Westinghouse Electric Company LLC

Nuclear Services Business Unit

Box 355
Pittsburgh, Pennsylvania 15230-0355

May 16, 2000
CAW-00-1397

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: WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at the Seabrook Station," (Proprietary)

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-00-1397 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 North Atlantic Energy Service Corporation.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-00-1397 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

0464s.doc

cc: S. Bloom, NRR

CAW-00-1397

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 17th day
of May, 2000

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

0464s.doc

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at the Seabrook Station", (Proprietary), April 2000 for Seabrook Unit 1, being transmitted by North Atlantic Energy Service Corporation letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel J. Collins. The proprietary information as submitted for use by North Atlantic Energy Service Corporation for the Seabrook Unit 1 Nuclear Power Plants is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of use of RCS flow verification using elbow taps.

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

(a) Provide elbow tap methodology for baseline flows.

(b) Establish appropriate procedures for baseline calorimetric flow with elbow tap measurements.

(c) Assist the customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

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

(b) Westinghouse can sell support and defense of RCS verification methodology using elbow taps to its customers in the licensing process.

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

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

In order for competitors of Westinghouse to duplicate this information, similar 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 testing and analytical methods and performing tests.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

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Westinghouse
Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

NAH-00-028
June 9, 2000

Mr. P. Gurney
North Atlantic Energy Service Corporation
Seabrook Station, U. S. Route 1
P.O. Box 300
Seabrook, NH 03874

Subject: **NORTH ATLANTIC ENERGY SERVICE CORPORATION**
Seabrook Unit 1
WCAPs 15404 and 15415 Appendix B

Dear Mr. Gurney:

Per your request, Westinghouse recommends that the Technical Specification markups from WCAPs 15404 and 15415, Appendix B, be revised as shown on the attachment to this letter. These changes are consistent with the five column to two column Reactor Trip Setpoint methodology reviewed and approved by the NRC in submittals from Millstone. Also, it is noted that the two column technical specification recommendation is consistent with the Improved Technical Specification (ITS) format under NUGEG 1431 and the recommended changes suggested by Reg. Guide 1.105, Rev. 3.

Very truly yours,

WESTINGHOUSE ELECTRIC COMPANY

Stephen P. Swigart

Stephen P. Swigart
Customer Projects Manager

Attachment

cc: D. Samara, Seabrook Station
K. Garner, ECE 478D

Westinghouse Non-Proprietary Class 3

WCAP - 15415
Revision 0

Justification of Elbow Taps for RCS Flow Verification at the Seabrook Station

Westinghouse Electric Company LLC

WCAP-15415

Justification of Elbow Taps for RCS Flow Verification at the Seabrook Station

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TABLE OF CONTENTS

LIST OF TABLES v

LIST OF FIGURES vii

REFERENCES ix

1.0 INTRODUCTION 1-1

2.0 SUMMARY 2-1

3.0 RCS HOT LEG TEMPERATURE STREAMING 3-1

3.1 Phenomenon 3-1

3.2 History 3-1

3.3 Hot Leg Streaming Impact on RCS Flow Measurements 3-2

4.0 ELBOW TAP FLOW MEASUREMENT APPLICATION 4-1

4.1 Elbow Tap Flow Measurements 4-1

4.2 Elbow Tap Flow Measurement Procedure 4-3

4.3 Baseline Parameters for Elbow Tap Flow Measurements 4-5

5.0 BEST ESTIMATE FLOW ANALYSIS 5-1

5.1 Background 5-1

5.2 Prairie Island Hydraulics Test Program 5-1

5.3 Additional Prairie Island Tests 5-3

5.4 System Flow Resistance Analysis 5-3

5.5 Best Estimate RCS Flow Calculations 5-4

6.0 EVALUATION OF SEABROOK FLOW PERFORMANCE 6-1

6.1 Introduction 6-1

6.2 Best Estimate Flow Predictions 6-1

6.3 Evaluation of Elbow Tap Flows 6-2

6.4 Evaluation of Calorimetric Flows 6-2

6.5 Flow Comparisons 6-3

TABLE OF CONTENTS (Continued)

7.0 ELBOW TAP FLOW MEASUREMENT SAFETY EVALUATION 7-1

7.1 Background 7-1

7.2 Licensing Basis 7-1

7.3 Evaluation 7-1

7.4 Determination of No Unreviewed Safety Question 7-2

7.5 Conclusions 7-3

APPENDICES

A Indicated RCS Flow and Reactor Coolant Flow-Low Reactor Trip Instrument Uncertainties A-1

B Suggested Modifications to Technical Specifications B-1

LIST OF TABLES

4-1 Comparisons of LEFM and Elbow Tap Flow Measurements 4-9

6-1 Best Estimate, Elbow Tap and Calorimetric Flows 6-4

6-2 Baseline Calorimetric Flow Development 6-4

A-1 Flow Calorimetric Instrumentation Uncertainties (BOC 1) A-2

A-2 Flow Calorimetric Sensitivities A-3

A-3 Calorimetric RCS Flow Measurement Uncertainties A-4

A-4 Cold Leg Elbow Tap Flow Uncertainty (Process Computer) A-5

A-5 Low Flow Reactor Trip A-6

LIST OF FIGURES

3-1 Upper Plenum and RCS Hot Leg Flow Patterns 3-4

3-2 Typical Core Exit Temperature Gradient and RCS Hot Leg Circumferential Temperature Gradient 3-5

3-3 Typical Core Exit Temperature Change 3-6

4-1 Elbow Tap Locations in RCS Piping 4-10

6-1 Seabrook RCS Flow History 6-5

REFERENCES

1. North Atlantic Energy Service Corporation Letter CE-00-002, February 16, 2000.

2. "Fluid Meters, Their Theory and Application", 6th Edition, Howard S. Bean, ASME, New York, 1971.

3. WCAP-13466, Revision 1, "Bases Document for Westinghouse Setpoint Methodology for Protection System, 24 Month Cycle Evaluation, Seabrook Station," July 1998.

1.0 INTRODUCTION

Reactor Coolant System (RCS) secondary calorimetric-based flow measurements at many pressurized water reactor (PWR) plants, including Seabrook Unit 1, have been affected by increases in hot leg temperature streaming. The increases are related to changes in the reactor core radial power distribution, resulting from implementation of low leakage loading patterns (LLLPs). In some cases, measured flow appears to have decreased to, or below, the flow required by the Technical Specifications, which presently require that RCS flow be confirmed by measurement once per fuel cycle. Such occurrences require licensee actions to either account for the apparent flow reduction in the plant safety analyses or to confirm by other means that RCS flow has not decreased below the specified limit. In many cases, utilities have relied on the repeatability of RCS elbow tap flow meters to demonstrate that RCS flow has not decreased.

The current RCS calorimetric flow measurement method based on RCS temperature and secondary calorimetric power measurements has inherent limitations imposed by LLLPs. This report was prepared in response to a North Atlantic Energy Service Corporation (NAESCO) request (Reference 1), presenting the justification of an alternate method to measure RCS flow, and evaluating RCS flow performance at Seabrook Unit 1. The proposed alternate method uses elbow tap flow measurements normalized to a measured baseline calorimetric flow to minimize the LLLP impact.

The following sections present information on:

- Hot leg temperature streaming phenomenon;
- Elbow tap flow measurement application and justification;
- Best estimate hydraulics analysis used to predict RCS flow;
- Evaluation of elbow tap and calorimetric flows at Seabrook Unit 1;
- Elbow tap flow measurement licensing considerations;
- Measurement uncertainty using elbow taps; and
- Modifications to Seabrook Technical Specifications.

2.0 SUMMARY

The procedure described in this report for verifying RCS total flow with normalized elbow tap flow measurements is similar to the Westinghouse procedure approved by the Nuclear Regulatory Commission (NRC) for application at Westinghouse 3-loop nuclear power plants. Applicability of the procedure is confirmed by comparing measured RCS elbow tap flow trends with best estimate flow trends, based on analysis and application of RCS hydraulic test data.

The evaluation of operating data from Seabrook Unit 1 has defined sufficiently accurate baseline parameters for both the elbow tap and calorimetric flow measurements. Flow changes measured by elbow taps over several fuel cycles are consistent with the predicted flow changes due to changes in RCS hydraulics, as shown on Figure 6-1. Application of the flow measurement procedure using normalized elbow tap measurements, described in Section 4.2, would result in the recovery of more than 3% flow, the apparent loss attributed to changes in hot leg temperature streaming.

