TOLEDO EDISON

RICHARD P. CROUSE
Vice President
Nuclear
(419) 259-5221

Docket No. 50-346

License No. NPF-3

Serial No. 923

March 21, 1983

Mr. John F. Stolz, Director
Nuclear Reactor Regulation
Operating Reactor Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

Toledo Edison acknowledges receipt of your January 26, 1983 letter (Log No. 1197) requesting additional information for the proposed change to Appendix A, Technical Specification 3.2.5 - Flow/Thermal Power Setpoint Change - for the Davis-Besse Nuclear Power Station, Unit 1. Please find enclosed the requested information.

Very truly yours,

RCP:JAE

Enclosure

dh e/4

A001

Serial 923

Question 1. Your "Safety Evaluation" in support of the proposed change on Technical Specification 3.2.5 indicates the reactor coolant measurement uncertainty of 2.5%. Provide a detailed description on how the RC flow measurement uncertainty is obtained, including a detailed breakdown of measurement components and uncertainty associated with each component.

Response:

The attached report entitled "Determination of Total RC Flowrate and its Accuracy for Davis-Besse 1" provides a detailed description of how the RC flow measurement uncertainty is obtained. A detailed breakdown of measurement components and the uncertainty associated with each component is included.

This report was transmitted via the attached letter from Lowell E. Roe to John F. Stolz, dated May 26, 1978. The flow measurement accuracy determined in the report is 2.2%. The value of 2.5% which is indicated in the "Safety Evaluation" mentioned above is conservative with respect to the calculated value.

Question 2. The same Safety Evaluation states that B&W has performed calculations to determine the DNBR margin gain for the proposed RC flow and power tradeoff. Is the B&W analysis done specifically for Davis Besse? Provide the B&W analysis report.

Question 3. Figure 1 of your submittal gives a relationship between MDNBR, calculated with BAW-2 correlations, and the thermal power reduction factor N. Also, the proposed Technical Specification change uses N=2(%). Are the figure and the associated analysis based on current Davis Besse fuel design and loading? Do the analysis and the N=2 bound all fuel loading and fuel design for the future cycles?

Response:

INTRODUCTION

Section 3.2.5 of the Davis Besse Technical Specifications (Reference 1) sets acceptance criteria on the DNB related parameters of RCS pressure, hot leg flow temperature, and RCS flow rate. If any of these parameters exceed the prescribed limits, the parameter must be restored to within its limits within two hours or the reactor must be reduced to 5% of rated thermal power within the next four hours (rated thermal power = 2772 MWt).

The minimum reactor flow rate specified in Reference 1 is 396880 GPM with 4 RC pumps operating (includes a 2.5% flow rate uncertainty). This minimum flow criterion is based upon the 110% of design flow assumed for DNB analyses (design flow = 88000 GPM/pump). The minimum RC flow rates measured at DB-1 are typically in the range of 111 to 112% of design flow (including the 2.5% flow uncertainty reduction). Toledo Edison desires to modify the action requiring a reduction in reactor thermal power to 5% of rated power in the event measured flow is outside its allowable value. This modification will require a reduction of the maximum allowable thermal power by 2% for every 1% the RC flow rate is below the Tech. Spec. 3.2.5 limit (which would be unchanged). This Tech. Spec. modification is recommended only to cover small changes in measured steady state flow, on the order of ~1 to 2 percent, and not large changes in measured flow rate which obviously would indicate more serious concerns. This report describes the criteria, methods, results, and conclusions of the analyses supporting the proposed Tech. Spec. change.

CRITERIA

The purpose of Tech. Spec. 3.2.5 is to ensure that each of the DNB-related parameters is maintained within the normal steady state envelope that was assumed as an initial condition in transient and accident analysis. Ultimately then the purpose of Tech. Spec. 3.2.5 is to ensure that the DNBR safety limit is not violated during steady state or moderate frequency transient conditions. Since any changes to the transient initial (time = 0) DNBR (i.e., steady state DNBR) will tend to be carried through to the transient minimum DNBR, then the appropriate basis for judging the proposed Tech. Spec. change is the effect of the proposed combinations of reactor power and RC flow rate on the steady state minimum DNBR. The datums for these comparisons should be the minimum DNBR's at DB-1 design over power conditions for 4 RC pump and 3 RC pump operation. The minimum DNBR's to be compared to the design over power cases should be calculated at sets of conditions defined by the following formulas:

