Westinghouse Non-Proprietary Class 3

WCAP-14754

# RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs

Westinghouse Energy Systems

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14754

# RCS FLOW VERIFICATION USING ELBOW TAPS AT WESTINGHOUSE 3-LOOP PWRs

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## 1.0 INTRODUCTION

Reactor Coolant System (RCS) secondary calorimetric-based flow measurements at many pressurized water reactor (PWR) plants 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 the implemetation of low leakage core loading patterns. In some cases, measured flow appears to have decreased to, or below, the minimum measured flow required by the Technical Specifications. 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, plants have relied on the repeatibility of RCS elbow tap flow meters to demonstrate that RCS flow has not decreased. This alternate approach confirms RCS flow by a normalization process using both calorimetric and elbow tap flow measurements obtained during the initial plant startup or subsequent cycles which have not been affected by changes in core radial power distribution.

A WOG Minigroup representing 15 3-loop plants was formed in 1995 to address the hot leg temperature streaming issue and its effect on calorimetric-based flow measurements. This report presents the results of one of the objectives of this program: to provide the justification of an alternate method for measuring total RCS flow using RCS elbow tap flow meters in 3-loop Westinghouse PWRs. Presently the Standardized Technical Specifications require the measurement of total RCS flow once per fuel cycle to demonstrate that the actual flow is greater than the minimum flow assumed in the plant safety analysis. The accepted measurement method based on RCS temperature and secondary calorimetric power measurements has inherent limitations. The proposed alternative method using elbow tap measurements normalized to early cycle calorimetric flow measurements minimizes these limitations.

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 flow measurements at Farley Nuclear Plant (FNP), Units 1 and 2;
- Elbow tap flow measurement licensing considerations;
- Measurement uncertainty using elbow taps;
- Modifications to Standard Technical Specifications;
- Modifications to Farley Technical Specifications; and
- Elbow tap flow measurement procedure.

## 2.0 SUMMARY

Westinghouse has defined the procedure for verifying RCS total flow with elbow tap flow measurements, normalized to calorimetric flow measurements obtained during early fuel cycles when core radial power distributions had minimal impact on the hot leg temperature streaming biases. The applicability of the procedure is confirmed by comparing RCS flow trends with best estimate flow based on analyses and application of RCS hydraulic test data.

The Farley Nuclear Plant (FNP) was selected to be the lead plant for the WOG Minigroup program. The evaluation of plant operating data from Farley Units 1 and 2 has defined sufficiently accurate baseline parameters for both the RCS elbow tap and calorimetric flow measurements. Flow changes measured by elbow taps obtained over several fuel cycles are consistent with the predicted flow changes due to changes in RCS hydraulics, as shown on Figures 6-1 and 6-2. Application of the procedure using normalized elbow tap measurements, described in Appendix D, will result in recovery of the apparent decrease in flow attributed to changes in hot leg temperature streaming. For Farley, the flow measurement uncertainty for this procedure is the same as the current Nuclear Regulatory Commission (NRC) licensed value listed in the FNP Technical Specifications. While modifications to the FNP Technical Specification Bases will be necessary to allow use of the alternate RCS flow measurement procedure, no unreviewed safety questions have been identified.

Although Farley has determined that changes to the FNP Technical Specifications Limiting Conditions for Operation or Surveillance Requirements are not necessary, other Westinghouse 3-loop PWRs may require changes depending upon plant design, instrumentation, and calibration practices. As such, Section 7 describes the evaluation process required to prepare a licensing submittal based on Standard Technical Specifications. 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 determine the RCS flow. The uncertainty of the hot leg temperature measurement 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 low leakage core loading patterns (LLLP), the largest difference in fuel assembly exit temperatures when operating at full power was typically no more than 30°F (17°C), with the lowest temperatures measured at the exit of assemblies 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 the hot leg nozzle.

Since hot leg flow is highly turbulent, additional mixing occurs in the pipe, and the maximum gradient at the point where temperature is measured, 7 to 17 feet (2 to 5 meters) from the

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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. Typically, the core exit and hot leg gradients remain relatively stable, changing only slightly as the radial power distribution changes during a fuel 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.

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### 3.3 Hot Leg Streaming Impact on RCS Flow Measurements

Before 1988, reports of hot leg temperature measurement problems were unusual, and no significant changes in streaming gradients were indicated. In 1988, the first significant 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 measured coolant temperature rise across the core ($T_{hot}$ - $T_{cold}$, or $\Delta T$) had increased from that measured in previous fuel cycles by as much as 3%. Since coolant $\Delta T$ is an input to the RCS calorimetric flow measurement, a $\Delta T$ increase of as much as 3% implied that RCS flow had apparently decreased by as much as 3%. Several other plants, including 3-loop plants, also reported apparent flow reductions. In some cases, the apparent flow was just at or above the minimum flow requirement in the Technical Specifications, raising a concern that measured flows could be lower in future cycles. In all cases, however, RCS elbow tap flows indicated that the actual flow had not changed.

Both units at one plant site in 1990 reported that calorimetric flows appeared to be below Technical Specification requirements. After additional data had been evaluated, data from elbow taps confirmed that RCS flow was adequate. The Nuclear Regulatory Commission (NRC) was advised of the apparent low calorimetric flow indication and the elbow tap flow data, and concurred with the licensee's conclusion that RCS flow was adequate for safe operation at full power for the remainder of the cycle.

### 3.4 Correlation of Power Distribution and Flow Changes

In the plants where apparent flow reductions were measured, it was noted that in all cases the core exit thermocouples measured much larger temperature gradients, approaching 60°F (33°C), as shown on Figure 3-3, due to much lower exit temperatures at the edge of the core. A review of core radial power distributions showed that the power generated in outer row fuel assemblies had decreased significantly from powers measured in earlier cycles, confirming the core exit temperature data.

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FIGURE 3-1

UPPER PLENUM and RCS HOT LEG FLOW PATTERNS

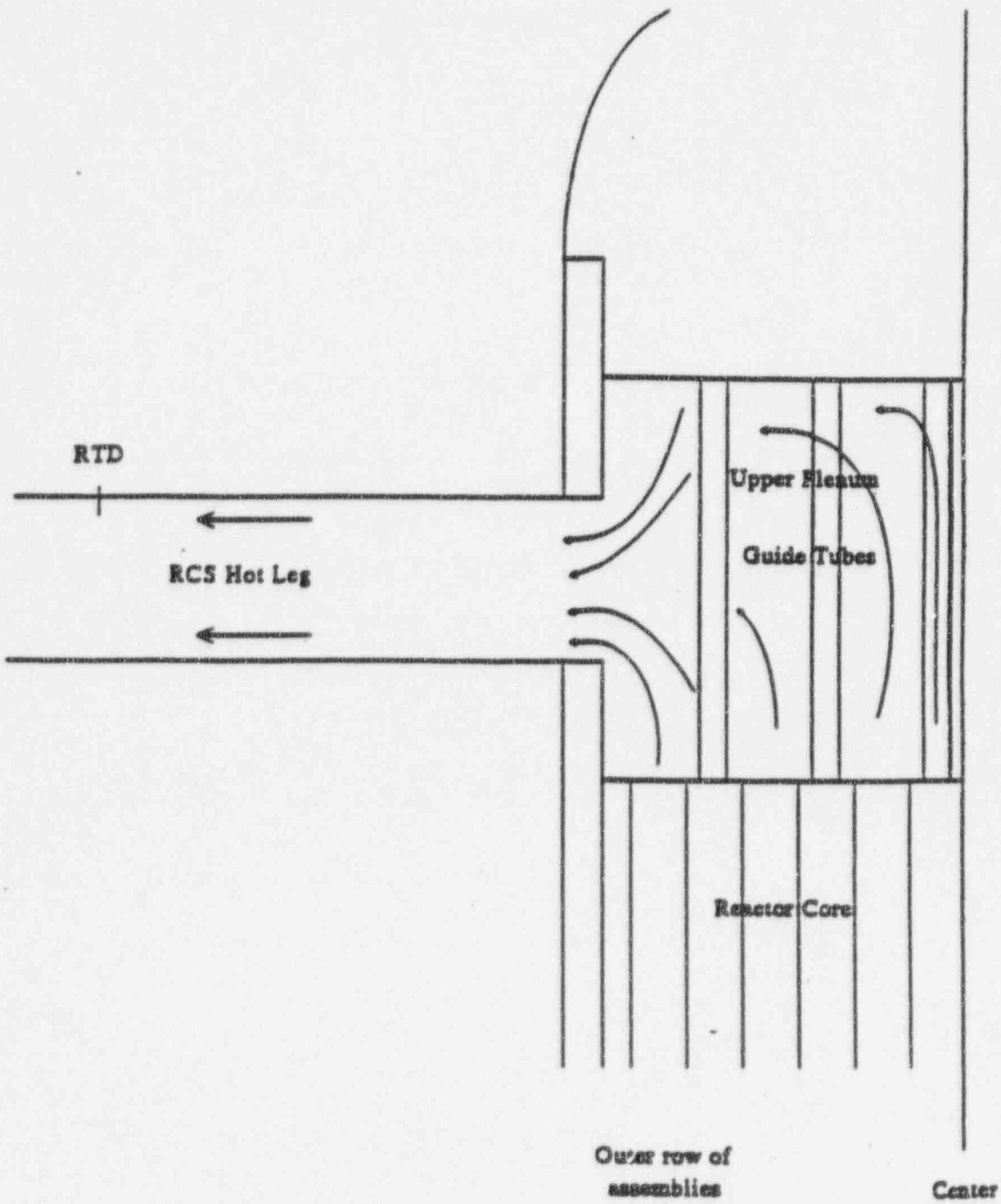
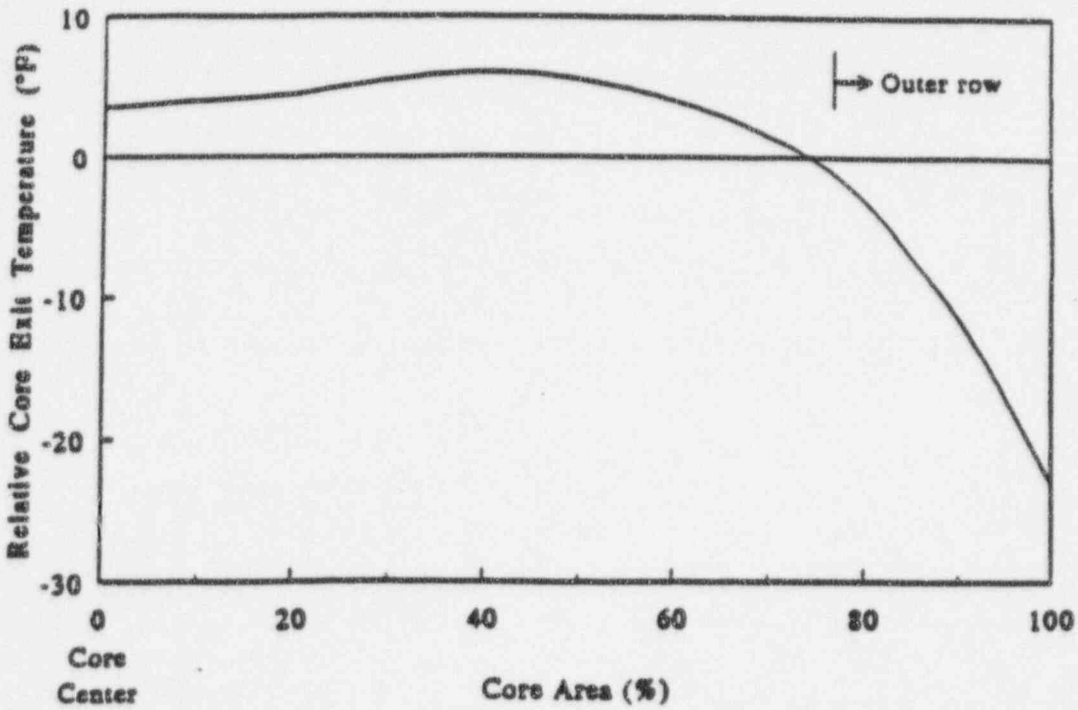


