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

February 5, 2002
NOC-AE-02001259
File No.: G25
10CFR50.90
STI:31402206

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
Additional Information to Support the Request for Approval
of Power Uprate and a Revision to the Technical Specifications

References:
1) Letter from J. J. Sheppard to NRC Document Control Desk, "Proposed Amendment to Facility Operating Licenses and Technical Specifications Associated with a 1.4-percent Core Power Uprate," August 22, 2001 (NOC-AE-01001162)
2) Letter from M. C. Thadani, NRC, to W. T. Cottle, STPNOC, "South Texas Project Units 1 and 2: Request for Approval of Power Uprate and Revision to the Technical Specifications Supporting the Power Uprate," December 20, 2001 (TAC Nos. MB2899 and MB2903) (ST-AE-NOC-02000907)
3) Letter from J. J. Sheppard to NRC Document Control Desk, "Additional Information to Support the Request for Approval of Power Uprate and a Revision to the Technical Specifications," January 21, 2002 (NOC-AE-02001249)

Reference 1 requested approval of increasing the plant operating power level by 1.4 percent and submitted a license amendment supporting associated revisions to Technical Specifications. Reference 2 requested that additional information from South Texas be submitted to the NRC in order for the staff to complete its evaluation. Reference 3 was a partial response to the NRC request. Attachment 1 to this letter is the response to the remainder of the NRC questions.

Reference 2 requested that WCAP-13441 be submitted to the NRC staff. Attachments 2 through 4 provide the requested WCAP and related documents. Attachment 3 contains information proprietary to the Westinghouse Electric Company LLC ("Westinghouse"). It is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the Westinghouse proprietary information may be withheld from public disclosure by the Commission and addresses, with specificity, the considerations listed in paragraph (b)(4) of

A001

10 CFR2.790 of the Commission's regulations. Accordingly, it is 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 regulations.

Correspondence with respect to the copyright on proprietary aspects of the item listed above or the supporting Westinghouse affidavit should reference CAW-02-1510 and should be addressed to Mr. H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Corporation, LLC ("Westinghouse"), P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

There are no licensing commitments in this letter. If you should have any questions concerning this matter, please contact Mr. Ken Taplett at (361) 972-8416 or me at (361) 972-8757.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2/5/02

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

KJT/

Attachments:
1. Additional Information
2. Westinghouse authorization letter, CAW-02-1510, accompanying affidavit, Proprietary Information Notice and Copyright Notice.
3. WCAP-13441, Revision 1 (proprietary), "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for South Texas Units 1 and 2 Project", July 1999
4. WCAP-15803, Revision 0 (non-proprietary), "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for South Texas Units 1 and 2 Project", January 2002

cc: Without Attachments 2 -4 unless noted by *

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U. S. Nuclear Regulatory Commission
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ATTACHMENT 1

ADDITIONAL INFORMATION

By letter dated August 22, 2001, STP Nuclear Operating Company (STPNOC), the licensee for South Texas Project Units 1 & 2, requested a license amendment to raise the plant operating power level by 1.4 percent. The NRC staff reviewed the application and determined that it needs additional information to complete its review. Responses to some of the NRC questions in a letter dated December 20, 2001 were provided in a letter dated January 21, 2002. The following are responses to the remaining questions in the December 20, 2001 letter.

Note: The numbers below correspond to the question numbers in the December 20th letter.

<u>**Instrumentation and Controls**</u>

3. **Power Range Neutron Flux High setpoint values are revised in Table 3.7-1 of the STP Technical Specifications to reflect the power uprate. Please confirm that the reactor trip setpoint of this function does not need revision.**

   Response:

   The calculational method used to determine the maximum allowable power range neutron flux high setpoints of Technical Specification Table 3.7-1 (with inoperable steam line safety valves during four-loop operation) requires the nominal NSSS power level as input. Therefore, these setpoints had to be revised. The revised setpoints for the uprate are provided in revised Table 3.7-1 of the Technical Specification amendment request.

   The power range neutron flux high and low nominal trip setpoints (109% and 25% of rated thermal power) presented in Technical Specification Table 2.2-1 are applicable when all main steam safety valves (MSSVs) are operable. These setpoints were not revised as part of the power uprate program because neither the safety analysis setpoints (118% and 35% of rated thermal power) nor the setpoint uncertainties were revised.

<u>**Reactor Systems**</u>

1. **Attachment 6, Section 2, Table 2.1-1 presents 4 steady-state plant conditions. However, the table does not describe the cases. Please state the 4 cases presented and describe the major methodologies and assumptions used to generate its calculated values. Also include the current design parameters and assumptions for STP Units 1 and 2 as presented in Table 2.1-1.**

Response:

Please refer to the response to Question #2 from the Reactor Systems branch that follows this response for the method used to calculate the information provided in the four cases.

Four cases are provided for South Texas Units 1 & 2 with the following assumptions:

- Model Delta 94 SGs,
- Nuclear Steam Supply System (NSSS) uprated power level of 3874 MWt (3853 MWt core power + 21 MWt pump heat),
- Nominal and reduced feedwater temperatures (T_{feed}) of 441.8°F and 390°F,
- Thermal design flow of 98,000 gpm/loop,
- Robust Fuel Assembly (RFA) and V5H fuel types, and
- Core bypass flow of 8.5% that accounts for upper head T_{cold} conversion and thimble plug removal (TPR).

The T_{avg} parameter window was adjusted slightly from the current analysis to maintain the current maximum vessel T_{hot} (624.8°F) and minimum T_{cold} (549.8°F) conditions as found in the current replacement steam generator analysis of record.

Cases 1 and 2 vary from 0% and 10% steam generator tube plugging (SGTP), respectively, while maintaining a T_{avg} of 592.6°F which was calculated based on maintaining a maximum vessel T_{hot} of 624.8°F.

Cases 3 and 4 vary from 0% and 10% SGTP, respectively, while maintaining a T_{avg} of 582.7°F which was calculated based on maintaining a minimum T_{cold} of 549.8°F.

The current NSSS design parameters of record for South Texas Units 1 and 2 are those documented for the Model Delta 94 replacement steam generator program (plant change was made pursuant to 10CFR50.59). The key current parameters are found in the Updated Final Safety Analysis Report (UFSAR), Table 5.1-1.

2. **Attachment 6, Section 6.1, Nuclear Steam Supply System Fluid Systems, states that various reactor coolant system (RCS) parameters remain unaffected by the power uprate because they are bounded by the values calculated for the Model Delta 94 steam generators. Please provide a reference to the approvals or show that the parameters were calculated using methods or processes that were previously approved by the NRC.**

Response:

The code used to determine the NSSS design parameters was SGPER (Steam Generator PERformance). There is no explicit NRC-approval for the code since it is used to

facilitate calculations that could be performed by hand. That is, the code and method used to calculate these values have been successfully used to license all previous uprates for Westinghouse plants. They use basic thermal/hydraulic calculations, along with first principles of engineering, to generate the temperatures, pressures, and flows shown in Table 2.1-1 of Attachment 6 to the licensing application.

3. **Attachment 6, Section 6.2.1, Main Steam System - Steam Generator (SG) Power-Operated Relief Valves (PORV), states that an evaluation of the installed capacity of the SG PORVs indicates that the original design basis cooldown capacity can still be achieved for the uprated conditions, however, sufficient bases were not provided to support that conclusion. Please provide the original design basis cooldown capacity and the uprated capacity or a reference to this information.**

Response:

The installed capacity of 68,000 lb/hr at 100 psia for the steam generator PORVs has not changed. The original design basis cooldown capacity was to ensure that the auxiliary feedwater storage tank had sufficient capacity to allow the plant to maintain hot standby conditions for four hours and then cool down to residual heat removal (RHR) conditions for the limiting cooldown event with a single active failure. This design basis has not changed for the power uprate condition. The analysis for the Delta 94 steam generators without the power uprate was performed assuming the initial power was at 102% of 3800 MWt (3876 MWt). The results of this analysis show that the limiting event is a main feedwater line break and the limiting single failure is the failure of an auxiliary feedwater controller. The amount of auxiliary feedwater required to maintain hot standby conditions for four hours and then cooldown to RHR conditions is less than 387,172 gallons. This is less than the available auxiliary feedwater storage tank capacity of 465,000 gallons. The uprate power level is 3853 MWt with an uncertainty of 0.6% (3876 MWt), which is the same as the previous analysis. No other conditions assumed in the cooldown analysis have changed. Therefore, the cooldown capacity is not impacted for the uprate condition.

5. **Attachment 6, Section 6.2.4, Auxiliary Feedwater System (AFS), states that evaluations of the limiting transients and accidents have confirmed that the current AFWS design basis performance remains acceptable, however, sufficient bases were not provided to support your conclusion. At the uprated conditions, state the limiting transients for the AFWS design basis, the limiting minimum flow requirements of the AFWS for the limiting transients, and the AFWS performance for these transients.**

Response:

The limiting transients for the auxiliary feedwater system (AFWS) design basis are loss of non-emergency A/C power to the plant auxiliaries (UFSAR Section 15.2.6), loss of normal feedwater (UFSAR 15.2.7), and feedwater system pipe break (UFSAR 15.2.8). As stated in Sections 8.3.1.2 and 8.3.1.3 of Attachment 6 to the licensing amendment request, these transients did not require explicit re-analyses. This is because the power level assumed in the current analyses (non-uprated nominal power of 3800 MWt plus 2 percent uncertainty) is equivalent to the uprated nominal power of 3,853 MWt plus 0.6 percent uncertainty. Initially, all three transients require a minimum flow of 500 gpm to each of two steam generators 60 seconds after a steam generator low-low signal. In addition, the loss of normal feedwater transient requires 500 gpm to a third steam generator within 15 minutes after the initiation of the event. The feedwater system pipe break requires 500 gpm to a third steam generator within 30 minutes after the initiation of the event.

6. **Attachment 6, Section 8.3.1.9, Chemical and Volume Control System Malfunction, states that an evaluation of the Mode 1 analysis showed that the power uprate has an insignificant impact on the automatic reactor trip time used in the analysis. However, sufficient bases were not provided to support this conclusion. Provide the technical bases which lead to this conclusion.**

Response:

A sensitivity study performed by Westinghouse showed that a 1.4-percent power uprate would increase the reactor trip (due to over-temperature delta-temperature condition) time by less than one second. The assumed reactor trip time is based on an uncontrolled rod control cluster assembly (RCCA) withdrawal at power analysis in which the reactivity insertion rate is equivalent to that expected for the Mode 1 boron dilution scenario. Although the Westinghouse sensitivity study did not explicitly examine the South Texas Project, it was judged to be applicable. Also considered was the fact that the calculated operator action time was rounded down to the nearest whole minute, which can account for more than five seconds of margin.

8. **Attachment 6, Section 7.10.1, Nuclear Design, states that "...adequate margin to the limits associated with all reload safety analysis parameters that are evaluated for each cycle have been confirmed..." For the power uprated cycle, please provide the overall peaking factor (vs the old) and the departure from nucleate boiling ratio (DNBR) (vs the old). Will the new cycle contain different fuels? Is the new cycle a transition (mixed fuel) cycle? Will the new cycle contain a lead test assembly?**

Response:

For the power uprated cycle, the maximum core enthalpy rise hot channel factor (F-delta-h) for the non-uprated portion of the cycle is 1.408 and the maximum for the uprated portion is 1.406. The thermal/hydraulic (T/H) departure from nucleate boiling ratio (DNBR) safety analysis is based on F-delta-H limits that are 1.62 for non-Revised Thermal Design Procedure (RTDP) and 1.557 for RTDP. The margin between these and the cycle-specific values in the core operating limits report (COLR) can be used for cycle-specific DNB margin, if necessary. Neither unit's cycle will contain different fuels than previously used in STP cores, as there are no changes being made to the fuel loading for the uprating. Neither unit's cycle is a transition cycle. However, both the Unit 1 core, which is planned for uprating during the current fuel cycle, and the Unit 2 core, which is planned for uprating following the replacement steam generator outage, will contain both RFA and V5H fuel assemblies. This mixture of fuel types is consistent with fuel loadings in previous cycle cores. Neither unit will contain a lead test assembly.

