes an increase in feedwater flow event as discussed in UFSAR Section 15.1.2. The feedwater system malfunction that causes a reduction in feedwater temperature event discussed in UFSAR Section 15.1.1 continues to be bounded by the excessive increase in secondary steam flow event and was not re-analyzed for the 1.4-percent uprating.

The maximum feedwater flow to one steam generator due to a control system malfunction that causes the feedwater control valves to fail in the full-open position is assumed. Cases with and without automatic rod control initiated at hot full-power conditions were considered in support of the 1.4-percent uprating. The licensing-basis analysis also addresses cases that are initiated at HZP conditions, but these are not impacted by an increase in the nominal full-power rating. Therefore, the conclusions of the feedwater malfunction analysis at HZP conditions continue to remain valid for the 1.4-percent uprating.

The results of the analysis show that the minimum DNBR calculated is above the safety analysis limit value for an excessive feedwater addition at power. Therefore, the DNB design basis is met. With regard to the RCS and main steam system (MSS) overpressure criteria, this event is bounded by the turbine trip analysis documented in Section 8.3.2.3.

Since all applicable acceptance criteria have been satisfied, the failure of any of the feedwater control valves will not challenge the RCS and MSS pressure boundaries, nor will the integrity of the fuel cladding be compromised due to DNB.

8.3.2.2 Spectrum of Steam System Piping Failures Inside and Outside Containment

Steam system piping failures, such as ruptures, result in steam being discharged from the steam generators. This escaping steam causes an increase in steam flow, which results in an increase in the heat extraction rate and a consequential reduction in primary-system temperature and pressure. Due to the negative moderator temperature coefficient and fuel temperature reactivity feedback at end of cycle conditions, the core reactivity increases, as the primary coolant temperature decreases. If no automatic or manual actions are taken, the core power will eventually rise to a level that corresponds to the increased steam flow rate.

The main steam line rupture event is analyzed at conditions consistent with both zero- and full-power conditions. The zero-power case is analyzed using non-statistical DNB methods, assuming a double-ended guillotine rupture of the main steam line on one steam generator. Uncontrolled steam releases could also result from the inadvertent opening of a steam generator relief valve, steam generator safety valve, or steam dump valve. The zero-power case is analyzed to demonstrate that any return to power resulting from the uncontrolled steam release, does not result in a violation of the DNB design basis.

Based on the fact that the zero-power steam line rupture event is analyzed using non-statistical DNB methods, and that it is analyzed from a shutdown condition, the analysis results are not impacted by the 1.4-percent power uprate. Therefore, the licensing-basis zero-power steam line rupture analysis presented in Section 15.1.5 of the South Texas UFSAR remains valid. Additionally, the results of the licensing-basis inadvertent opening of a steam generator relief or safety valve event presented in Section 15.1.4 of the South Texas UFSAR are bounded by the results of the zero-power double-ended rupture of a main steam line.

The hot-full-power steam line rupture event is analyzed to demonstrate that the RPS provides sufficient protection to prevent both DNB and fuel centerline melting resulting from an overpower condition initiated by an uncontrolled steam release. The full-power steam line rupture event is analyzed over a spectrum of break sizes to identify the limiting break size that results in the worst overpower condition. The limiting break is typically identified as the largest break size that would not result in a reactor trip on low steam line pressure, but instead relies on the OPΔT reactor trip. Any break larger than this would result in an earlier reactor trip on low steam line pressure, and subsequently, a less limiting overpower condition.

The full-power steam line rupture event is analyzed using statistical DNB methodology. Therefore, it was specifically re-analyzed for the South Texas 1.4-Percent Power Uprate Program. While the results of the analysis are not presented in the UFSAR, both the calculated minimum DNBR and peak fuel-rod power have been confirmed to meet the applicable acceptance criteria.

8.3.2.3 Turbine Trip

8.3.2.3.1 Introduction/Event Description

The analysis for this event was analyzed as a turbine trip from full-power conditions. This event bounds the following events:

- A loss of external electrical load (UFSAR Section 15.2.2)
- An inadvertent closure of the main steam isolation valves (UFSAR Section 15.2.4)
- A loss of condenser vacuum (UFSAR Section 15.2.5)
- Other events causing a turbine trip

With respect to pressure effects, the turbine trip event is more limiting than any other partial or complete loss-of-load event, since it results in the most rapid reduction in steam flow. This causes the most limiting increase in pressure and temperature in the RCS and the MSS, due to the very rapid decrease in secondary steam flow.

The analysis conservatively assumes that the reactor trip is actuated by the RPS, and not by the turbine trip signal. This assumption is made because the UFSAR analysis is performed to show that the RPS signals are capable of providing a reactor trip in sufficient time following the event initiation, to satisfy the acceptance criteria for the event, and to conservatively bound the other events listed above.

For this event, the reactor may be tripped by any of the following RPS trip signals:

- OTΔT
- Pressurizer high pressure

In the event that the steam dump valves fail to open following a large loss of load, the sudden reduction in steam flow results in an increase in pressure and temperature in the steam generator secondary side. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise. This causes coolant expansion, a pressurizer insurge, and a rise in RCS pressure. Throughout the event, power is available for the continued operation of plant components, such as the RCPs.

Unless the transient RCS response to the turbine trip event is terminated by manual or automatic action, the resultant reactor coolant temperature rise could eventually result in DNB and/or the resultant pressure increases could challenge the integrity of the reactor coolant pressure boundary, or the main steam system pressure boundary. To avoid the potential damage that might otherwise result from this event, the RPS is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value, and before the RCS and/or MSS pressures exceed the values at which the integrity of the pressure boundaries would be jeopardized.

The major challenges associated with the turbine trip are overpressurization of the RCS and MSS, and possible fuel cladding damage resulting from the increase in RCS temperature.

The transient responses for a turbine trip from full-power conditions are presented in the UFSAR as two cases: one case for the scenario with pressurizer pressure control, and a second case without pressurizer pressure control.

The results of this analysis demonstrate that: 1) the fuel design limits are maintained by the RPS, since the DNBR is maintained above the limit value, and 2) the plant design is such that a turbine trip presents no hazard to the integrity of the RCS or the MSS pressure boundary. Therefore, all ANS Condition II acceptance criteria are satisfied.

As stated earlier in this section, the turbine trip analysis bounds several other events that are applicable to South Texas Units 1 and 2. Specifically, the analysis documented in this section bounds a loss of external electrical load (UFSAR 15.2.2), an inadvertent closure of main steam isolation valves (UFSAR 15.2.4), and a loss of condenser vacuum event (UFSAR 15.2.5), and other events causing a turbine trip.

8.3.2.4 Uncontrolled RCCA Bank Withdrawal at Power

An uncontrolled RCCA bank withdrawal at power that causes an increase in core heat flux may result from an operator error, or a malfunction in the rod control system. Immediately following the initiation of the accident, the steam generator heat removal rate lags behind the core power generation rate, until the steam generator pressure reaches the setpoint of the steam generator relief or safety valves. This imbalance between heat removal and heat generation rate causes the reactor coolant temperature to rise. Unless terminated, the power mismatch and resultant coolant temperature rise could eventually result in DNB and/or fuel centerline melt. Therefore, to avoid damage to the core, the RPS is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value, or the fuel rod linear heat generation rate (kw/ft) limit is exceeded.

The automatic features of the RPS that prevent core damage for an RCCA bank withdrawal event at power include the following:

1. The power range neutron flux instrumentation initiates a reactor trip on neutron flux if two-out-of-four channels exceed an overpower setpoint.

2. A reactor trip is initiated if any two-out-of-four channels in any two loops exceed a rate lag setpoint on the high positive neutron flux rate.

3. A reactor trip is initiated if any two-out-of-four ΔT channels in any two loops exceed an OTΔT setpoint. This setpoint is automatically varied with the axial power distribution, coolant average temperature, and coolant average pressure to protect against DNB.

4. A reactor trip is initiated if any two-out-of-four ΔT channels in any two loops exceed an OPΔT setpoint. This setpoint is automatically varied with the coolant average temperature, so that the allowable heat generation rate (kw/ft) is not exceeded.

5. A high pressurizer pressure reactor trip, is initiated if any two-out-of-four pressure channels, exceeds the setpoint. This reactor trip on high pressurizer pressure occurs at a pressure that is less than the set pressure for the pressurizer safety valves.

6. A high pressurizer water level reactor trip is initiated if any two-out-of-four level channels exceed the setpoint.

The high neutron flux, OTΔT, high positive flux rate, and high pressurizer pressure reactor trip functions provide adequate protection over the entire range of potential reactivity insertion rates. The minimum value of DNBR is always greater than the safety analysis limit value, and the RCS and MSS are maintained below 110 percent of the design pressures. Therefore, the results of the analysis show that an uncontrolled RCCA bank withdrawal at power does not adversely affect the core, the RCS, or the MSS.

8.4 STEAM LINE BREAK EVALUATION

The licensing-basis safety analyses related to the steam line break mass and energy releases were evaluated to determine the effect of a 1.4-percent power uprating for South Texas Units 1 and 2. The evaluation determined that the NSSS design parameters for South Texas, as shown in Table 2.1-1, remain unchanged or bounded by the current safety analysis values.