1. For 4 RC Pump Operation:

   RC Flow Rate = (110 - N)% of design 4 pump flow
   Reactor Power Level = (112 - 2N)% of rated power

2. For 3 RC Pump Operation:

   RC Flow Rate = (110 - N)% of design 3 pump flow
   Reactor Power Level = (90 - 2N)% of rated power

   where N = arbitrary integer (1, 2, 3, etc.)

If the minimum DNBR's increase with increasing values of N, then the trade-off of power for flow (2 for 1) will have been shown to be conservative in terms of minimum DNBR, thus demonstrating that the proposed Tech. Spec. change is conservative.

METHODS

The methods employed in the analyses supporting the proposed Tech. Spec. change are as described in Chapter 4.4 of the DB-1 FSAR (Reference 2). Briefly, there were two steady state DNBR analyses performed specifically for Davis Besse: one with 4 RC pump operation and one with 3 RC pump operation. These analyses conservatively bound the current Davis Besse fuel design and all expected loadings. No fuel design changes that would impact the results of these analyses are planned for Davis Besse.

Maximum design conditions were assumed for both DNBR analyses. Maximum design conditions assume that the most conservative nuclear, thermal, and mechanical conditions exist simultaneously in a particular subchannel. The maximum design conditions at DB-1 are represented by the following assumptions:

1. A maximum fuel pin radial - local power factor ($F\Delta h$) of 1.714.
2. A symmetric cosine axial flux shape with a max./avg. peak (F_z) of 1.5.
3. The limiting fuel assembly is assumed to receive only 95% of the average fuel assembly flow.
4. The limiting fuel assembly is assumed to have a reduced peripheral flow area because of adjacent fuel assembly proximity.
5. A maximum core pressure error of -65 psi is assessed against the nominal RCS pressure.
6. A maximum RC inlet temperature error of +2^{O}F is assessed against the nominal inlet temperature based on heat balance.
7. Three engineering hot channel factors are applied to the subchannel types with maximum $F\Delta h$ values to account for as-built variations of key parameters:
   A. The subchannel flow area is reduced by a factor (FA) of .97 or .98 depending on subchannel type.
   B. The fuel pin local surface heat flux is increased by a factor (FQ") of 1.014.
   C. The fuel pin heat output (i.e., subchannel enthalpy rise) is increased by a factor (FQ) of 1.011.

8. A maximum core bypass flow fraction of 10.7% which includes the absence of all orifice rods.

9. A minimum fuel stack height resulting from fuel densification.

The computer codes used in the two DNBR analyses are the CHATA code (Reference 3) for core flow distribution on an assembly by assembly basis and the TEMP code (Reference 4) for detailed DNBR analysis of the limiting fuel assembly. The CHATA code calculates the bundle by bundle flow distribution for a core given the power level, inlet temperature and pressure, total core flow rate, radial power distribution, and hydraulic characteristics of the fuel assemblies, assuming a uniform pressure drop across the core. The flow rate of the limiting fuel assembly calculated with CHATA is input to the TEMP model. TEMP will distribute this input flow rate among the fuel assembly subchannels according to the local power distribution and the hydraulic characteristics of the different subchannels assuming a constant pressure drop across the fuel assembly. TEMP allows the transfer of energy (enthalpy) between subchannels by turbulent mixing but allows no mass interchange. CHATA and TEMP are the steady state thermal hydraulic codes that were used for DB-1 licensing.

RESULTS

The analysis with 4 RC pump operation assumed 112% of rated power and 110% of design flow as base case conditions. These are the design over power conditions for DB-1. The minimum DNBR of this base case is compared with the minimum DNBR's of the other 4 RC pump cases in Figure 1. The "N" parameter on the abscissa of Figure 1 relates to the power and flow conditions of each case as defined by equation in the "Criteria" section of this report and as listed below for illustration.

N	% Rated Power	% Design Flow*
0	112	110
2	108	108
4	104	106
6	100	104

*100% design flow is the RC pump design flow rate of 88000 GPM/pump.