10. **As described in Section 3 of Attachments 2 and Section 7.10.3 of Attachment 6, the core thermal-hydraulic analyses and evaluations for the power uprate were performed with: (1) the assumption of core designs composed of RFAs, (2) use of the WRB-2M correlation for the DNB analysis, (3) use of the revised thermal design procedure (RTDP) DNB methodology, and (4) use of the WRB-1 DNB correlation for the standard and Vantage 5 Hybrid (V5H) fuel type for which the WRB-2M DNB correlation is not applicable.**

    a. **The NRC staff safety evaluation for the acceptance of the WRB-2M correlation described in WCAP-15025-P-A states that the WRB-2M correlation may only be used for the Modified 17x17 Vantage 5H (V5H) fuel without further justification. Provide a comparison of the RFA and modified V5H fuel designs, and justification for the use of WRB-2M for the RFA design.**

Response:

For DNB analysis, the 17x17 RFA fuel is identical to the fuel design with the Modified V5H mixing vane grids described in WCAP-15025-P-A. The use of the WRB-2M correlation was approved in WCAP-15025-P-A.

    b. **Provide justification for application of the WRB-1 correlation to the V5H fuel.**

Response:

The following letters provide justification:

(1) Letter from A. C. Thadani (NRC) to R. A. Wieseman (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-10444PA, Addendum 2, "VANTAGE 5H Fuel Assembly," dated April 1988 (TAC NO. 68240)
(2) Letter from N. J. Liparulo (Westinghouse) to L. E. Philips (NRC), "WRB-1 Correlation Applicability, Revision 1," NTD-NRC-94-4186, July 1,1994
(3) Safety Evaluation for South Texas Project, Units 1 & 2 – Amendment Nos. 61 and 50 to Facility Operating License Nos. NPF-76 and NPF-80 (TAC Nos. M86688 and M86689), dated May 27, 1994.

c. **How do you assure that the WRB-2M and WRB-1 correlations are not applied outside their ranges of applicability, including pressure, local mass velocity, local quality, and fuel design?**

Response:

The results of the VIPRE-01 and THINC DNBR calculations provide the thermal hydraulic parameters such as pressure, local mass velocity and local quality. Westinghouse verifies that these are within the range of applicability of the DNB correlations used. If a parameter is outside the range of applicability, then an alternate licensed correlation is used. For example, for the analysis of hot zero power steamline break, the W-3 DNB correlation is used because the pressure is below the range of applicability of the WRB-2M or WRB-1 correlations.

11. It appears that there will be mixed cores of standard, V5H, and RFA fuel designs in the future fuel cycles.

a. **What is the basis for assuming the core designs are composed only of RFAs?**

Response:

South Texas does plan to utilize older fuel types in future core designs. However for the DNB analysis, RFA is considered because it is the most limiting fuel type. 17x17XL STD or V5H fuel that may be used in future uprate (3853 MWt) cores will have a significant amount of burnup and thus reduced F-delta-H values. These fuel assemblies will be analyzed using RTDP, with DNBR design limits equal to 1.26 for thimble cells and 1.24 for typical cells. The analysis will use DNBR safety analysis limits equal to 1.43 for typical cells and 1.38 for thimble cells, the improved THINC-IV modeling, and a value of F-delta-H in the COLR that gives DNBRs that meet the DNBR design basis.

b. **How is it determined that the limiting channel having the minimum DNBR would not occur in the standard or V5H fuel?**

Response:

If there is a mixed core, then separate cycle-specific DNBR analysis is performed for each fuel type. It is not expected that any STD or V5H fuel will be limiting due to the significant amount of burnup that this fuel would have since they are only used as reload assemblies.

c. **What is the mixed-core DNBR penalty value used in the thermal-hydraulic calculation assuming RFAs?**

Response:

There will be no mixed core DNBR penalty since the RFA fuel design is hydraulically fully compatible with the STD and V5H fuel. Note that the XL RFA does not have intermediate flow mixers.

d. **How is the mixed core penalty value determined?**

Response:

There will be no mixed core DNB penalty.

e. **Has the mixed core penalty been applied in the safety analysis assuming the RFA cores? If not, why not?**

Response:

There will be no mixed core DNBR penalty.

12. **Attachment 6, Section 7.10.3 states that to support operation at the uprated conditions with the use of WRB-2M DNB correlation, revised RTDP DNBR design limits were calculated. In addition, the safety analysis limits were revised to create an increased DNB margin.**

a. **Provide WCAP-13441 which was stated to provide the basis for the RTDP uncertainties.**

Response:

The proprietary and non-proprietary versions of WCAP-13441 are included as Attachments 3 and 4 of this letter.

The purpose of WCAP-13441 "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for South Texas Units 1 and 2 project" is to

establish instrument uncertainties used in the determination of reactor thermal power. The primary parameters evaluated are (1) pressurizer pressure control, (2) T_{avg} control, (3) primary side calorimetric reactor coolant system (RCS) flow, (4) cold elbow tap RCS flow, and (5) secondary side (daily) power calorimetric.

Revision 0 was issued in February 1993 and addressed the parameters noted above. Revision 1 was issued in July 1999 to accommodate the replacement of various transmitters with Rosemount Model 1154. Specifically, pressurizer pressure, T_{avg} control, and secondary side (daily) power calorimetric values were re-evaluated. The primary side calorimetric RCS flow measurements were unaffected by the transmitter replacement. Elbow tap RCS flow parameters were not revised because transmitter characteristics are normalized using this methodology during the performance of the precision calorimetric.

Currently, WCAP-13441 Rev. 1 uncertainties are applicable for pressurizer pressure control and T_{avg} control. WCAP-13441 Rev. 0 uncertainties are applied to the primary side calorimetric RCS flow, elbow tap RCS flow and secondary side (daily) power calorimetric. Revision 1 is not applied to the secondary side (daily) power calorimetric because the steam generator steam pressure transmitters have not yet been replaced with Rosemount transmitters.

It should be noted that once the 1.4-percent implementation is complete, the secondary side (daily) power calorimetric uncertainties will be based on WCAP-15633 Rev. 0 or WCAP-15697 Rev 0. (Crossflow Out of Service).

b. Provide the revised RTDP DNBR design limits and safety analysis limits for both typical and thimble cells.

Response:

The RTDP DNBR design limits for the RFA with the WRB-2M correlation are 1.24 for typical cells and 1.23 for thimble cells. The safety analysis limits are 1.52 for typical cells and 1.52 for thimble cells. Any 17x17XL STD or V5H fuel that is present will have RTDP design limits equal to 1.26 for typical cells and 1.24 for thimble cells and DNBR safety analysis limits of 1.43 for typical cells and 1.38 for thimble cells.

c. Describe how these DNBR design limits are derived based on the power measurement uncertainty using the CROSSFLOW UFM for the feedwater flow.

Response:

The power uncertainty is combined with the uncertainties in flow, pressure, temperature and the other RTDP uncertainties as described in WCAP-11397-P-A, "Revised Thermal Design Procedure."

d. What are the RTDP design DNBR limits when the CROSSFLOW UFM is out of service? How are these limits accounted for in the safety analyses?

Response:

The in-service power uncertainty is 0.6-percent. The out of service power uncertainty is 1.0%. This increase in the power uncertainty is sufficiently small that the DNBR design limits are not affected. The DNBR safety analysis limits (SAL) are set to values higher than the DNBR design limits (DL). The safety analyses are performed to meet these limits. Analysis in this manner provides DNBR margin, M = 1 - DL/SAL. This margin is used to offset penalties such as rod bow and to give margin to allow design flexibility and to cover unanticipated penalties.

The DNBR limits were recalculated with VIPRE-01 for RFA using the WRB-2M correlation at the uprate conditions. The resulting DNBR design limits are 1.24/1.23 typical cell/thimble cell. The original design limits were calculated using THINC-IV and the WRB-1 correlation.

e. If the limiting hot channel occurs in the standard or V5H fuel design, and the WRB-1 correlation is used for these fuel designs, what are the values for the RTDP design and safety analysis DNBR limits?

Response:

Any 17x17XL STD or V5H fuel that is present will have RTDP design limits equal to 1.26 for typical cells and 1.24 for thimble cells. The DNBR safety analysis limits are 1.43 for typical cells and 1.38 for thimble cells.

f. Describe how the analysis conforms to the restrictions stated in the NRC staff safety evaluation accepting the use of the RTDP methodology described in WCAP-11397-P-A.

Response:

Seven restrictions were imposed. These were satisfied as follows:

(1) The limiting RTDP sensitivity factors are:

Variable	Sensitivity (typical cell)	Sensitivity (thimble cell)
power	-2.742	-2.649
temperature	-10.578	-10.175
pressure	2.791	2.584
flow	1.830	1.750
F-delta-H	-2.846	-2.719
F-delta-H,1-E	-0.503	-0.471

(2) The sensitivity factors were calculated based on the WRB-2M DNB correlation and the VIPRE-01 code.

(3) The limiting cases were recalculated using the VIPRE-01 code and the WRB-2M correlation. The RTDP linearity assumption was validated for this code and correlation.

(4) Variances and distributions applicable to the South Texas units were used.

(5) RTDP was applied only where applicable. This methodology was not applied where not applicable, e.g., to overpressure and hot zero power steamline break transient calculations.

(6) The chosen conditions bound all applications.

(7) Code uncertainties of +/- 4 percent for VIPRE (the THINC-IV replacement) and +/- 1 percent for the transient code were used.

13. Attachment 6, Section 7.10.3 states that the DNBR analyses at the uprate conditions showed that the DNB design basis continues to be met. However, sufficient technical bases were not provided to support this conclusion. Please provide the analyses and evaluations that were performed which lead to your conclusion. Also, provide the input from the THINC-IV calculations used in the RTDP analysis.

Response:

THINC-IV was replaced with the VIPRE-01 code for the DNBR calculations of the RFA. This code was licensed for use by WCAP-14565-P-A, "VIPRE-01 Modeling and

Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," April 1989, Sung,Y.X., et al.

The RTDP DNBR design limits and core and axial offset limits were calculated with VIPRE-01 at the uprate conditions. The limiting DNBR transients, i.e. loss of flow, locked rotor, dynamic dropped rod, static rod misalignment, rod withdrawal from subcritical, and steamline break were recalculated at the 1.4-percent uprate conditions using VIPRE and the WRB-2M correlation, where applicable. The VIPRE input was based on the following design parameters:

Reactor core heat output, MWt	3853
Pressurizer pressure, psia	2250
Minimum measure flow, gpm (RTDP)	403,000
Thermal design flow, gpm (STDP)	392,000
Core bypass flow, %	
Statistical (RTDP)	7.6
Non-statistical (STDP)	8.5
F-delta-H (RTDP)	1.557(1- 3(1-P))
F-delta-H (STDP)	1.62(1- 3(1-P))
(P=power fraction)	
Vessel average temperature, deg F	592.6

All DNBR limits were met on a 95/95 basis. Thus the DNB design basis continues to be met.

14. **Attachment 6, Section 8.3, non-loss-of-coolant-accident (Non-LOCA) Analysis, states that nominal values of initial conditions are assumed in accident analyses that are performed to demonstrate meeting the DNB acceptance criteria. However, the same section states that some non-LOCA analyses are currently analyzed with an explicit 2-percent power measurement uncertainty, thus not requiring re-analysis for the power uprate. These transients include: (1) loss of alternating current (AC) power and loss of normal feedwater, (2) startup of an inactive reactor coolant loop, and (3) chemical and volume control system malfunction that results in increasing reactor coolant inventory. Please clarify whether these three transients were analyzed with the RTDP methodology or the deterministic methodology.**

Response:

These three transients were analyzed using the deterministic methodology.

15. Attachment 6, Section 8.3 states that the core thermal limits and the resulting overtemperature ΔT (OT ΔT) and overpower ΔT (OP ΔT) setpoints are essential inputs to the non-LOCA analyses. It also states that a revised set of core thermal limits was developed because of the power uprate, and that the OT ΔT and OP ΔT setpoints need not be revised.

a. Clarify how the core thermal limits are input in the non-LOCA safety analyses.

Response:

The core thermal limits are inputs for the calculation of the OT ΔT and OP ΔT reactor protection setpoints. The method used by Westinghouse to calculate these setpoints is described in WCAP-8745-P-A "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986. In addition, a partial derivative approximation of the DNB core thermal limit lines (DNBR model) is input to the RETRAN (or LOFTRAN) code to conservatively approximate the change in the DNBR during certain DNB-related transients (primarily those in which the reactor coolant flow is constant). This partial derivative approximation is possible because the DNB core thermal limit lines are relatively linear with respect to changes in reactor coolant temperature, pressure and thermal power. A more detailed discussion of the Westinghouse RETRAN DNBR model is provided in WCAP-14882-P-A "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

b. Specify the core thermal limits and provide the technical bases which supported your conclusion that the trip setpoints need not be revised.

Response:

The core thermal limits are presented in Table 1 on the following page. Applying the method described in WCAP-8745-P-A, the current OT ΔT and OP ΔT setpoints were found to protect the revised set of core thermal limits. Thus, no changes to the OT ΔT or OP ΔT setpoints were required.