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

The critical parameters for the long-term steam line break event include the following conditions on the primary and secondary sides: NSSS power level, reactivity feedback characteristics including the minimum shutdown margin, the initial value for the steam generator water mass, main feedwater flow, auxiliary feedwater flow, main and auxiliary feedwater enthalpies, and the times at which steam line and feed line isolation occur. The input assumptions related to these critical parameters dictate the quantity and rate of the mass and energy releases.

The power increase of 1.4 percent will be offset by an equivalent reduction in the calorimetric uncertainty. The Analyses of Record applicable to both units for the inside containment long-term steam line breaks assume a 2-percent power calorimetric uncertainty added to the NSSS power of 3,821 MWt. A minimum 0.6-percent power calorimetric uncertainty applied to a maximum 1.4-percent power increase, is equivalent to the licensing-basis safety Analyses of Record. Therefore, as long as the sum of the power increase and power calorimetric uncertainty does not exceed 2 percent, there is no impact on either the current licensing basis long-term steam line break mass and energy release analyses or the UFSAR conclusions.

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

The critical parameters for the short-term steam line break event are defined at no-load conditions. At this power level, the steam generator pressure is high. Also, the steam generator inventory is greatest at no-load conditions. Since the 1.4-percent power uprating does not affect the no-load thermodynamic conditions, there is no effect on either the current licensing basis analysis or the UFSAR conclusions.

The critical parameters for the short-term steam line break in the IVC are defined at no-load conditions. At these conditions, the steam generator pressure is high, as is the critical mass flow rate. In addition, the steam generator inventory is greatest at no-load conditions. Since the 1.4-percent power uprating does not affect the no-load thermodynamic conditions, there is no impact on the current licensing-basis analysis or the UFSAR conclusions.

The bounding analysis for the MSLB forcing functions used for the design of the main steam piping supports and restraints is based on full-power steam conditions at 1,100 psia and 556.6°F. The full-power main steam conditions are 1,057 psia and 551.4°F (see Table 2.1-1) associated with the 1.4-percent uprating. Since the 1.4-percent power uprating conditions are less limiting than the values used in design calculations, the results of the original calculations remain bounding.

8.4.3 Short-Term Steam Generator Blowdown Line Mass and Energy Releases Outside of Containment

The critical parameters for the short-term steam generator blowdown line break are defined at no-load conditions. At these conditions, the steam generator pressure is high, as is the critical mass flow rate. Since the 1.4-percent power uprate does not affect the no-load thermodynamic conditions, there is no effect on the current licensing-basis analysis or the UFSAR conclusions.

8.5 POST-LOCA CONTAINMENT HYDROGEN GENERATION

An evaluation of the hydrogen generation in containment following a LOCA for South Texas Units 1 and 2 was performed based on updated parameters that reflect current plant conditions.

Without recombination, a containment concentration of 3.5 v/o is reached in 4.2 days and a containment concentration of 4.0 v/o is reached in 6.3 days.

Assuming the operation of a single recombiner at 90-percent efficiency and recombination beginning at 24 hours after LOCA, the peak concentration is 3.73 v/o. If recombination begins when the hydrogen concentration is 3.50 v/o, the peak containment concentration is 3.95 v/o, which is reached at about 14–17 days after a LOCA.

The 1.4-percent uprate has no impact on the post-LOCA containment hydrogen generation, since the hydrogen concentration remains below the 4.00 v/o limit specified in Regulatory Guide 1.7.

8.6 LOCA MASS AND ENERGY RELEASES

8.6.1 Long-Term LOCA Mass and Energy Release Analysis

This analysis demonstrates the ability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA. The methodology for the most limiting LOCA mass and energy release calculation for the Model Δ94 steam generators is contained in WCAP-10325-P-A up to the point of steam generator depressurization. After this point, the methodology described in STP Nuclear Operating Company letter to the NRC of September 29, 1998 and endorsed by the NRC in a Safety Evaluation, dated May 20, 1999, to Amendments 110 and 97 to Facility Operating Licenses NPF-76 and NPF-80 respectively is used.

Based on this methodology, the Analysis of Record presently assumes a core thermal power of 3,876 MWt. This value is the current licensed core power of 3,800 MWt, plus an additional 2-percent power measurement uncertainty. The improved thermal power measurement accuracy obviates the need for the full 2-percent power margin assumed in the analysis and reduces it to 0.6 percent.

The power measurement margin is one of many conservative assumptions used in the analysis. Taken together, the improved power measurement uncertainty and the other conservative assumptions provide substantial conservatism such that the margin of safety would not be reduced for the 1.4-percent uprate.

8.6.2 Short-Term LOCA Mass and Energy Release Analysis

Short-term LOCA mass and energy release calculations are performed to support the reactor cavity and loop subcompartment pressurization analyses. These analyses are performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) that accompanies a LOCA within the region.

South Texas Units 1 and 2 have been approved for Leak Before Break methods. The only break locations that need to be considered are the pressurizer spray line and the RHR line from the hot leg to the first isolation valve. The analysis inputs that may potentially change with the uprate are the initial RCS fluid temperatures. Since the critical portion of this event last for less than 3 seconds, the single effect of reactor power is not significant.

The critical flow correlation used in the mass and energy releases for this analysis will provide an increase in the mass and energy release for a slighty lower fluid temperature. For the uprate conditions, the RCS cold leg temperature remains the same as the current analysis. Therefore, the mass and energy releases for cold leg breaks are not impacted. However, the hot leg temperature increases approximately 0.7°F. This increase in hot leg temperature will result in a lower mass and energy release from hot leg breaks, which will result in lower compartment pressures. Therefore, the current licensing basis remains bounding for the short-term LOCA sub-compartment pressurization analysis.

8.7 Steam Generator Tube Rupture Analysis

The analysis for the steam generator tube rupture event is used to demonstrate that the ruptured steam generator will go not go water-solid, and that the offsite does remains below the 10 CFR part 100 limits. The analysis for the water-solid condition was performed assuming an initial condition of 102-percent power. This assumption will bound the proposed 1.4-percent power uprating when considering the improved calorimetric uncertainty measurement. The analysis for the offsite dose considers the mass of steam released as a result of the event, and the source term. The analysis for the mass of steam release was also performed assuming an initial condition of 102 percent. The source term for this analysis was developed assuming an initial power level of 4,100 MWt. Therefore, the current analysis for the steam generator tube rupture event will bound the proposed power uprate of 1.4 percent.

9 ELECTRICAL POWER

9.1 ELECTRICAL DISTRIBUTION SYSTEM

The normal distribution system, supplying power to the ESF buses for startup and normal operation, is distributed through the associated ESF bus transformers.

The electrical distribution system connects the ESF buses to the unit auxiliary transformer and standby auxiliary transformers.

Additionally, there is another offsite power source, the 13.8 kV emergency transformer, capable of supplying power to one ESF bus of each unit.

The onsite standby power supply consists of three independent stand-by diesel generators (SBDGs) for each unit.

The onsite electrical power supplies include the Class 1E battery system. For each unit, this system consists of four independent separated buses. Each bus is energized by one of the two available battery chargers and one battery.

None of the above systems are impacted by the 1.4-percent uprating. Therefore, it is concluded that the ESF distribution system is not affected by the 1.4-percent uprating.

9.2 TURBINE GENERATOR

The capability of the turbine generators to perform at the 1.4-percent uprated power conditions (3,853 MWt) was evaluated. The review included the throttles, high-pressure and low-pressure turbines, the generators and exciters, as well as associated auxiliary equipment including moisture separator reheaters (MSRs) and relief valves. All turbine-generator components have sufficient margin to support operation at the 1.4-percent uprated power conditions.

No changes to the equipment protection relay settings for the turbine generator are required for the 1.4-percent uprate.

The effect of the uprated steam conditions results in a small increase in the LP turbine disc temperatures when compared to the original disc design operating temperatures. This increase in the LP turbine disc temperatures results in a small increase in the turbine missile probability. However, the turbine maintenance program will ensure that the turbine missile generation requirement of less than 1E-04 will still be met for the 1.4-percent uprating.

Based on a revised heat balance, the 1.4-percent uprating will result in a generator gross output power of approximately 1,344 MWe for South Texas Units 1 and 2. This increased power output is within the turbine generator nameplate rating of 1,505 Mva @ a 0.9 power factor (PF) therefore, the turbine generator will continue to operate at the 1.4-percent uprated power level.

9.3 ISOPHASED PHASE BUSES

The isophased bus' main section is forced-air-cooled and is rated at 25 kV, 36,600 amps. The maximum current in the main generator terminals assuming the maximum rated generator power output of 1504.8 MVA at 23.75 kV (95 percent of 25kV), is 36,581 amps, which is less than the isophase bus' rated permanent current carrying capability. The main generator is not going to be modified for the 1.4-percent uprating. Since the maximum generator output current that can be reached after the 1.4-percent uprating will not exceed the rated values, the isophase buses will support the 1.4-percent uprating.