The flow measurement uncertainty for this procedure is slightly less than the current NRC licensed value. While modifications to the Seabrook Technical Specifications will be needed to apply the elbow tap flow measurement procedure, no unreviewed safety questions have been identified.

Section 7 provides input to the evaluation process required to prepare a licensing submittal.

Appendix B provides the supporting significant hazards evaluation and marked-up Technical Specification changes.

3.0 RCS HOT LEG TEMPERATURE STREAMING

3.1 PHENOMENON

The RCS hot leg temperature measurements are used in control and protection systems to ensure temperature is within design limits, and in a surveillance procedure with secondary plant calorimetric power measurements to confirm RCS flow. The hot leg temperature measurement uncertainty can have a significant impact on PWR performance. A precise measurement of hot leg temperature is difficult due to the phenomenon defined as hot leg temperature streaming, i.e., large temperature gradients within the hot leg pipe resulting from incomplete mixing of the coolant leaving fuel assemblies at different temperatures. The magnitude of these hot leg temperature gradients where the temperatures are measured is a function of the core radial power distribution, mixing in the reactor vessel upper plenum, and mixing in the hot leg pipe.

Prior to application of LLLPs, the largest difference in fuel assembly exit temperatures at full power was no more than 30°F. The lowest temperatures were measured at fuel assembly exits on the outer row of the core. Flow from a fuel assembly in the center of the core mixes with coolant from nearby fuel assemblies as it flows around control rod guide tubes and support columns toward the hot leg nozzles. Flow from a fuel assembly on the outer row, separated from the center region flows by the outer row of guide tubes, has little opportunity to mix with hotter flows before reaching the nozzles, so a significant temperature gradient can exist at the nozzle.

Since hot leg flow is highly turbulent, additional mixing occurs in the hot leg pipe, and the maximum gradient where temperature is measured, 7 to 17 feet downstream, is less than at the nozzle. In 1968, gradients measured on the circumference of the pipe were as high as 7 to 10°F, so turbulent mixing in the pipe did not eliminate the gradient introduced at the core exit.

The 1968 measurements and subsequent measurements showed that the highest temperatures are in the top half of the pipe. The lowest temperatures are in the bottom half, as expected, since the colder water from the outer row of fuel assemblies is closest to the bottom of the hot leg nozzle.

Figure 3-1 illustrates a postulated flow pattern in the reactor vessel upper plenum between the core exit and the hot leg nozzle. Figure 3-2 illustrates typical temperature gradients at the core exit and on the hot leg circumference at the point where the temperatures are measured. The core exit and hot leg temperature gradients change only slightly (typically less than 2 % of the coolant temperature rise (ΔT) across the core) as the radial power distribution changes during a cycle.

3.2 HISTORY

Prior to 1968, there were no multiple temperature measurements on hot leg pipes, so temperature streaming gradients were undetected and resistance temperature detector (RTD)

locations were based on other criteria. During startup of a Westinghouse-designed 3-loop plant in 1968, RTDs on opposite sides of the hot leg pipes measured different temperatures. Recalibrations and special tests confirmed that the RTD measurements were valid, so it was concluded that the hot leg temperature differences resulted from incomplete mixing of flows leaving fuel assemblies at different temperatures. To confirm this conclusion, thermocouples were strapped to the outside of two hot leg pipes, and circumferential temperature gradients were detected that increased as core power increased. The maximum, full power temperature gradient was 10°F in one loop and 7°F in the other loop. Since only one RTD was used to measure hot leg temperature for the control and protection systems, the temperature measurement was not as accurate as intended.

With additional analyses and development, a new hot leg temperature measurement system was designed and installed at other plants after 1968 to compensate for hot leg temperature streaming gradients. The new system, called the RTD Bypass System, employed scoops in the hot leg piping at three uniformly spaced locations on the pipe circumference. Holes on the upstream side of the scoop collected small sample flows. The three sample flows, which were at different temperatures, were combined and directed through an RTD manifold where the measured temperature more closely represented the average hot leg temperature.

To eliminate personnel radiation exposure to the RTD Bypass System piping during plant shutdowns, many systems were replaced after 1988 with a system having three thermowell RTDs in each hot leg. The RTDs were installed at uniformly spaced locations, like the RTD bypass scoops, to retain the three measurements on the hot leg. In many cases the thermowell RTDs were installed inside the bypass scoops, so the average thermowell RTD measurement was the same as the temperature measured by the RTD Bypass System.

After 1968, additional hot leg streaming measurements were performed at 2-loop, 3-loop and 4-loop plants. The results of these measurements were used in several analyses to define hot leg temperature streaming uncertainties for protection setpoint calculations and safety analyses. Gradients measured in these tests varied from 7 to 9°F. After 1988, the thermowell RTD systems provided hot leg streaming data from the three RTDs in each hot leg. The gradients measured prior to 1991 varied from 2 to 9°F with most of the gradients measured at 5 to 7°F.

3.3 HOT LEG STREAMING IMPACT ON RCS FLOW MEASUREMENTS

Prior to 1988, no hot leg temperature measurement problems were reported, and no significant changes in streaming gradients were indicated. In 1988, the first indication of a streaming change occurred at a 4-loop plant, followed by similar occurrences in 1989 and 1990 at three more 4-loop plants. In all four cases, the coolant temperature rise across the core (ΔT) had increased from that measured in previous fuel cycles. Since coolant ΔT is a major input to RCS calorimetric flow measurement, the increased ΔT indicated that RCS flow had apparently decreased. Conversely, RCS elbow tap flow measurements indicated that flow had not changed. It was also noted that core exit temperature gradients had increased, with lower temperatures being measured at the edge of the core, as shown on Figure 3-3.

No additional analyses were performed in 1988 or 1989, since the calorimetric flow at those plants was still above the Technical Specification requirement. However, calorimetric flow measured at both units at a plant in 1990 was below the Technical Specification requirement. After additional data had been evaluated, the appropriate data from elbow taps and core exit thermocouples confirmed that RCS flow was adequate. The NRC was advised of the apparent low flow and the elbow tap flow and core exit thermocouple data, and concurred with the utility's conclusion that RCS flow was adequate for safe operation at 100% power for the cycle.

Other 4-loop plants and some 3-loop plants subsequently reported apparent reductions in RCS calorimetric flow. The reductions occurred at plants measuring hot leg temperatures with either an RTD bypass system or with thermowell RTDs. In some cases, the apparent flow was just at the minimum Technical Specification requirement, raising a concern that measured flows could be lower in future cycles, requiring additional analyses or alternate flow measurements to justify that flow is adequate.

The alternate flow measurement employing elbow tap flow meters to verify adequate flow has been reviewed and approved by the NRC for 3-loop plants. Elbow tap flow measurements are compared with an elbow tap measurement obtained concurrently with early cycle calorimetric flow measurements, when the effects of core exit and hot leg temperature gradients on the temperature measurement were minimal. If the comparison of elbow tap measurements shows that the flow has not changed, the flow is considered to be the same as determined by the initial calorimetric (baseline) flow.