Table 1
South Texas Project Units 1 and 2 Core Limits for 1.4-percent Power Uprate
17x17 XL Robust Fuel Assembly, WRB-2M Correlation, RTDP,
$F_{\Delta H}$ = 1.56(1+0.3(1-P)), 1.61 cosine
3853 MWt, MMF = 403000 gpm (2.8% flow uncertainty), 7.6% bypass flow
DNBR SAL = 1.52/1.52, typ/thm

Pressurizer Pressure (psia)	Nominal Power (%)	Vessel Exit Boiling Limit (°F)	DNBR Limiting Temperature (°F)
1775	120	543.02	536.3
	110	549.39	553.6
	100	555.77	570.9
	90	562.14	
	80	568.52	
	60	581.27	
	40	593.98	
2000	120	562.31	549.6
	110	568.55	566.2
	100	574.78	582.7
	90	581.01	
	80	587.23	
	60	599.62	
	40	611.90	
2250	120	582.66	564.8
	110	588.73	580.9
	100	594.79	597.1
	90	600.82	
	80	606.83	
	60	618.74	
	40	630.44	
2400	120	594.53	573.8
	110	600.48	590.1
	100	606.41	606.2
	90	612.30	618.5
	80	618.16	
	60	629.74	
	40	641.04	
2450	120	598.68	576.7
	110	604.59	593.1
	100	610.47	609.1
	90	616.31	621.8
	80	622.11	
	60	633.56	
	40	644.70	

18. **For accidents and transients that the existing analyses of record do not bound plant operation at the proposed uprated power level:**

I. **This section covers the transient and accident analyses that are included in the plant's Updated Final Safety Analysis Report (UFSAR) (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing of its plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, and flooding).**

II. **For analyses that are covered by the NRC approved reload methodology for the plant:**

a. **Identify the transient/accident that is the subject of the analysis;**

Response:

See tables below

b. **Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate;**

Response:

The subject transients/accidents were re-analyzed to support the August 22, 2001 licensing amendment request as discussed in Section 8.3 of Attachment 6 to this licensing amendment request. Section 8.3.2 of Attachment 6 of the licensing application provided a discussion of each analysis. These analyses were performed consistent with the Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985.

c. **Provide an explicit commitment to submit the analysis for NRC review, prior to implementation of the power uprate, if NRC review is deemed necessary by the criteria in 10 CFR 50.59;**

Response:

Section 8.3.2 of Attachment 6 of the licensing application provides a discussion of each analysis.

d. Provide a reference to the NRC's approval of the plant's reload methodology; and;

Response:

This methodology was approved by the NRC in the South Texas Technical Specifications, Section 6.9.1.6.b.1.

e. Provide tables containing the following information:

1. A summary of the initial conditions and assumptions for all transients reanalyzed that will differ from the NRC approved UFSAR due to uprated power operations. This table should identify the conditions or assumptions that have changed, its values used in the cycle before and after the uprate, and a brief justification for any values that move in an non-conservative direction.

Response:

See tables below. The only key input that changed was the assumed core thermal power from 3800 MWt to 3853 MWt.

2. A summary of the results of all transients reanalyzed at the uprated power level. Include the applicable safety limit values used as the acceptance criteria and provide the values obtained at the current rated thermal power (RTP) and the uprated RTP. Provide a discussion for any transients that result in a reduced margin of safety.

Response:

See tables below.

Transient: Feedwater Malfunction

Related UFSAR Section(s)	15.1.2
Key Inputs	• Initiating event: accidental opening of one feedwater control valve with the reactor at full power. This results in a feedwater flow increase to 200% of nominal flow to one steam generator. • Two cases were examined: one with the rod control system in automatic mode and one with the rod control system in manual mode. • Initial steam generator water level was minimized at the value that corresponds to the nominal level minus instrument uncertainties [62.3% narrow range span (NRS)]. • The high-high steam generator water level turbine trip setpoint was conservatively maximized at 98.3% NRS. • Most-negative moderator and Doppler temperature coefficients • Least-negative Doppler power defect
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the Model Δ94 steam generators. As the RETRAN code was utilized in the analysis, the Westinghouse RETRAN methodology described in WCAP-14882-P-A "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel et al., April 1999 was applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 was also applied.
Safety Analysis Limits	The minimum DNBR safety analysis limit is 1.52 for the 1.4-percent uprate program, corresponding to the WRB-2M DNBR correlation. In comparison, the DNBR SAL for the Δ94 RSG program is 1.38, corresponding to the WRB-1 DNBR correlation..
Calculated Results	The minimum DNBR values calculated using RETRAN for the two cases are listed as follows: Automatic rod control - 1.954 (1.4-percent uprate) - 1.712 (Δ94 RSG) Manual rod control - 1.814 (1.4-percent uprate) - 1.604 (Δ94 RSG)

Transient: Steamline Rupture at Full Power

Related UFSAR Section(s)	15.1.4 and 15.1.5
Key Inputs	• Initiating event: a rupture in a steamline. • A spectrum of break sizes ranging from 0.6 ft^2 to 1.4 ft^2 was examined in order to identify the most limiting overpower condition. • A conservatively high over power ΔT (OP ΔT) reactor protection setpoint was assumed [K4 (constant term in OP ΔT setpoint equation) = 1.151]. • The low steamline pressure safety injection actuation setpoint was conservatively minimized at a value of 520 psia (unchanged for uprate). • Most-negative moderator and Doppler temperature coefficients • Least-negative Doppler power defect
Methodology	The applied methodology is consistent with the methodology described in WCAP-9226 "Reactor Core Response to Excessive Secondary Steam Release," January 1978. As the RETRAN and VIPRE codes were utilized in the analysis, the Westinghouse RETRAN methodology described in WCAP-14882-P-A ("RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel et al., April 1999) and the Westinghouse VIPRE methodology described in WCAP-14565-P-A "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., October 1999 were applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 was also applied.
Safety Analysis Limits	• The minimum DNBR safety analysis limit (SAL) is 1.52 for the 1.4-percent uprate program, corresponding to the WRB-2M DNBR correlation. In comparison, the DNBR SAL for the Δ94 RSG program is 1.38, corresponding to the WRB-1 DNBR correlation. • The peak fuel rod power limit is 22.45 kW/ft (1.4-percent uprate and Δ94 RSG).
Calculated Results	The limiting break size was determined to be 1.07 ft^2. The corresponding minimum DNBR and peak fuel rod power values, calculated using the VIPRE code, are listed as follows: Minimum DNBR - 2.044 (1.4-percent uprate) - 1.629 (Δ94 RSG) Peak Fuel Rod Power - 19.50 kW/ft (1.4-percent uprate) - 15.98 kW/ft (Δ94 RSG)

Transient: Loss of External Electrical Load / Turbine Trip

Related UFSAR Section(s)	15.2.2 and 15.2.3
Key Inputs	• Initiating event: turbine trip • A conservatively high over temperature ΔT (OT ΔT) reactor protection setpoint was assumed [K1 (constant term in OT ΔT setpoint equation) = 1.30]. (1.4-percent Uprate and Δ94 RSG) • The pressurizer sprays and power-operated relief valves are assumed to be available. • Least-negative moderator temperature coefficient (0.0 pcm/°F). • Least-negative Doppler power defect.
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the Model Δ94 steam generators. As the RETRAN code was utilized in the analysis, the Westinghouse RETRAN methodology described in WCAP-14882-P-A "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel et al., April 1999 was applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 was also applied.
Safety Analysis Limits	The minimum DNBR safety analysis limit (SAL) is 1.52 for the 1.4-percent uprate program, corresponding to the WRB-2M DNBR correlation. In comparison, the DNBR SAL for the Δ94 RSG program is 1.38, corresponding to the WRB-1 DNBR correlation.
Calculated Results	The minimum DNBR values calculated using RETRAN are 2.1395 (1.4-percent uprate) and 1.5587 (Δ94 RSG).

Transient: Uncontrolled RCCA Bank Withdrawal at Power

Related UFSAR Section(s)	15.4.2
Key Inputs	• Initiating event: RCCA bank withdrawal • A spectrum of reactivity insertion rates ranging from 0.6 pcm/sec to 100 pcm/sec were examined at 10%, 60%, and 100% of nominal power in order to demonstrate that the applicable acceptance criteria, namely the minimum DNBR safety analysis limit, are satisfied over a wide range of conditions. • Both maximum and minimum reactivity feedback conditions were examined. • A conservatively high over temperature ΔT (OT ΔT) reactor protection setpoint was assumed [K1 (constant term in OT ΔT setpoint equation) = 1.30] (1.4-percent Uprate) and 1.42 (Δ94 RSG). • A conservatively high neutron flux reactor protection setpoint of 118% of uprated RTP was assumed. • Assumed statistical core bypass flow was 6.5% for 1.4-percent Uprate and 3.5% for Δ94 RSG.
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the Model E steam generators. As the LOFTRAN code was utilized in the analysis, the Westinghouse LOFTRAN methodology described in WCAP-7907-P-A "LOFTRAN Code Description," T. W. T. Burnett, October 1972 was applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 was also applied.
Safety Analysis Limits	The minimum DNBR safety analysis limit (SAL) is 1.52 for the 1.4-percent uprate program, corresponding to the WRB-2M DNBR correlation. In comparison, the DNBR SAL for the Δ94 RSG program is 1.38, corresponding to the WRB-1 DNBR correlation.
Calculated Results	• For the 1.4-percent uprate, the minimum DNBR calculated using LOFTRAN is 1.5747 and corresponds to a case initiated from 100% power assuming minimum reactivity feedback conditions and a reactivity insertion rate of 2.5 pcm/sec. • For the Δ94 RSG, the minimum DNBR calculated using LOFTRAN is 1.499 and corresponds to a case initiated from 100% power assuming minimum reactivity feedback conditions and a reactivity insertion rate of 1.0 pcm/sec.

III. For analyses that are not covered by the reload methodology for the plant, provide a detailed discussion of each analysis:

Response:

All analysis conducted to support the licensing amendment request of August 22, 2001 are covered by the reload methodology for the plant.

20. To show that the referenced generically approved LOCA analysis methodologies apply specifically to the South Texas plants, provide a statement that the South Texas plants and its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature sensitive parameters bound the as-operated plant values for those parameters.

Response:

Westinghouse has processes in place which ensure that the PCT-sensitive parameters used as input to the large break LOCA (LBLOCA) and small break LOCA (SBLOCA) analyses bound the as-operated plant values for South Texas. The LBLOCA and SBLOCA analyses employ Appendix K methodology and require the use of the most conservative value for parameters which are PCT-sensitive. If a direction of conservatism is not apparent, nominal values are typically used. As a result, the LOCA analyses are based on conservative, bounding input parameters relative to where the plant will operate.

South Texas Units 1 and 2 are operated in accordance with their Technical Specification requirements. This helps to ensure that the LOCA analysis input values for peak cladding sensitive parameters bound the as-operated values for those parameters.

In addition, parameters that may be sensitive to fuel reloads are reviewed and confirmed prior to each reload as part of the reload safety analysis checklist (RSAC) process documented in WCAP-9272-P-A. This requires transmitting a list of LOCA analysis parameter limits to the core design group. The core design group compares the known and predicted parameters for the upcoming cycle to the analysis limits, to ensure the LOCA analyses will remain bounding.

21. What is the decay heat source assumed in the design of the emergency core cooling system (ECCS) switchover from the injection mode to the ECCS sump recirculation mode for the current power rating? Does this assumed heat source change for the uprated power? Is the timing of the switchover affected? Please explain.

Response:

The Westinghouse SBLOCA analysis methodology is capable of modeling flow interruptions, flow reductions, or changes in SI enthalpy during the switchover from injection phase to sump recirculation. The Westinghouse long-term cooling analysis methodology assumes no significant reduction in SI during the switchover to recirculation. For these reasons, the ECCS sump recirculation mode can be considered and extension of the SBLOCA and LBLOCA analyses and decay heat assumptions are embodied in these analyses.

Consistent with the requirements outlined in Appendix K of 10 CFR 50, the decay heat model assumed in the LOCA analyses is 1.2 times the values for infinite operating time in the 1971 American Nuclear Society (ANS) Standard.

The current licensing basis LBLOCA and SBLOCA analyses employ a nominal core power of 3800 MWt. The licensing basis methodology includes a 2-percent calorimetric power measurement uncertainty (yielding an assumed core power 3876 MWt) in accordance with the original requirements of 10 CFR 50, Appendix K. South Texas proposes to reduce the power measurement uncertainty to 0.6-percent based on the use of Crossflow device. The existing 2-percent uncertainty margin in the LBLOCA and SBLOCA analyses is re-allocated and applied to the increase in licensed core power level and 0.6-percent retained to account for power measurement uncertainty. The total core power (including uncertainties) assumed in the analyses remains unchanged at 3876 MWt. Therefore, the assumed heat source in the LOCA analyses will not change for the uprated power.