FIGURE 3-2

TYPICAL CORE EXIT TEMPERATURE GRADIENT and
RCS HOT LEG CIRCUMFERENTIAL TEMPERATURE GRADIENT



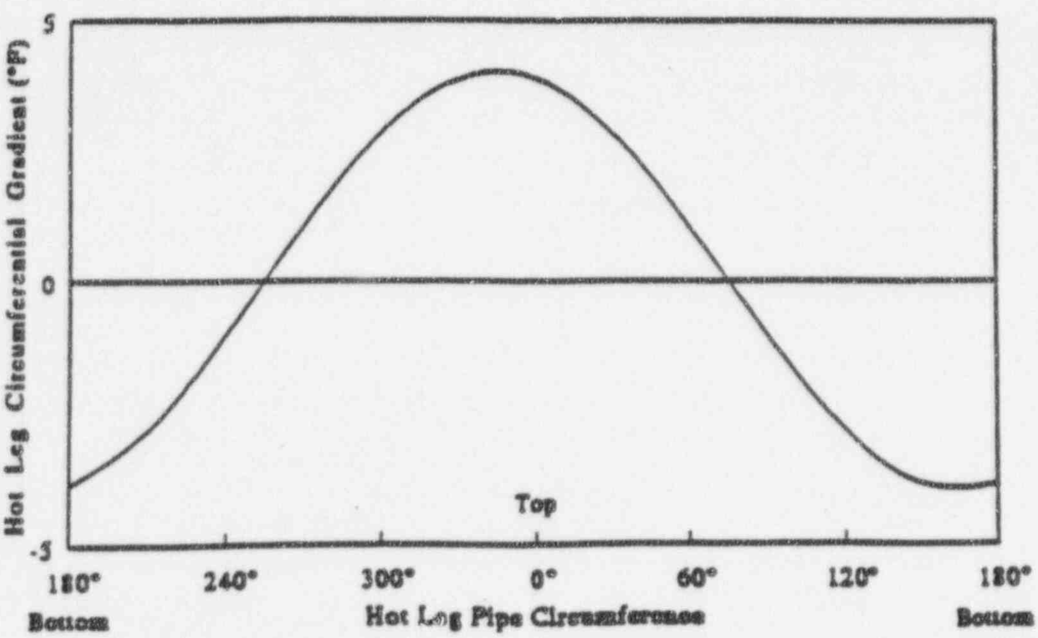



FIGURE 3-3

TYPICAL CORE EXIT TEMPERATURE CHANGE

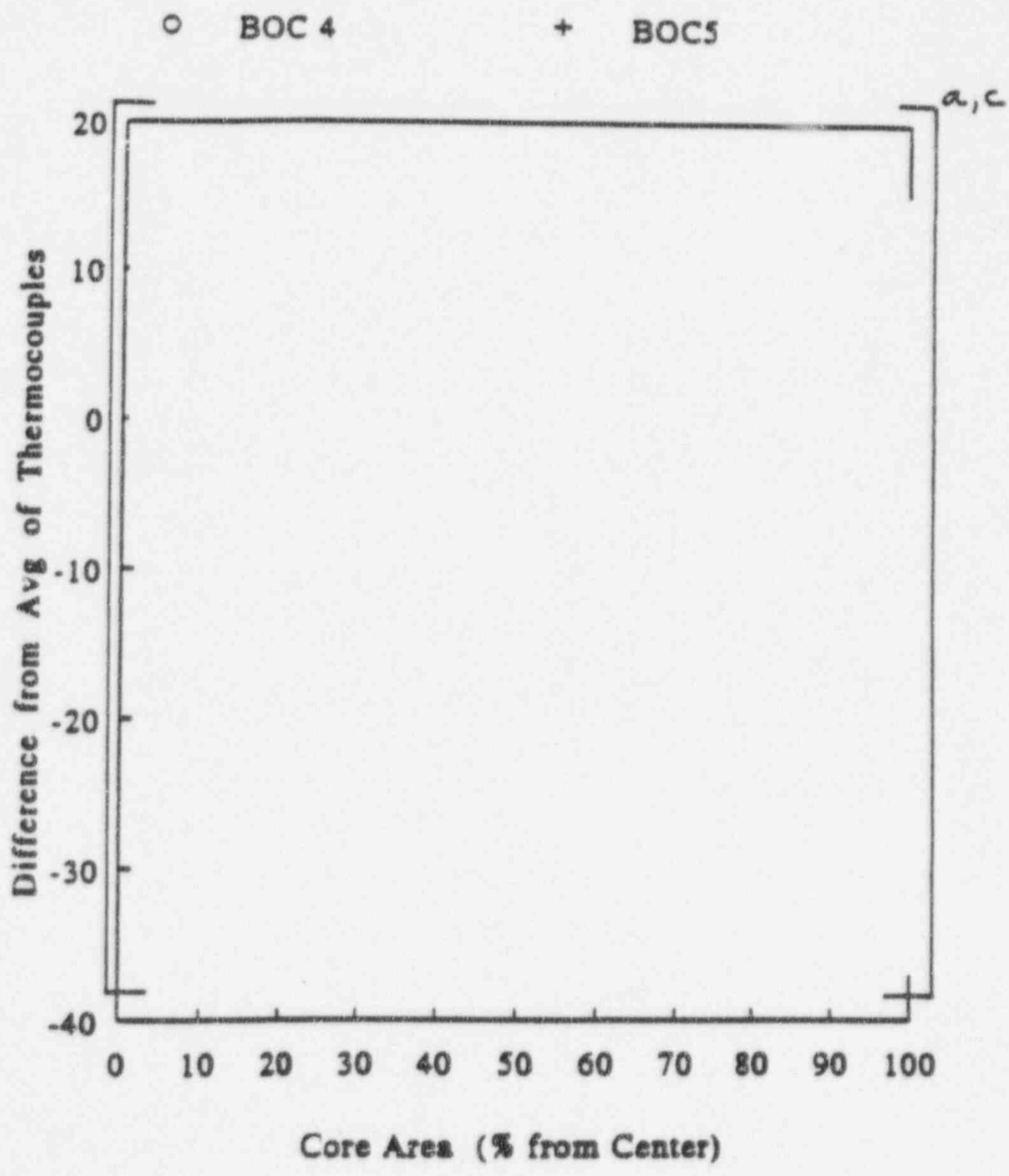
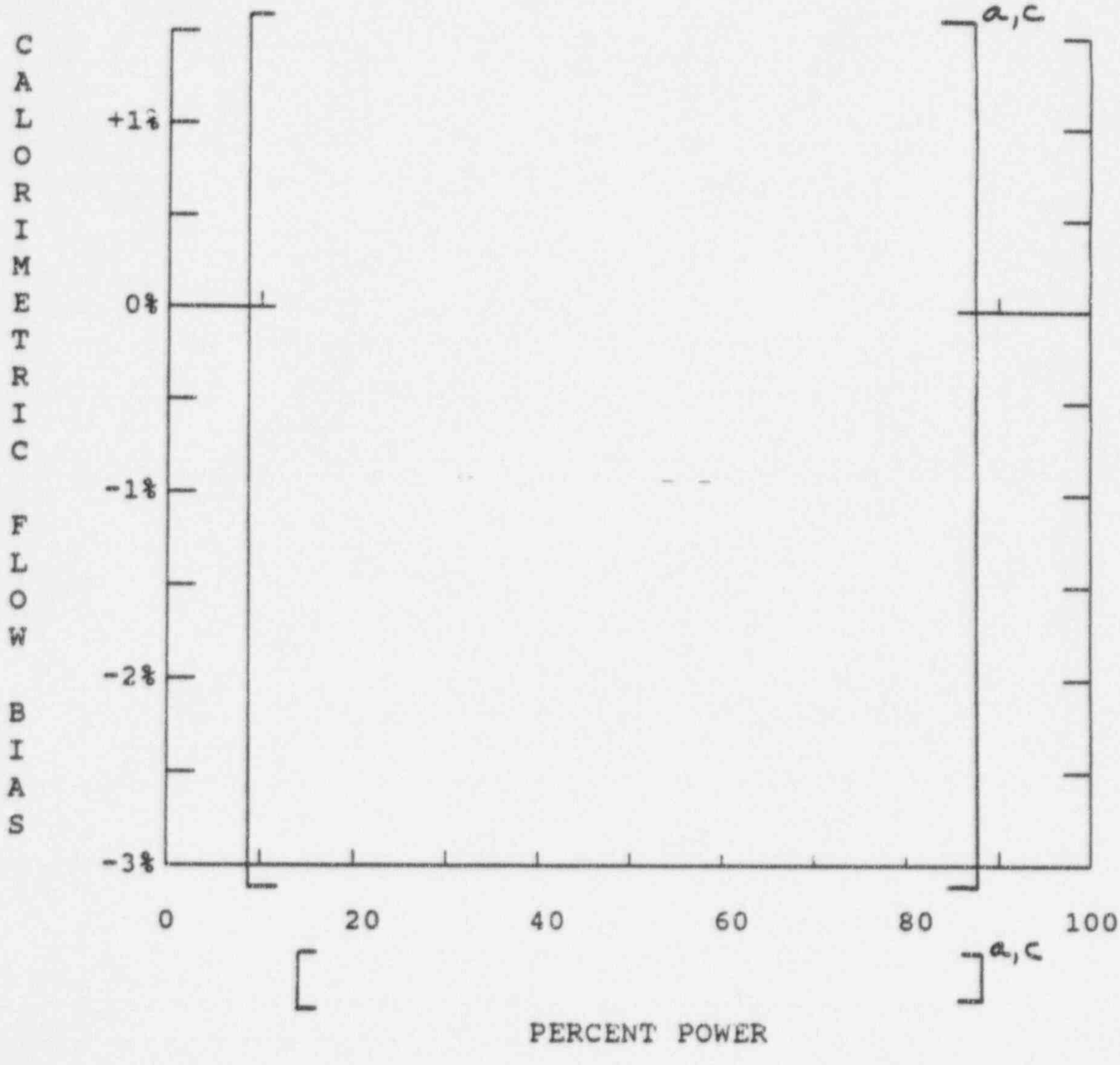


FIGURE 3-4

CORRELATION OF CORE POWER DISTRIBUTION

WITH CALORIMETRIC FLOW BIASES

BASED ON 97 FUEL CYCLES AT 12 3-LOOP PLANTS

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## 4.0 ELBOW TAP FLOW MEASUREMENT APPLICATION

### 4.1 Elbow Tap Flow Measurements

Elbow tap differential pressure ($\Delta P$) 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 for Prairie Island on Figure 4-1. The $\Delta P$ taps are located on a plane 22.5° around the first 90° elbow. Each elbow has three low pressure taps and one high pressure tap connected to three redundant $\Delta P$ transmitters. Elbow taps in this configuration are not used to define absolute flows due to the lack of straight piping lengths. The $\Delta P$ measurements are repeatable, however, providing accurate indications of changes in flow during plant operation and from cycle to cycle.

The RCS elbow tap flow meters[1] are a form of centrifugal meter measuring momentum forces developed by the change in direction around the 90° elbow. The principal parameters defining the $\Delta P$ for a specified flow are the radius of curvature of the elbow and the diameter of the flow channel through the elbow. Tests[1] have demonstrated that elbow tap flow measurements have a high degree of repeatability and that the flow measurements are not affected by changes in roughness of the elbow surface.

Specific phenomena that have affected other types of flow meters or that might affect the elbow tap flow meters in the RCS piping application have been evaluated to determine if these phenomena would affect repeatability of the flow measurement. In addition, measurements at Prairie Island Unit 2, where the highly accurate ultrasonic Leading Edge Flow Meter (LEFM) is installed, were compared with elbow tap measurements to confirm elbow tap flow measurement repeatability. The results of these evaluations and comparisons are summarized in the following paragraphs.

#### Venturi Fouling

Venturi flow meters in feedwater systems are affected by crud deposits (i.e., fouling) that affect surface roughness, local pressures, and flow area through the venturi throat. Fouling is apparently caused by an electro-chemical ionization plating of copper and magnetite particles in the feedwater on the venturi surfaces. The fouling process is directly related to the velocity increase as flow approaches the smaller venturi flow area. This condition is not present in an elbow since there is no change in cross section to produce a velocity increase and ionization. In addition, surface roughness changes as experienced in venturi flow meters do not affect the elbow tap flow measurement.

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

### Meter Dimensional Changes

The elbow tap flow meter is part of the RCS pressure boundary, so there are only minimal dimensional changes associated with pipe stresses, and pressure and temperature are the same (full power conditions) whenever flow measurements are made. Erosion of the stainless steel elbow surface is unlikely, and velocities are not large (42 fps) relative to erosion. The effects of a dimensional change or erosion could only affect flow by changing elbow radius or pipe diameter, and these dimensions are very large relative to a possible dimensional change. Therefore, elbow tap flow meters are considered to be a highly stable flow measurement element.

### Upstream Velocity Distribution Effects

The velocity distribution entering the steam generator outlet nozzle may be skewed by its off-center location relative to the tube sheet, and the velocity distribution entering the 90° elbow where the flow meter taps are located will be skewed by the out-of-plane upstream 40° elbow on the steam generator outlet nozzle. However, these velocity distributions, including the distribution in the elbow tap flow meter, will remain constant so the elbow tap flow meter $\Delta P$/flow relationship does not change.

Another upstream effect that was considered was steam generator tube plugging. However, tube plugging is typically distributed randomly across the tube sheet, so the velocity distribution approaching the outlet nozzle does not change as additional tubes are plugged. The velocity distribution could change if extensive tube plugging were to occur in one area of the tube sheet. However, the plenum velocity head approaching the outlet nozzle is small compared to the pipe velocity head (0.6 ft versus 27 ft), and the large change in flow area significantly reduces or flattens an upstream velocity gradient. Therefore, any tube plugging, even if asymmetrically distributed, does not impact elbow tap flow measurement repeatability.