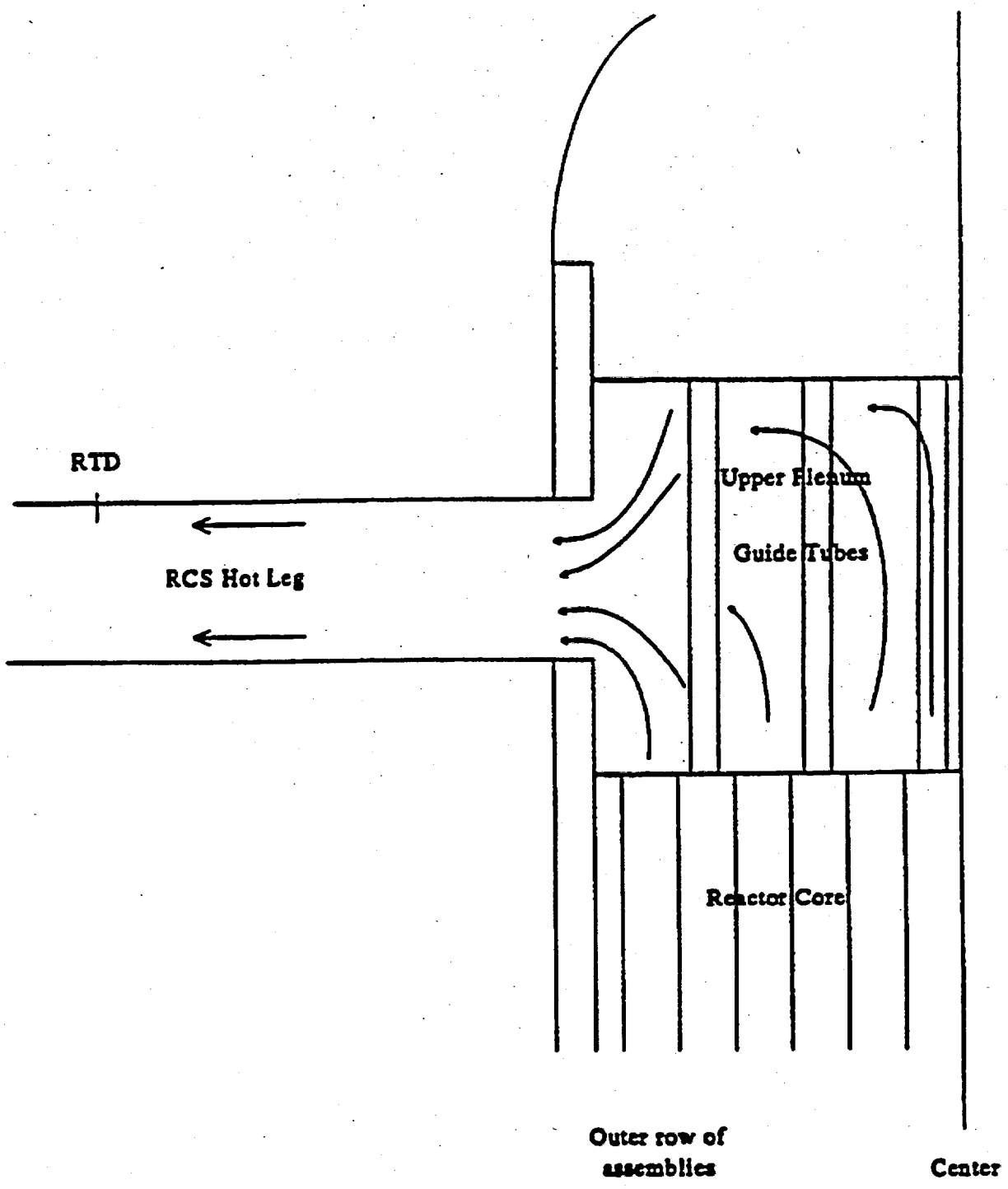
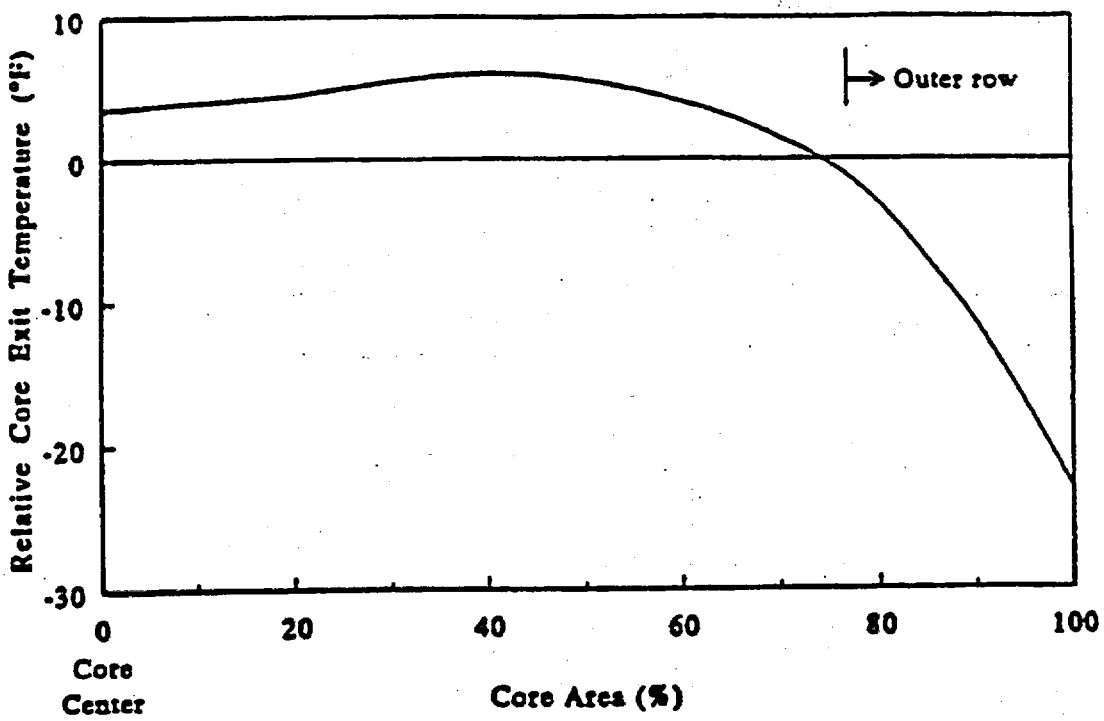


Figure 3-1
Upper Plenum and RCS Hot Leg Flow Patterns



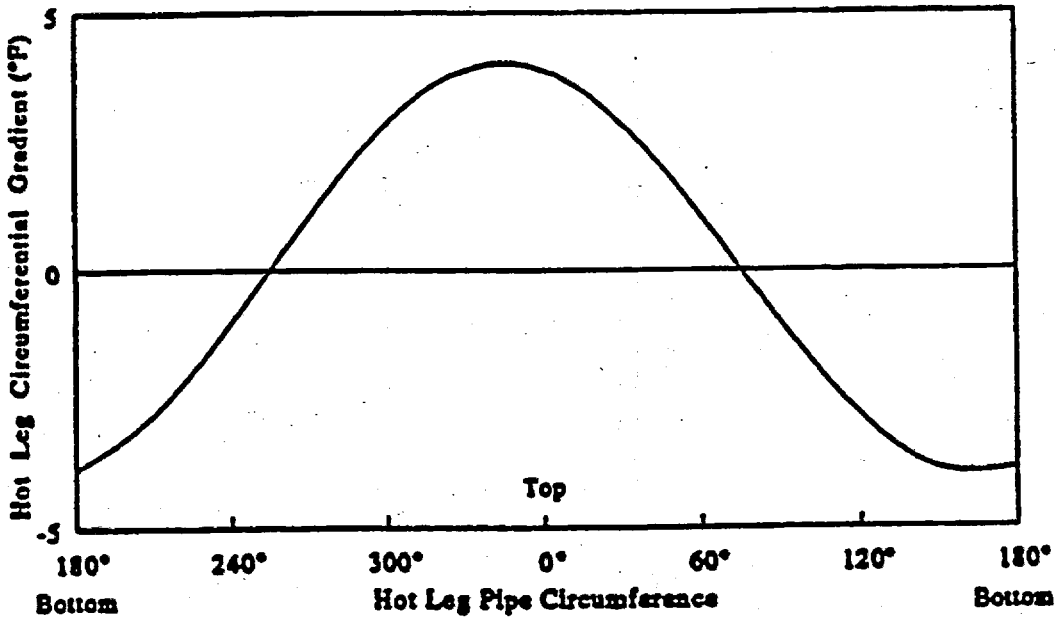


Figure 3-2
Typical Core Exit Temperature Gradient and
RCS Hot Leg Circumferential Temperature Gradient