There are no changes to the RWST drain down rate, the ECCS flow rates, or containment spray flow rates due to the uprate. Therefore, the timing of the switchover to recirculation will not be impacted.

22. **Attachment 6, Section 8.2.2, "Post-LOCA Long-Term Core Cooling," states that the boron concentration in the recirculating coolant is maintained at adequate levels to keep the core subcritical post-LOCA. It also states that this detail will be confirmed in the core reload licensing process. What is the basis for calculating the boron concentration during the core reload licensing process? In addition, what are the assumptions used for the calculations? In particular, please identify what the reactor power (at time of LOCA initiation) is assumed to be. Also, during this process, how are issues with boron precipitation handled?**

Response:

The time in core life of greatest core reactivity (without xenon) is selected as the point at which the pre-LOCA critical boron concentration is determined. This will generally be at

beginning of cycle (BOC) based on the low end of the previous cycle burnup window, but for cores that use large numbers of burnable absorbers, this time in core life of maximum reactivity may occur later than BOC after some absorber burnout is achieved.

If the pre-LOCA RCS boron concentration is lowered, the boron concentration of the long-term cooling water will subsequently be lowered. Therefore, it is conservative to determine the lowest all rods out (ARO) RCS boron concentration that will occur at the maximum reactivity time in core life. The lowest ARO RCS boron is conservatively taken as the hot full power (HFP), ARO peak xenon condition at the most reactive time in life. Even if the most reactive time in core life is determined to be 0 MWD/MTU, a non-zero xenon concentration should be modeled. It is conservative to use the 150 MWD/MTU HFP boron concentration in this case. Additional work to account for the uprated power has not been performed, as the slight power increase would make an insignificant difference in the results, and there is adequate margin to the limit. That is, the impact on post-LOCA boron concentration from increasing the power level by 1.4-percent for a given cycle is insignificant (<5 ppm) compared with the typical margin to the limit for this analysis (~200 ppm).

For a given pre-LOCA boron concentration, there is a corresponding post-LOCA sump boron concentration, as defined by the coolant system characteristics. Corresponding to the pre-LOCA conditions, there is a post-LOCA (68°F to 212°F, ARO) critical boron concentration. In the reload analysis, it is confirmed that the calculated post-LOCA critical boron concentration is less than the post-LOCA sump boron concentration, ensuring that post-LOCA criticality will not occur.

Boron precipitation, as it may affect core cooling, is addressed in the hot leg switchover (HLSO) analysis. The HLSO analysis calculates a switchover time (to hot leg recirculation) that precludes boron precipitation and resulting core blockage that would otherwise interfere with core cooling. This analysis is performed at 102% of 3800 MWt, or 3876 MWt, and therefore is not impacted by the 1.4-percent uprate.

Attachment 2

Westinghouse authorization letter, CAW-02-1510, accompanying affidavit, Proprietary Information Notice and Copyright Notice.

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

January 30, 2002

CAW-02-1510

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

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

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report which is further identified in Affidavit CAW-02-1510 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-02-1510 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: D. Holland/NRC

0585s

CAW-02-1510

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric 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:

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 31st day
of January, 2002

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

0585s

(1) I am Manager, Regulatory and Licensing Engineering, in Nuclear Services at 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 Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR 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 constitute Westinghouse policy and provide 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, and 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 that 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 that 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 10 CFR Section 2.790, it is to be received in confidence by the Commission.

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

(v) The proprietary information sought to be withheld in this submittal is that which is Appropriately marked in "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for South Texas Units 1 and 2 Project", WCAP-13441, Revision 1 (Proprietary) July 1999 for information in support of South Texas Project Nuclear Operating Company's submittal to the Commission, transmitted via South Texas Project Nuclear Operating Company letter and Application for Withholding Proprietary Information from Public Disclosure, Mr. H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse 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 responses to NRC questions on Power Uprate for South Texas Units 1 and 2.

(b) Assist its customer to obtain a license.

Further this information has substantial commercial value as follows:

(a) The information reveals the distinguishing aspects of a process 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, 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, and assurance of quality, or licensing a similar product.

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

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

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing 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 of 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 subscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower cases 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 3

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

Attachment 4

WCAP-15803, Revision 0 (non-proprietary), "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for South Texas Units 1 and 2 Project", January 2002

Westinghouse Non-Proprietary Class 3

WCAP - 15803
Revision 0

Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for South Texas Units 1 and 2 Project

Westinghouse Electric Company LLC

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15803

WESTINGHOUSE REVISED THERMAL DESIGN PROCEDURE INSTRUMENT UNCERTAINTY METHODOLOGY FOR SOUTH TEXAS UNITS 1 AND 2 PROJECT

January, 2002

T. P. Williams
J. L. Seese

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

TABLE OF CONTENTS

WESTINGHOUSE REVISED THERMAL DESIGN PROCEDURE
INSTRUMENT UNCERTAINTY METHODOLOGY FOR
SOUTH TEXAS PROJECT UNITS 1 AND 2
ROSEMOUNT REPLACEMENT TRANSMITTER PROGRAM

I. INTRODUCTION

Four operating parameter uncertainties are used in the uncertainty analysis of the Revised Thermal Design Procedure (RTDP). These parameters are Pressurizer Pressure, Primary Coolant Temperature (T_{avg}), Reactor Power, and Reactor Coolant System Flow. They are frequently monitored and several are used for control purposes. Reactor power is monitored by the performance of a secondary side heat balance (calorimetric) measurement once every 24 hours. RCS flow is monitored by the performance of a calorimetric RCS flow measurement at the beginning of each cycle. Pressurizer pressure is a controlled parameter and the uncertainty reflects the control system. T_{avg} is a controlled parameter via the temperature input to the rod control system and the uncertainty reflects this control system. This WCAP update reflects changes to the Pressurizer Pressure control, T_{avg} Rod Control, and Daily Power calorimetric calculation as a result of the Rosemount Replacement Transmitter Program. The RCS flow calorimetric calculation is not affected by this program. Therefore there is no update to the previous calculation.

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, two sided probability distributions.[4] This approach is used to substantiate the acceptability of the protection system setpoints for many Westinghouse plants, e.g., D. C. Cook 2[5], V. C. Summer, Wolf Creek, and others. The second approach is now utilized for the determination of all instrumentation uncertainties for both RTDP parameters and protection functions.

II. METHODOLOGY

The methodology used to combine the uncertainty components for a channel is the square root of the sum of the squares (SRSS) of those groups of components which are statistically independent. Those uncertainties that are dependent are combined arithmetically into independent groups, which are then systematically combined. The uncertainty components 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 uncertainty components and the channel statistical allowance are variations of the basic Westinghouse Setpoint Methodology[12] and are based on STPNOC specific procedures and processes and are defined as follows:

1. For parameter indication utilizing the plant process computer:

$$\begin{aligned} CSA = \{ & (PMA)^2 + (PEA)^2 + (SMTE+SD)^2 + (SPE)^2 + (STE)^2 + (SRA)^2 + (SEISMIC)^2 + \\ & (SMTE+SCA)^2 + (RMTE+RD)^2_{COMP} + (RTE)^2_{7300} + (RMTE+RCA)^2_{COMP} + (RTE)^2_{COMP} + \\ & (RMTE+RD)^2_{7300} + (RMTE+RCA)^2_{7300}\}^{1/2} + BIAS \qquad \text{Eq. 1} \end{aligned}$$

2. For parameters which have control systems, the control board indicators are used as the verification method for proper control system operation:

$$\begin{aligned} CSA = \{ & (PMA)^2 + (PEA)^2 + (SMTE+SD)^2 + (SPE)^2 + (STE)^2 + (SRA)^2 + \\ & (SMTE+SCA)^2 + (RMTE+RD)^2_{7300} + (RMTE+RCA)^2_{7300} + (RMTE+RD)^2_{IND} + (RTE)^2 + \\ & (RMTE+RCA)^2_{IND} + (RDOUT)^2_{IND} + (CA)^2\}^{1/2} + BIAS \qquad \text{Eq. 2} \end{aligned}$$

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
RDOUT	=	Readout Device Accuracy
CA	=	Controller Accuracy

The parameters above are as defined in references 5 and 12 and are based on ISA S51.1-1979 (R93)[13]. However, for ease in understanding they are paraphrased below:

PMA	-	non-instrument related measurement uncertainties, e.g., temperature stratification of a fluid in a pipe,
PEA	-	uncertainties 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 based on plant calibration procedures,
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 Δp cell,
STE	-	change in input-output relationship due to a change in ambient temperature for a sensor/transmitter,
SD	-	change in input-output relationship over a period of time at reference conditions for a sensor/transmitter,
CA	-	the accuracy of the controller,
RCA	-	reference (calibration) accuracy for all rack modules in loop or channel assuming the loop or channel is string calibrated,
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,
RDOUT	-	the accuracy of a special, local test gauge, a digital voltmeter or multimeter on its most accurate applicable range, or 1/2 of the smallest increment on an indicator,
BIAS	-	a non-random uncertainty for a sensor/transmitter or a process parameter,

A/D - the uncertainty component associated with a computer readout,

IND - the uncertainty component associated with an analog indicator.

A more detailed explanation of the Westinghouse methodology noting the interaction of several parameters is provided in references 5 and 12.

III. INSTRUMENTATION UNCERTAINTIES

The instrumentation uncertainties will be discussed first for the two parameters which are controlled by automatic systems, Pressurizer Pressure and T_{avg} (through Rod Control).

Pressurizer Pressure Uncertainty

Pressurizer Pressure uses a closed-loop control system with a comparison of the measured vapor space pressure to a reference value. Proper operation of the control system is verified by the main control board indicators. Uncertainties are from the transmitter and the process racks/indicators as shown in Table 1. The controller uncertainty determined for this function is [$]^{+a,c}$.

The nominal pressure assumed in the Non-LOCA safety analysis for operation in MODES 1 and 2 is 2235 psig, with a total fluctuation (control variation and instrument uncertainty) of ±46 psi. This value represents an initial condition which is used to minimize the initial system pressure assumed for DNBR and LOCA events and to maximize the initial system pressure assumed for overpressurization events.

In addition to the controller accuracy, an allowance is made for pressure overshoot or undershoot due to the interaction and thermal inertia of the heaters and spray. This allowance is based on engineering judgement from an informal survey of 2, 3, and 4 loop plants conducted in the late 70's. It was concluded that an allowance of [$]^{+a,c}$ would bound the effects due to normal interaction of the spray and thermal inertia of the heaters. Therefore, a total control system uncertainty of [$]^{+a,c}$ is calculated as noted on Table 1, which is bounded by the accident analyses.

Table 1
Pressurizer Pressure Control System Uncertainty

+a,c

PMA	=
PEA	=
SRA	=
SCA	=
SMTE	=
SPE	=
STE	=
SD	=
SEISMIC	=
BIAS	=
RCA_I	=
RCA_{7300}	=
$RMTE_I$	=
$RMTE_{7300}$	=
RTE	=
RD_I	=
RD_{7300}	=
RDOUT	=
CA	=

All above values in % of instrument span. Span = 800 psi.

$CSA =$ +a,c

Eq.3

$CSA =$ +a,c

Eq.4

$$
\begin{array}{rcl}
\text{CSA} & = & \\
\text{CSA} & = & \\
\text{CONTROLLER UNC} & &
\end{array}
\left[\qquad\qquad\qquad\qquad \right]^{+a,c}
$$

Tavg Uncertainty

Tavg uses a closed-loop control system that compares the auctioneered high Tavg from the loops to a reference derived from the First Stage Turbine Impulse Chamber Pressure. Proper operation of the control system is verified by the main control board indicators. Tavg is the average of the narrow range Thot and Tcold values, and the highest loop Tavg is used in the controller. Uncertainties are from hot leg and cold leg streaming, the RTDs, the turbine pressure transmitter, and the process racks/indicators (as noted on Table 2). Based on the assumption that 2 Thot and 1 Tcold cross-calibrated RTDs are used to calculate Tavg (assuming one failed Thot RTD per loop) and that the RTDs are located in the hot and cold legs, the electronics uncertainty is calculated to be [$]^{+a,c}$. Assuming a normal, two sided probability distribution results in an electronics standard deviation (s_1) of [$]^{+a,c}$.