### Flow Measurement Comparisons

The LEFMs installed at Prairie Island Unit 2 provided data to confirm repeatability of elbow tap flow meters. The comparisons, listed in Table 4-1, covered 11 years of plant operation, during which a significant change in system hydraulics was made. A reactor coolant pump impeller was replaced, and the replacement impeller produced additional flow. The LEFM data after pump replacement was in agreement with the predicted flow change, and the elbow tap flow meters indicated similar changes. The 11 year flow comparison shows that the average difference between elbow taps and LEFMs was less than 0.3% flow. Another comparison of data obtained before and after impeller replacement showed that measurements agreed to within 0.2% flow on the ratio of flows when one and two pumps were operating, thus further confirming the relative flow measurements from elbow tap flow meters.

## 4.2 Elbow Tap Flow Measurement Procedure

The elbow tap flow measurement procedure relies on repeatability of elbow tap measurements to obtain an accurate verification of RCS flow. Comparison of elbow tap measurements at or near full power from one cycle to the next provides an accurate indication of any change in flow. When normalized to early fuel cycle calorimetric flows, the elbow tap ΔP measurements provide the means to accurately verify flow for any future fuel cycle. The elbow tap procedure for verifying RCS flows is described in detail in the following paragraphs.

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## Imbalanced Loop Flows

Imbalanced loop flows can occur primarily due to steam generator tube plugging imbalances. The RCS elbow tap flow measurement procedure properly accounts for these imbalances in determining the RCS total flow.

The elbow tap flow measurement procedure defines only RCS total flow, even when loop flows may differ. Loop flow imbalances have no impact on reactor core heat transfer performance, since the reactor core flow resistance and open channel geometry results in an even flow distribution across and through the core. Due to the arrangement of the reactor vessel downcomer and lower plenum, it is unlikely that a loop flow imbalance would exist at the reactor core inlet.

In the RCS flow confirmation procedure described above, only the total elbow tap flows and total best estimate flows are compared. Analyses have shown that best estimate flows based on imbalanced loop flow calculations incorporating imbalances in tube plugging result in essentially the same flows as calculations based on an average number of plugged tubes. Estimates of total flow with large tube plugging imbalances have been found to agree well with measurements of total elbow tap flows. A comparison of estimated and measured flows for several cycles at a 3-loop plant with tube plugging as high as 19.5% in one loop, concurrent with plugging of 12.5% in another loop, is shown on Figure 4-2. The comparison shows that the estimated and measured flows agree well as average tube plugging progressed from 4% to 16%, and as the differences in plugging approached a maximum of 7%. Therefore, comparison of total flows in the procedure is justified.

TABLE 4-1

COMPARISONS of LEFM and ELBOW TAP FLOW MEASUREMENTS

RCS FLOW MEASUREMENT COMPARISONS AT FULL POWER
gpm/loop

| LOOP/METER: | A/LEFM | A/ELBOW | B/LEFM | B/ELBOW |
|---|---|---|---|---|
| DATE | | | | |
| Feb 1980 | 97519 | * | 97950 | * |
| Jul 1981 | 98673 | 98309 | 97763 | 97267 |
| Aug 1991 | 98724 | 98557 | 97543 | 97607 |

* - Normalized to LEFM Flow

RATIO OF FLOW WITH 1 PUMP OPERATING
TO FLOW WITH 2 PUMPS OPERATING

| LOOP/METER: | A/LEFM | A/ELBOW | B/LEFM | B/ELBOW |
|---|---|---|---|---|
| DATE | | | | |
| Dec 1974 | 1.0819 | 1.0777 | 1.0852 | 1.0875 |
| Jul 1981 | 1.0794 | 1.0816 | 1.0820 | 1.0820 |

FIGURE 4-1

LEADING EDGE FLOW METER AND ELBOW TAP FLOW METER LOCATIONS

AT PRAIRIE ISLAND UNIT 2

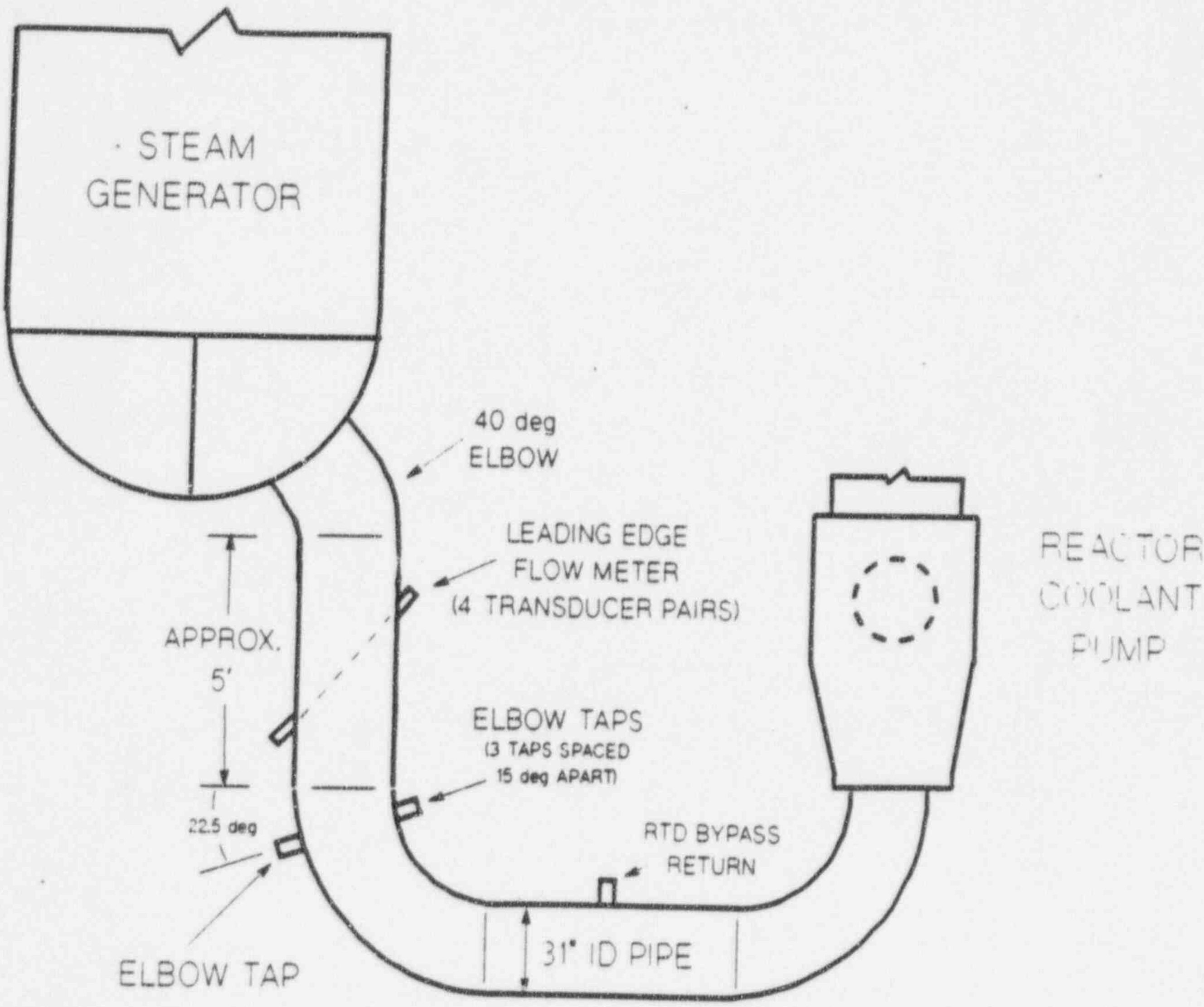
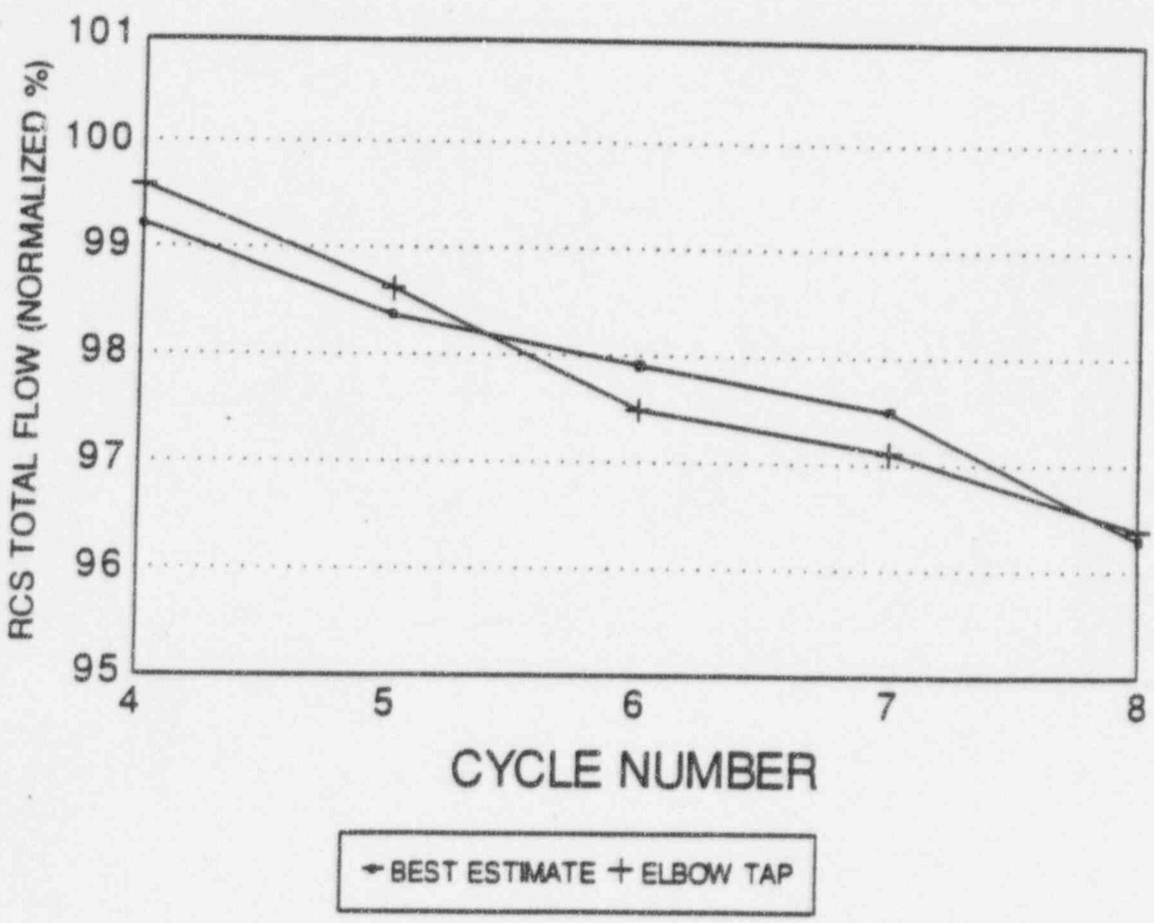


FIGURE 4-2

COMPARISON OF BEST ESTIMATE AND ELBOW TAP FLOWS

WITH IMBALANCED STEAM GENERATOR TUBE PLUGGING



## 5.0 BEST ESTIMATE RCS FLOW ANALYSIS

### 5.1 Background

Westinghouse developed the best estimate RCS flow calculational procedure in 1974 and has applied the procedure to estimate RCS flows 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 an accuracy of ±2% flow, indicating that the actual flow is within 2% of the calculated best estimate flow.

The best estimate hydraulics analysis was developed and confirmed by numerous component flow resistance tests and analyses. The most significant input was the test data collected at Prairie Island Unit 2, where ultrasonic Leading Edge Flow Meters (LEFMs) were installed. This program and other tests are described in the following sections.

### 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 LEFM flows, concurrent measurements of reactor vessel and steam generator flow resistances were also obtained, as well as pump dynamic head, input power and speed. The program collected data during 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 measurements 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 analyses of dimensional tolerances to be ±0.67% of measured flow. The Alden test modelled the piping configuration both upstream and downstream from the metered pipe section. Tests performed at several circumferential locations of the ultrasonic transducers defined the optimum location for the transducers in the pipe section relative to the upstream and downstream elbows.

The component $\Delta P$ accuracy for the Prairie Island measurements was established by calibrations to be within ±1% of the measured $\Delta P$. The sum of $\Delta P$s measured across the reactor and steam generator were within 1% of the pump $\Delta P$, confirming measurement accuracy.