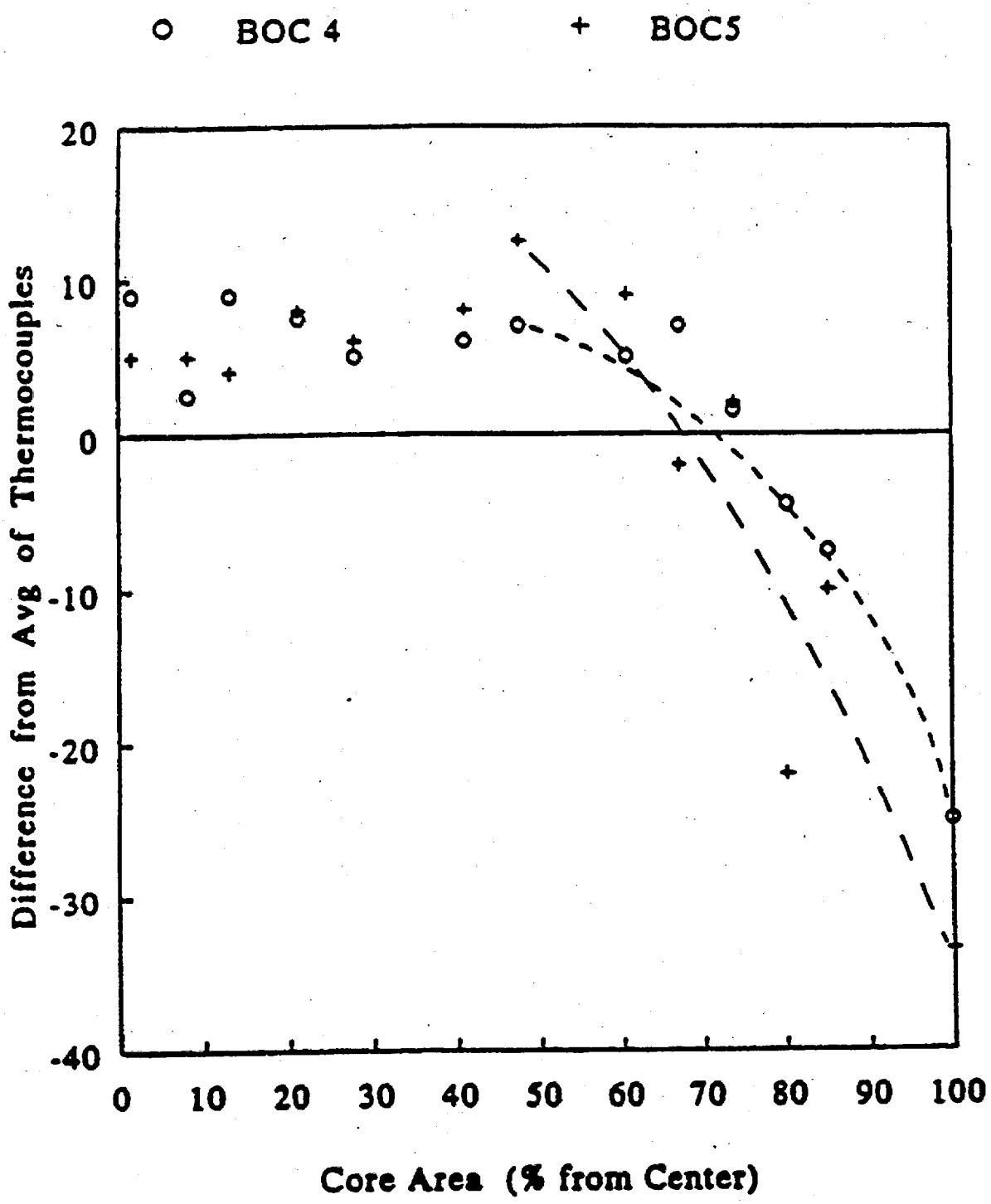


Figure 3-3
Typical Core Exit Temperature Change

4.0 ELBOW TAP FLOW MEASUREMENT APPLICATION

4.1 ELBOW TAP FLOW MEASUREMENTS

Elbow tap differential pressure (Δp) measurements are being used more frequently to determine if, or by how much, RCS flow has changed from one fuel cycle to the next. Elbow tap flow meters are installed in all Westinghouse PWRs on the RCS pump suction piping on each loop, as shown on Figure 4-1. The Δp taps are located on a plane 22.5° around the first 90° elbow. Each elbow has one high pressure and three low pressure taps connected to three redundant Δp transmitters. Elbow taps in this arrangement are used to define relative rather than absolute flows, due to the lack of upstream straight piping lengths. The Δp measurements are repeatable and thus provide accurate indications of flow changes during a cycle or from cycle to cycle.

Elbow tap flow meters (Reference 2) are a form of centrifugal meter, measuring momentum forces developed by the change in direction around the 90° elbow. The principal parameters defining the Δp for a specified flow are the elbow's radius of curvature and the flow channel diameter. Hydraulic tests described in Reference 2 demonstrated that elbow tap flow measurements have a high degree of repeatability and that the flow measurements are not affected by changes in the elbow surface roughness.

Phenomena that have affected other types of flow meters, or that might affect the elbow tap flow meters have been evaluated to determine if any of these phenomena would affect repeatability of the elbow taps. In addition, measurements at an operating plant equipped with a highly accurate RCS flow meter were compared with elbow tap flow measurements to demonstrate repeatability of the elbow taps. The results of these evaluations and comparisons are summarized below.

Venturi Fouling

Deposits (fouling) that collect on the surface and reduce flow area through the venturi throat affect Venturi flow meters. The fouling is caused by an electro-chemical ionization plating of copper and magnetite particles in the feedwater on the venturi surface, a process related to the velocity increase as flow approaches the smaller venturi flow area. There is no change in cross section to produce a velocity increase and ionization in an elbow, and surface roughness changes as experienced in venturi flow meters do not affect the elbow tap flow measurement.

Meter Dimensional Changes

The elbow tap flow meter is part of the RCS pressure boundary, so there would be only minimal dimensional changes associated with pipe stresses. Pressure and temperature would be essentially the same (full power conditions) whenever the flow is measured. Erosion of the elbow surface is unlikely since stainless steel is used, and velocities are low (42 fps) relative to erosion. The effects of dimensional change or erosion could only affect flow by changing elbow

radius or pipe diameter, both very large relative to any possible dimensional change. Therefore, the elbow tap flow meter is considered to be a highly stable flow measurement element.

Upstream Velocity Distribution Effects

The velocity distribution entering the steam generator outlet nozzle is skewed by its off-center location relative to the tube sheet. The out-of-plane upstream 40° elbow on the steam generator outlet nozzle skews the velocity distribution entering the 90° elbow with Δp taps. These velocity distributions, including the distribution in the elbow tap flow meter, will remain constant, so the elbow tap flow meter Δp/flow relationship would not change.

Steam generator tube plugging is usually randomly distributed across the tube sheet, so the velocity distribution approaching the outlet nozzle would not change. The velocity distribution in the outlet plenum could change if extensive tube plugging were to occur in one area of the tube sheet. However, the outlet plenum velocity approaching the outlet nozzle is small compared to the pipe velocity (6 fps vs. 42 fps), and this large change in flow area would significantly reduce or flatten an upstream velocity gradient. Therefore, any tube plugging, even if asymmetrically distributed, would not affect the elbow tap flow measurement repeatability.

Flow Measurement Comparisons

Leading Edge Flow Meters (LEFMs), ultrasonic devices installed in both reactor coolant loops at Prairie Island Unit 2 provide the data to confirm repeatability of the elbow tap flow meters. The comparisons covered 11 years of operation, during which a significant change in system hydraulics was made. One of the reactor coolant pump impellers was replaced, and the replacement impeller produced additional flow. The LEFM measurements after pump replacement were in agreement with the predicted change, and the elbow tap flow meters indicated similar changes, but slightly lower flows than measured by the LEFM.

The 11-year flow comparison showed that the average difference between elbow tap flows and LEFM flows was less than 0.3% flow. Another comparison performed before and after the pump replacement showed that the LEFM and elbow tap measurements agreed to within an average of 0.2% on the ratio of flows when one and two pumps were operating, thus further confirming the relative flow accuracy of elbow tap flow meters. These comparisons are listed on Table 4-1.