The range of power and flow investigated was deemed adequate because of the clearly evidenced trend in Figure 1, that minimum DNBR increases with increasing "N" value. This trend indicates that with the proposed Tech. Spec. change is place, an RC flow rate below the Tech. Spec. 3.2.5 limit would restrict allowable reactor power to a level where greater thermal margin would exist than assumed in the initial conditions for transient analysis. This strongly implies, that greater thermal margin would exist during a transient initiated from N $\geq$ 0 conditions than from N = 0 conditions. Therefore the consequences of transients analyzed in Chapter 15 of the DB-1 FSAR will remain bounding with the proposed Tech. Spec. change enacted, so long as measured flow is 100 percent or greater of the pump design flow rate.

The analyses with 3 RC pump operation assumed 90% of rated power and 110% of design 3 pump flow as base case conditions. The 90% power level was arbitrarily chosen as a bounding 3 pump allowable power. The minimum DNBR's of this base case and other 3 RC pump cases are compared in Figure 2. The "N" parameter on the abscissa again relates power and flow as defined by equation in the "Criteria" section of this report and as listed below for illustration.

N	% Rated Power	% Design 3 Pump Flow
0	90	110
2	86	108
4	82	106
6	78	104

As with 4 RC pump operation, the minimum DNBR increases with increasing "N" value with 3 RC pump operation. For the same reasons as with 4 RC pump operation, therefore, the proposed Tech. Spec. change will not adversely affect the consequences of transients analyzed with 3 RC pumps operating.

CONCLUSIONS

The conclusion of the two analyses supporting the proposed Tech. Spec. 3.2.5 change, and of this report, is that the proposed Tech. Spec. change is conservative in terms of minimum DNBR. The improvement in DNBR margin over that associated with the assumed transient initial conditions leads to the further conclusion that DB-1 FSAR Chapter 15 analytical results would not be adversely affected by the proposed Tech. Spec. change. Thus the intent of Tech. Spec. 3.2.5, to ensure that the consequences of existing transient analyses remain bounding by preserving the transient initial conditions, is clearly satisfied.

REFERENCES

1) 05-0011-16, "Davis Besse Nuclear Power Station Unit 1 Technical Specifications", October 2, 1980.

2) Davis Besse Unit 1 Final Safety Analysis Report, Docket #50-346.

3) CHATA - Core Hydraulic and Thermal Analysis, J. M. Alcorn and R. H. Wilson, BAW-10110, Babcock & Wilcox, January 1976.

4) TEMP - Thermal Enthalpy Mixing Program, BAW-10021, Babcock & Wilcox, April 1970.

FIGURE 1

Minumum DNBR as a Function of Flow and Power Parameter "N" with 4 RC Pump Operation