However, this does not include the controller deadband of [$]^{+a,c}$. The controller uncertainty is the combination of the electronics uncertainty and the deadband. The probability distribution for the deadband has been determined to be [$].^{+a,c}$ The variance for the deadband uncertainty is then:

$$(s_2)^2 = [\qquad\qquad]^{+a,c} \qquad \text{Eq. 5}$$

Combining the variance for the electronics and the variance for the deadband results in a controller variance of:

$$(s_C)^2 = (s_1)^2 + (s_2)^2 = [\qquad\qquad]^{+a,c} \qquad \text{Eq. 6}$$

The controller standard deviation s_C = [$]^{+a,c}$ for a total uncertainty of [$]^{+a,c}$ and a total bias value of [$]^{+a,c}$, which are bounded by the RTDP analysis.

Table 2
Tavg Control System Uncertainty

		Tavg*	TURB PRES**
PMA_{HOT}	=		
SRA	=		
SCA	=		
SMTE	=		
SPE	=		
STE	=		
SD	=		
SEISMIC	=		
RCA_{TURB}	=		
RCA_{TAVG}	=		
RCA_{COLD}	=		
RCA_{QDPS}	=		
RCA_{IND}	=		
$RMTE_{TURB}$	=		
$RMTE_{TAVG}$	=		
$RMTE_{COLD}$	=		
$RMTE_{QDPS}$	=		
$RMTE_{IND}$	=		
RD_{TURB}	=		
RD_{TAVG}	=		
RD_{COLD}	=		
RD_{QDPS}	=		
RD_{IND}	=		
RTE_{TURB}	=		
RTE_{TAVG}	=		
RTE_{COLD}	=		
RTE_{QDPS}	=		
RTE_{IND}	=		
RE_{LIN}	=		
$RDOUT_{IND}$	=		
$BIAS_{TAVG}$	=		
$BIAS_{QDPS}$	=		
CA	=		
TP_Sen	=		

+a,c

* % of instrument span. Span = 100°F.
** % of instrument span. Span = 1000 psi.

@[ ] +a,c

Table 2 (continued)
Tavg Control System Uncertainty

[ ]+a,c

csa = [ ]+a,c

Eq. 7

CSA	=	[ ]+a,c
ELECTRONICS SIGMA	=	
CONTROLLER SIGMA	=	
CONTROLLER BIAS	=	
CONTROLLER UNC (random)	=	
TOTAL CONTROLLER UNC.		

CONTROLLER DEADBAND OF [ ]+a,c INCORPORATED

Reactor Power Measurement Uncertainty Using A Feedwater Venturi Measurement

The daily power measurement assumes the measurement of the feedwater flow using the Δp transmitters and the flow venturis placed in the feedwater lines. This method of measurement is sensitive to fouling in the venturi throat which results in an indication of higher-than-actual flow which results in a conservative over-estimate of power.

Assuming that the primary and secondary sides are in equilibrium; the core power is determined by summing the thermal output of the steam generators, correcting the total secondary power for steam generator blowdown, subtracting the reactor coolant pump (RCP) heat addition, adding the primary side system net heat losses, and dividing by the core rated power (Btu/hr). The equation for this calculation is:

$$RP = \frac{\{(\sum Q_{SG}) + Q_L - Q_p\}}{H}(100) \qquad Eq\,8$$

where:

RP	=	Core power (% RTP)
Q_{SG}	=	Steam generator thermal output (BTU/hr)
Q_p	=	RCP heat addition (Btu/hr)
Q_L	=	Primary system net heat losses (Btu/hr)
H	=	Core rated 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 larger uncertainty values.

The thermal output of the steam generator is determined by a secondary side power (calorimetric) measurement, which is defined as:

$$Q_{SG} = (h_s - h_f)W_f \qquad \text{Eq. 9}$$

where:

h_s	=	Steam enthalpy (Btu/lb)
h_f	=	Feedwater enthalpy (Btu/lb)
W_f	=	Feedwater flow (lb/hr).

The steam enthalpy is based on the measurement of steam generator outlet steam pressure, assuming saturated conditions. At STPNOC, the feedwater enthalpy is based on the measurement of feedwater temperature and inferred feedwater pressure (from steam pressure). The feedwater flow is determined by multiple measurements and the following calculation:

$$W_f = (K)(F_a)\{(\rho_f)(\Delta p)\}^{1/2} \qquad \text{Eq. 10}$$

where:

K	=	Feedwater venturi flow coefficient
F_a	=	Feedwater venturi correction for thermal expansion
ρ_f	=	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). The venturi pressure drop is obtained from the output of the differential pressure transmitter connected to the venturi.

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

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

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

A single calculated sum for 100% RTP operation is used for these losses or heat inputs.

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

Steamline pressure (P_s)
Feedwater temperature (T_f)
Feedwater venturi differential pressure (Δp)
Steam generator blowdown (not included based on insignificant blowdown effect on uncertainty calculation)

and on the following calculated values:

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

The derivation of the measurement uncertainties is shown on Table 3. Since it is necessary to make a daily power measurement, STPNOC instructed Westinghouse to assume the plant computer is used for the measurements. Table 4 provides the daily power measurement sensitivities based on 100% RTP design conditions, and Table 5 provides the power measurement uncertainty components as discussed below.

Secondary Side

The secondary side uncertainties are in three principal areas, feedwater flow, feedwater enthalpy, and steam enthalpy. These three areas are specifically identified on Table 5.

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 [$]^{+a,c}$ span. Since NSSS loop power is proportional to steam generator thermal output which is proportional to feedwater flow, the flow coefficient uncertainty is expressed as [$]^{+a,c}$ flow. It should be noted that no allowance is made for venturi fouling. The venturis should be inspected and cleaned, if necessary, during the refueling outage. The effect of fouling results in an indicated power higher-than-actual power which is conservative.

The uncertainty applied to the feedwater venturi thermal expansion correction (F_a) is based on the uncertainties of the measured feedwater temperature and the coefficient of thermal expansion for the venturi material, 300 series stainless steel. For this material, a change of ±1°F in the nominal feedwater temperature range changes F_a by [$]^{+a,c}$ and the steam generator thermal output by the same amount.

An uncertainty in F_a of [$]^{+a,c}$ for 300 series stainless steel is used in this analysis. This results in an additional uncertainty of [$]^{+a,c}$ in feedwater flow. For conservatism, a value of [$]^{+a,c}$ was used in these calculations.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 3 notes the instrument uncertainties for the hardware used to perform the measurements. Table 4 lists the various sensitivities. As can be seen on Table 4,

feedwater temperature uncertainties have an impact on venturi F_a, feedwater density and feedwater enthalpy. Feedwater pressure uncertainties impact feedwater density and feedwater enthalpy.

Feedwater venturi Δp uncertainties are converted to % feedwater flow using the following conversion factor (which presumes the nominal feedwater flow is 100% at 100% RTP):

$$\% \text{ flow} = (\Delta p \text{ uncertainty})(1/2)(\text{transmitter span}/100)^2$$

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

Using the NBS/NRC Steam Tables again, it is possible to determine the sensitivity of steam enthalpy to changes in steam pressure and steam quality. Table 3 notes the uncertainty in steam pressure and Table 4 provides the sensitivity. For steam quality, the Steam Tables were used to determine the sensitivity at a moisture content of [$]^{+a,c}$.

The net pump heat uncertainty is derived from the combination of the primary system net heat losses and RCP heat addition and are summarized for the STPNOC plants as follows:

	$[\quad]^{+a,c}$
System heat losses	
Component conduction and convection losses	
Pump heat adder	
Net Heat input to RCS	

(difference between rated reactor power and rated NSSS power)

The uncertainty on system heat losses, which is essentially all due to charging and letdown flows, has been estimated to be [$]^{+a,c}$. Since direct measurements are not possible, the uncertainty on component conduction and convection losses has been assumed to be [$]^{+a,c}$. Reactor coolant pump hydraulics are known to a relatively high confidence level, and are supported by system hydraulics tests performed at Prairie Island II and by input power measurements from several plants. Therefore, the uncertainty for the pump heat addition is estimated to be [

]$^{+a,c}$. Considering these parameters as one quantity, which is designated the net pump heat uncertainty, the combined uncertainties are less than [ ]$^{+a,c}$ of core power.

Parameter dependent effects are identified on Table 5. Westinghouse has determined the dependent sets in the calculation and the direction of interaction, i.e., whether components in a dependent set are additive or subtractive with respect to a conservative calculation of power. The same work was performed for the instrument bias values. As a result, the calculation explicitly accounts for dependent effects and biases with credit taken for sign (or direction of impact). The basic equation used for this calculation is:

$$\text{POWER} = \left[\qquad \right]^{+a,c} \qquad \text{Eq.11}$$

where:

FW_V = Feedwater flow venturi (basic accuracy)

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

h_{f_T} = Feedwater flow enthalpy (as a function of temperature)

F_{a_T} = Feedwater flow F_a (as a function of temperature)

F_{a_m} = Feedwater flow F_a (as a function of material)

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

h_{f_p} = Feedwater flow enthalpy (as a function of pressure)

d/p_f = Feedwater flow Δp

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

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

NPHA = Net pump heat addition

N = Number of primary side loops

Using the power uncertainty values noted on Table 5, the 4 loop uncertainty (with bias values) equation is as follows:

$$\text{Power} = \left[\qquad\qquad\qquad \right]^{+a,c} \qquad \text{Eq. 12}$$

Based on four 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 3
Daily Power Measurement Instrumentation Uncertainties
(Using Feedwater Venturi Measurement)

(% span)		FW TEMP	FW PRESS	FW Δp	STM PRESS
					+a,c
SRA	=				
SCA	=				
SMTE	=				
SPE	=				
STE	=				
SD	=				
SEISMIC	=				
RCA_{7300}	=				
RCA_{COMP}	=				
$RMTE_{7300}$	=				
$RMTE_{COMP}$	=				
RTE_{7300}	=				
RTE_{COMP}	=				
RD_{7300}	=				
RD_{COMP}	=				
CSA	=				

NUMBER OF INSTRUMENTS USED					
		1/LOOP	1/LOOP	1/LOOP	1/LOOP
		°F	psi	% Δp	psi
INST SPAN	=	300	2500	120% flow	1400
					+a,c
INST UNC (RANDOM)	=				
INST UNC (BIAS)	=				
NOMINAL	=	440°F	1181 PSIA	100% flow	1081 PSIA

All parameters are read by the process computer
* Feedwater Pressure can be inferred from Steam Pressure. A conservative uncertainty is assumed for this calculation.
** % flow

Table 4
Daily Power Measurement Sensitivities
(Using Feedwater Venturi Measurement)

FEEDWATER FLOW			
F_a			+a,c
TEMPERATURE	=		
MATERIAL	=		
DENSITY			
TEMPERATURE	=		
PRESSURE	=		
DELTA P	=		
FEEDWATER ENTHALPY			
TEMPERATURE	=		
PRESSURE	=		
h_s	=	1189.0 BTU/LBM	
h_f	=	419.5 BTU/LBM	
$\Delta h(SG)$	=	769.5 BTU/LBM	
STEAM ENTHALPY			+a,c
PRESSURE	=		
MOISTURE	=		

Table 5
Daily Power Measurement Uncertainty
(Using Feedwater Venturi Measurement)

COMPONENT	INSTRUMENT ERROR	POWER UNCERTAINTY
		+a,c
FEEDWATER FLOW		
VENTURI		
THERMAL EXPANSION COEFFICIENT		
TEMPERATURE		
MATERIAL		
DENSITY		
TEMPERATURE		
PRESSURE		
FEEDWATER ENTHALPY		
TEMPERATURE		
PRESSURE		
STEAM ENTHALPY		
PRESSURE		
MOISTURE		
NET PUMP HEAT ADDITION		
BIAS VALUES		
STEAM PRESSURE ENTHALPY		
POWER BIAS TOTAL VALUE		
*, ** INDICATE SETS OF DEPENDENT PARAMETERS		
SINGLE LOOP UNCERTAINTY (WITHOUT BIAS VALUES)		
4 LOOP UNCERTAINTY (WITHOUT BIAS VALUES)		
4 LOOP UNCERTAINTY (WITH BIAS VALUES)		

CONCLUSIONS

The preceding sections provide the methodology to account for pressure, temperature, and power uncertainties for the safety analysis. The plant-specific instrumentation and procedures have been reviewed for STPNOC and the uncertainty calculations are completed to reflect STPNOC specific details. The following or more conservative values are used in the safety analysis.

	+a,c
Pressurizer pressure uncertainty	
Temperature (Tavg) uncertainty	
Power measurement uncertainty (using feedwater venturi)	

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13. Instrument Society of America Standard S51.1-1979, Reaffirmed 1993, "Process Instrumentation Terminology".