Elbow tap flow measurements have also been compared with flows based on the best estimate hydraulics analysis described in Section 5. The comparisons showed that elbow tap and best estimate flow trends were in close agreement at many plants, including plants that experienced changes in flow due to RCS hydraulics changes, including pump impeller replacement as described above, and steam generator tube plugging and replacement. The close agreement between elbow tap total flow and best estimate total flow occurs even where tube plugging and loop flows are significantly imbalanced. Elbow tap measurements over five cycles from a plant with tube plugging increasing from 4% to over 19%, and with a loop-to-loop plugging spread of 7% were well within the repeatability allowance (0.4%) when compared with the best estimate

flows. RCS flows measured by elbow taps after replacing the steam generators at this plant were also in good agreement with the predicted flow.

4.2 ELBOW TAP FLOW MEASUREMENT PROCEDURE

The elbow tap flow measurement procedure relies on the repeatability of elbow tap Δp measurements to accurately verify RCS flow. Comparison of the elbow tap Δp measurements obtained from one cycle to the next provides an accurate indication of the actual change in flow. When normalized to an early cycle calorimetric flow measurement, elbow tap Δp measurements define an accurate flow for all future cycles.

The elbow tap flow measurement procedure is described on the next page. Acronyms used in the procedure are listed and defined below. The baseline parameters for the procedure and their development (baseline calorimetric flow and baseline elbow tap flow coefficient) are presented in Section 4.3.

Acronyms used in Elbow Tap Flow Measurement Procedure

B — Baseline Flow Coefficient: defined by the elbow tap Δp and specific volume at T_{cold} measured at the beginning of the baseline cycle.

BCF — Baseline Calorimetric Flow: defined by calorimetric flows measured in early cycles with minimal impact from core radial power distribution.

BEF — Best Estimate Flow: estimated RCS flow for the baseline cycle, based on the best estimate hydraulics analysis.

CCF — Current Cycle (calorimetric) Flow: correction to the Baseline Calorimetric Flow (BCF) to account for changes in flow, using the elbow tap flow ratio (R) or the estimated flow ratio (R'). CCF defines the RCS flow for the current cycle.

CEF — Cycle Estimated Flow: estimated RCS flow for the current cycle, based on actual RCS hydraulics changes.

K — Elbow Tap Flow Coefficient: current cycle flow coefficient defined by the elbow tap Δp and specific volume at Tcold measured at the beginning of the current cycle.

R — Measured Flow Ratio: elbow tap Δp ratio, defines the actual change in flow for the current cycle, used to define the Current Cycle Flow (CCF).

R' — Estimated Flow Ratio: defines the current cycle estimated change in flow relative to the baseline cycle Best Estimate Flow (BEF).

TSF — Technical Specification Flow: specified flow that must be confirmed by a flow measurement.

Flow Measurement Procedure

1. Determine the current cycle estimated flow ratio R′ (typically provided by Westinghouse; refer to the background information below) which is defined by dividing the Cycle Estimated Flow (CEF) by the baseline cycle Best Estimate Flow (BEF), using equation 1:

$$R' = CEF/BEF \qquad \text{(Eq. 1)}$$

where BEF and CEF are calculated flows based on analyses of baseline cycle and current cycle hydraulics.

2. Determine the current cycle elbow tap flow coefficient (K) with equation 2:

$$K = \Delta p * v \qquad \text{(Eq. 2)}$$

where: Δp = elbow tap Δp (inches H_2O, average of 12 elbow taps),

v = cold leg specific volume (ft^3/lb, average for 4 loops).

3. Determine the elbow tap measured flow ratio R with equation 3:

$$R = (K/B)^{1/2} \qquad \text{(Eq. 3)}$$

where: B = baseline elbow tap flow coefficient (from Section 4.3)

4. Compare the estimated flow ratio R′ with the elbow tap measured flow ratio R. If R is less than or equal to [1.004 * R′], R is within its repeatability allowance. Use R to calculate the Current Cycle Flow (CCF) with equation 4:

$$CCF = R * BCF \qquad \text{(Eq. 4)}$$

where: BCF= baseline calorimetric flow (gpm, from Section 4.3)

If R is greater than [1.004 * R′], R is not within its repeatability allowance. Use [1.004 * R′] to calculate the CCF, using equation 5 instead of equation 4:

$$CCF = [1.004 * R'] * BCF \qquad \text{(Eq. 5)}$$

If CCF does not meet or exceed TSF, reverify CCF including elbow tap measurements and CEF to confirm that RCS flow is below TSF. If the reverification confirms that flow is below TSF, the plant must enter the Technical Specification Action.

Background Information

During the refueling outage, the estimated flow ratio R′ is calculated by Westinghouse for the new cycle, accounting for known changes, e.g., core Δp changes, and steam generator tube plugging estimated by plant personnel. If the actual steam generator tube plugging differs from the estimated plugging, the R′ calculation must be reviewed and R′ must be redefined. For small differences in tube plugging, the following conservative correction to R′ may be applied: [0.2% decrease in total flow per 1% increase in total tube plugging].

The multiplier (1.004) applied to R' is an allowance for repeatability of the elbow tap flow measurements. The elbow tap flow measurement uncertainty presented in Appendix A includes elements (e.g., sensor and rack calibration allowances) that define a repeatability allowance for the flow measurement that is larger than 0.4%. A measured flow ratio R that is no greater than 0.4% above the estimated flow ratio R' will still define a conservative flow. Application of this acceptance criterion results in definition of a conservative future cycle flow, confirmed by both the elbow tap measurements and the best estimate hydraulics analysis.

4.3 BASELINE PARAMETERS FOR ELBOW TAP FLOW MEASUREMENTS

Baseline Calorimetric Flow

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Table 4-1

Comparisons of LEFM and Elbow Tap Flow Measurements

RCS Flow Measurement Comparisons at Full Power

gpm/loop

Loop Meter Date	A LEFM	A Elbow	B LEFM	B Elbow
February 1980	97519	(same)	97950	(same)
July 1981	98673	98309	97763	97267
August 1991	98724	98557	97543	97607

Ratio of Loop Flow with 1 Pump Operating to Loop Flow with 2 Pumps Operating

Loop meter date	A LEFM	A Elbow	B LEFM	B Elbow
December 1974	1.0819	1.0777	1.0852	1.0875
July 1981	1.0794	1.0816	1.0820	1.0820