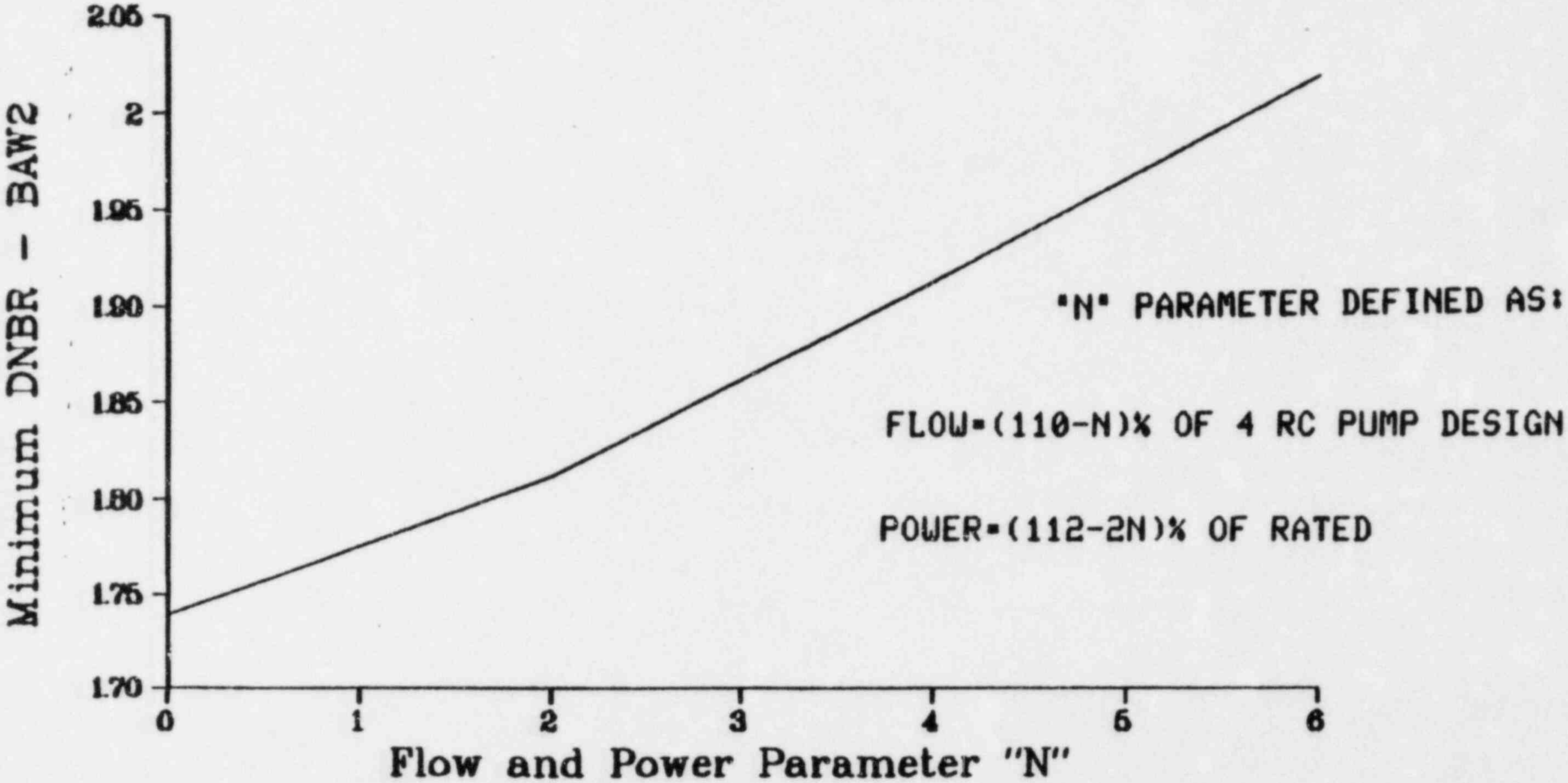
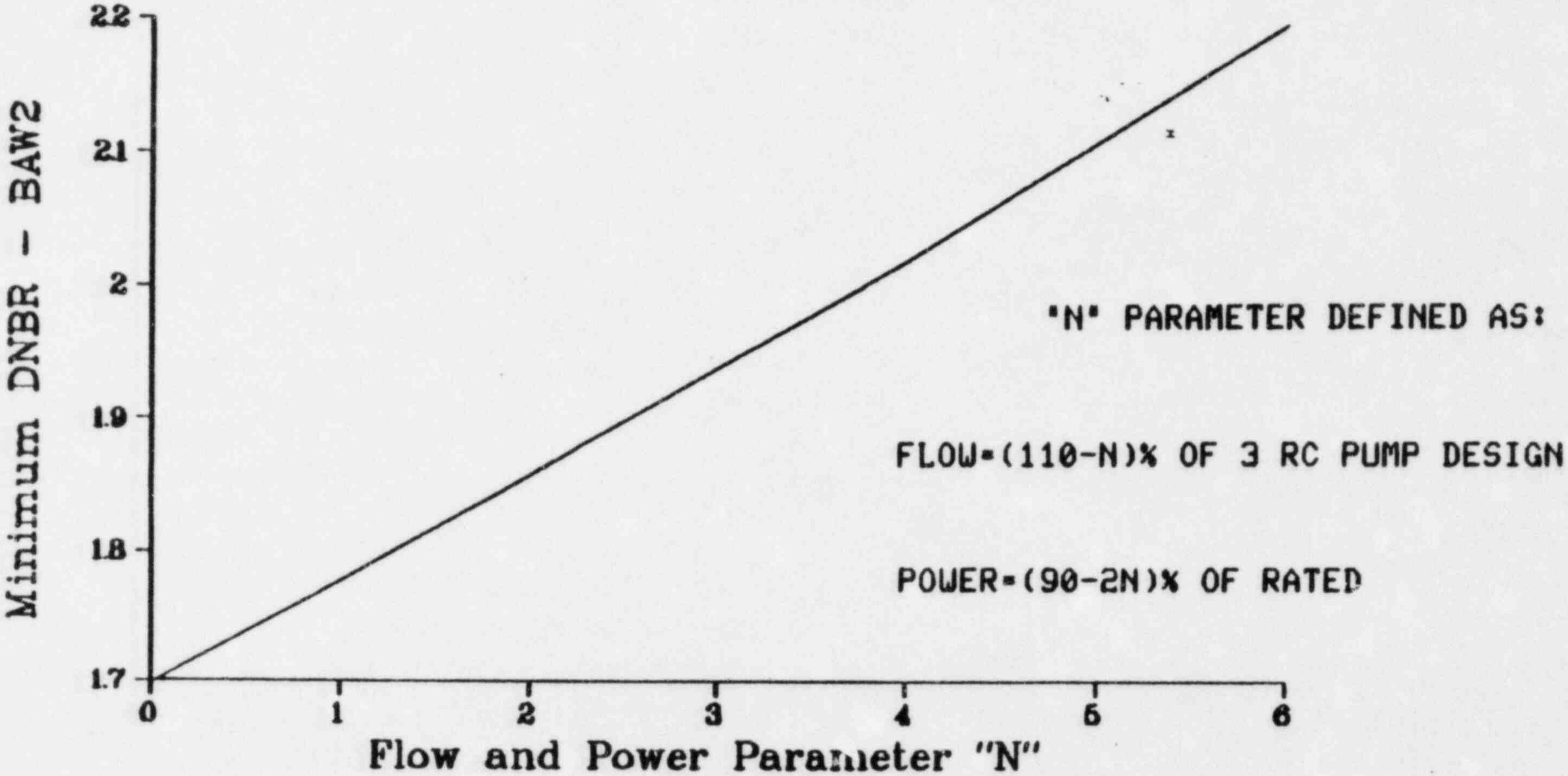