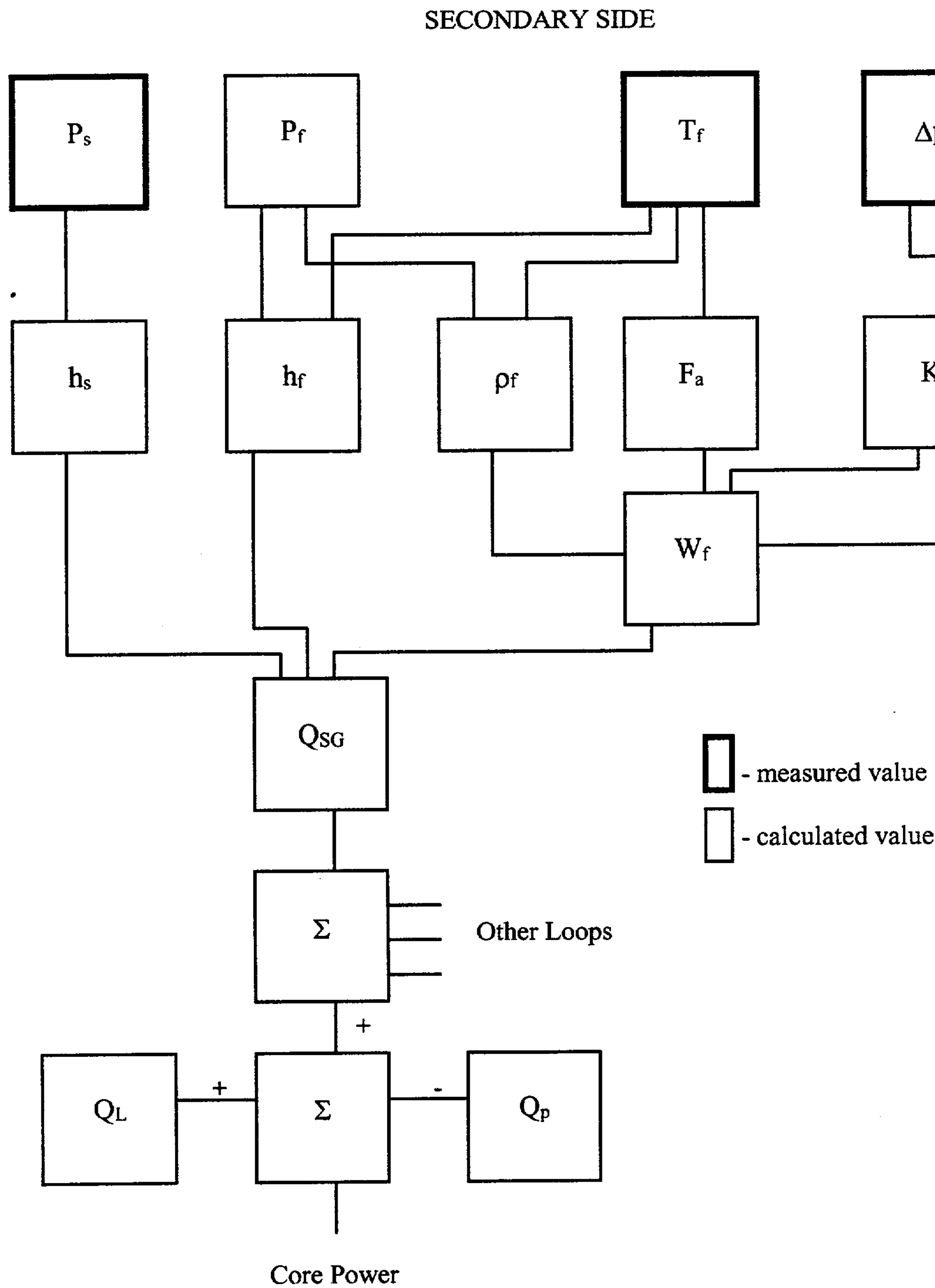

Figure 1 Calorimetric Power Measurement

(Using Feedwater Venturi Measurement)

APPENDIX A

WCAP 13441 REV. 0

The information contained in this Appendix is an exact copy of the information contained in WCAP-13441 dated February 1993, and is included here as information only at the specific request of STPNOC. The information contained in this Appendix has not been evaluated, validated, or updated by Westinghouse to address the technical issues presented in any Westinghouse Technical Bulletins or 10CFR Part 21 notifications published since February 1993.

WCAP-13441

WESTINGHOUSE REVISED THERMAL DESIGN PROCEDURE

INSTRUMENT UNCERTAINTY METHODOLOGY

South Texas Project Units 1 & 2

February, 1993

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

WESTINGHOUSE REVISED THERMAL DESIGN PROCEDURE INSTRUMENT UNCERTAINTY METHODOLOGY

I. INTRODUCTION

Four operating parameter uncertainties are used in the uncertainty analysis of the Revised Thermal Design Procedure (RTDP). These parameters are Pressurizer Pressure, Primary Coolant Temperature (T_{avg}), Reactor Power, and Reactor Coolant System Flow. They are frequently monitored and several are used for control purposes. Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) once every 24 hours. RCS flow is monitored by the performance of a precision flow calorimetric at the beginning of each cycle. The RCS Cold Leg elbow taps are normalized against the precision calorimetric and used for the 12-hour surveillance (with a small increase in uncertainty). Pressurizer pressure is a controlled parameter and the uncertainty reflects the control system. T_{avg} is a controlled parameter via the temperature input to the rod control system and the uncertainty reflects this control system.

Westinghouse has been involved with the development of several techniques to treat instrumentation uncertainties. An early version (for D. C. Cook 2 and Trojan) 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 (for McGuire and Catawba) is based on the more realistic assumption that the uncertainties can be described with random, normal, and two sided probability distributions.(4) This approach is used to substantiate the acceptability of the protection system setpoints for many Westinghouse plants, e.g., D. C. Cook 2(5), V. C. Summer, Wolf Creek, Millstone Unit 3 and others. The second approach is now utilized for the determination of all instrumentation errors for both 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 of those groups of components which are statistically independent. Those errors 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. The sum of both sides is equal to the range for that parameter, e.g., Rack Drift is typically [ ]$^{+a,c}$, the range for this parameter is [ ]$^{+a,c}$. 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 defined as follows:

1. For precision parameter indication using Special Test Equipment or a DVM at the input to the racks;

$$CSA = \{(SCA + SMTE + SD)^2 + (SPE)^2 + (STE)^2 + (RDOUT)^2\}^{1/2} + BIAS \qquad \text{Eq. 1}$$

2. For parameter indication utilizing the plant process computer;

$$CSA = \{(SCA + SMTE + SD)^2 + (SPE)^2 + (STE)^2 + (RCA + RMTE + RD)^2 + (RTE)^2 + (ID)^2 + (A/D)^2\}^{1/2} + BIAS \qquad \text{Eq. 2}$$

3. For parameters which have control systems;

$$CSA = \{(PMA)^2 + (PEA)^2 + (SCA + SMTE + SD)^2 + (SPE)^2 + (STE)^2 + (RCA + RMTE + RD + CA)^2 + (RTE)^2\}^{1/2} + BIAS \qquad \text{Eq. 3}$$

where:

CSA	=	Channel Allowance
PMA	=	Process Measurement Accuracy
PEA	=	Primary Element 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
RDOUT	=	Readout Device Accuracy (DVM or gauge)
ID	=	Computer Isolator Drift
A/D	=	Analog to Digital Conversion Accuracy
CA	=	Controller Accuracy

PMA and PEA terms are not included in equations 1 and 2 since they determine instrumentation uncertainties only. PMA and PEA terms are included in the determination of control system uncertainties.

The parameters above are as defined in references 5 and 12 and are based on SAMA Standard PMC 20.1, 1973[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.
SCA	=	reference (calibration) accuracy for a sensor or 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	=	reference (calibration) accuracy for all rack modules in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy.
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.
RDOUT	=	the measurement accuracy of a special test local gauge, digital voltmeter or multimeter on it's most accurate applicable range for the parameter measured.
ID	=	change in input-output relationship over a period of time at reference conditions for a control or protection signal isolating device.
A/D	=	allowance for conversion accuracy of an analog signal to a digital signal for process computer use.
CA	=	allowance for the accuracy of a controller, not including deadband.
BIAS	=	a non-random uncertainty for a sensor or transmitter or a process parameter.

A more detailed explanation of the Westinghouse methodology noting the interaction of several parameters is provided in references 5 and 12.

III. INSTRUMENTATION UNCERTAINTIES

The instrumentation uncertainties will be discussed first for the two parameters which are controlled by automatic systems, Pressurizer Pressure, and T_{avg} (through Rod Control).

1. PRESSURIZER PRESSURE CONTROL

Pressurizer Pressure is controlled by comparison of the measured vapor space pressure and a reference value. Allowances are made for the transmitter and the process racks and controller. As noted on Table 1, the electronics uncertainty for this function is [ ]$^{+a,c}$ which corresponds to an accuracy of [ ]$^{+a,c}$. In addition to the controller accuracy, an allowance is made for pressure overshoot or undershoot due to the interaction and thermal inertia of the heaters and spray. Based on an evaluation of plant operation, an allowance of [ ]$^{+a,c}$ was made for this effect. An additional bias of [ ]$^{+a,c}$ was included for the temperature compensation of the Veritrak transmitters. Therefore, a total control system uncertainty of [ ]$^{+a,c}$ is calculated, which, after accounting for the bias terms, results in a standard deviation of [ ]$^{+a,c}$ (assuming a normal, two sided probability distribution).

TABLE 1

PRESSURIZER PRESSURE CONTROL SYSTEM ACCURACY

$^{+a,c}$

SCA	=	
M&TE	=	
STE	=	
SD	=	
BIAS	=	
RCA	=	
M&TE	=	
RTE	=	
RD	=	
CA	=	

$^{+a,c}$

ELECTRONICS UNCERTAINTY =
PLUS

ELECTRONICS UNCERTAINTY =
PLUS

CONTROLLER UNCERTAINTY =

2. TAVG (ROD CONTROL)

T_{avg} is controlled by a system that compares the auctioneered high T_{avg} from the loops with a reference, usually derived from the First Stage Turbine Impulse Chamber Pressure. T_{avg} is the average of the narrow range T_H and T_C values. The highest loop T_{avg} is then used in the controller. Allowances are made (as noted on Table 2) for the RTDs, transmitter and the process racks, and controller. The CSA for this function is dependent on the type of RTD, pressure transmitter, and the location of the RTDs, i.e., in the RTD bypass manifold or in the Hot and Cold Legs. Based on the assumption that two T_H and one T_C cross-calibrated Rdf RTD is used to calculate T_{avg} and the RTDs are located in the hot and cold legs, the CSA for the electronics is [ ]$^{+a,c}$. Assuming a normal, two sided probability distribution results in an electronics standard deviation (σ_1) of [ ]$^{+a,c}$.

However, this does not include the controller deadband of ± 1.5 °F or any bias terms. For T_{avg} the controller accuracy is the combination of the instrumentation accuracy and the deadband. The probability distribution for the deadband has been determined to be [ ]$^{+a,c}$. The variance for the deadband uncertainty is then:

$$(\sigma_2)^2 = [\qquad]^{+a,c}.$$

Combining the variance for instrumentation and deadband results in a controller variance of:

$$(\sigma_T)^2 = (\sigma_1)^2 + (\sigma_2)^2 = [\qquad]^{+a,c}$$

The controller σ_T = [ ]$^{+a,c}$ is combined with the controller bias of [ ]$^{+a,c}$ for a total uncertainty of [ ]$^{+a,c}$.

TABLE 2

ROD CONTROL SYSTEM ACCURACY

		Tavg	TURB PRES	QDPS
PMA	=			
SCA	=			
M&TE	=			
STE	=			
SD	=			
BIAS	=			
RCA	=			
M&TE	=			
M&TE	=			
RTE	=			
RD	=			
CA	=			
BIAS	=			

+a,c

RTDs USED - TH = 2 TC = 1

ELECTRONICS CSA	=	
ELECTRONICS SIGMA	=	
CONTROLLER SIGMA	=	
CONTROLLER BIAS	=	
CONTROLLER CSA	=	

+a,c

3. RCS FLOW

RTDP and plant Technical Specifications require an RCS flow measurement with a high degree of accuracy. It is assumed for this error analysis that the flow measurement is performed within ninety days of completing the cross-calibration of the hot leg and cold leg narrow range RTDs. Therefore, a small drift effect is included. It is also assumed that the calorimetric flow measurement is performed at the beginning of a cycle, i.e., no allowances have been made for Feedwater venturi fouling, and above 90% RTP.