9.4 MAIN GENERATOR CIRCUIT BREAKER

The main generator circuit breaker is rated at 37,500 amps continuous at 25 kV. The expected main generator continuous current will not exceed 36,581 amps after the 1.4-percent uprate, which is less than the main generator circuit breaker continuous rating. Therefore the main generator circuit breaker is capable of carrying the current associated with the 1.4-percent uprate.

9.5 MAIN TRANSFORMERS

The size and features of the transformers in the step-up bank of Unit 1 (700/784 MVA and 650/728 MVA in parallel) and Unit 2 (two 700/784 MVA transformers in parallel), are adequate to deliver the electric output power supplied by the turbine-generator set of Units 1 and 2, after the 1.4-percent power uprating. This assumes a generator continuous real power output of 1,348 MW (based on a maximum winter output for the 1.4-percent uprating), and the corresponding reactive power, according to the generator capability curve.

9.6 SWITCHYARD

The South Texas Units 1 and 2 switchyard has eight 345 kV transmission lines connecting the 345 kV switchyard to the electrical system. The transmission lines are capable of accepting the additional electrical load associated with the 1.4-percent uprate.

9.7 345 KV GRID STABILITY

Steady-state and transient-stability studies have been performed to demonstrate that the offsite power system is in compliance with BTP-ICSB11.

These studies show that the loss of both South Texas Units 1 and 2 does not endanger the ability of the system to supply power to the engineered safety features (ESF) electrical system. These studies further demonstrate that the loss of external transmission circuits does not jeopardize the supply of power to the ESF electrical system.

A evaluation of the load flow and transient stability analysis was performed. It was determined that there is sufficient margin to accommodate the 1.4-percent uprating without impacting the grid stability.

10 BALANCE OF PLANT

The South Texas Units 1 and 2 balance-of-plant (BOP) systems were reviewed for impact due to the 1.4-percent power uprating to 3,853 MWt reactor core power. The BOP systems that could potentially be impacted due to the 1.4-percent uprating are the turbine and main generator, main steam and reheat steam, steam dump, steam generator 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 from a thermal-hydraulic, equipment, piping, and instrumentation and control (I&C) review of the design basis calculations.

The conclusions of the BOP systems evaluation are as follows:

1. Based on the uprated heat balance for the 1.4-percent uprating, the BOP plant systems can accommodate the 1.4-percent uprating within the existing design margins of the original system designs.

2. There will be no changes in engineered safeguard features (ESF) system requirements.

3. The existing BOP plant system components are adequate for the 1.4-percent uprating and continue to comply with all their original design requirements.

4. The feedwater temperature and flow measurement equipment has been modified to support the 1.4-percent uprating. Steam pressure instruments will be modified to Rosemount transmitters to support the 1.4-percent uprate.

5. No increased flows are required for any intermediate cooling systems or heat sinks, including the ultimate heat sink. This includes the residual heat removal (RHR) shutdown cooling, and spent fuel pool cooling. The increased heat load can be accommodated within the existing designs of these cooling systems.

6. New electrical loads were generated based on the uprated brake horsepower requirements for the pumps. The evaluation has determined that the change in electrical loads is minor and is bounded by the design capacity of the distribution system. The station auxiliary loads will remain within the existing design loads.

Based on the above, it is concluded that the 1.4-percent uprating of South Texas Units 1 and 2 to 3,853 MWt reactor core power is acceptable, and can be accommodated with the existing BOP plant systems, while maintaining continued compliance with all design requirements.

11 RADIOLOGICAL CONSEQUENCES

The current licensed core power level is 3,800 MWt. In accordance with the guidance of Regulatory Guide 1.49 to address possible instrument error in determining the power level, post-accident radiological analyses were performed to demonstrate compliance with 10 CFR 100 should be based on at least 1.02 times the proposed licensed core level. However, power levels used in STP analyses have ranged from 3,800 MWt to 4,100 MWt. An evaluation was performed to determine the power levels used in the various radiological analyses and to determine the impact of the proposed 1.4 percent power uprating.

For analyses that were performed at 3800 MWt, it was determined that the proposed 1.4-percent increase in assumed power level would cause changes in the results that are within the calculational error of the analysis. The proposed uprate has negligible impact on the radiological analyses for STP.

11.1 NORMAL OPERATION ANALYSES

Radiation Source Terms

The NSSS power uprate to 3,853 MWt will increase the concentration of fission products in the primary and secondary coolant by approximately 1.4 percent. The expected source terms, which are generated based on the power level, will increase by approximately the same percentage. The technical specification source terms will not change since:

- The power uprate increases the concentration of tritium and fission products in the coolant, but does not significantly impact the mix of radionuclides.
- The source terms are calculated based on normalization to the I-131 dose equivalent, which is fixed via the Technical Specifications.

The impact of a change in the assumed reactor power on the plant's radiological analyses is generally in proportion to the power change. The power change will increase the production of fission products. As related to the design basis, this increase will then be reflected in proportional increases in plant system isotopic inventories, plant dose rates, normal releases and the resulting offsite doses, and the doses resulting from postulated accidents.

Gaseous and Liquid Releases

The prediction of normal releases from the plant is based upon some plant-specific design features, parameters, and assumptions on operations. The NRC GALE computer code (NUREG-0017) is then used to determine the expected offsite releases. The assumptions used in this generic code have at least as important an impact on the predicted release values as a 1.4 percent power increase. The effect of the proposed 1.4-percent increase in power is within the expected error of this type of analysis. Therefore, the proposed uprating would have a negligible impact on the prediction of normal releases.

The impact of the proposed power increase on the resultant population doses is also negligible. The dose analyses use the projected normal plant releases as input. In addition, assumptions are made on population distribution, eating habits (amounts and types of food), and recreational habits (swimming, sunbathing, etc). Again, the accuracy of these assumptions has at least as important an impact on the predicted dose values as a 1.4-percent power increase. The effect of the proposed 1.4-percent increase in power is within the expected error of this type of analysis. Therefore, the proposed uprating would have a negligible impact on the prediction of normal doses.

The analyses to determine the environmental impact of an accidental liquid release on the area groundwater were performed at 3,800 MWt. These analyses assume the liquid contents of a large tank in a radioactive system are spilled on the ground. The radionuclides are then assumed to diffuse though the earth to reach the groundwater. The resultant groundwater isotopic concentrations would be expected to increase by 1.4 percent due to the proposed power uprating. However, given the assumptions on isotopic migration through the earth, this increase is within the error of the analyses themselves. Therefore, the proposed uprating would have a negligible impact on the groundwater isotopic concentrations determined in this manner.

In addition, release concentrations and offsite doses are controlled by the STP Offsite Dose Calculation Manual (ODCM). This manual provides the information and methodologies to be used at STP to assure compliance with the Administrative Controls Section (i.e., 6.8.3g) of the Technical Specifications. Compliance with these controls further ensures both the accuracy and reliability of effluent dose calculations, and effluent alarm setpoint calculations.

Shielding

The plant's shielding analyses were primarily done at 3800 MWt and 1-percent failed fuel. An increase of power by 1.4 percent would, theoretically, increase plant area dose rates by 1.4 percent. Typical shielding analyses make use of several assumptions on the photo interaction properties of materials, on geometric arrangement of components, buildup factors, scattering surface areas and albedos. The isotopic inventory of the component to be shielded is also based on assumptions on plant operation. The dose rate increases associated with the proposed power uprating is within the error of the calculations themselves. Therefore, the proposed uprating would have a negligible impact on the plant's shielding design.

Normal Operation Analyses - Summary

Based on the discussions provided above, an NSSS power uprating to 3,853 MWt will not cause radiological exposure in excess of the dose criteria (for restricted and unrestricted access) provided in the current 10 CFR 20. From an operations perspective, radiation levels in most areas of the plant are expected to increase no more than the percentage increase in power level. Individual worker exposures will be maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. Gaseous and liquid effluent releases are also expected to increase by no more than the percentage increase in power level. Offsite release concentrations and doses will be maintained within the limits of the current 10 CFR 20 and 10 CFR 50, Appendix I by the site Radwaste Effluent Control Program.

11.2 ACCIDENT ANALYSES

The radiological analyses for the UFSAR Chapter 15 accidents were evaluated for the 1.4-percent power uprating. The radiological source terms for all the analyses (except Small Line Failure Outside the Containment, for UFSAR Chapter 15.6.2) were determined at 4,100 MWt. Also, the reactor coolant system and secondary-side parameters used in the analyses bound those for the proposed power uprating. The steam release rates to the environment were also evaluated and found to bound those for the proposed power uprate.