The flows measured in 1974-75 were 5% higher than predicted. Analysis of the data led to the identification of the following changes to the hydraulics 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 the conservatism in the hydraulics scaleup from the model tests. With the flow, head, input power and speed data, hydraulic and electrical efficiency were verified. Since the LEFM also measures reverse flows, the 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 flow resistance measurements, producing an additional flow of almost 3%. Tests with one pump operating 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 the measured component flow resistances, accounting for reduced $\Delta p$ due to close coupling of components and elbows in the piping. Part of an elbow $\Delta p$ 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 RCS 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 (due to volumetric expansion (10-12%) of coolant in the core, hot leg, and steam generator tubes) increasing the total system flow resistance.

## 5.3 Additional Prairie Island Tests

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

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%. It was also noted that pump input power had decreased by about 2%. After evaluating this data and considering other available information, the pump hydraulics engineers concluded that the flow reduction was due to impeller "smoothing", where the impeller surface roughness decreases due to wear or crud buildup between high points on the impeller surfaces. The "smoothing" effect occurs within one or two fuel cycles after initial plant startup. This small flow reduction during the initial cycles has also been detected with elbow tap flow measurements at several other 3-loop and 4-loop plants.

Pump Impeller Replacement

LEFM measurements were obtained at Prairie Island in 1981 to confirm RCS flows after replacement of a pump impeller. The replacement impeller was predicted to have a higher performance than that of the original impeller, and an increase in loop flow was predicted. The LEFM data confirmed the 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 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 analyses of test data, such as, but not limited to, the Prairie Island hydraulics test program. The component flow resistances are combined to define total system 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 in the following paragraphs.

Reactor Vessel

The reactor vessel flow resistance is defined in three parts.

a. The reactor core flow resistance is based on a full size reactor fuel assembly hydraulic test, including the $\Delta$Ps at RCS total flow through the 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 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 overall flow resistance was confirmed by the Prairie Island hydraulics test program (Section 5.2). 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 loss coefficients, using the results of industry hydraulics tests. The total flow resistance was consistent with the measurements from the Prairie Island hydraulics test program (Section 5.2).

## 5.5 Best Estimate RCS Flow Calculations

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## 6.0 EVALUATION OF FARLEY RCS PERFORMANCE

RCS calorimetric flow and elbow tap flow measurements from Farley Units 1 and 2 were evaluated and compared with best estimate flow predictions to determine RCS flow performance. Calorimetric data from early fuel cycles at each unit established a baseline flow to compare with flows measured in later fuel cycles and to define the flow changes caused by hot leg temperature streaming biases. The elbow tap flows provide an indication of actual flow changes to compare with predicted changes due to known modifications which affect the system hydraulics, such as steam generator tube plugging or fuel design changes. The Farley RCS flow measurement evaluation is described in the following sections.

### 6.1 Evaluation of Calorimetric Flows

To apply an RCS flow measurement procedure based on elbow tap $\Delta p$, a baseline RCS calorimetric flow must be defined. Ideally, the baseline flow is based on early operating cycles, when the effect of hot leg temperature streaming on measured calorimetric flow is minimal. Cold leg temperature streaming, caused by incomplete mixing of the temperature gradient at the exit of long and short steam generator U-tubes, generally has a conservative impact on calorimetric flow measurements for 3-loop plants. At both Farley units, the Loop 3 cold leg RTD location differs from that in the other two loops, in a direction predicted to bias the Loop 3 flow measurement slightly higher. Although the resulting measurement may be conservative, the Loop 3 cold leg temperature was adjusted in this evaluation to account for the difference, so the adjusted Loop 3 flow is 1.5% less than indicated by the calorimetric data.

Farley Unit 1

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Farley Unit 2

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## 6.2 Best Estimate Flow Predictions

Several hydraulics changes occurred at the Farley units after the initial plant startup. The hydraulics changes and the associated best estimate flow changes are described below.

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### Steam Generator Tube Plugging

An increasing number of steam generator tubes were plugged at the Farley units after the initial fuel cycle. In some cases, some tubes were sleeved instead of plugged, resulting in a smaller increase in steam generator flow resistance. In later cycles, some plugs were removed and sleeves were installed, so the flow resistance could be reduced. Table 6-3 lists percent plugging equivalent to the flow resistance for the combination of plugged and sleeved tubes. Table 6-3 also lists the calculated best estimate flow reduction for each fuel cycle at both Farley units.

Reactor Core Hydraulics Changes

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<u>Farley Unit 1</u>

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<u>Farley Unit 2</u>

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## 6.4 <u>Flow Comparisons</u>

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### 6.5 Application: Elbow Tap Flow Measurements

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TABLE 6-1

FARLEY UNIT 1 CALORIMETRIC AND BASELINE FLOWS

| Fuel Cycle# | Loop 1 gpm | Loop 2 gpm | Loop 3 gpm | Average gpm | Percent of Baseline |
|---|---|---|---|---|---|
| 1 | 93583 | 95801 | 97501 | 95628 | - |
| 4 | 93836 | 94646 | 97166 | 95216 | - |
| 5 | 93148 | 95054 | 96642 | 94948 | - |
| [ | | | | | ]a,c |
| 6 | 93621 | 93608 | 95393 | 94207 | [ ]a,c |
| 7 | 94035 | 93855 | 94551 | 94147 | |
| 8 | 94470 | 92952 | 95094 | 94172 | |
| 9 | 94294 | 92375 | 94715 | 93795 | |
| 10 | 93069 | 91768 | 93443 | 92760 | |
| 12 | 94476 | 92997 | 93015 | 93496 | |
| 13 | 92600 | 91175 | 93953 | 92576 | |
| 14 | 92141 | 91296 | 91259 | 91565 | |

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# Data from Cycles 2, 3 and 11 are intentionally omitted.

TABLE 6-2

FARLEY UNIT 2 CALORIMETRIC AND BASELINE FLOWS

| Fuel Cycle# | Loop 1 gpm | Loop 2 gpm | Loop 3 gpm | Average gpm | Percent of Baseline |
|---|---|---|---|---|---|
| 2 | 98033 | 92455 | 94678 | 95055 | - |
| 3 | 98139 | 92157 | 94943 | 95080 | - |
| 5 | 97602 | 91966 | 94756 | 94775 | - |
| | | | | | a,c |
| 6 | 96407 | 90472 | 92837 | 93238 | a,c |
| 7 | 95590 | 89014 | 93126 | 92577 | |
| 8 | 94915 | 90601 | 92851 | 92789 | |
| 9 | 93256 | 89150 | 91881 | 91429 | |
| 10 | 94376 | 89702 | 90824 | 91634 | |
| 11 | 91891 | 87751 | 89864 | 89835 | |
| | | | | | a,c |

# Data from Cycles 1 and 4 are intentionally omitted.

TABLE 6-3

EQUIVALENT TUBE PLUGGING

AND RESULTING FLOW REDUCTIONS

| Fuel Cycle | Farley Unit 1 | | Farley Unit 2 | |
|---|---|---|---|---|
| | Plug % | ΔFlow % | Plug % | ΔFlow % |
| 2 | - | - | 2.83 | -0.57 |
| 3 | - | - | 2.86 | -0.58 |
| 4 | 2.77 | -0.56 | - | - |
| 5 | 2.77 | -0.56 | 3.69 | -0.76 |
| 6 | 2.86 | -0.58 | 4.76 | -0.99 |
| 7 | 2.87 | -0.58 | 5.03 | -1.05 |
| 8 | 2.93 | -0.60 | 7.66 | -1.65 |
| 9 | 3.10 | -0.64 | 8.11 | -1.76 |
| 10 | 3.34 | -0.68 | 7.10 | -1.52 |
| 11 | - | - | 7.11 | -1.52 |
| 12 | 3.59 | -0.73 | NA | NA |
| 13 | 3.75 | -0.77 | NA | NA |
| 14 | 6.91 | -1.47 | NA | NA |

TABLE 6-4

COMPARISON OF BEST ESTIMATE AND ELBOW TAP FLOWS

| Fuel Cycle | Farley Unit 1 | | Farley Unit 2 | |
|---|---|---|---|---|
| | BEF % | ETF % | BEF % | ETF % |
| 1 | | | | |
| 2 | | | | |
| 3 | | | | |
| 4 | | | | |
| 5 | | | | |
| 6 | | | | |
| 7 | | | | |
| 8 | | | | |
| 9 | | | | |
| 10 | | | | |
| 11 | | | | |
| 12 | | | | |
| 13 | | | | |
| 14 | | | | |

a,c

* Elbow tap flow normalized to best estimate for comparison.