The flow measurement is performed by determining the Steam Generator thermal output (corrected for the RCP heat input and the loop's share of primary system heat losses) and the enthalpy rise (Delta-h) of the primary coolant. Assuming that the primary and secondary sides are in equilibrium, the RCS total vessel flow is the sum of the individual primary loop flows, i.e.,

$$W_{RCS} = N(W_L). \qquad \text{Eq. 4}$$

The individual primary loop volumetric flows are determined by correcting the thermal output of the Steam Generator for Steam Generator blowdown (if not secured), subtracting the RCP heat addition, adding the loop's share of the primary side system losses, dividing by the primary side enthalpy rise and multiplying by the Cold Leg specific volume. The equation for this calculation is:

$$W_L = \frac{(A)\,[Q_{SG} - Q_P + (\frac{Q_L}{N})]\,(V_C)}{(h_H - h_C)} \qquad \text{Eq. 5}$$

where;

W_L	=	Loop flow (gpm)
A	=	0.1247 gpm/(ft^3/hr)
Q_{SG}	=	Steam Generator thermal output (Btu/hr)
Q_P	=	RCP heat addition (Btu/hr)

Q_L = Primary system net heat losses (Btu/hr)
V_C = Specific volume of the Cold Leg at T_C (ft^3/lb)
N = Number of primary side loops
h_H = Hot Leg enthalpy (Btu/lb)
h_C = Cold Leg enthalpy (Btu/lb).

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

$$Q_{SG} = (h_s - h_f)W_f \qquad \text{Eq. 6}$$

where; h_s = Steam enthalpy (Btu/lb)
h_f = Feedwater enthalpy (Btu/lb)
W_f = Feedwater flow (lb/hr).

The Steam enthalpy is based on the measurement of Steam Generator outlet Steam pressure, assuming saturated conditions. The Feedwater enthalpy is based on the measurement of Feedwater temperature and Feedwater pressure. The Feedwater flow is determined by multiple measurements and the following calculation:

$$W_f = (K)(F_a)\{(p_f)(d/p)\}^{1/2} \qquad \text{Eq. 7}$$

where; K = Feedwater venturi flow coefficient
F_a = Feedwater venturi correction for thermal expansion
p_f = Feedwater density (lb/ft^3)
d/p = 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. Feedwater density is based on the measurement of Feedwater temperature and Feedwater pressure. The venturi

pressure drop is obtained from the output of the differential pressure cell connected to the venturi.

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

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

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

A single calculated sum for 100% RTP operation is used for these losses or heat inputs.

The Hot Leg and Cold Leg enthalpies are based on the measurement of the Hot Leg temperature and Cold Leg temperature, and an assumed Pressurizer pressure. The Cold Leg specific volume is based on measurement of the Cold Leg temperature and the assumed Pressurizer pressure.

The RCS flow measurement is thus based on the following plant measurements:

Steamline pressure (P_s)
Feedwater temperature (T_f)
Feedwater pressure (P_f)
Feedwater venturi differential pressure (d/p)
Hot Leg temperature (T_H)
Cold Leg temperature (T_C)
Steam Generator blowdown (if not secured)

and on the following calculated or assumed values:

Feedwater venturi flow coefficients (K)
Feedwater venturi thermal expansion correction (F_a)
Feedwater density (ρ_f)
Feedwater enthalpy (h_f)
Steam enthalpy (h_s)
Moisture carryover (impacts h_s)
Primary system net heat losses (Q_L)
RCP heat addition (Q_p)
Hot Leg enthalpy (h_H)
Cold Leg enthalpy (h_C).
Pressurizer pressure (P_p)

These measurements and calculations are presented schematically on Figure 1.

The derivation of the measurement errors and flow uncertainties on Table 5 are noted below.

Secondary Side

The secondary side uncertainties are in four principal areas, Feedwater flow, Feedwater enthalpy, Steam enthalpy and RCP heat addition. These four areas are specifically identified on Table 5.

For the measurement of Feedwater flow, each Feedwater venturi is calibrated by the vendor in a hydraulics laboratory under controlled conditions to an accuracy of [$]^{+a,b,c}$. The calibration data which substantiates this accuracy is provided to the plant by the vendor. An additional uncertainty factor of [$]^{+a,c}$ is included for installation effects, resulting in a conservative overall flow coefficient (K) uncertainty of [$]^{+a,c}$. Since RCS loop flow is proportional to Steam Generator thermal output which is proportional to Feedwater flow, the flow coefficient uncertainty is expressed as [$]^{+a,c}$. **It should be noted that no allowance is made for venturi fouling.** The venturis should be inspected, and cleaned if necessary, prior to performance of the precision measurement. If

fouling is present but not removed, its effects must be treated as a flow bias.

The uncertainty applied to the Feedwater venturi thermal expansion correction (F_a) is based on the uncertainties of the measured Feedwater temperature and the coefficient of thermal expansion for the venturi material, usually 304 stainless steel. For this material, a change of ± 1°F in the nominal Feedwater temperature range changes F_a by ± 0.002 % and the Steam Generator thermal output by the same amount.

An uncertainty in F_a of ± 5 % for 304 stainless steel is used in this analysis. This results in an additional uncertainty of [$]^{+a,c}$ in Feedwater flow. Westinghouse uses the conservative value of [$]^{+a,c}$.

Using the 1967 ASME Steam Tables it is possible to determine the sensitivities of various parameters to changes in Feedwater temperature and pressure. Table 3 notes the instrument uncertainties for the hardware used to perform the measurements. Table 4 lists the various sensitivities. As can be seen on Table 4, Feedwater temperature uncertainties have an impact on venturi F_a, Feedwater density and Feedwater enthalpy. Feedwater pressure uncertainties impact Feedwater density and Feedwater enthalpy.

Feedwater venturi d/p uncertainties are converted to % Feedwater flow using the following conversion factor:

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

Typically, the Feedwater flow transmitter span is [$]^{+a,c}$ of nominal flow.

Using the 1967 ASME Steam Tables again, it is possible to determine the sensitivity of Steam enthalpy to changes in Steam pressure and Steam quality. Table 3 notes the uncertainty in Steam pressure and Table 4 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 4.

The net pump heat uncertainty is derived from the combination of the primary system net heat losses and pump heat addition and are summarized for a four loop plant as follows:

System heat losses	- 2.0 MWt
Component conduction and convection losses	- 1.4
Pump heat adder	<u>+17.0</u>
Net Heat input to RCS	+13.6 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 II and by input power measurements from several plants, therefore, the uncertainty for the pump heat addition is estimated to be [ ]$^{+a,c}$ of the best estimate value. Considering these parameters as one quantity, which is designated the net pump heat uncertainty, the combined uncertainties are less than [ ]$^{+a,c}$ of the total, which is [ ]$^{+a,c}$ of core power.

Primary Side

The primary side uncertainties are in three principal areas, Hot Leg enthalpy, Cold Leg enthalpy and Cold Leg specific volume. These are specifically noted on Table 5. Two primary side parameters are actually measured, T_H and T_C, and Pressurizer pressure is assumed. Hot Leg enthalpy is influenced by T_H, Pressurizer pressure and Hot Leg temperature streaming. The uncertainties for the instrumentation are noted on Table 3 and the sensitivities are provided on Table 4. The Hot Leg streaming is split into random and bias (systematic)

components. For plants with RTDs located in thermowells placed in the scoops (bypass manifolds eliminated), the streaming uncertainty is, [ ]$^{+a,c}$ random and [ ]$^{+a,c}$ systematic.

The Cold Leg enthalpy and specific volume uncertainties are impacted by T_c and Pressurizer pressure. Table 3 notes the T_c instrument uncertainty and Table 4 provides the sensitivities.

Noted on Table 5 is the plant specific RTD cross-calibration systematic allowance. When necessary, an allowance is made for a systematic temperature error due to the RTD cross-calibration procedure. No allowance was necessary for South Texas.

Parameter dependent effects are identified on Table 5. Westinghouse has determined the dependent sets in the calculation and the direction of interaction, i.e., whether components in a dependent set are additive or subtractive with respect to a conservative calculation of RCS flow. The same work was performed for the instrument bias values. As a result, the calculation explicitly accounts for dependent effects and biases with credit taken for sign (or direction of impact).

Using Table 5, the 4 loop uncertainty equation (with biases) is as follows:

[ ]$^{+a,c}$

Based on the number of loops, number, type and measurement method of RTDs, and the vessel Delta-T, the flow uncertainty is:

# of loops	flow uncertainty (% flow)
4	[ ]$^{+a,c}$

<u>**TABLE 3**</u>

FLOW CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

(% SPAN)	FW TEMP	FW PRES	FW d/p	STM PRESS	T_H	T_C	PRZ PRESS +a,c
SCA =							
M&TE=							
SPE =							
STE =							
SD =							
R/E =							
RDOT=							
BIAS=							
CSA =							
# OF INST USED					3	1	1
	DEG F	PSIA	% DP	PSIA	DEG F	DEG F	PSIA
INST SPAN =	300.	1400.	120.	1400.	100.	100.	800.
INST UNC. (RANDOM) =							+a,c
INST UNC. (BIAS) =							
NOMINAL =	440.	1181.		1081.	625.6	560.4	2250.

TABLE 4

FLOW CALORIMETRIC SENSITIVITIES

FEEDWATER FLOW

+a,c

FA	
TEMPERATURE	=
MATERIAL	=
DENSITY	
TEMPERATURE	=
PRESSURE	=
DELTA P	=
FEEDWATER ENTHALPY	
TEMPERATURE	=
PRESSURE	=

hS	=	1189.8 BTU/LBM
hF	=	419.6 BTU/LBM
Dh(SG)	=	770.2 BTU/LBM

STEAM ENTHALPY

+a,c

PRESSURE	=
MOISTURE	=

HOT LEG ENTHALPY

TEMPERATURE	=
PRESSURE	=

hH	=	651.9 BTU/LBM
hC	=	560.2 BTU/LBM
Dh(VESS)	=	91.7 BTU/LBM
Cp(TH)	=	1.617 BTU/LBM-DEGF

COLD LEG ENTHALPY

+a,c

TEMPERATURE	=
PRESSURE	=

Cp(TC)	=	1.270 BTU/LBM-DEGF

COLD LEG SPECIFIC VOLUME

+a,c

TEMPERATURE	=
PRESSURE	=

TABLE 5

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTIES

+a,c

COMPONENT	INSTRUMENT ERROR	FLOW UNCERTAINTY
FEEDWATER FLOW		
VENTURI		
THERMAL EXPANSION COEFFICIENT		
TEMPERATURE		
MATERIAL		
DENSITY		
TEMPERATURE		
PRESSURE		
DELTA P		
FEEDWATER ENTHALPY		
TEMPERATURE		
PRESSURE		
STEAM ENTHALPY		
PRESSURE		
MOISTURE		
NET PUMP HEAT ADDITION		
HOT LEG ENTHALPY		
TEMPERATURE		
STREAMING, RANDOM		
STREAMING, SYSTEMATIC		
PRESSURE		
COLD LEG ENTHALPY		
TEMPERATURE		
PRESSURE		
COLD LEG SPECIFIC VOLUME		
TEMPERATURE		
PRESSURE		
RTD CROSS-CAL SYSTEMATIC ALLOWANCE		

*,**,+,++ INDICATE SETS OF DEPENDENT PARAMETERS

TABLE 5 (CONTINUED)

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTIES

COMPONENT	INSTRUMENT ERROR	FLOW UNCERTAINTY
BIAS VALUES		+a,c
FEEDWATER PRESSURE	DENSITY	
	ENTHALPY	
STEAM PRESSURE	ENTHALPY	
PRESSURIZER PRESSURE	ENTHALPY - HOT LEG	
	ENTHALPY - COLD LEG	
	SPECIFIC VOLUME - COLD LEG	
FLOW BIAS TOTAL VALUE		

*,**,+,++ INDICATE SETS OF DEPENDENT PARAMETERS

		+a,c
SINGLE LOOP UNCERTAINTY	(WITHOUT BIAS VALUES)	
4 LOOP UNCERTAINTY	(WITHOUT BIAS VALUES)	
4 LOOP UNCERTAINTY	(WITH BIAS VALUES)	

As noted earlier, the precision flow calorimetric is used as the reference for the normalization of the Cold Leg elbow taps. Assuming that the elbow tap d/p transmitters are used to feed the plant process computer, it is a simple matter to perform Technical Specification required surveillance. Table 6 notes the instrument uncertainties for normalization of the elbow taps, assuming one elbow tap per loop. The d/p transmitter uncertainties are converted to % flow on the same basis as the Feedwater venturi d/p. The elbow tap uncertainty is then combined with the precision flow calorimetric uncertainty. This combination of uncertainties results in the following total flow uncertainty:

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

The corresponding values used in RTDP are:

# of loops	standard deviation (% flow)
4	[ ]+a,c

TABLE 6

COLD LEG ELBOW TAP FLOW UNCERTAINTY

INSTRUMENT UNCERTAINTIES

	% DP SPAN	% FLOW
PMA =		
PEA =		
SCA =		
SPE =		
STE =		
SD =		
RCA =		
M&TE=		
RTE =		
RD =		
ID =		
A/D =		
RDOT=		
BIAS=		
FLOW CALORIM. BIAS =		
FLOW CALORIMETRIC =		
INSTRUMENT SPAN =		

+a,c

+a,c

SINGLE LOOP ELBOW TAP FLOW UNC =

N LOOP ELBOW TAP FLOW UNC =

N LOOP RCS FLOW UNCERTAINTY (WITHOUT BIAS VALUES) =

N LOOP RCS FLOW UNCERTAINTY (WITH BIAS VALUES) =

4. REACTOR POWER

Generally a plant performs a primary/secondary side heat balance once every 24 hours when power is above 15% Rated Thermal Power. This heat balance is used to verify that the plant is operating within the limits of the Operating License and to adjust the Power Range Neutron Flux channels when the difference between the NIS and the heat balance is greater than that required by the plant Technical Specifications.