The proposed 1.4-percent power uprating is bounded by the existing analyses for the following accidents:

- Main Steam Line Break (UFSAR Chapter 15.1.5)
- Reactor Coolant Pump Shaft Seizure (UFSAR Chapter 15.3.3)
- Control Rod Ejection (UFSAR Chapter 15.4.8)
- Steam Generator Tube Rupture Accident (UFSAR Chapter 15.6.3)
- LOCA (including Control Room, and Technical Support Center) (UFSAR Chapter 15.6.5)
- Fuel Handling Accident (UFSAR Chapter 15.7.4)

The analysis for the Small Line Failure Outside the Containment (UFSAR 15.6.2) considers pre-existing and concurrent iodine spikes. As such, the analysis is based upon the Technical Specification limits for iodine in the reactor coolant system and in the secondary-side systems. The noble gas concentrations in these systems corresponding to 1-percent failed fuel were developed at 3,800 MWt. A 1.4-percent increase in the noble gas concentrations in the reactor coolant system and in the secondary-side systems would result in a proportional increase in the contributions of the noble gases to the offsite whole body and skin doses. The contribution from the iodines would not be impacted. Overall, however, the doses would remain a small fraction of the guideline values of 10 CFR 100.

The analysis that estimates doses to operators carrying out post-accident duties specified in the emergency operating procedures was reviewed for this effort. An increase of power by 1.4 percent would, theoretically, increase operator action doses by 1.4 percent. The analyses assume operator travel paths, walking times, and times required to perform postulated actions. The dose increase associated with the proposed power uprating is within the error of the calculations themselves. Therefore, the proposed uprating would have negligible impact on the operator action doses.

11.3 EQUIPMENT QUALIFICATION

The analyses were performed in one of three manners:

- Doses based on calculated radiation fields from an assumed reactor power of 4,100 MWt
- Doses determined from a radiation area zone based on operation at 3,800 MWt
- Component-specific doses based on 3,800 MWt, used to determine if a component's dose is below some specified limit (e.g., 10^5, 10^6 rads, etc.)

The analyses performed at 4100 MWt bound the case for the 1.4-percent uprating. The equipment doses determined on the basis of post-accident area dose rates are acceptable for use with the 1.4-percent uprating. The doses would increase by 1.4 percent if they were to be recalculated at the new power. However, this increase is within the error of the analyses themselves. Therefore, the proposed uprating would have negligible impact on the doses determined in this manner.

The component-specific analyses were examined and it was determined that a 1.4-percent increase in the resultant doses would not change the conclusion of the analyses (e.g., the revised dose would remain below the applicable limit in the calculation). Therefore, these analyses remain correct for the proposed uprating.

12 PLANT OPERATIONS

12.1 PROCEDURES

Plant procedures will not require significant changes for the power uprate. Procedural limitations on power operation 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 the power uprating will be treated in a manner consistent with any other plant modification.

Procedures required for the operation and maintenance of the CROSSFLOW system have been implemented.

Specific operator actions to be taken when the Crossflow system is inoperable are discussed in Section 3.3 and will be addressed in procedural guidance.

12.2 EFFECT ON OPERATOR ACTIONS AND TRAINING

ESF System design and setpoints, and procedural requirements already bound the proposed 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, controls and displays for a 1.4-percent uprating. The CROSSFLOW system will have alarms in the control room to alert operators to conditions that impair its availability or accuracy. No other alarm impacts are expected. It is not anticipated that any existing alarms will be modified or deleted. Alarms will be recalibrated as necessary to reflect small setpoint changes. However, no significant or fundamental setpoint 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 Technical Specification 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.

12.3 QUALIFIED DISPLAY PROCESSING SYSTEM

Process parameter scaling changes will be made, as required, to the Qualified Display Parameter System (QDPS). There are no other impacts to the QDPS from the 1.4-percent uprate.

13 OTHER EVALUATIONS

13.1 10CFR50, APPENDIX R

The fire hazards/cold shutdown analysis cases 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 uprating to 3,853 MWt, with allowance for up to 0.6-percent uncertainty. Since reactor coolant system (RCS) temperatures are approximately the same as those previously analyzed, and since the decay heat is bounded, there is no adverse impact on the fire hazards/cold shutdown analysis due to the 1.4-percent uprating.

13.2 ENVIRONMENTAL QUALIFICATION

The 1.4-percent uprating has no impact on the environmental qualification criteria. 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.

13.3 STATION BLACKOUT

There are no changes required for station blackout coping and mitigation due to the 1.4-percent power uprating. The condensate inventory requirements are bounded by the current station blackout analysis. All other associated parameters are either independent of the core power level, or have margin available to accommodate the 1.4-percent power increase.

13.4 PIPE BREAK EVALUATION

All fluid systems that experienced a change in energy (pressures, temperatures, or flow rates) due to the 1.4-percent uprating were reviewed to evaluate the consequences of a pipe break or crack. These systems include the main steam system, the feedwater system, and the steam generator 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 nuclear steam supply system (NSSS) temperatures contained in Table 2.1-1.

The 1.4-percent uprating will have a negligible impact on the pipe stress. Therefore, no pipe stress re-analysis was required for the 1.4-percent power uprating.

The 1.4-percent (3,853 MWt) power uprating 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 loss-of-coolant accident (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.

13.5 FLOW-ACCELERATED CORROSION

South Texas Project Nuclear Operating Company (STPNOC) has developed a surveillance program for ensuring the integrity of piping systems that may be subject to single-phase or two-phase flow erosion-corrosion induced wall thinning. The Flow-Accelerated Corrosion (FAC) Program identifies, monitors, and mitigates the effects of degradation in carbon steel and alloy steel piping systems based on an accepted industry methodology and NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants."

Susceptible safety-related and non-safety-related systems are modeled at South Texas using the Electric Power Research Institute's (EPRI's) CHECWORKS software. CHECWORKS models were revised to incorporate flow and process system conditions that are determined for the 1.4-percent power uprate conditions. The results of these upgraded models will be factored into future surveillance/pipe repair plans.

13.6 SAFETY-RELATED MOTOR OPERATED VALVES

The inputs discussed in NRC Generic Letters 89-10 and 96-05 regarding motor operated valve thrust and torque requirements calculations and discussed in NRC Generic Letter 95-07 regarding motor operated valve pressure locking thermal binding requirement calculations are based on the following:

1. Safety related pump shutoff head's,
2. Valve and tank elevations,
3. Tank pressurization values,
4. Safety and relief valve set points,
5. Reactor Coolant System pressure and temperature limits during Residual Heat Removal System operations,
6. Pressure/ temperature calculations for various accident scenarios.

A review of (1) through (6) listed above concluded that the 1.4% uprate will not require any changes to the parameters listed. The pressure/ temperature calculations for various accident scenarios are not effected by the 1.4% power up-rate since these calculations used conservative inputs that bound the inputs for the uprate. Therefore, the 1.4% power uprate will not impact the motor operated valve calculations discussed in the NRC Generic Letters 89-10, 95-07 or 96-05.

13.7 IMPACT ON PROBABILISTIC SAFETY ASSESSMENT RESULTS

The proposed power uprate has the potential to affect several areas in the South Texas Project Probabilistic Risk Analysis (PRA). These areas are:

- Initiating Event Frequency
- System Success Criteria
- Operator Recovery Timing
- Fission Product Inventory

Initiating Event Frequency

The likelihood of occurrence of an initiating event is not significantly affected as a result of the proposed power uprate and is bounded by the uncertainty in the initiating event frequency.

System Success Criteria

The success criteria for the systems modeled in the PRA are based on UFSAR criteria or specific calculations (e.g., station black-out, room heat-up, steam generator boil-off, etc,) performed to support alternative criteria. Calculated system success criteria will be reevaluated to ensure criteria assumed in the PRA are still valid. Based on available system margins, no changes in system success criteria are expected as a result of the proposed power uprate.

Operator Recovery Timing

Operator actions to recover from potential core damaging scenarios are included in the PRA where appropriate. The time available to perform these actions is based on the particular scenario and the equipment available. As a result of the proposed power uprate, the calculations that support these operator actions will be reevaluated. Because of the uncertainty in the operator actions included in the PRA, no change in likelihood of operator success or failure is expected.

Fission Product Inventory

The Level II analysis included in the South Texas Project PRA is based on the fission product inventory of the present core. The containment release categories will be slightly affected by the proposed power uprate. Large Early Release fractions will be recalculated, however, based on the uncertainties in the Large Early Release results and the small change in rated power, no significant change in Large Early Release frequency is expected and the change will remain below current NRC acceptance criteria.

13.8 SPENT FUEL POOL COOLING SYSTEM

The primary function of the spent fuel pool cooling system (SFPCS) is to maintain the SFP and the in-containment storage area (ICSA) water temperatures below prescribed limits by removing decay heat generated by the stored spent fuel. The spent fuel pool cooling system licensing limits are discussed in the UFSAR Section 9.1.3. The limiting calculations for water temperature and gamma heating on the concrete walls were performed with initial conditions at 102-percent power, 40,000 effective full-power hours (EFPH) of burnup (>60,000 MWD/MTU), and a 105°F CCW temperature. These conditions envelop the 1.4-percent power uprate conditions with reduced uncertainty. Therefore, the current results are bounded by the assumptions used in the calculation.

The SFP boiling dose analysis is discussed in UFSAR Section 9.1.3.3.4. The analysis assumes that a complete loss of SFP cooling occurs 100 hours after shutdown with a decay heat load of 8×10^7 Btu/hr. This assumption exceeds the end-of-life maximum SFP decay heat load at 102-percent power by more than 25 percent. Using these assumptions, the dose consequences of spent fuel pool boiling are well below the requirements of 10 CFR 100. Therefore, the 1.4-percent power uprate conditions with the reduced uncertainty are bounded by the assumptions used in the current analysis.