FIGURE 6-1

COMPARISON OF FARLEY UNIT 1 RCS FLOWS

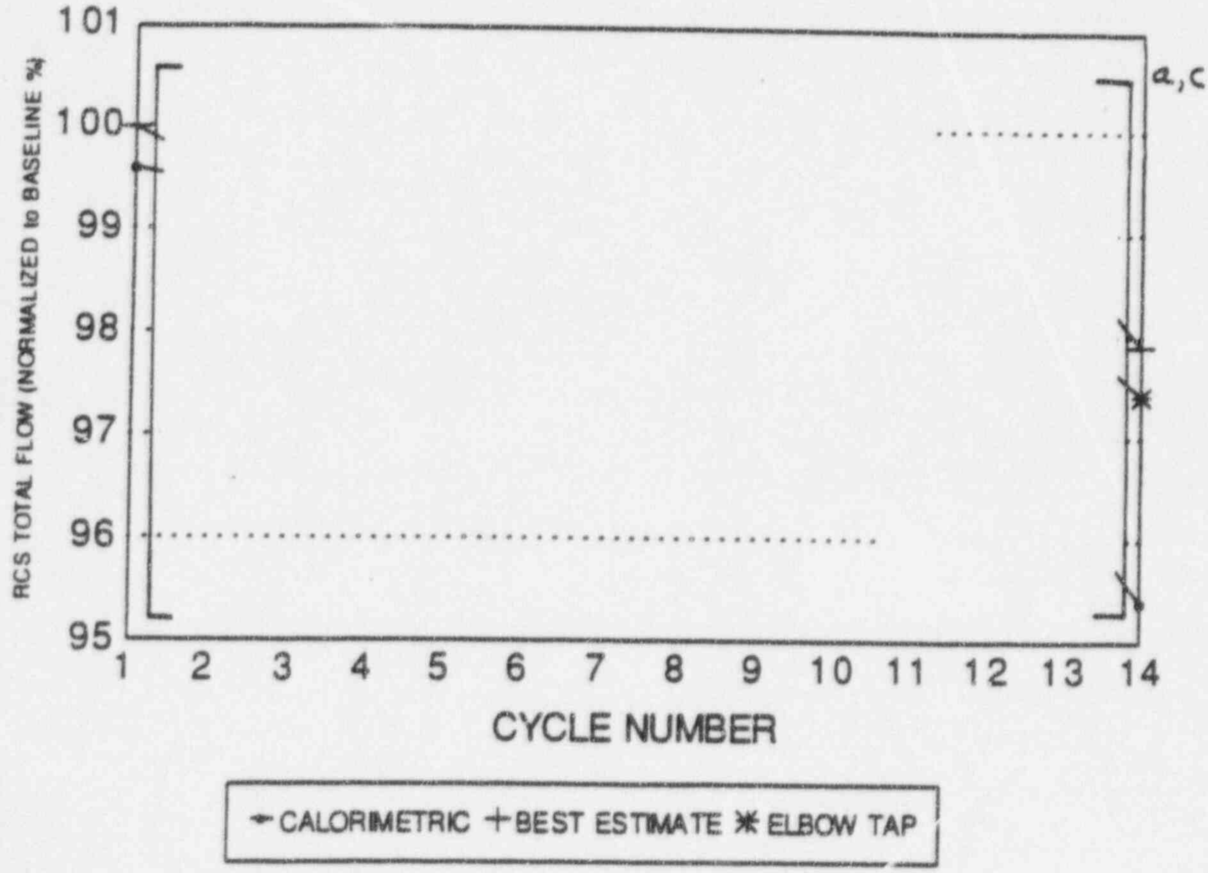


FIGURE 6-2

COMPARISON OF FARLEY UNIT 2 RCS FLOWS

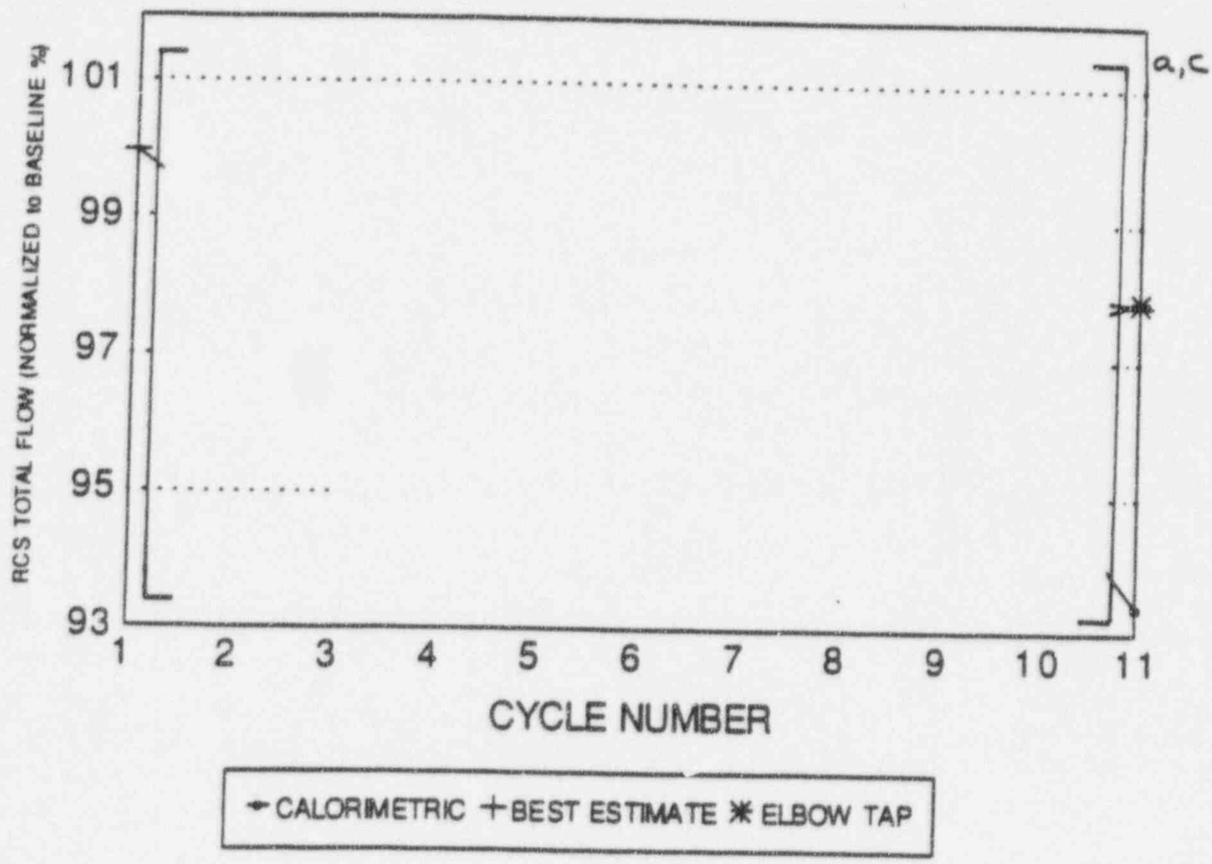


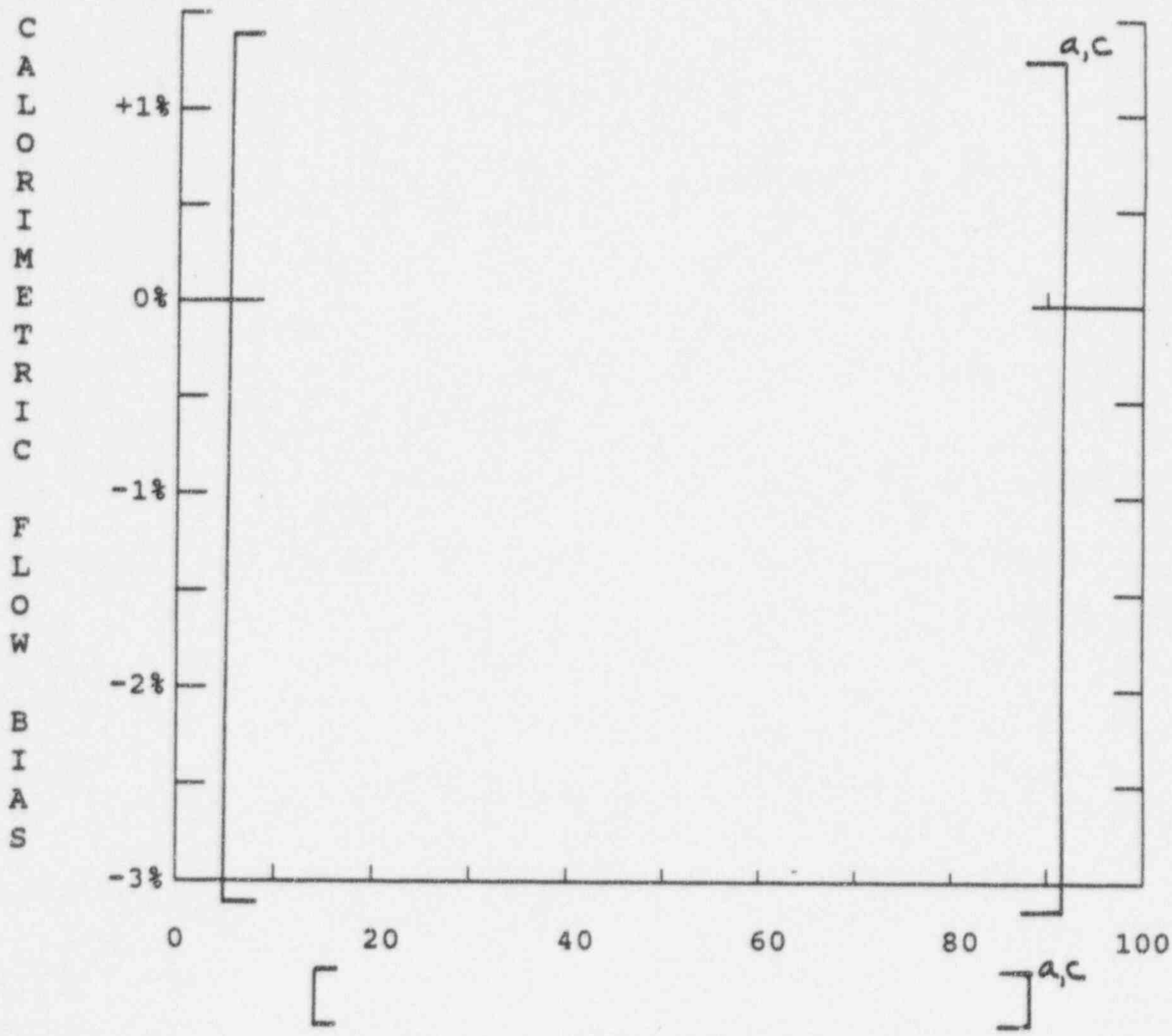
FIGURE 6-3

CORRELATION OF CORE POWER DISTRIBUTION

WITH CALORIMETRIC FLOW BIASES

FARLEY UNIT 1 o

FARLEY UNIT 2 +




PERCENT POWER

[ ]a,c

## 7.0 ELBOW TAP FLOW MEASUREMENT LICENSING CONSIDERATIONS

### 7.1 Background

Plant Technical Specifications require that an RCS total flow measurement be performed every 18 months to verify that sufficient RCS flow is available to satisfy the safety analysis assumptions. This surveillance is normally performed at the beginning of each operating cycle. Technical Specifications also require that a qualitative RCS flow verification (i.e., channel check) be performed every 12 hours during Mode 1. In addition, some plant Technical Specifications require a monthly total flow measurement as a periodic re-verification throughout the cycle. These surveillances ensure RCS flow is maintained within the assumed safety analysis value, i.e., Minimum Measured Flow (MMF).

The 18-month RCS flow surveillance is typically satisfied by a secondary power calorimetric-based RCS flow measurement; the monthly RCS flow surveillance is satisfied by a process computer algorithm using inputs from the RCS elbow tap $\Delta p$ instrument channels; and the 12 hour RCS flow surveillance is satisfied by control board RCS flow indicator readings. These surveillances and the RCS Low Flow reactor trip are interrelated since the calorimetric RCS flow measurement is used to correlate elbow tap $\Delta p$ measurements to flow, and the flow at the $\Delta p$ for the RCS Low Flow reactor trip is verified to be at or above the flow assumed in the safety analysis. The process computer output is normalized to the calorimetric flow. The uncertainty associated with the 18-month precision calorimetric is, therefore, included in the uncertainty calculations for the monthly RCS flow surveillance criterion and the RCS Low Flow trip.

The purpose of this evaluation is to support the use of elbow tap $\Delta p$ measurements as an alternate method for performing the 18-month RCS flow surveillance. Many plants in recent cycles have experienced apparent decreases in flow rates which have been attributed to variations in hot leg streaming, as discussed in previous sections of this report. These effects directly impact the hot leg temperatures used in the precision calorimetric, resulting in the calculation of apparently low RCS flow rates. In using the elbow tap $\Delta p$ method, the RCS elbow tap measurements are correlated (as described in Appendix D) to precision calorimetric measurements performed during an earlier cycle (or cycles) when the hot leg streaming effects were minimal.

### 7.2 Supporting Calculations

In order to implement the elbow tap $\Delta p$ method of measuring RCS flow, calculations must be performed to determine the uncertainty associated with the precision RCS flow calorimetric(s) for the baseline cycle(s). These calculations must account for the plant instrumentation, test equipment, and procedures which were in place at the time the calorimetric was performed.

In addition, uncertainty calculations must be performed for the indicated RCS flow (computer) and the RCS low flow reactor trip. These calculations must reflect the correlation of the elbow taps to the baseline precision RCS flow calorimetric(s) noted above. Additional instrument uncertainties are required to reflect this correlation. Appendix A contains uncertainty calculations which were performed using Farley specific inputs.

These uncertainty calculations may require revisions to certain plant specific safety analyses and associated protection and/or control system setpoint calculations which may be documented in plant specific WCAPs and/or engineering reports. In particular, an increase in the RCS total flow uncertainty due to the elbow tap $\Delta p$ method may impact the Westinghouse Improved Thermal Design Procedure (or the Revised Thermal Design Procedure) instrumentation uncertainties, which are used in deriving the Technical Specifications reactor core safety limits and the corresponding DNB limits. The low flow reactor trip setpoint uncertainty may also be adversely affected, requiring changes to the Technical Specifications nominal trip setpoint and allowable value.

## 7.3 Potential Document Impacts

Depending on the results of the uncertainty analyses described above, there are several Technical Specifications which may be impacted. Since the format and wording of the Technical Specifications can vary significantly between plants, the discussion here is generic in nature. Specific changes for a particular plant will need to be identified based on that plant's Technical Specifications.

One potential impact is on the RCS low flow reactor trip setpoint and allowable value in the Reactor Trip System Instrumentation table. For some plants the bases for the low flow trip setpoint and footnotes relating to the trip function may also be impacted.

Depending on the specific plant Technical Specifications format, the RCS total flow rate requirement may be in either the DNB Parameters or the RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor section. The specific minimum total flow rate limit and the associated measurement uncertainty may need to be changed in the appropriate section, depending on the results of the uncertainty calculations. For the Improved Standard Technical Specifications, it is recommended that the limiting condition for operation (LCO) and the surveillance requirements include two different DNB flow requirements - one to be used if a precision calorimetric measurement is performed and one to be used with an elbow tap $\Delta p$ measurement. The applicable Bases section will also require revisions to include a description of the elbow tap $\Delta p$ method of flow measurement.

Appendix B contains a sample markup of the Improved Standard Technical Specifications which may be used as a guide in making plant specific Technical Specifications revisions. This appendix also contains a sample 50.92 for use in preparing licensing documentation.

In the case of the Farley-specific instrument uncertainty analyses shown in Appendix A, the RCS flow uncertainty associated with the elbow tap $\Delta p$ method was the same as the current Technical Specification value. In addition, the DNB flow surveillance requirement in the current Farley Technical Specifications does not explicitly refer to either a precision heat balance or a calorimetric measurement method. RCS low flow reactor trip setpoint uncertainty calculations also verify that the current nominal trip setpoint and allowable value remain valid. Therefore, the only changes required are in the Bases of the DNB Specification, which must be revised to reflect the use of either the precision heat balance or the elbow tap measurement method to verify the RCS total flow requirement. Appendix C contains the Farley Technical Specifications markups reflecting only the Bases revisions, as well as the supporting 50.59.

In addition to these Technical Specification changes, each plant should review their FSAR (Chapter 7) and PLS documents to assure that any discussions of flow measurement are consistent with the elbow tap measurement method.