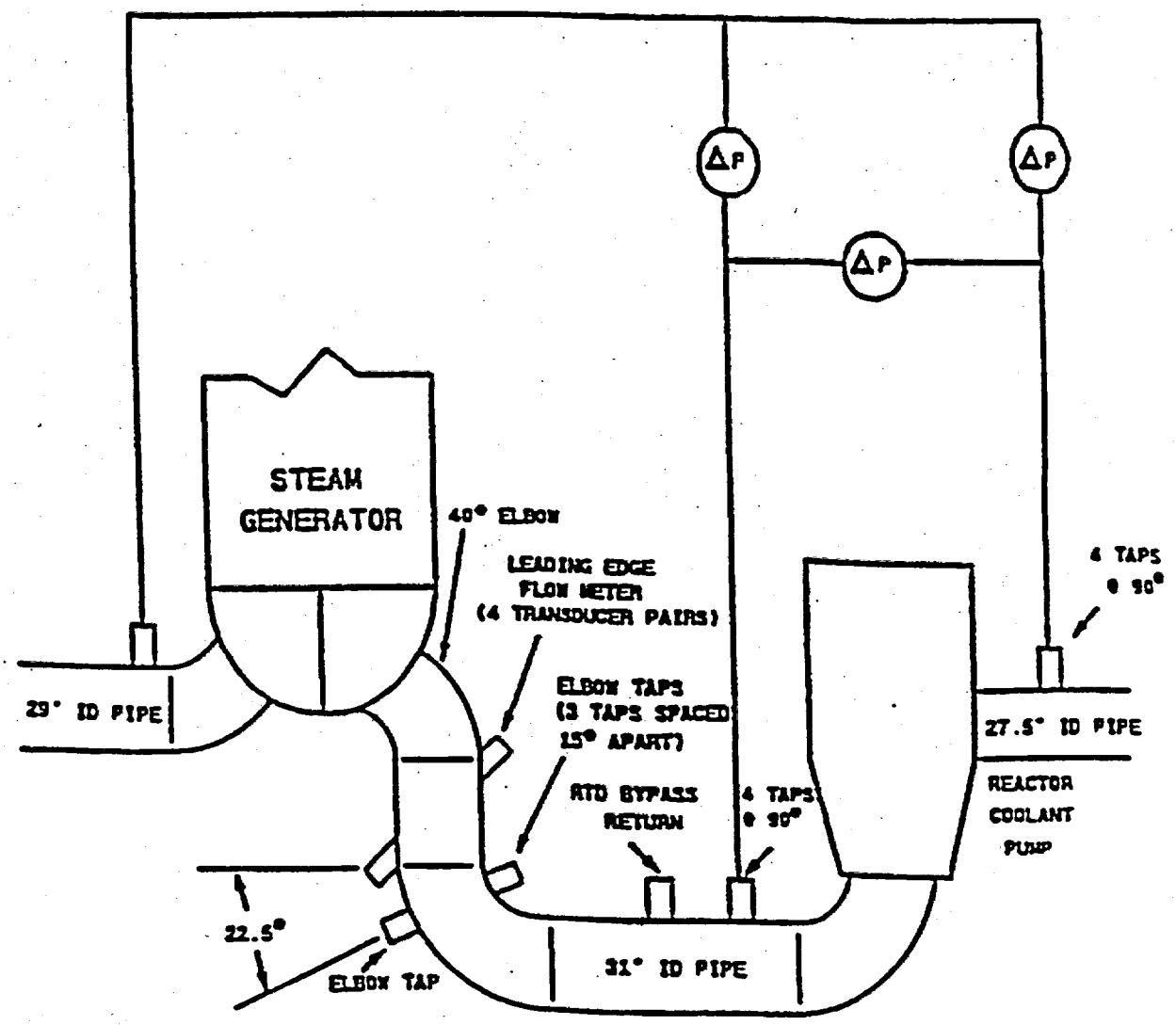

Figure 4-1
LEFM and Elbow Tap Locations in RCS Piping

5.0 BEST ESTIMATE RCS FLOW ANALYSIS

5.1 BACKGROUND

The procedure for calculating best estimate RCS flow was developed in 1974 and has been used to estimate RCS flow at all Westinghouse-designed plants. The procedure uses component flow resistances and pump performance with no margins applied, so the resulting flow calculations define a true best estimate of the actual flow.

Uncertainties in the best estimate hydraulics analysis, based on both plant and component test data, define a flow uncertainty of ±2% flow, indicating that actual flow is expected to be within 2% of the calculated best estimate flow. Since the uncertainty of a component flow resistance contributes only a fraction of the ±2% best estimate flow uncertainty, the uncertainty of a change in flow due to a known hydraulics change is smaller than ±2%, estimated to be no more than 10% of the predicted change in flow.

The most significant input to the best estimate hydraulics analysis was the test data collected at Prairie Island Unit 2, where ultrasonic LEFMs were installed. This program and other tests are described below.

5.2 PRAIRIE ISLAND HYDRAULICS TEST PROGRAM

The LEFM was installed in 1973 at Prairie Island Unit 2, on both loops as shown on Figure 4-1. Measurements were obtained during the hot functional and plant startup tests in 1974. In addition to the LEFM flows, component Δp taps were provided as shown on Figure 4-1 to obtain concurrent measurements of reactor vessel and steam generator Δps as well as reactor coolant pump dynamic head. Pump input power and speed measurements were also obtained.

The program collected data during plant heatup from 200°F to normal operating temperatures with one and two pumps operating. Full power flow measurements were obtained early in 1975. Subsequent flow and pump input power data were obtained in 1979, 1980, 1981 and 1991.

The LEFM accuracy for the Prairie Island plant measurements was established by a calibration test at Alden Laboratories, and by analysis of dimensional tolerances, to be ±0.67% of measured flow. The Alden test modeled the piping configuration both upstream and downstream from the metered pipe section. Tests performed with the ultrasonic transducers installed at several locations on the pipe circumference defined the optimum location for the transducers in the pipe section relative to the angular orientations of the upstream and downstream elbows.

The Prairie Island component Δps were based on measurements at the locations shown on Figure 4-1: hot leg, pump suction and pump discharge piping. The accuracy of the measurements was established by calibrations to be within ±1% of the measured Δp. Since the Δps are measured with common taps, the sum of the reactor and steam generator Δps should equal the pump Δp; these comparisons agreed to within 1%, further confirming the Δp measurement accuracy.

The RCS flows measured in 1974-75 were 5% higher than predicted, due to the following effects, evaluated in additional analyses.

Reactor Coolant Pump Performance

Reactor coolant pump performance was higher than predicted from hydraulic model tests, producing an additional 2% flow, partly due to pump impeller thermal expansion and partly due to conservatism in the hydraulics scaleup from model tests. With flow, head, input power and speed data, hydraulic and electrical efficiency were verified. Since the LEFM also measured reverse flows, the flow resistance of the pump impeller to reverse flow was confirmed to be as originally specified.

Reactor Vessel Flow Resistance

The reactor vessel flow resistance was lower than predicted from reactor vessel model tests and fuel assembly Δp measurements, producing an additional flow of almost 3%. Tests with one pump in operation provided additional data to confirm the division of flow resistances between vessel internals (total flow) and vessel nozzles (loop flow).

Steam Generator Flow Resistance

The steam generator flow resistance was the same as predicted from analysis, so changes in the analysis were not required. The large change in the predicted flow resistance resulting from the change in tubing Reynolds Number and friction factor during plant heatup was also confirmed by the flow resistance measurements.

Piping Flow Resistance

The reactor coolant piping flow resistance, 6% of the total system resistance, was reduced by about 25% to be consistent with measured component flow resistances, accounting for reduced Δp due to close coupling of components and elbows in the piping. Part of an elbow Δp loss occurs as increased turbulence in the downstream piping, but the loss is reduced if a component or another elbow is located at or close to the elbow outlet.

Flow vs Power

LEFM measurements at full power indicated that the Prairie Island Unit 2 RCS cold leg volumetric flow decreased by about 0.8% as the reactor was brought from zero to full power. This result confirmed the predicted effect of higher velocities in the core, hot leg, and steam generator tubes as temperatures at these locations increase above cold leg temperature. The RCS flow velocity in these regions increases by 5 to 12%, causing an increase in the total RCS flow resistance applied to the reactor coolant pumps. The resulting decrease in flow as reactor power increases from zero to 100% is plant specific, differing from 0.8% to 1.2%, depending on the plant specific hot leg and cold leg temperatures, and flow resistances of the affected components.

5.3 ADDITIONAL PRAIRIE ISLAND TESTS

The flow measurements in later years contributed additional data on system hydraulics performance, used to revise and further validate the hydraulics analyses, as described below.

Impeller Smoothing

LEFM and pump input power measurements were obtained at Prairie Island in 1979 and 1980 to reconfirm RCS flows and hydraulic performance. LEFM data indicated that RCS flows had decreased slightly, by 0.6 to 0.8%. Pump input power had also decreased by about 2%. After evaluating this data and other available information, it was concluded that the flow decrease was due to impeller smoothing, where the impeller surface roughness decreases due to wear or deposit buildup between high points on the impeller surfaces. The smoothing effect occurs within one or two fuel cycles after initial startup. This small flow decrease during early cycles has also been measured by elbow tap flow meters at several 3-loop and 4-loop plants.

Pump Impeller Replacement

The LEFMs were used at Prairie Island in 1981 to confirm RCS flows after replacement of a pump impeller. The new impeller performance was predicted to be higher than the original impeller, and a loop flow increase was predicted. The LEFM confirmed this prediction.

Elbow Tap Flow Comparison

LEFM measurements obtained in 1991 were compared with the 1980 data to confirm that the elbow taps measured the same flow changes over the same period. The comparison indicated that the elbow tap and LEFM loop flows were in good agreement, with an average difference in flow of less than 0.3% over 11 years.

5.4 SYSTEM FLOW RESISTANCE ANALYSES

Flow resistances are calculated for each component, based on the component hydraulic design data and on hydraulics coefficients resulting from analysis of test data such as, but not limited to, the Prairie Island test program. The component flow resistances are combined to define total system flow resistance, and then combined with the predicted pump head-flow performance to define individual loop and total RCS flow. The background and bases for the flow resistance calculations are described below.