Attachment 7

Westinghouse Letter,
"Application for Withholding Proprietary Information from Public Disclosure"
(CAW-01-1466)
with Affidavit CAW-01-1466

Attachment 9

WCAP-15696, Revision 0,
"Power Calorimetric for the 1.4% Uprating
for the
South Texas Project Nuclear Operating Company
Units 1 and 2"
(Non-Proprietary)

Westinghouse Non-Proprietary Class 3

WCAP - 15696
Revision 0

Power Calorimetric for the 1.4% Uprating for South Texas Project Nuclear Operating Company Units 1 and 2

Westinghouse Electric Company LLC

WCAP-15696

POWER CALORIMETRIC FOR THE 1.4 % UPRATING FOR SOUTH TEXAS PROJECT NUCLEAR OPERATING COMPANY UNITS 1 AND 2

June, 2001

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POWER CALORIMETRIC FOR THE 1.4 % UPRATING

INSTRUMENT UNCERTAINTY METHODOLOGY

I. INTRODUCTION

The purpose of this analysis is to determine the uncertainty in the Daily Power Calorimetric for the 1.4% Uprating. This report assumes the use of CROSSFLOW to measure feedwater flow and that the plant is at or near 100% RTP. Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) at least once every 24 hours. The Daily Power Calorimetric uncertainty must be a value sufficiently small enough to account for the increase in nominal operating power without exceeding the 102% RTP initial power assumption for Condition 3 and 4 events.

Westinghouse has been involved with the development of several techniques to treat instrumentation uncertainties. An early version used the methodology outlined in WCAP-8567 "Improved Thermal Design Procedure",[1,2,3] which is based on the conservative assumption that the uncertainties can be described with uniform probability distributions. Another approach is based on the more realistic assumption that the uncertainties can be described with random, normal, two sided probability distributions.[4] This approach is used to substantiate the acceptability of the protection system setpoints for many Westinghouse plants, e.g., Millstone Unit 3, Diablo Canyon, Farley and others. The second approach is now utilized for the determination of all instrumentation uncertainties for the RTDP parameters and protection functions.

II. METHODOLOGY

The methodology used to combine the error components for a channel is the square root of the sum of the squares (SRSS) of those groups of components that are statistically independent. Those uncertainties that are dependent are combined arithmetically into independent groups, which are then systematically combined. The uncertainties used are considered to be random, two sided distributions. This technique has been utilized before as noted above, and has been endorsed by the NRC staff [6,7,8,9] and various industry standards [10,11].

The relationships between the error components and the channel instrument error allowance are variations of the basic Westinghouse Setpoint Methodology [12] and are based on STPNOC Units 1 & 2 specific procedures and processes and are defined as follows:

For parameter indication utilizing the plant process computer:

$$\begin{aligned} CSA = \{ & (PMA)^2 + (PEA)^2 + (SMTE+SCA)^2 + (SPE)^2 + (STE)^2 + (SRA)^2 + \\ & (SMTE + SD)^2 + (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + \\ & (COMPMTE + COMPCAL)^2 + (COMPTE)^2 + (COMPMTE + COMPDRIFT)^2\}^{1/2} + \\ & BIAS \end{aligned}$$

Eq. 1

For parameter indication utilizing the control board indication:

$$\begin{aligned} CSA = \{ & (PMA)^2 + (PEA)^2 + (SMTE+SCA)^2 + (SPE)^2 + (STE)^2 + (SRA)^2 + \\ & (SMTE + SD)^2 + (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + \\ & (INDMTE + INDCAL)^2 + (INDMTE + INDDRIFT)^2 + (INDREAD)^2 \}^{1/2} + \\ & BIAS \end{aligned}$$

Eq. 2

where:

CSA	=	Channel Statistical Allowance
PMA	=	Process Measurement Accuracy
PEA	=	Primary Element Accuracy
SRA	=	Sensor Reference Accuracy
SCA	=	Sensor Calibration Accuracy
SMTE	=	Sensor Measurement and Test Equipment Accuracy
SPE	=	Sensor Pressure Effects
STE	=	Sensor Temperature Effects
SD	=	Sensor Drift
RCA	=	Rack Calibration Accuracy
RMTE	=	Rack Measurement and Test Equipment Accuracy
RTE	=	Rack Temperature Effects
RD	=	Rack Drift
COMPCAL	=	Plant Computer Calibration Accuracy
COMPTE	=	Plant Computer Temperature Effects
COMPDRIFT	=	Plant Computer Drift
COMPMTE	=	Plant Computer Measurement and Test Equipment Accuracy
INDCAL	=	Indicator Calibration Accuracy
INDDRIFT	=	Indicator Drift
INDMTE	=	Indicator Measurement and Test Equipment Accuracy
INDREAD	=	Indicator Readability

Many of the parameters above are defined in Reference 12 and are based on ANSI/ISA 51.1-1979 (Reaffirmed 1993)[13]. However, for ease in understanding they are paraphrased below:

- PMA - non-instrument related measurement errors, e.g., temperature stratification of a fluid in a pipe
- PEA - errors due to a metering device, e.g., elbow, venturi, orifice
- SRA - reference (calibration) accuracy for a sensor/transmitter
- SCA - calibration tolerance for a sensor/transmitter
- SMTE - measurement and test equipment used to calibrate a sensor/transmitter
- SPE - change in input-output relationship due to a change in static pressure for a differential pressure (d/p) cell.
- STE - change in input-output relationship due to a change in ambient temperature for a sensor or transmitter
- SD - change in input-output relationship over a period of time at reference conditions for a sensor or transmitter
- RCA - calibration accuracy for all rack modules in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy
- RMTE - measurement and test equipment used to calibrate rack modules
- RTE - change in input-output relationship due to a change in ambient temperature for the rack modules
- RD - change in input-output relationship over a period of time at reference conditions for the rack modules
- COMPCAL - calibration accuracy for plant computer in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy
- COMPDRIFT - change in input-output relationship over a period of time at reference conditions for the plant computer
- COMPTE - change in input-output relationship due to a change in ambient temperature for the plant computer
- COMPMTE - measurement and test equipment used to calibrate plant computer
- BIAS - a one directional uncertainty for a sensor/transmitter or a process parameter with a known magnitude

INDCAL - Indicator Calibration Accuracy

INDDRIFT - change in input-output relationship over a period of time at reference conditions for the indicator

INDMTE - measurement and test equipment used to calibrate the indicator

INDREAD - Readability is based on ½ of the smallest increment on an indicator

A more detailed explanation of the Westinghouse methodology noting the interaction of several parameters is provided in Reference 12.

III. INSTRUMENTATION UNCERTAINTIES

The Reactor Power Measurement algorithm will be discussed first, followed by the results of the power calorimetric calculations.

Reactor Power Measurement

The daily power measurement assumes the measurement of the feedwater flow using the CROSSFLOW system. The results of this measurement are used to adjust the feedwater flow venturi measurement as indicated in the plant process computer.

Assuming that the primary and secondary sides are in equilibrium; the core power is, in general, determined by summing the thermal output of the steam generators, correcting for appropriate heat additions and losses, and dividing by the core Btu/hr at rated full power. The equation for this calculation is:

$$RP = \frac{\{(\sum Q_{SG}) + Q_L - Q_P\}(100)}{H} \qquad \text{Eq. 3}$$

Where:

RP	= Core power (% RTP)
Q_{SG}	= Steam generator thermal output (BTU / hr)
Q_P	= heat additions (BTU / hr)
Q_L	= net heat losses (BTU / hr)
H	= Rated core power (BTU / hr).

For the purposes of this uncertainty analysis (and based on H noted above) it is assumed that the plant is at 100 % RTP when the measurement is taken. Measurements performed at lower power levels will result in different uncertainty values.

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

$$Q_{SG} = (h_s - h_f)*W_f - (h_s - h_{bd})*W_{bd} \qquad \text{Eq. 4}$$

Where:

h_s	=	Steam enthalpy (BTU/lb)
h_f	=	Feedwater enthalpy (BTU/lb)
h_{bd}	=	Steam generator blowdown enthalpy (BTU/lb)
W_f	=	Feedwater flow (lb/hr)
W_{bd}	=	Steam generator blowdown flow (lb/hr)

The Steam enthalpy is based on the measurement of Steam Generator outlet Steam pressure assuming wet steam. The Feedwater enthalpy is based on the measurement of Feedwater temperature and Feedwater pressure. Blowdown enthalpy is based on the measurement of Steam Generator outlet steam pressure assuming wet steam.

The loop feedwater flow is effectively determined by the CROSSFLOW system and the following calculation:

$$W_f = (C_f)(A_p)(\rho_{fw})(L/\Delta t) \qquad \text{Eq. 5}$$

Where:

W_f	= Feedwater loop flow (lb/hr)
C_f	= CROSSFLOW velocity profile correction factor
A_p	= Cross sectional area of pipe flow path
ρ_{fw}	= Feedwater density (lb/ft^3)
L	= Length of pipe between transducer points
Δt	= Time required for signature to travel length of L

- The feedwater flow profile correction factor is the product of a number of constants including as-built dimensions of the CROSSFLOW and calibration tests performed by the vendor.
- Feedwater density is based on the measurement of feedwater temperature and feedwater pressure.
- The pipe length between transducer points is a fixed value once the CROSSFLOW system is installed.
- Time required for signature to travel between transducers is obtained from the CROSSFLOW electronics.