# APPENDIX A

# INDICATED RCS FLOW

# and

# REACTOR COOLANT FLOW - LOW REACTOR TRIP

# INSTRUMENT UNCERTAINTIES

TABLE A-1 BASELINE FLOW CALORIMETRIC
INSTRUMENTATION UNCERTAINTIES

+a,c

| (% SPAN) | $T_{FW}$ | $P_{FW}$ | $\Delta P_{FW}$ | $P_{STM}$ | $T_{HOT}$ | $T_{COLD}$ | $P_{RCS}$ |
|---|---|---|---|---|---|---|---|
| SCA = | | | | | | | |
| M&TE= | | | | | | | |
| SRA = | | | | | | | |
| SPE = | | | | | | | |
| STE = | | | | | | | |
| SD = | | | | | | | |
| R/E = | | | | | | | |
| RDOUT= | | | | | | | |
| BIAS= | | | | | | | |
| CSA = | | | | | | | |
| # INST USED | 2 | * | 1 | 3 | 1 | 1 | 3 |
| | DEG F | PSIA | % DP | PSIA | DEG F | DEG F | PSIA |
| INST SPAN | 500 | 2000 | 114 | 1200 | 100+ | 100+ | 800 |
| INST UNC. (RANDOM) = | | | | | | | |
| INST UNC. (BIAS) = | | | | | | | |
| NOMINAL = | 435 | 880 | | 845 | 605.9 | 543.3 | 2250 |

+a,c

* Feedwater pressure is not measured, but is assumed based on steam pressure. A conservative uncertainty value is used.

\+ $T_{AVG}$ span

TABLE A-2  FLOW CALORIMETRIC SENSITIVITIES

| | | | |
|---|---|---|---|
| **FEEDWATER FLOW** | | | |
| FA | | | +a,c |
| TEMPERATURE | = | | %/DEGF |
| MATERIAL | = | | % |
| DENSITY | | | |
| TEMPERATURE | = | | %/DEGF |
| PRESSURE | = | | %/PSI |
| DELTA P | = | | %/%DP |
| **FEEDWATER ENTHALPY** | | | |
| TEMPERATURE | = | | %/DEGF |
| PRESSURE | = | | %/PSI |
| hS | = | 1197.7 | BTU/LBM |
| hF | = | 413.8 | BTU/LBM |
| Dh(SG) | = | 783.9 | BTU/LBM |
| **STEAM ENTHALPY** | | | +a,c |
| PRESSURE | = | | %/PSI |
| MOISTURE | = | | %/0.25% MOISTURE |
| **HOT LEG ENTHALPY** | | | |
| TEMPERATURE | = | | %/DEGF |
| PRESSURE | = | | %/PSI |
| hH | = | 621.2 | BTU/LBM |
| hC | = | 538.6 | BTU/LBM |
| Dh(VESS) | = | 82.6 | BTU/LBM |
| Cp(TH) | = | 1.453 | BTU/LBM-DEGF |
| **COLD LEG ENTHALPY** | | | +a,c |
| TEMPERATURE | = | | %/DEGF |
| PRESSURE | = | | %/PSI |
| Cp(TC) | = | 1.225 | BTU/LBM-DEGF |
| **COLD LEG SPECIFIC VOLUME** | | | +a,c |
| TEMPERATURE | = | | %/DEGF |
| PRESSURE | = | | %/PSI |

TABLE A-3 CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTIES

| COMPONENT | | INSTRUMENT ERROR | FLOW UNCERTAINTY |
|---|---|---|---|
| | | +a,c | +a,c |
| FEEDWATER FLOW | | | |
| VENTURI | | % K | % FLOW |
| THERMAL EXPANSION COEFFICIENT | | | |
| TEMPERATURE | | DEGF | |
| MATERIAL | | % | |
| DENSITY | | | |
| TEMPERATURE | | DEGF | |
| PRESSURE | | PSI | |
| DELTA P | | % DP | |
| FEEDWATER ENTHALPY | | | |
| TEMPERATURE | | DEGF | |
| PRESSURE | | PSI | |
| STEAM ENTHALPY | | | |
| PRESSURE | | PSI | |
| MOISTURE | | % MOISTURE | |
| NET PUMP HEAT ADDITION | | % | |
| HOT LEG ENTHALPY | | | |
| TEMPERATURE | | DEGF | |
| STREAMING, RANDOM | | DEGF | |
| STREAMING, SYSTEMATIC | | DEGF | |
| PRESSURE | | PSI | |
| COLD LEG ENTHALPY | | | |
| TEMPERATURE | | DEGF | |
| PRESSURE | | PSI | |
| COLD LEG SPECIFIC VOLUME | | | |
| TEMPERATURE | | DEGF | |
| PRESSURE | | PSI | |
| BIAS VALUES | | | |
| 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 | (NO BIAS) | % FLOW |
| N LOOP UNCERTAINTY | (NO BIAS) | FLOW |
| N LOOP UNCERTAINTY | (WITH BIAS) | % FLOW |

TABLE A-4 COLD LEG ELBOW TAP FLOW UNCERTAINTY (PROCESS COMPUTER)

INSTRUMENT UNCERTAINTIES

| | % DP SPAN | % FLOW |
|---|---|---|
| PMA = | | |
| PEA = | | |
| SCA = | | |
| M&TE= | | |
| SRA = | | |
| SPE = | | |
| STE = | | |
| SD = | | |
| RCA = | | |
| M&TE= | | |
| RTE = | | |
| RD = | | |
| BIAS= | | |
| FLOW CALORIM BIAS = | | |
| FLOW CALORIMETRIC = | | |

+a,c

INSTRUMENT SPAN = 125.0 % Flow

NUMBER TAPS PER LOOP= 2

N LOOP RCS FLOW UNCERTAINTY = 2.3 % FLOW

TABLE A-5 LOW FLOW REACTOR TRIP

% DP SPAN % FLOW SPAN

+a,c

PMA1 =
PMA2 =
PEA =
SCA =
M&TE =
SRA =
SPE =
STE =
SD =
BIAS1=
RCA =
M&TE =
RTE =
RD =
BIAS =

INSTRUMENT RANGE = 0 TO 125.0 % FLOW

FLOW SPAN = 125.0 % FLOW

SAFETY ANALYSIS LIMIT = 85.0 % FLOW

NOMINAL TRIP SETPOINT = 90.0 % FLOW

TA = 4.0 % FLOW SPAN

CSA = [ ] +a,c % FLOW SPAN

MAR = [ ] +a,c % FLOW SPAN

APPENDIX B

SAMPLE 50.92 AND
SUGGESTED MODIFICATIONS TO
IMPROVED STANDARD TECHNICAL SPECIFICATIONS

1) NUCLEAR PLANT(S): PLANT NAME

2) SUBJECT: ELBOW TAP FLOW MEASUREMENT

3) The written safety evaluation of the revised procedure, design change or modification required by 10 CFR 50.59 (b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A 10CFR50.59(a)(1)

| | | | |
|---|---|---|---|
| 3.1 | Yes___ | No_X_ | A change to the plant as described in the FSAR? |
| 3.2 | Yes___ | No_X_ | A change to procedures as described in the FSAR? |
| 3.3 | Yes___ | No_X_ | A test or experiment not described in the FSAR? |
| 3.4 | Yes_X_ | No____ | A change to the plant Technical Specifications? (See Note on Page 2.) |

4) CHECK LIST - PART B 10CFR50.59(a)(2) (Justification for Part B answers is included on Page 2.)

| | | | |
|---|---|---|---|
| 4.1 | Yes___ | No_X_ | Will the probability of an accident previously evaluated in the FSAR be increased? |
| 4.2 | Yes___ | No_X_ | Will the consequences of an accident previously evaluated in the FSAR be increased? |
| 4.3 | Yes___ | No_X_ | May the possibility of an accident which is different than any already evaluated in the FSAR be created? |
| 4.4 | Yes___ | No_X_ | Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased? |
| 4.5 | Yes___ | No_X_ | Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased? |
| 4.6 | Yes___ | No_X_ | May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created? |
| 4.7 | Yes___ | No_X_ | Will the margin of safety as defined in the Bases to any Technical Specification be reduced? |

NOTES:

If the answer to any of the above questions is unknown, indicate under Section 5.0 REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written Safety Evaluation, the change review would require an application for license amendment as required by 10CFR50.59(c) and submitted to the NRC pursuant to 10CFR50.90.

5) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation. Due to the requirement for a change to the Technical Specifications, a Significant Hazards Consideration Evaluation has been prepared and is included as Attachment 2.

FOR FSAR UPDATE

Section: N/A Pages: N/A Tables: N/A Figures: N/A

Reason for/Description of Change:

6) SAFETY EVALUATION APPROVAL LADDER:

Prepared By: ____________________________ Date: ______________

Reviewed By: ____________________________ Date: ______________

## ELBOW TAP FLOW MEASUREMENT SAFETY EVALUATION

### 1.0 INTRODUCTION AND BACKGROUND

The purpose of this evaluation is to assess the impact of using elbow tap Δp measurements as an alternate method for performing the 18-month RCS flow surveillance on the licensing basis and demonstrate that it will not adversely affect the subsequent safe operation of the plant. The 18-month RCS flow surveillance is typically satisfied by a secondary power calorimetric-based RCS flow measurement. Many plants in recent cycles have experienced apparent decreases in flow rates which have been attributed to variations in hot leg streaming. These effects directly impact the hot leg temperatures used in the precision calorimetric, resulting in the calculation of apparently low RCS flows. In using the elbow tap Δp method, the RCS loop elbow tap measurements are correlated to precision calorimetric measurements performed during an earlier cycle (or cycles) when the hot leg streaming effects were minimal. This evaluation supports the conclusion that implementation of elbow tap Δp measurement as an alternate method of determining RCS total flow rate does not represent an unreviewed safety question as defined in 10 CFR 50.59.

### 2.0 LICENSING BASIS

Title 10 of the Code of Federal Regulations, Part 50, Section 59 (10 CFR 50.59) allows the holder of a license, authorizing operation of a nuclear power facility the capacity to initiate certain changes, tests and experiments not described in the Final Safety Analysis Report (FSAR), and to evaluate these types of changes. Prior Nuclear Regulatory Commission (NRC) approval is not required to return the plant to power as long as the situation does not involve an unreviewed safety question or result in a change to the plant Technical Specifications incorporated in the license. It is, however, the obligation of the licensee to maintain a record of the change or modification to the facility, as a result of any given situation, to the extent that such a change impacts the FSAR.

The implementation of the elbow tap Δp measurement as an alternate method for measuring RCS flow represents a change to the **plant** Technical Specifications and will require an amendment to the operating license.

The current Technical Specification 3.4.1 (**Page X-X Amendment XX**) "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits", describes the RCS total flow rate requirements. In addition, the RCS Low Flow Nominal Trip Setpoint and Allowable Value are defined in Table 3.3.1-1, "Reactor Trip System Instrumentation". The implementation of the elbow tap Δp measurement as an alternate method of measuring RCS total flow rate requires changes to these sections. The revised Technical Specifications are provided in Attachment 1.

Title 10 CFR 50.59 further stipulates that these records shall include a written safety evaluation which provides the basis for the determination that the situation does not involve an unreviewed safety question. It is the purpose of this document to support the requirement for a written safety evaluation.

3.0 EVALUATIONS

Use of elbow tap $\Delta$ps to determine RCS total flow requires that the $\Delta$p measurements for the present cycle be correlated to the precision calorimetric flow measurement performed during the baseline cycle(s). A calculation has been performed to determine the uncertainty in RCS total flow using this method. Included in this calculation is the uncertainty associated with the baseline calorimetric measurement of RCS total flow, as well as uncertainties associated with $\Delta$p transmitters and the process computer. The uncertainty calculation performed for this method of flow measurement is consistent with the methodology recommended by the NRC (NUREG/CR-3659, PNL-4973, 2/85). The only significant difference is the assumption of correlation to previously performed RCS flow calorimetrics. However, this has been accounted for by the addition of certain instrument uncertainties previously considered to be zeroed out by the assumption of normalization to a calorimetric performed each cycle. Based on these calculations, the uncertainty on the RCS flow measurement using the elbow tap method is x.x% flow (without accounting for feedwater venturi fouling). Including the 0.1% feedwater venturi fouling allowance results in a minimum RCS total flow of xxx,xxx gpm which must be measured in the plant at 100% RTP.

The calculations are documented in Tables 1 through 5. Specific calculations performed were for Precision RCS Flow Calorimetrics for the specified baseline cycles, Indicated RCS Flow (computer), and the Reactor Coolant Flow - Low reactor trip. The calculations for Indicated RCS Flow and Reactor Coolant Flow - Low reflect the correlation of the elbow taps to the baseline precision RCS Flow Calorimetrics. As discussed above, additional instrument uncertainties were required to reflect this correlation.

The uncertainty associated with the RCS Low Flow trip increased to x.x% flow span. It was determined that an increase in the Nominal Trip Setpoint to xx% flow, along with the current Safety Analysis Limit (yy% flow) will be sufficient to allow for the increased instrument uncertainties associated with the $\Delta$p to flow correlation.

Note for plants using this evaluation:
A plant specific evaluation from Core Analysis which demonstrates that the increased flow uncertainty does not impact the reactor core safety limits must also be included in the plant specific evaluation, as well as a determination that the Minimum Measured Flow (MMF) assumed in the safety analyses is conservative with respect to the MMF calculated for the elbow tap method.

## 4.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

The use of elbow tap $\Delta p$ measurements as an alternate method of measuring RCS total flow rate has been evaluated using the guidance of NSAC-125. On the basis of the following justification, the use of this alternate method does not involve an unreviewed safety question per the criteria of 10CFR50.59(a)(2).