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

$$RP = \frac{\{(N)\,[Q_{SG} - Q_P + (\frac{Q_L}{N})]\}(100)}{H} \qquad \text{Eq. 8}$$

where;

RP	=	Core power (% RTP)
N	=	Number of primary side loops
Q_{SG}	=	Steam Generator thermal output (BTU/hr) as defined in Eq. 6
Q_p	=	RCP heat adder (Btu/hr) as defined in Eq. 5
Q_L	=	Primary system net heat losses (Btu/hr) as defined in Eq. 5
H	=	Core rated Btu/hr at full power.

For the purposes of this uncertainty analysis (and based on H noted above) it is assumed that the plant is at 100% RTP when the measurement is taken. Measurements performed at lower power levels will result in different uncertainty values. However, operation at lower power levels results in increased margin to DNB far in excess of any margin losses due to increased measurement uncertainty.

The secondary side power calorimetric equations and effects are the same as those noted for the precision flow calorimetric (secondary side portion), equations 6 and 7. The measurements and calculations are presented schematically on Figure 2. Table 7 provides the instrument uncertainties for those measurements performed. Since it is necessary to make this determination daily, it has been assumed that the plant process computer will be used for the measurements. The sensitivities calculated are the same as those noted for the secondary side on Table 4. As noted on Table 8, Westinghouse has determined the dependent sets in the calculation and the direction of interaction. This is the same as that performed for the RCS flow calorimetric, but applicable only to power. The same was performed for the bias values noted. It should be noted that Westinghouse does not include any allowance for Feedwater venturi fouling. The effect of fouling is to result in an indicated power higher than actual, which is conservative.

Using the power uncertainty values noted on Table 8, the 4 loop uncertainty (with bias values) equation is as follows:

[ ]+a,c

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

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

TABLE 7

POWER CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

(% SPAN)	FW TEMP	FW PRES	FW d/p	STM PRESS +a,c
SCA =				
M&TE=				
SPE =				
STE =				
SD =				
BIAS=				
RCA =				
M&TE=				
RTE =				
RD =				
ID =				
A/D =				
CSA =				
	DEG F	PSIA	% DP	PSIA
INST SPAN =	300.	1400.	120.	1400.
INST UNC (RANDOM) =				+a,c
INST UNC (BIAS) =				
NOMINAL =	440.	1181.		1081.

TABLE 8

SECONDARY SIDE POWER CALORIMETRIC MEASUREMENT UNCERTAINTIES

+a,c

COMPONENT		INSTRUMENT ERROR	POWER UNCERTAINTY
FEEDWATER FLOW			
VENTURI			
THERMAL EXPANSION COEFFICIENT			
TEMPERATURE			
MATERIAL			
DENSITY			
TEMPERATURE			
PRESSURE			
DELTA P			
FEEDWATER ENTHALPY			
TEMPERATURE			
PRESSURE			
STEAM ENTHALPY			
PRESSURE			
MOISTURE			
NET PUMP HEAT ADDITION			
BIAS VALUES			
FEEDWATER DELTA P			
FEEDWATER PRESSURE	DENSITY		
	ENTHALPY		
STEAM PRESSURE	ENTHALPY		
POWER BIAS TOTAL VALUE			

*,** INDICATE SETS OF DEPENDENT PARAMETERS

SINGLE LOOP UNCERTAINTY (WITHOUT BIAS VALUES)		
N LOOP UNCERTAINTY (WITHOUT BIAS VALUES)		
N LOOP UNCERTAINTY (WITH BIAS VALUES)		

IV. CONCLUSIONS

The preceding sections provide the methodology to account for pressure, temperature, power and RCS flow uncertainties for the RTDP analysis. The plant specific instrumentation data and procedures supplied by HL&P have been reviewed and the uncertainty calculations completed using this data.

V. REFERENCES

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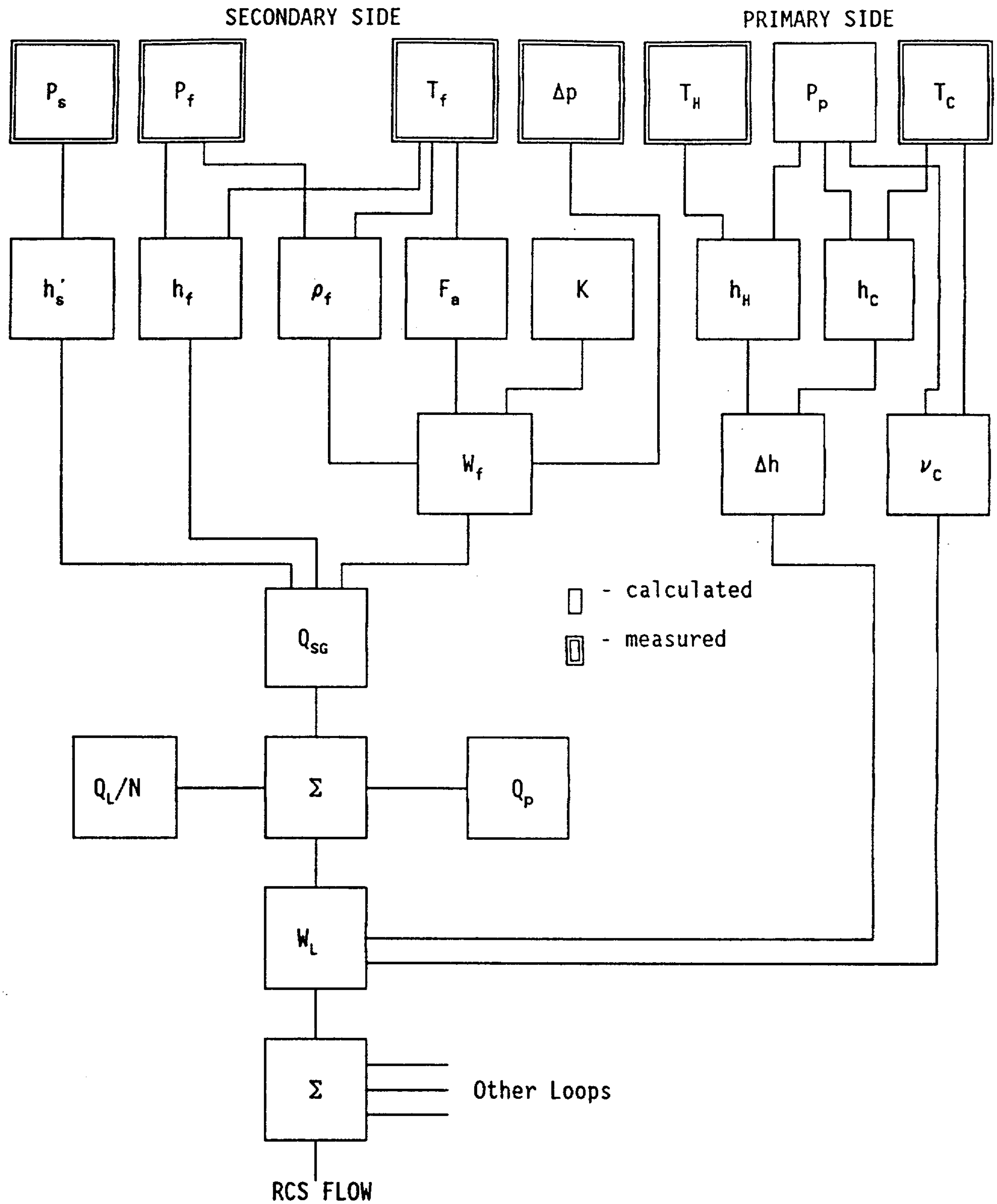

Figure 1
RCS Flow Calorimetric Schematic

SECONDARY SIDE

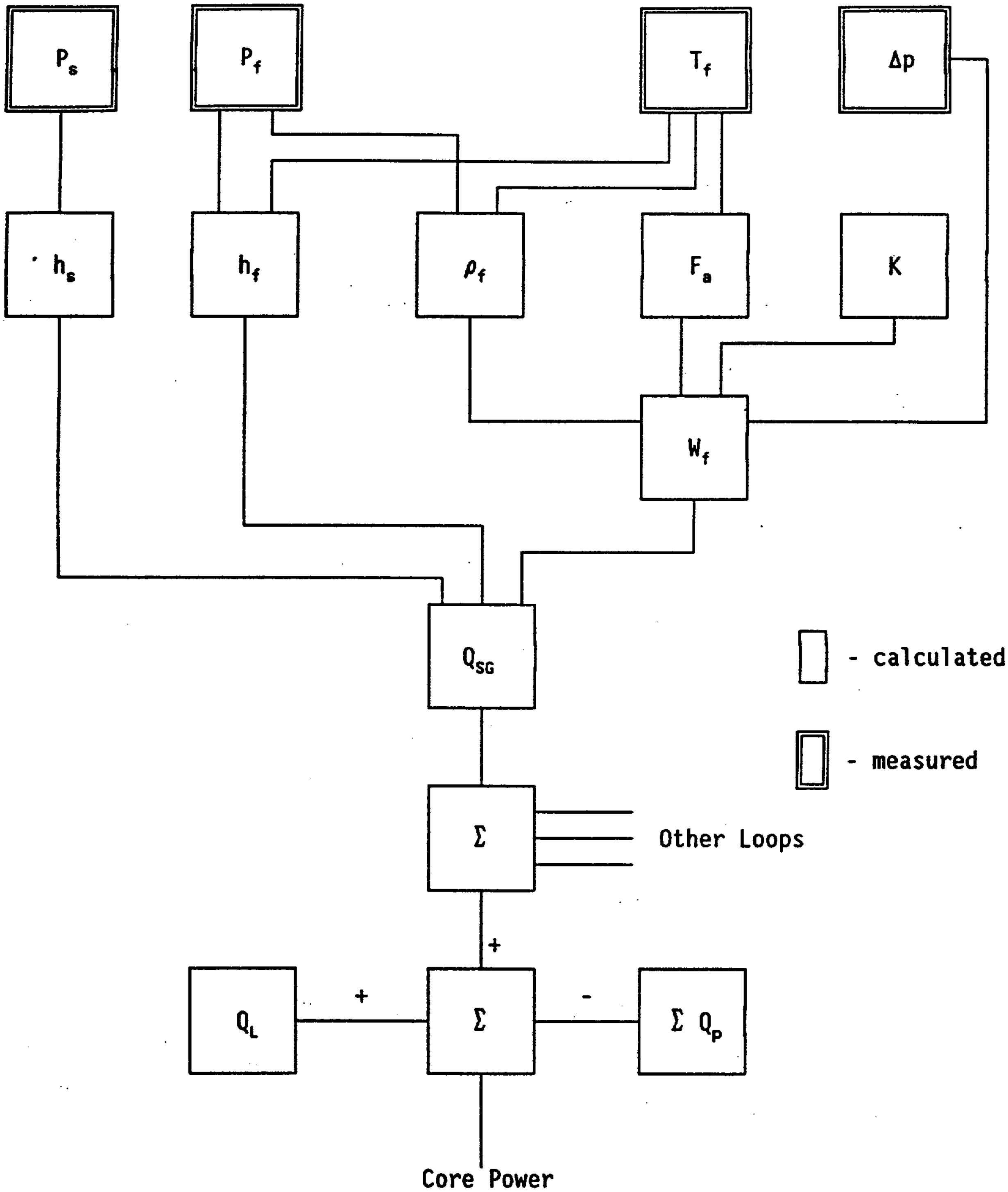

Figure 2
Power Calorimetric Schematic