The power measurement uncertainties are thus based on the following plant measurements:

- Steamline pressure (P_s)
- Feedwater temperature (T_f)
- Feedwater pressure (P_f)
- Steam generator blowdown
- Feedwater flow (W_f) (from the CROSSFLOW system)

and on the following calculated values:

- Feedwater density (ρ_f)
- Feedwater enthalpy (h_f)
- Steam enthalpy (h_s)
- Moisture carryover (affects h_s)
- Primary system net heat losses (Q_L)
- RCP heat addition (Q_p)

Uncertainties

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

For the measurement of feedwater flow, the CROSSFLOW allowance is [$]^{+a,c}$. Since the calculated steam generator thermal output is proportional to feedwater flow, the flow coefficient uncertainty is expressed as [$]^{+a,c}$.

An allowance of [$]^{+a,c}$ was used for the Steam Generator Blowdown orifice plate flow coefficient. This resulted in an uncertainty of [$]^{+a,c}$ power.

The uncertainty applied to the Steam Generator Blowdown orifice plate thermal expansion correction (F_a) is based on the uncertainties of the temperature and the coefficient of thermal expansion for the orifice plate material, assumed to be 304 stainless steel. For this material, a change of ± 1.0 °F in the nominal temperature range changes F_a by [$]^{+a,c}$ but the change in steam generator thermal output is negligible.

An uncertainty of 5.0 % in F_a for 304 stainless steel is used in this analysis. This results in an additional uncertainty bounded by [$]^{+a,c}$ power. This allowance is included to account for the variations in material composition that could exist for the orifice plate.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 1 notes the instrument uncertainties for the hardware used to perform the parameter measurements. Table 2 lists the various parameter sensitivities. As can be seen on Table 2, both feedwater temperature uncertainties and feedwater pressure uncertainties have an effect on feedwater density and feedwater enthalpy.

Steam Generator Blowdown orifice plate d/p uncertainties are converted to % Steam Generator Blowdown flow using the following conversion factor:

$$\% \text{ flow} = (\text{d/p uncertainty})(1/2)(\text{transmitter span} / 100)^2. \qquad \text{Eq. 6}$$

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

The net pump heat addition uncertainty is derived from the combination of the primary system net heat losses and pump heat addition and are summarized for the South Texas Project as follows:

System heat losses	- 2.0 MWt
Component conduction and convection losses	- 1.4 MWt
Pump heat adder	+ 23.4 MWt
Net Heat input to RCS	+ 21.0 MWt

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

The calorimetric power measurement determination is performed using the plant computer or a manual calculation. For purposes of this calculation, the plant computer uncertainties are conservative and bounding for a manual calorimetric that uses a digital voltmeter, (DVM), at the process racks. Also, a conservative allowance has been assigned for feed pressure uncertainties for those times when feed pressure may be inferred from steam pressure. Steam generator blowdown uncertainties are based on the use of the control board indicators. As noted in Table 3, Westinghouse has determined the dependent sets in the calculation and the direction of interaction.

Using the power uncertainty values noted on Table 3, the 4-loop uncertainty equation is as follows:

+a,c

Power = [ ] Eq. 7

Where:

CF = Feedwater Flow (mass flow accuracy of the CROSSFLOW system)

$SGBF_{\Delta P}$ = Steam Generator Blowdown flow Delta P

$SGBF_V$ = Steam Generator Blowdown flow orifice plate (basic accuracy)

ρ_t = Feedwater flow density (as a function of temperature)

h_t = Feedwater flow enthalpy (as a function of temperature)

Fa_t = Steam Generator Blowdown flow F_a (as a function of temperature, inferred from steam pressure)

Fa_m = Steam Generator Blowdown flow F_a (as a function of material)

ρ_p = Feedwater flow density (as a function of pressure)

h_p = Feedwater flow enthalpy (as a function of pressure)

h_{SP} = Steam enthalpy (as a function of pressure)

$h_{s\ moist}$ = Steam enthalpy (as a function of moisture)

h_{SG_LIQ} = Steam Generator Blowdown flow enthalpy (as a function of steam pressure)

ρ_{SG_P} = Steam Generator Blowdown flow density (as a function of steam pressure)

NPHA = Net pump heat addition

N = Number of primary side loops

Power = [ ] + a,c

Based on the number of loops and the instrument uncertainties for the four parameters, the uncertainty for the secondary side power calorimetric measurement is:

# of loops	power uncertainty (% RTP)
4	[ ] +a,c

TABLE 1
POWER CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

	FW TEMP	FW PRESS	FW FLOW	S/G BLDN	STM PRESS	
	°F	% Span ***	%Mass Flow	% Flow	%Span	
CF						+a,c
SRA						
SCA						
SMTE						
SPE						
STE						
SD						
SEISMIC						
RCA						
COMPCAL						
RMTE						
COMPMTE						
RTE						
COMPTE						
RD						
COMPDRFT						
INDCAL						
INDDRIFT						
INDMTE						
INDREAD						
CSA						
# Inst Used	3	1/Loop	1/Loop	1/Loop	1/Loop	
Units	°F	Psi	Mass Flow	% Flow	psi	
Inst Span	N/A	2500		90,000 lb/hr	1400	
Inst Unc. (Random)						+a,c
Nominal	441.8°F	1057 psia		35,000 lb/hr	957 psia	

* Provided by STPNOC

** Provided by CROSSFLOW

*** Inferred from Steam Pressure

TABLE 2
POWER CALORIMETRIC SENSITIVITIES AT 3853 MW THERMAL

+a,c

FEEDWATER FLOW	=	
FEEDWATER DENSITY		
TEMPERATURE	=	
PRESSURE	=	
FEEDWATER ENTHALPY		
TEMPERATURE	=	
PRESSURE	=	
h_s	=	
h_f	=	
Δh (SG)	=	
STEAM ENTHALPY		
PRESSURE	=	
MOISTURE	=	
SG BLOWDOWN ENTHALPY		
PRESSURE	=	
SG BLOWDOWN FLOW		
Fa		
TEMPERATURE	=	
MATERIAL	=	
DENSITY		
PRESSURE	=	
DELTA P	=	

TABLE 3
SECONDARY SIDE POWER CALORIMETRIC MEASUREMENT UNCERTAINTY

+a,c

COMPONENT	INSTRUMENT UNCERTAINTY	POWER UNCERTAINTY
FEEDWATER FLOW		
CROSSFLOW (CF)		
SG BLOWDOWN FLOW		
ORIFICE PLATE ($SGBF_V$)		
THERMAL EXPANSION COEFFICIENT		
TEMPERATURE (Fa_t)		
MATERIAL (Fa_m)		
DENSITY		
PRESSURE (ρ_{SG_P})		
DELTA P ($SGBF_{\Delta P}$)		
SG BLOWDOWN LIQUID ENTHALPY		
PRESSURE (h_{SG_LIQ})		
FEEDWATER DENSITY		
TEMPERATURE (ρ_t)		
PRESSURE (ρ_p)		
FEEDWATER ENTHALPY		
TEMPERATURE (h_t)		
PRESSURE (h_p)		
STEAM ENTHALPY		
PRESSURE (h_{sp})		
MOISTURE ($h_{s\ moist}$)		
NET PUMP HEAT ADDITION (NPHA)		
4 LOOP UNCERTAINTY		

*, * *, * * * Indicates sets of dependent parameters

IV. RESULTS/CONCLUSIONS

The preceding sections provide the methodology to account for the Power Calorimetric uncertainties for the 1.4 % Uprating. The uncertainty calculations have been performed for STPNOC Units 1 and 2 utilizing plant specific instrumentation and calibration procedures. A power calorimetric uncertainty value of [ ]$^{+a,c}$ will be used in the STPNOC Units 1 and 2 safety analysis.

V. REFERENCES

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2. Westinghouse letter NS-PLC-5111, T. M. Anderson to E. Case, NRC, dated 5/30/78.

3. Westinghouse letter NS-TMA-1837, T. M. Anderson to S. Varga, NRC, dated 6/23/78.

4. Westinghouse letter NS-EPR-2577, E. P. Rahe Jr. to C. H. Berlinger, NRC, dated 3/31/82.

5. Not Used.

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

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

8. Regulatory Guide 1.105 Rev. 3, "Instrument Setpoints for Safety-Related Systems", dated 12/99.

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

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

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12. Tuley, C. R., Williams T. P., "The Significance of Verifying the SAMA PMC 20.1-1973 Defined Reference Accuracy for the Westinghouse Setpoint Methodology", Instrumentation, Controls, and Automation in the Power Industry, June 1992, Vol.35, pp. 497-508.