4.1 Will the probability of an accident previously evaluated in the FSAR 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.

4.2 Will the consequences of an accident previously evaluated in the FSAR 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.

4.3 May the possibility of an accident which is different than any previously evaluated in the FSAR 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.

4.4 Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR 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 uncertainty. There is no increase in the probability of a malfunction of this equipment.

4.5 Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR 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.

4.6 May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR 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.

4.7 Will the margin of safety as defined in the Bases to any Technical Specifications be reduced?

No changes to the Safety Analysis assumptions were required; therefore, the margin of safety as defined in the Bases will remain the same.

## 5.0 CONCLUSION

The evaluation of the elbow tap $\Delta p$ method of measuring RCS total flow rate concludes that it will not result in a potential unreviewed safety question, as defined in 10 CFR 50.59, since it does not increase the probability or occurrence or the consequences of an accident in the FSAR. Nor has any mechanism for an accident or malfunction, which has not been previously evaluated in the FSAR, been identified. Also, the change does not decrease the margin of safety as identified in the basis for any Technical Specification.

## 6.0 REFERENCES

** References will be plant specific, but may include Setpoint Study and/or ITDP/RTDP Instrument Uncertainty WCAPs.

## 7.0 ATTACHMENTS

Attachment 1: Technical Specification Markups

Attachment 2: Significant Hazards Consideration Evaluation

# ATTACHMENT 1

STANDARD TECHNICAL SPECIFICATION MARKUPS

Table 3.3.1-1 (page 3 of 8)
Reactor Trip System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE | TRIP SETPOINT[a] |
|---|---|---|---|---|---|---|
| 8. Pressurizer Pressure | | | | | | |
| a. Low | 1[g] | [4] | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.16 | ≥ [1886] psig | ≥ [1900] psig |
| b. High | 1,2 | [4] | E | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.16 | ≤ [2396] psig | ≤ [2385] psig |
| 9. Pressurizer Water Level – High | 1[g] | 3 | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10 | ≤ [93.8]% | ≤ [92]% |
| 10. Reactor Coolant Flow – Low | | | | | | |
| a. Single Loop | 1[h] | 3 per loop | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.16 | XX<br>≥ ~~[89.2]~~% | YY<br>≥ ~~[90]~~% |
| b. Two Loops | 1[i] | 3 per loop | M | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10<br>SR 3.3.1.16 | XX<br>≥ ~~[89.2]~~% | YY<br>≥ ~~[90]~~% |

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.

(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

(h) Above the P-8 (Power Range Neutron Flux) interlock.

(i) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Pressurizer Water Level—High

The Pressurizer Water Level—High trip Function provides a backup signal for the Pressurizer Pressure—High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level—High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level—High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

Note: The low flow setpoint is given in percent of indicated flow. The indicated flow is normalized based on the measured Δp at 100% RTP.

10. Reactor Coolant Flow—Low

a. Reactor Coolant Flow—Low (Single Loop)

The Reactor Coolant Flow—Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Coolant Flow—Low (Single Loop) (continued)

The LCO requires three Reactor Coolant Flow—Low channels per loop to be OPERABLE in MODE 1 above P-8.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip (Function 10.b) because of the lower power level and the greater margin to the design limit DNBR.

b. Reactor Coolant Flow—Low (Two Loops)

The Reactor Coolant Flow—Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow—Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow—Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

(continued)

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ [2200] psig;

b. RCS average temperature ≤ [581]°F; and

c. RCS total flow rate ≥ [XXX ~~284,000~~] gpm when using the precision heat balance method or [yyy] gpm when using the elbow tap method.

APPLICABILITY: MODE 1.

------------------------------NOTE------------------------------

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

----------------------------------------------------------------

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more RCS DNB parameters not within limits. | A.1 | Restore RCS DNB parameter(s) to within limit. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Be in MODE 2. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.1.1 | Verify pressurizer pressure is ≥ [2200] psig. | 12 hours |
| SR 3.4.1.2 | Verify RCS average temperature is ≤ [581]°F. | 12 hours |
| SR 3.4.1.3 | Verify RCS total flow rate is ≥ [284,000] gpm. xxx | 12 hours |
| SR 3.4.1.4 | --------------------NOTE--------------------<br>Not required to be performed until 24 hours after ≥ [90]% RTP.<br>---------------------------------------------<br><br>Verify by precision heat balance that RCS total flow rate is ≥ [284,000] gpm. xxx | [18] months |

when using the precision heat balance method or [yyy] gpm when using the elbow tap method

or by elbow taps that RCS total flow rate is ≥ [yyy] gpm

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits. e

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will

(continued)

BASES

**APPLICABLE SAFETY ANALYSES (continued)**

result in meeting the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of [2200] psig and the RCS average temperature limit of [581]°F correspond to analytical limits of [2205] psig and [595]°F used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

**LCO**

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

of [xxx] gpm — uncertainty — at the beginning of the current cycle

RCS total flow rate contains a measurement error of [2.0]% based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of [0.1]% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to [2.1]% for no fouling.

Insert A

Any fouling that might bias the flow rate measurement greater than [0.1]% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

action shall be taken before performing subsequent precision heat balance measurements, i.e.,

(continued)

## INSERT A

RCS total flow rate of [yyy] gpm contains a measurement uncertainty of [z.z] % based on cold leg elbow taps correlated to past precision heat balance measurements. Correlation of the flow indication channels with these past heat balance measurements is documented in WCAP-14750. Use of this method removes the requirement for performance of a precision RCS flow calorimetric measurement for that cycle. Potential fouling of the feedwater venturi, which might not have been detected, could bias the results from the past precision heat balance measurements in a nonconservative manner. Therefore, a penalty of [0.1] % for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to [z.z + 0.1] %.

BASES

LCO (continued)

The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

(continued)

BASES

ACTIONS

A.1 (continued)

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

is a qualitative verification of significant flow degradation and

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

or by cold leg elbow taps which have been correlated to past precision heat balances

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every [18] months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of [18] months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

exceeding

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ② [90%] RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of [90%] RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching [90%] RTP.

72 hours

100%

and the elbow tap measurement methods both

REFERENCES

1. FSAR, Section [15].

The intent is that this Surveillance be performed near the beginning of the cycle, as close to 100% RTP as possible.

# ATTACHMENT 2

Significant Hazards Consideration Evaluation

1) NUCLEAR PLANT: PLANT NAME

2) SUBJECT: ELBOW TAP FLOW MEASUREMENT

3) TECHNICAL SPECIFICATIONS CHANGED:

Table 3.3.1-1 Reactor Trip System Instrumentation, Functions 10.a and 10.b Reactor Coolant Flow-Low

Specification 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

4) A written evaluation of the significant hazards consideration, in accordance with the three factor test of 10 CFR 50.92, of a proposed license amendment to implement the subject change has been prepared and is attached. On the basis of the evaluation the checklist below has been completed.

Will operation of the plant in accordance with the proposed amendment:

4.1) Yes___ No_X_ Involve a significant increase in the probability or consequences of an accident previously evaluated?

4.2) Yes___ No_X_ Create the possibility of a new or different kind of accident from any accident previously evaluated?

4.3) Yes___ No_X_ Involve a significant reduction in a margin of safety?

5) Reference Documents:

***Will be plant specific

6) Significant Hazards Consideration Approval:

Prepared By: ____________________ Date: ____________________

Reviewed By: ____________________ Date: ____________________

## 10 CFR 50.92 EVALUATION

Pursuant to 10 CFR 50.92 each application for amendment to an operating license must be reviewed to determine if the proposed change involves a Significant Hazards Consideration. The amendment, as defined below, describing the Technical Specification (T/S) change associated with the change has been reviewed and deemed not to involve Significant Hazards Considerations. The basis for this determination follows.

**Proposed Change:** The current Technical Specification Table 3.3.1-1 (**Page x-x Amendment xx**), "Reactor Trip System Instrumentation", provides the Nominal Trip Setpoint and Allowable Value for the RCS Flow-Low trip. These values will be changed to reflect the increased uncertainty associated with the correlation of the elbow taps to a previous baseline calorimetric. In addition, Technical Specification 3.4.1 (**Page x-x Amendment x**), "RCS Temperature, Pressure, and Flow Departure from Nucleate Boiling (DNB) Limits", will be changed to include an alternate minimum flow requirement for use when RCS total flow is measured by the elbow tap $\Delta p$ method. Changes will also be made to SR 3.4.1.3 and SR 3.4.1.4 to include this alternate minimum flow requirement. Appropriate Technical Specification Bases sections will also be revised to reflect the alternate method for flow measurement and to provide clarification. In addition, the Bases to SR 3.4.1.4 will be revised to allow 72 hours after reaching 100% RTP for the flow measurement to be performed. The revised Technical Specifications and Bases sections are provided in Attachment 1.

**Background:** The 18-month total RCS flow surveillance is typically satisfied by a secondary power calorimetric-based RCS flow measurement. **PLANT NAME** in recent cycles has experienced apparent decreases in flow rates which have been attributed to variations in hot leg streaming effects. These effects directly impact the hot leg temperatures used in the precision calorimetric, resulting in the calculation of low RCS flow rates. **PLANT NAME**, in addition to several other three loop plants, participated in a Westinghouse Owner's Group (WOG) minigroup that extensively evaluated the phenomenon of hot leg streaming. The effect of an apparent RCS flow reduction, as indicated by precision calorimetric flow measurements performed at the beginning of each cycle, was present to some degree in every plant that participated in the program. The apparent flow reduction was even more pronounced in plants which have implemented aggressive Low Leakage Loading Patterns (LLLPs). Evidence that the flow reduction was apparent and not actual was provided by elbow tap measurements. The results of this evaluation, including a detailed description of the hot leg streaming phenomenon, are documented in WCAP-14750 (RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs).

**PLANT NAME** intends to begin using an alternate method of measuring flow using the elbow tap Δp measurements as described in WCAP-14750. For this alternate method, the RCS elbow tap measurements are correlated to precision calorimetric measurements performed during an earlier cycle (or cycles) when the hot leg streaming effects were minimal.

The purpose of this evaluation is to assess the impact of using the elbow tap Δp measurements as an alternate method for performing the 18-month RCS flow surveillance on the licensing basis and demonstrate that it will not adversely affect the subsequent safe operation of the plant. This evaluation supports the conclusion that implementation of the elbow tap Δp measurement as an alternate method of determining RCS total flow rate does not represent a significant hazards consideration as defined in 10 CFR 50.92.

**Evaluation:** Use of the elbow tap Δp method to determine RCS total flow requires that the Δp measurements for the present cycle be correlated to the precision calorimetric flow measurement which was performed during the baseline cycle(s). A calculation has been performed to determine the uncertainty in the RCS total flow using this method. This calculation includes the uncertainty associated with the RCS total flow baseline calorimetric measurement, as well as uncertainties associated with Δp transmitters and the process computer. The uncertainty calculation performed for this method of flow measurement is consistent with the methodology recommended by the NRC (NUREG/CR-3659, PNL-4973, 2/85). The only significant difference is the assumption of correlation to a previously performed RCS flow calorimetric. However, this has been accounted for by addition of instrument uncertainties previously considered to be zeroed out by the assumption of normalization to a calorimetric performed each cycle. Based on these calculations, the uncertainty on the RCS flow measurement using the elbow tap method is x.x% flow (without accounting for feedwater venturi fouling). Including the 0.1% feedwater venturi fouling allowance results in a minimum RCS total flow of xxx,xxx gpm which must be measured at 100% RTP.

The calculations are documented in Tables 1 through 5. The specific calculations performed were for Precision RCS Flow Calorimetrics for the specified baseline cycles, Indicated RCS Flow (computer) if applicable, and the Reactor Coolant Flow - Low reactor trip. The calculations for Indicated RCS Flow and Reactor Coolant Flow - Low reflect correlation of elbow taps to baseline precision RCS Flow Calorimetrics. As discussed above, additional instrument uncertainties were included for this correlation.

The uncertainty associated with the RCS Low Flow trip increased to x.x% flow span. It was determined that an increase in the Nominal Trip Setpoint to xx% flow, along with the current Safety Analysis Limit (yy% flow) will be sufficient to allow for the increased instrument uncertainties associated with the Δp to flow correlation.

The note associated with SR 3.4.1.4 specifies that the flow measurement may not be performed until 24 hours after exceeding 90% RTP. The Basis to this surveillance requirement previously stated that the surveillance be performed within 24 hours after reaching 90% RTP. Since a more accurate flow measurement can be obtained at 100% RTP, it is proposed that this statement in the Bases be revised such that the surveillance is performed within 72 hours after reaching 100% RTP. The 72 hour period will allow sufficient time to perform an accurate measurement, and is consistent with typical allowed outage times for equipment to be returned to service.