Reactor Vessel

The reactor vessel flow resistance is defined in three parts.

a. The reactor core flow resistance is based on a full size fuel assembly hydraulic test, including Δps at RCS total flow through inlet and outlet core plates as well as the core.

b. The vessel internals flow resistance accounts for the Δps with total flow through the downcomer, lower plenum, and upper plenum. The flow resistances are determined from hydraulic model test data for each type of reactor vessel, based on Δp measurements within the model.

c. The vessel nozzle flow resistances include Δps based on loop flow through the inlet and outlet nozzles.

In addition, the overall analysis accounts for small flows that bypass the reactor core through the upper head, hot leg nozzle gaps, baffle-barrel gaps, and control rod drive thimbles.

Steam Generator

The steam generator flow resistance is defined in five parts: inlet nozzle; tube inlet; tubes; tube outlet; and outlet nozzle. The Prairie Island test program (Section 5.2) confirmed the overall flow resistance. The analysis accounts for the plugged or sleeved tubes in each steam generator, so loop specific flows can be calculated when different numbers of tubes are plugged or sleeved in each loop.

Reactor Coolant Piping

The reactor coolant piping flow resistance combines the flow resistances for the hot leg, crossover leg, and cold leg piping. The flow resistance for each section is based on an analysis of the effect of upstream and downstream components on elbow hydraulic loss coefficients, using the results of industry hydraulics tests. The total flow resistance was consistent with the measurements from the Prairie Island test program (Section 5.2).

5.5 BEST ESTIMATE RCS FLOW CALCULATIONS

The best estimate flow analysis defines baseline best estimate flow (BEF) and future cycle estimated flow (FEF) for the elbow tap flow measurement procedure. The calculation combines component flow resistances and pump performance predictions based on hydraulic model tests, and defines RCS loop flows at the desired power or temperature with any combination of pumps operating, with any fuel assembly design, and with different tube plugging in each steam generator. Estimated flows were in good agreement with calorimetric flow measurements from many plants before LLLPs were implemented. The calculated best estimate changes in flow from cycle to cycle have been in good agreement with changes measured by elbow taps.

6.0 EVALUATION OF SEABROOK FLOW PERFORMANCE

6.1 INTRODUCTION

RCS elbow tap flow and calorimetric flow measurements were obtained from NAESCO (Reference 1) and evaluated to determine RCS flow performance. The elbow tap data provided an accurate indication of the actual flow changes for comparison with predicted changes due to known modifications which affect the system hydraulics, such as steam generator tube plugging or changes in fuel design. The calorimetric flow data established a baseline flow and indicated the magnitude of the flow bias caused by changes in LLLP and hot leg temperature streaming. The Seabrook flow measurement evaluation is summarized in the following sections.

6.2 BEST ESTIMATE FLOW PREDICTIONS

Best estimate flow analyses defined flows for the seven Seabrook Unit 1 fuel cycles. The RCS hydraulics changes that affected flows after Cycle 1 are described below.

Impeller Smoothing

As stated in Section 5.3, impeller smoothing is expected to cause a flow decrease of about 0.6% to 0.8% flow after initial plant startup. For this analysis, the impeller smoothing flow decrease was applied as a -0.6% flow decrease prior to Cycle 2.

Steam Generator Tube Plugging:

As stated in Reference 2, the total number of tubes plugged through the first seven cycles was 74 tubes. The estimated impact on RCS flow due to this level of tube plugging is negligible, so an RCS flow decrease for tube plugging was not applied.

Fuel Design Changes

Standard Westinghouse fuel was used in the first three cycles. In Cycle 4, V5H fuel began to be loaded. Since standard fuel and V5H fuel have essentially the same hydraulic flow resistance, the change in fuel type had no impact on RCS flow. Thimble plugs have been installed in all seven cycles, so there has been no impact on RCS flow due to thimble plug removal.

Hydraulic Impact Summary

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6.3 EVALUATION OF ELBOW TAP FLOWS

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]+a,c

The elbow tap Δp measurements were corrected to account for the following effects:

Impact of Reactor Power on RCS Flow

[

]+a,c

Impact of RTD Bypass Elimination on Elbow Tap Flow

[

]+a,c

6.4 EVALUATION OF CALORIMETRIC FLOWS

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[

]+a,c

6.5 FLOW COMPARISONS

[

]+a,c

Table 6-1

Best Estimate, Elbow Tap and Calorimetric Flows

Baseline Best Estimate Flow = 402,384 gpm

Cycle	Best Est Flow %	Elbow Tap Flow %	Calorimetric Flow Meas gpm	Calorimetric Flow %
1	[	] +a,c	410,252	[ ] +a,c
2			409,189	
3			404,107	
4			397,388	
5			394,809	
6			395,890	
7			394,101	

Table 6-2

Baseline Calorimetric Flow Development

Cycle	Power%	Best EST Flow %	Calorimetric Flow Meas GPM	Calorimetric Flow Corr GPM	Calorimetric Flow Corr %
1	90		410,211	410,211	99.99
1	100		410,294	410,294	100.01
avg 1		100.00	410,252	410,252	100.00
2	100	99.40	409,189	411,659	100.34
avg 1&2				410,721	100.17

[ ] +a,c

+a,c

Figure 6-1
Seabrook RCS Flow History

7.0 ELBOW TAP FLOW MEASUREMENT SAFETY EVALUATION

7.1 BACKGROUND

Westinghouse was requested by North Atlantic Energy Service Corporation to modify the Seabrook Technical Specifications to reflect normalization of the elbow tap Δp transmitters to precision RCS flow calorimetric measurements performed at the Beginning Of Cycle (BOC) 1 and BOC2. The calculated instrument uncertainty is 2.3 % flow (without accounting for feedwater venturi fouling) and is based on the following:

1. the current plant configuration,

2. inclusion of the effects of RTD Bypass Elimination,

3. performance of a normalization of the elbow taps to the average of the RCS Flow Calorimetrics performed BOC 1 and BOC2, and

4. indication of RCS flow via the plant process computer or the control board meters.

This uncertainty is slightly less than the current NRC licensed value of 2.4 % Flow. Westinghouse determined that the instrument uncertainty calculation is reasonable and consistent with the Westinghouse approach approved by the NRC. The only significant difference is the assumption of normalization to previously performed RCS flow calorimetrics for cycles 1 and 2. This has been accounted for by the addition of instrument uncertainties usually considered to be zeroed out by normalization performed each cycle. Based on these calculations, the minimum RCS flow that must be measured at 100 % reactor thermal power (RTP) is maintained at 392,800 gpm.

7.2 LICENSING BASIS

The work performed is consistent with the requirements of 10CFR50.36 and information documented in WCAP-13181 for flow uncertainty calculations previously performed by Westinghouse. As noted above, the uncertainty calculations are essentially the same as those performed previously for Seabrook. Differences from previous calculations lie in the assumption of the normalization of the elbow taps to previously performed RCS flow calorimetric measurements (BOC 1 and BOC2) which requires inclusion of additional uncertainties in the determination of the indicated RCS flow uncertainty.

7.3 EVALUATION

The calculations performed are documented in Appendix A. The specific calculations performed were for the Precision RCS Flow Calorimetric for BOC 1 (BOC2 was provided by NAESCO), the Indicated RCS Flow (computer) and the Reactor Coolant Flow - Low reactor trip. The calculations for Indicated RCS Flow and Reactor Coolant Flow - Low reflect performance of

normalization of the elbow taps to the precision RCS flow calorimetrics performed BOC1 and BOC2. Additional instrument uncertainties were required to reflect the normalization.

It was determined that the difference between the current Safety Analysis Limit (87 % flow) and Nominal Trip Setpoint for Reactor Coolant Flow - Low (90 % flow) is sufficient to allow for the increased instrument uncertainties due to the normalization. The revised Allowable Value reflects the allowed calibration tolerance of the protection racks.

7.4 DETERMINATION OF NO UNREVIEWED SAFETY QUESTION

While modifications to the plant technical specifications have been determined to be necessary, no unreviewed safety questions have been identified. The seven questions typically answered for a 10CFR50.59 evaluation are noted as follows.

a. Will the probability of an accident previously evaluated in the SAR be increased?

An evaluation has not noted any increase in the probability of an accident. Sufficient margin exists to account for all reasonable instrument uncertainties, therefore no changes to installed equipment or hardware in the plant are required, thus the probability of an accident occurring remains unchanged.

b. Will the consequences of an accident previously evaluated in the SAR be increased?