13. ANSI/ISA-51.1-1979 (Reaffirmed 1993), "Process Instrumentation Terminology".

Attachment 12

WCAP-15698, Revision 0
"CROSSFLOW Out of Service Power Calorimetric Uncertainties for the South Texas Project Nuclear Operating Company Units 1 and 2"
(Non-Proprietary)

Westinghouse Non-Proprietary Class 3

WCAP - 15698
Revision 0

Crossflow Out of Service Power Calorimetric Uncertainties for South Texas Project Nuclear Operating Company Units 1 and 2

Westinghouse Electric Company LLC

WCAP-15698

CROSSFLOW OUT OF SERVICE POWER CALORIMETRIC UNCERTAINTIES FOR SOUTH TEXAS PROJECT NUCLEAR OPERATING COMPANY UNITS 1 AND 2

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POWER CALORIMETRIC INSTRUMENT UNCERTAINTY
METHODOLOGY FOR CROSSFLOW OUT OF SERVICE

I. INTRODUCTION

The purpose of this analysis is to determine the uncertainty in the Daily Power Calorimetric when the CROSSFLOW system is out of service. Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) at least once every 24 hours.

Westinghouse has been involved with the development of several techniques to treat instrumentation uncertainties. An early version used the methodology outlined in WCAP-8567 "Improved Thermal Design Procedure",[1,2,3] which is based on the conservative assumption that the uncertainties can be described with uniform probability distributions. Another approach is based on the more realistic assumption that the uncertainties can be described with random, normal, two sided probability distributions.[4] This approach is used to substantiate the acceptability of the protection system setpoints for many Westinghouse plants, e.g., Millstone Unit 3, Diablo Canyon, Farley and others. The second approach is now utilized for the determination of all instrumentation uncertainties for the RTDP parameters and protection functions.

II. METHODOLOGY

The methodology used to combine the error components for a channel is the square root of the sum of the squares (SRSS) of those groups of components that are statistically independent. Those uncertainties that are dependent are combined arithmetically into independent groups, which are then systematically combined. The uncertainties used are considered to be random, two sided distributions. This technique has been utilized before as noted above, and has been endorsed by the NRC staff [6,7,8,9] and various industry standards [10,11].

The relationships between the error components and the channel instrument error allowance are variations of the basic Westinghouse Setpoint Methodology [12] and are based on STPNOC Units 1 & 2 specific procedures and processes and are defined as follows:

For parameter indication utilizing the plant process computer:

$$\begin{aligned}CSA = \{&(PMA)^2 + (PEA)^2 + (SMTE+SCA)^2 + (SPE)^2 + (STE)^2 + (SRA)^2 + \\ &(SMTE + SD)^2 + (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + \\ &(COMPMTE + COMPCAL)^2 + (COMPTE)^2 + (COMPMTE + COMPDRIFT)^2\}^{1/2} + \\ &BIAS\end{aligned} \qquad \text{Eq. 1}$$

For parameter indication utilizing the control board indication:

$$\begin{aligned}CSA = \{&(PMA)^2 + (PEA)^2 + (SMTE+SCA)^2 + (SPE)^2 + (STE)^2 + (SRA)^2 + \\ &(SMTE + SD)^2 + (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + \\ &(INDMTE + INDCAL)^2 + (INDMTE + INDDRIFT)^2 + (INDREAD)^2\}^{1/2} + \\ &BIAS\end{aligned} \qquad \text{Eq. 2}$$

where:

CSA	=	Channel Statistical Allowance
PMA	=	Process Measurement Accuracy
PEA	=	Primary Element Accuracy
SRA	=	Sensor Reference Accuracy
SCA	=	Sensor Calibration Accuracy
SMTE	=	Sensor Measurement and Test Equipment Accuracy
SPE	=	Sensor Pressure Effects
STE	=	Sensor Temperature Effects
SD	=	Sensor Drift
RCA	=	Rack Calibration Accuracy
RMTE	=	Rack Measurement and Test Equipment Accuracy
RTE	=	Rack Temperature Effects
RD	=	Rack Drift
COMPCAL	=	Plant Computer Calibration Accuracy
COMPTE	=	Plant Computer Temperature Effects
COMPDRIFT	=	Plant Computer Drift
COMPMTE	=	Plant Computer Measurement and Test Equipment Accuracy
INDCAL	=	Indicator Calibration Accuracy
INDDRIFT	=	Indicator Drift
INDMTE	=	Indicator Measurement and Test Equipment Accuracy
INDREAD	=	Indicator Readability

Many of the parameters above are defined in Reference 12 and are based on ANSI/ISA 51.1-1979 (Reaffirmed 1993)[13]. However, for ease in understanding they are paraphrased below:

PMA -	non-instrument related measurement errors, e.g., temperature stratification of a fluid in a pipe
PEA -	errors due to a metering device, e.g., elbow, venturi, orifice
SRA -	reference (calibration) accuracy for a sensor/transmitter
SCA -	calibration tolerance for a sensor/transmitter
SMTE -	measurement and test equipment used to calibrate a sensor/transmitter
SPE -	change in input-output relationship due to a change in static pressure for a differential pressure (d/p) cell.
STE -	change in input-output relationship due to a change in ambient temperature for a sensor or transmitter
SD -	change in input-output relationship over a period of time at reference conditions for a sensor or transmitter
RCA -	calibration accuracy for all rack modules in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy
RMTE -	measurement and test equipment used to calibrate rack modules
RTE -	change in input-output relationship due to a change in ambient temperature for the rack modules
RD -	change in input-output relationship over a period of time at reference conditions for the rack modules
COMPCAL -	calibration accuracy for plant computer
COMPDRIFT -	change in input-output relationship over a period of time at reference conditions for the plant computer
COMPTE -	change in input-output relationship due to a change in ambient temperature for the plant computer
COMPMTE -	measurement and test equipment used to calibrate plant computer
BIAS -	a one directional uncertainty for a sensor/transmitter or a process parameter with a known magnitude
INDCAL -	Indicator Calibration Accuracy

INDDRIFT - change in input-output relationship over a period of time at reference conditions for the indicator

INDMTE - measurement and test equipment used to calibrate the indicator

INDREAD - Readability is based on ½ of the smallest increment on an indicator

A more detailed explanation of the Westinghouse methodology noting the interaction of several parameters is provided in Reference 12.

III. INSTRUMENTATION UNCERTAINTIES

The Reactor Power Measurement algorithm will be discussed first, followed by the results of the power calorimetric calculations.

Reactor Power Measurement

The daily power measurement assumes the measurement of the feedwater flow using the ΔP transmitters and the flow venturis in the feedwater lines because the CROSSFLOW system is out of service. This method assumes the feedwater flow venturi measurement as indicated in the plant process computer has not been previously adjusted via previous CROSSFLOW correction factors.

Assuming that the primary and secondary sides are in equilibrium; the core power is, in general, determined by summing the thermal output of the steam generators, correcting for appropriate heat additions and losses, and dividing by the core Btu/hr at rated full power. The equation for this calculation is:

$$RP = \frac{\{(\sum Q_{SG}) + Q_L - Q_P\}(100)}{H} \qquad \text{Eq. 3}$$

Where:

RP	= Core power (% RTP)
Q_{SG}	= Steam generator thermal output (BTU / hr)
Q_P	= heat additions (BTU / hr)
Q_L	= net heat losses (BTU / hr)
H	= Rated core power (BTU / hr).

For the purposes of this uncertainty analysis (and based on H noted above) it is assumed that the plant is at 100 % RTP when the measurement is taken. Measurements performed at lower power levels will result in different uncertainty values.

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

$$Q_{SG} = (h_s - h_f)*W_f - (h_s - h_{bd})*W_{bd} \qquad \text{Eq. 4}$$

Where:

h_s	=	Steam enthalpy (BTU/lb)
h_f	=	Feedwater enthalpy (BTU/lb)
h_{bd}	=	Steam generator blowdown enthalpy (BTU/lb)
W_f	=	Feedwater flow (lb/hr)
W_{bd}	=	Steam generator blowdown flow (lb/hr)

The Steam enthalpy is based on the measurement of Steam Generator outlet Steam pressure assuming wet steam. The Feedwater enthalpy is based on the measurement of Feedwater temperature and Feedwater pressure. Blowdown enthalpy is based on the measurement of Steam Generator outlet steam pressure assuming wet steam.

The loop feedwater flow is determined by the venturi measurement and the following calculation:

$$W_f = (K)(F_a)\{(\rho_{fw})(\Delta P)\}^{1/2} \qquad \text{Eq. 5}$$

Where:

W_f = Feedwater loop flow (lb/hr)

K = Feedwater venturi flow coefficient

F_a = Feedwater venturi correction for thermal expansion

ρ_{fw} = Feedwater density (lb/ft^3)

ΔP = Feedwater venturi pressure drop (inches H_2O)

The feedwater venturi flow coefficient is the product of a number of constants including as-built dimensions of the venturi and calibration tests performed by the vendor. The thermal expansion correction is based on the coefficient of expansion of the venturi material and the difference between feedwater temperature and calibration temperature. At SPNOC, feedwater density is based on the measurement of feedwater temperature and inferred feedwater pressure (from steam pressure, which is conservative relative to the direct feedwater pressure measurement). The venturi pressure drop is obtained from the output of the differential pressure transmitter connected to the venturi.