**Note for plants using this evaluation:**
**A plant specific evaluation from Core Analysis which demonstrates that the increased flow uncertainty does not impact the reactor core safety limits must also be included in the plant specific 50.92, as well as a determination that the Minimum Measured Flow (MMF) assumed in the safety analyses is conservative with respect to the MMF calculated for the elbow tap method.**

Based on these evaluations, the proposed change would not invalidate the conclusions presented in the FSAR.

1. Does the proposed modification involve a significant increase in the probability or consequences of an accident previously evaluated?

An evaluation determined that the probability of an accident will not increase. 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.

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.

2. Does the proposed modification create the possibility of a new or different kind of accident from any accident previously evaluated?

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.

3. Does the proposed modification involve a significant reduction in a margin of safety.

No changes to the Safety Analysis assumptions were required; therefore, the margin of safety will remain the same.

**Conclusion:** Based on the preceding information, it has been determined that this proposed change to allow an alternate RCS total flow measurement based on elbow tap $\Delta p$ measurements does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92(c).

APPENDIX C

SAMPLE 50.59 AND

SUGGESTED MODIFICATIONS TO

FARLEY TECHNICAL SPECIFICATION BASES

1) NUCLEAR PLANT(S): Farley Nuclear Plant Units 1 and 2

2) SUBJECT: ELBOW TAP FLOW MEASUREMENT

3) The written safety evaluation of the revised procedure, design change or modification required by 10 CFR 50.59 (b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A 10CFR50.59(a)(1)

3.1 Yes___ No_X_ A change to the plant as described in the FSAR?

3.2 Yes___ No_X_ A change to procedures as described in the FSAR?

3.3 Yes___ No_X_ A test or experiment not described in the FSAR?

3.4 Yes___ No_X_ A change to the plant Technical Specifications? (See Note on Page 2.)

4) CHECK LIST - PART B 10CFR50.59(a)(2) (Justification for Part B answers is included on Page 2.)

4.1 Yes___ No_X_ Will the probability of an accident previously evaluated in the FSAR be increased?

4.2 Yes___ No_X_ Will the consequences of an accident previously evaluated in the FSAR be increased?

4.3 Yes___ No_X_ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

4.4 Yes___ No_X_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

4.5 Yes___ No_X_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

4.6 Yes___ No_X_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

4.7 Yes___ No_X_ Will the margin of safety as defined in the Bases to any Technical Specification be reduced?

NOTES:

If the answer to any of the above questions is unknown, indicate under Section 5.0 REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written Safety Evaluation, the change review would require an application for license amendment as required by 10CFR50.59(c) and submitted to the NRC pursuant to 10CFR50.90.

5) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

FOR FSAR UPDATE

Section: N/A Pages: N/A Tables: N/A Figures: N/A

Reason for/Description of Change:

6) SAFETY EVALUATION APPROVAL LADDER:

Prepared By: ____________________ Date: __________

Reviewed By: ____________________ Date: __________

## ELBOW TAP FLOW MEASUREMENT SAFETY EVALUATION

### 1.0 INTRODUCTION AND BACKGROUND

The purpose of this evaluation is to assess the impact of using the elbow tap Δp measurements as an alternate method for performing the 18-month RCS flow surveillance on the licensing basis and demonstrate that it will not adversely affect the subsequent safe operation of the plant. The 18-month RCS flow surveillance is typically satisfied by a secondary power calorimetric-based RCS flow measurement. Many plants in recent cycles have experienced apparent decreases in flow rates which have been attributed to variations in hot leg streaming. These effects directly impact the hot leg temperatures used in the precision calorimetric, resulting in the calculation of low RCS flow rates. In using the elbow tap Δp method, the RCS loop elbow tap measurements are correlated to precision calorimetric measurements performed during an earlier cycle when the hot leg streaming effects were minimal. This evaluation supports the conclusion that implementation of elbow tap Δp measurements as an alternate method of determining RCS total flow does not represent an unreviewed safety question as defined in 10 CFR 50.59.

### 2.0 LICENSING BASIS

Title 10 of the Code of Federal Regulations, Part 50, Section 59 (10 CFR 50.59) allows the holder of a license, authorizing operation of a nuclear power facility, the capacity to evaluate and initiate changes in procedures described in the Final Safety Analysis Report (FSAR). Prior Nuclear Regulatory Commission (NRC) approval is not required to implement a change provided that the proposed change does not involve an unreviewed safety question or result in a change to the plant Technical Specifications. However, it is the obligation of the licensee to maintain a record of the changes, to the extent that such changes impact the Final Safety Analysis Report (FSAR). The code further stipulates that these records shall include a written safety evaluation that provides the basis for the determination that the change does not involve an unreviewed safety question. It is the purpose of this document to support the requirement for a written safety evaluation to justify a change to the Technical Specifications Bases.

### 3.0 EVALUATION

Farley Nuclear Plant (FNP) in recent cycles has experienced apparent decreases in flow rates which have been attributed to variations in hot leg streaming. These effects directly impact the hot leg temperatures used in the precision calorimetric, resulting in the calculation of low RCS flow rates. FNP, in addition to several other three loop plants, participated in a Westinghouse Owner's Group (WOG) minigroup that extensively evaluated the phenomenon of hot leg streaming. The effect of an

apparent RCS flow reduction, indicated by precision calorimetric flow measurements performed at the beginning of each cycle, was present to some degree in every plant that participated in the program. The apparent flow reduction was even more pronounced in plants which have implemented aggressive Low Leakage Loading Patterns (LLLPs). Evidence that the flow reduction was apparent and not actual was provided by elbow tap measurements. The results of this evaluation, including a detailed description of the hot leg streaming phenomenon and a description of an alternate method of flow measurement using elbow taps, are documented in WCAP-14750 (RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs).

Calculations have been performed to determine the impact of using elbow tap $\Delta p$ measurements as an alternate method of determining RCS total flow at Farley Units 1 and 2. This alternate method correlates the elbow tap $\Delta p$ measurements for the present cycle to a precision calorimetric flow measurement performed during a baseline cycle or cycles. For Farley Unit 1 baseline flow was determined using data from cycles 1, 4 and 5, while the Unit 2 baseline cycles were 2, 3, and 5.

The uncertainty calculations are documented in Tables A-1 through A-5 of WCAP-14750. The calculations for both Farley units were for Precision RCS Flow Calorimetrics for the specified baseline cycles, Indicated RCS Flow (computer), and the Reactor Coolant Flow - Low reactor trip. The measured flow using the elbow tap correlation to the baseline precision calorimetric is reflected in the uncertainty calculations for Indicated RCS Flow (computer output normalized to measured flow) and Reactor Coolant Flow - Low reactor trip. Additional instrument uncertainties were required to reflect the elbow tap correlation.

Based on these calculations, the RCS flow measurement uncertainty is 2.3% flow, equivalent to the current NRC licensed value (2.4 % flow) after including the feedwater venturi fouling allowance of 0.1% flow. The instrument uncertainty calculations are consistent with the methodology recommended by the NRC (NUREG/CR-3659, PNL-4973, 2/85). The only significant difference is the assumption of correlation to previously performed RCS Flow calorimetrics. However, this has been accounted for by addition of instrument uncertainties previously considered to be zeroed out by normalization to a calorimetric performed each cycle. Based on these calculations, the minimum RCS Flow that must be measured in the plant at 100 % RTP is maintained at the value currently listed in the Technical Specifications.

It was determined that the difference between the current Safety Analysis Limit (85 % flow) and Nominal Trip Setpoint for Reactor Coolant Flow - Low (90 % flow) is sufficient to allow for the increased instrument uncertainties due to this correlation. Therefore, no setpoint change is required and no new safety analyses must be performed.

## 4.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

The use of elbow tap $\Delta p$ measurements as an alternate method of measuring RCS total flow rate has been evaluated using the guidance of NSAC-125. On the basis of the following justification, the use of this alternate method does not involve an unreviewed safety question per the criteria of 10CFR50.59(a)(2).

### 4.1 Will the probability of an accident previously evaluated in the FSAR be increased?

No change to the probability of an accident can occur since this change affects how RCS flow is calculated; actual flow is not changed. 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.

### 4.2 Will the consequences of an accident previously evaluated in the FSAR 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.

### 4.3 May the possibility of an accident which is different than any previously evaluated in the FSAR 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.

### 4.4 Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR 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 uncertainty. There is no increase in the probability of a malfunction of this equipment.

### 4.5 Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR 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.

4.6 May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR 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.

4.7 Will the margin of safety as defined in the Bases to any Technical Specifications be reduced?

No changes to the Safety Analysis assumptions were required, therefore, the margin of safety as defined in the Bases will remain the same.

## 5.0 CONCLUSION

The evaluation of the elbow tap $\Delta p$ method of measuring RCS total flow rate concludes that it will not result in a potential unreviewed safety question, as defined in 10 CFR 50.59, since it does not increase the probability or occurrence or the consequences of an accident in the FSAR. Nor has any mechanism for an accident or malfunction, which has not been previously evaluated in the FSAR, been identified. Also, the change does not decrease the margin of safety as identified in the basis for any Technical Specification. Therefore, the change to the FNP Technical Specifications Bases is acceptable.

## 6.0 REFERENCES


Ref 1: WCAP-14750, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs", September, 1996.

Ref 2: WCAP-13751, "Westinghouse Setpoint Methodology for Protection Systems, Southern Nuclear Operating Company, Farley Nuclear Plant Units 1 and 2", June, 1993.

Ref 3: WCAP-12771, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Alabama Power Farley Nuclear Plant Units 1 and 2 (for RTD Bypass Loop Elimination)", May, 1991.


## 7.0 ATTACHMENTS

Attachment 1: Technical Specification Bases Markups

# ATTACHMENT 1

FARLEY TECHNICAL SPECIFICATION BASES MARKUPS

## BASES

The radial peaking factor $F_{xy}(Z)$, is measured periodically to provide additional assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The $F_{xy}$ limit for RATED THERMAL POWER ($F_{xy}^{RTP}$) as provided in the Radial Peaking Factor limit report per Specification 6.9.1.11 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

### 3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q$ is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8.

### 3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated $T_{avg}$ value of 580.7°F is based on the average of two control board readings and an indication uncertainty of 2.5°F. The indicated pressure value of 2205 psig is based on the average of two control board readings and an indication uncertainty of 20 psi. The indicated total RCS flow rate is based on one elbow tap measurement from each loop and an uncertainty of 2.4% flow (0.1% flow is included for feedwater venturi fouling).

two a

The 12 hour surveillance of Tavg and pressurizer pressure through the control board readings are sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

Insert A

e

The 18 month surveillance of the total RCS flow rate is a precision measurement that verifies the RCS flow requirement at the beginning of each fuel cycle and ensures correlation of the flow indication channels with the measured loop flows. The monthly surveillance of the total RCS flow rate is a reverification of the RCS flow requirement using loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of the fuel cycle. The 12 hour RCS flow surveillance is a qualitative verification of significant flow degradation using the control board indicators and the loop elbow tap measurements that are correlated to the precision RCS flow measurement at the beginning of each fuel cycle.

Insert A

The 18 month surveillance of the total RCS flow rate may be performed by one of two alternate methods. One method is a precision calorimetric performed at the beginning of each fuel cycle. The other method is based on the Δp measurements from the cold leg elbow taps, which are correlated to past precision heat balance measurements. Correlation of the flow indication channels with selected precision loop flow calorimetrics for this method is documented in WCAP-14750. Use of the elbow tap Δp measurement method removes the requirement for performance of a precision RCS flow calorimetric measurement for that cycle. The monthly surveillance of the total RCS flow rate is a reverification of the RCS flow requirement using process computer indications of loop elbow tap measurements that are correlated either to the precision RCS flow measurement or the elbow tap measurement at the beginning of the fuel cycle. The 12 hour RCS flow surveillance is a qualitative verification of significant flow degradation using the control board indicators fed by elbow tap measurements.

# APPENDIX D

# RCS ELBOW TAP FLOW MEASUREMENT PROCEDURE

## Introduction

In this procedure, RCS total flow is verified at the beginning of a new fuel cycle by correcting the baseline calorimetric flow for the actual change in flow, determined by comparing the new cycle elbow tap ΔPs with the baseline ΔPs. This procedure can be used in place of the RCS calorimetric flow measurement procedure to demonstrate that the measured flow meets the requirements of the Technical Specification DNB surveillance at the beginning of the new fuel cycle. The procedure is described below.

### LIST OF ACRONYMS USED IN PROCEDURE

a,c

## Procedure

a,c

## Background Information

The following comments provide additional background information on the detailed procedure presented in the attachment.

a. Technical Specification flow (TSF) includes an allowance for baseline calorimetric flow (BCF) having a non-conservative hot leg streaming uncertainty, based on streaming gradients that were present in early cycles. Recent fuel cycles have larger streaming gradients, but the resulting streaming uncertainty results in a conservative flow measurement. The BCF streaming uncertainty never exceeds the non-conservative allowance in the TSF.

a,c