The initial conditions for all accident scenarios modeled are the same and the conditions at the time of trip, as modeled in the various safety analyses are the same. Therefore, the consequences of an accident will be the same as those previously analyzed.

c. May the possibility of an accident which is different than any already evaluated in the SAR be created?

No new accident scenarios have been identified. Operation of the plant will be consistent with that previously modeled, i.e., the time of reactor trip in the various safety analyses is the same, thus plant response will be the same and will not introduce any different accident scenarios that have not been evaluated.

d. Will the probability of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

No significant changes to equipment installed in the plant are required. The Nominal Trip Setpoint for the Reactor Coolant Flow - Low reactor trip allows for the revised normalization process and associated increased uncertainties. There is no increase in the probability of a malfunction of this equipment.

e. Will the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

The plant conditions at the time of trip are unchanged. Therefore it is expected that the consequences of a malfunction of equipment important to safety will be the same as those currently modeled.

f. May the possibility of a malfunction of equipment important to safety different than any already evaluated in the SAR be created?

No significant changes to equipment installed in the plant are required. The setpoint remains well within normal operating bounds of the hardware, thus no failure mode not previously evaluated is introduced.

g. Will the margin of safety as defined in the BASES to any technical specifications be reduced?

No changes to the Safety Analysis assumptions or the Nominal Trip Setpoint for Reactor Coolant Flow - Low were required, therefore, the margin of safety as defined in the BASES will remain the same.

7.5 CONCLUSIONS

Based on the above it has been determined that the changes noted in Appendix B (attached) are acceptable for use at Seabrook. The changes noted allow RCS Flow to be measured by using recalibrated elbow tap Δp transmitters without the requirement to normalize to a precision RCS flow Calorimetric each cycle. The additional instrument uncertainties resulting from this process change have been accounted for and no change in the Nominal Trip Setpoint is required. The Allowable Value is changed to reflect the allowed calibration tolerance for the process racks. Therefore the time of reactor trip, as modeled in the various safety analyses is maintained, and the conclusions of the safety analyses remain unchanged.

APPENDIX A

INDICATED RCS FLOW
AND
REACTOR COOLANT FLOW - LOW REACTOR TRIP
INSTRUMENT UNCERTAINTIES

Table A-1

Flow Calorimetric Instrumentation Uncertainties (BOC1)

+a,c

	T_{FW}	P_{FW}	ΔP_{FW}	P_{STM}	T_{HOT}	T_{COLD}	P_{RCS}
SCA							
M&TE							
SRA							
SPE							
STE							
SD							
R/E							
READOUT1							
READOUT2							
BIAS							
CSA							
# INST USED							
UNITS	°F	PSIA	%DP	PSIA	°F	°F	PSIA
INST SPAN	720	1500	134	1300	100	100	800
NOMINAL	444	1033	100	966.5	612.2	556.3	2250

Table A-2

Flow Calorimetric Sensitivities

FEEDWATER FLOW	
FA	[ ] +a,c
TEMPERATURE	
MATERIAL	
DENSITY	
TEMPERATURE	
PRESSURE	
DELTA P	
FEEDWATER ENTHALPY	
TEMPERATURE	
PRESSURE	
hS	1193.5 BTU/LBM
hF	423.8 BTU/LBM
Δh(SG)	769.7 BTU/LBM
STEAM ENTHALPY	[ ] +a,c
PRESSURE	
MOISTURE	
HOT LEG ENTHALPY	
TEMPERATURE	
PRESSURE	
hH	634.5 BTU/LBM
hC	557.2 BTU/LBM
Δh(VESS)	77.3 BTU/LBM
Cp(TH)	1.59 BTU/LBM-°F
COLD LEG ENTHALPY	
TEMPERATURE	[ ] +a,c
PRESSURE	
Cp(TC)	1.32 BTU/LBM-°F
COLD LEG SPECIFIC VOLUME	
TEMPERATURE	[ ] +a,c
PRESSURE	

Table A-3

Calorimetric RCS Flow Measurement Uncertainties

Component	Instrument error	Flow uncertainty
FEEDWATER FLOW		+a,c
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		
BIAS VALUES		
STEAM PRESSURE ENTHALPY		
PRESSURIZER PRESSURE ENTHALPY – COLD LEG SPECIFIC VOLUME		
PRESSURIZER PRESSURE ENTHALPY – COLD LEG PRESSURE		
PRESSURIZER PRESSURE ENTHALPY – HOT LEG PRESSURE		
FLOW BIAS TOTAL VALUE		

Table A-3 (Cont)
Calorimetric RCS Flow Measurement Uncertainties

Component	Instrument Error	Flow Uncertainty
*, **, +, ++ INDICATE SETS OF DEPENDENT PARAMETER		+a,c
SINGLE LOOP UNCERTAINTY (NO BIAS)		[ ]
N LOOP UNCERTAINTY (NO BIAS)		
N LOOP UNCERTAINTY (WITH BIAS)		

Table A-4

Cold Leg Elbow Tap Flow Uncertainty (Process Computer)

INSTRUMENT UNCERTAINTIES	% DP SPAN	% FLOW
PMA		+a,c
PEA		
SCA		
SM&TE		
SRA		
SPE		
STE		
SD		
RCA		
RM&TE		
RTE		
RD		
A/D		
A/D drift		
BIAS		

FLOW CALORIMETRIC BIAS

FLOW CALORIMETRIC = 2.1

INSTRUMENT SPAN = 120 % FLOW

NUMBER CHANNELS PER LOOP = 3

N LOOP RCS FLOW UNCERTAINTY = 2.3 % FLOW

Table A-5
Low Flow Reactor Trip

	% DP SPAN	% FLOW SPAN
PMA1		
PEA		
SCA		
SM&TE		
SRA		
SPE		
STE		
SD		
RCA		
RM&TE		
RTE		
RD		
BIAS		
FLOW CALORIMETRIC BIAS		
FLOW CALORIMETRIC		

+a,c

INSTRUMENT RANGE = 0 TO 120.0 % FLOW

FLOW SPAN = 120.0 % FLOW

SAFETY ANALYSIS LIMIT = 87.0 % FLOW

ALLOWABLE VALUE = 89.6 % FLOW

NOMINAL TRIP SETPOINT = 90.0 % FLOW

+a,c

APPENDIX B

SUGGESTED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg}, ≤ 594.3°F

b. Pressurizer Pressure, ≥ 2185 psig*

c. Reactor Coolant System Flow shall be:

1. ≥ 382,800 gpm**; and,

2. ≥ 392,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined ~~by a precision heat balance measurement~~ to be within its limit ~~prior to operation above 95% of RATED THERMAL~~ POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

within 72 hours of exceeding 90% of Rated Thermal Power

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

***Minimum measured flow used in the Revised Thermal Design Procedure.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the updated FSAR assumptions and have been analytically demonstrated adequate to assure compliance with acceptance criteria for each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits of 594.3°F for T_{avg} and 2185 psig for pressurizer pressure are not exceeded.

RCS flow must be greater than or equal to, 1) the Thermal Design Flow (TDF) with an allowance for measurement uncertainty and, 2) the minimum measured flow used in place of the TDF in the analysis of DNB related events when the Revised Thermal Design Procedure (RTDP) methodology is utilized.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the specified limit.

LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip, thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure that could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full-power equivalent); and on increasing power, the Pressurizer High Water Level trip is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 50% of RATED THERMAL POWER), an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

replace "Nominal full" with "indicated"

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Emergency Feedwater System.

SEABROOK - UNIT 1 B 2-6

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TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level - High	8.0	4.20	0.84	≤92% of instrument span	≤93.75% of instrument span
12. Reactor Coolant Flow - Low	2.5	1.9	0.6	≥90% of ~~measured~~ indicated loop flow	≥89.6% of ~~measured~~ indicated loop flow
13. Steam Generator Water Level Low - Low	14.0	12.53	0.55	≥14.0% of narrow range instrument span	≥12.6% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	15.0	1.39	0	≥10,200 volts	≥9,822 volts
15. Underfrequency - Reactor Coolant Pumps	2.9	0	0	≥55.5 Hz	≥55.3 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥500 psig	≥460 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.