The power measurement uncertainties are thus based on the following plant measurements:

Steamline pressure (P_s)
Feedwater temperature (T_f)
Steam generator blowdown flow
Feedwater flow (W_f) (from the ΔP transmitters)

and on the following calculated values:

Feedwater venturi flow coefficient (K)
Feedwater venturi thermal expansion correction (F_a)
Feedwater pressure (P_f)
Feedwater density (ρ_f)
Feedwater enthalpy (h_f)
Steam enthalpy (h_s)
Moisture carryover (affects h_s)
Primary system net heat losses (Q_L)
RCP heat addition (Q_p)

Uncertainties

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

For the measurement of feedwater flow, each feedwater venturi was assumed to be calibrated by the vendor in a hydraulics laboratory under controlled conditions to an accuracy of [$]^{+a,c}$ span. The calibration data which substantiates this accuracy is provided to the plant by the vendor. An additional uncertainty factor of [$]^{+a,c}$ span is included for installation effects, resulting in a conservative overall flow coefficient (K) uncertainty of [$]^{+a,c}$ flow. Since NSSS loop power is proportional to steam generator thermal output which is proportional to feedwater flow, the flow coefficient uncertainty is expressed as [$]^{+a,c}$ power. It should be noted that no allowance is made for venturi fouling.

An allowance of [$]^{+a,c}$ was used for the Steam Generator Blowdown orifice plate flow coefficient. This resulted in an uncertainty of [$]^{+a,c}$ power.

The uncertainty applied to the Steam Generator Blowdown orifice plate thermal expansion correction (F_a) is based on the uncertainties of the temperature and the coefficient of thermal expansion for the orifice plate material, assumed to be 304 stainless steel. For this material, a change of ± 1.0 °F in the nominal temperature range changes F_a by [ ]$^{+a,c}$ but the change in steam generator thermal output is negligible.

An uncertainty of 5.0 % in F_a for 304 stainless steel is used in this analysis. This results in an additional uncertainty bounded by [ ]$^{+a,c}$ power. This allowance is included to account for the variations in material composition that could exist for the orifice plate.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 1 notes the instrument uncertainties for the hardware used to perform the parameter measurements. Table 2 lists the various parameter sensitivities. As can be seen on Table 2, both feedwater temperature uncertainties and feedwater pressure uncertainties have an effect on feedwater density and feedwater enthalpy.

Feedwater flow and Steam Generator Blowdown d/p uncertainties are converted to % flow using the following conversion factor:

$$\% \text{ flow} = (\text{d/p uncertainty})(1/2)(\text{transmitter span} / 100)^2. \qquad \text{Eq. 6}$$

The feedwater flow transmitter span is 120% of nominal flow.

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

The net pump heat addition uncertainty is derived from the combination of the primary system net heat losses and pump heat addition and are summarized for the South Texas Project as follows:

System heat losses	- 2.0 MWt
Component conduction and convection losses	- 1.4 MWt
Pump heat adder	+ 23.4 MWt
Net Heat input to RCS	+ 21.0 MWt

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

The calorimetric power measurement determination is performed using the plant computer or a manual calculation. For purposes of this calculation, the plant computer and/or indicator uncertainties are conservative and bounding when compared to a manual calorimetric that uses a digital voltmeter, (DVM), at the process racks. Also, a conservative allowance has been assigned for feed pressure uncertainties for those times when feed pressure may be inferred from steam pressure. Steam generator blowdown uncertainties are based on the use of the control board indicators. As noted in Table 3, Westinghouse has determined the dependent sets in the calculation and the direction of interaction.

Using the power uncertainty values noted on Table 3, the 4-loop uncertainty equation is as follows:

$$\text{Power} = \left[\qquad \right]^{+a,c} \qquad \text{Eq. 7}$$

Where:

$SGBF_V$ = Steam Generator Blowdown flow orifice plate (basic accuracy)

$SGBF_{\Delta P}$ = Steam Generator Blowdown flow Delta P

h_{SP} = Steam enthalpy (as a function of pressure)

Fa_t = Steam Generator Blowdown flow F_a (as a function of temperature, inferred from steam pressure)

h_{SG_LIQ} = Steam Generator Blowdown flow enthalpy (as a function of steam pressure)

ρ_{SG_P} = Steam Generator Blowdown flow density (as a function of steam pressure)

Fa_m = Steam Generator Blowdown flow F_a (as a function of material)

ΔP	= Feedwater venturi pressure drop (inches H_2O)
Match	= Match criterion (normalization) for venturi flow to CROSSFLOW
FW_v	= Feedwater venturi calibration uncertainty
VFa_m	= Venturi correction factor, F_a (as a function of material)
N	= Number of primary side loops
ρ_t	= Feedwater flow density (as a function of temperature)
h_t	= Feedwater flow enthalpy (as a function of temperature)
VFa_t	= Venturi correction factor, F_a (as a function of temperature)
NPHA	= Net pump heat addition
ρ_p	= Feedwater flow density (as a function of pressure)
h_p	= Feedwater flow enthalpy (as a function of pressure)
$h_{s\ moist}$	= Steam enthalpy (as a function of moisture)

Power = [ ] +a,c

Based on the number of loops and the instrument uncertainties for the four parameters, the uncertainty for the secondary side power calorimetric measurement is:

# of loops	power uncertainty (% RTP)
4	[ ] +a,c

TABLE 1
POWER CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

	FW TEMP °F	FW PRESS % Span ***	FW ΔP	S/G BLDN % Flow	STM PRESS %Span	
SRA						+a,c
SCA						
SMTE						
SPE						
STE						
SD						
SEISMIC						
RCA						
COMPCAL						
RMTE						
COMPMTE						
RTE						
COMPTE						
RD						
COMPDRFT						
INDCAL						
INDDRIFT						
INDMTE						
INDREAD						
CSA						
# Inst Used	3	1/Loop	1/Loop	1/Loop	1/Loop	
Units	°F	Psi	% Flow	% Flow	psi	
Inst Span	N/A	2500	120 % Flow	90,000 lb/hr	1400	
Inst Unc. (Random)						+a,c
Nominal	441.8°F	1057 psia	100 % Flow	35,000 lb/hr	957 psia	

* Provided by STPNOC

** % flow

*** Inferred from Steam Pressure

TABLE 2
POWER CALORIMETRIC SENSITIVITIES

+a,c

FEEDWATER FLOW	=
Fa	
TEMPERATURE	=
MATERIAL	=
FEEDWATER DENSITY	
TEMPERATURE	=
PRESSURE	=
FEEDWATER ENTHALPY	
TEMPERATURE	=
PRESSURE	=
h_s	=
h_f	=
Δh (SG)	=
STEAM ENTHALPY	
PRESSURE	=
MOISTURE	=
SG BLOWDOWN ENTHALPY	
PRESSURE	=
SG BLOWDOWN FLOW	
Fa	
TEMPERATURE	=
MATERIAL	=
DENSITY	
PRESSURE	=
DELTA P	=

TABLE 3
SECONDARY SIDE POWER CALORIMETRIC MEASUREMENT UNCERTAINTY

COMPONENT	INSTRUMENT UNCERTAINTY	POWER UNCERTAINTY
		+a,c
FEEDWATER FLOW		
ΔP		
Venturi (FW_V)		
VENTURI THERMAL EXPANSION		
TEMPERATURE (VFa_t)		
MATERIAL (VFa_m)		
SG BLOWDOWN FLOW		
ORIFICE PLATE ($SGBF_V$)		
THERMAL EXPANSION COEFFICIENT		
TEMPERATURE (Fa_t)		
MATERIAL (Fa_m)		
DENSITY		
PRESSURE (ρ_{SG_P})		
DELTA P ($SGBF_{\Delta P}$)		
SG BLOWDOWN LIQUID ENTHALPY		
PRESSURE (h_{SG_LIQ})		
FEEDWATER DENSITY		
TEMPERATURE (ρ_t)		
PRESSURE (ρ_p)		
FEEDWATER ENTHALPY		
TEMPERATURE (h_t)		
PRESSURE (h_p)		
STEAM ENTHALPY		
PRESSURE (h_{sp})		
MOISTURE ($h_{s\ moist}$)		
NET PUMP HEAT ADDITION (NPHA)		
4 LOOP UNCERTAINTY		

*, * *, * * * Indicates sets of dependent parameters

IV. RESULTS/CONCLUSIONS

The preceding sections provide the methodology to account for the Power Calorimetric uncertainties when CROSSFLOW is out of service. The uncertainty calculations have been performed for STPNOC Units 1 and 2 utilizing plant specific instrumentation and calibration procedures. The resultant power calorimetric uncertainty value is [ ]$^{+a,c}$.

V. REFERENCES

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4. Westinghouse letter NS-EPR-2577, E. P. Rahe Jr. to C. H. Berlinger, NRC, dated 3/31/82.

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

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

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13. ANSI/ISA-51.1-1979 (Reaffirmed 1993), "Process Instrumentation Terminology".