FIGURE 2

Minumum DNBR as a Function of Flow and Power Parameter "N" with 3 RC Pump Operation



4. When operating with the reduced PC flow and power in accordance with the proposed Technical Specification, are your current design safety analyses with respect to all anticipated operational occurrences and accidents still valid?

Response:

Background:

If during power operation, measured RC flow decreases, power will be reduced according to the following scheme set in the Technical Specifications:

For every 1% drop in measured RC flow, power will be reduced by 2%.

This rule will be applied for both three and four pump initial operation.

The FSAR analysis was performed assuming an initial condition of 102% power and 100 % design flow. With the above scheme, the plant may run at lower power and flow than what was assumed in the FSAR. The FSAR analysis has been reviewed to assess the impact of the different initial conditions. The results of this assessment are summarized in Table 1 and the following paragraphs.

Results of the Assessment:

An important assumption in this review is that flow will not drop below 100% of design flow. At the present time Davis-Besse 1 is operating at 110% of design flow. Thus, measured flow is calibrated to 110% of design flow. This review will apply only to measured flows $\geq$ 91%. Since it is assumed that RC flow rate will be $\geq$ design flow, only reduced power levels need be addressed in this review.

Each FSAR transient was examined to determine the impact of a lower initial power level. The results are summarized in Table 1. Some general conclusions can be drawn from this table:

a. For overheating transients, a lower initial power level means less heat added to the RCS and thus lower peak RCS pressure.

b. For overcooling transients, a lower initial power level may give a delay in reactor trip. However, for most transients sensitivity studies have been performed that cover the variation in power level.

Conclusion:

Based upon a review of the FSAR analysis, the proposed Tech Spec is within the current design safety analyses with respect to all anticipated operational occurrences and accidents.

Table 1

Transient	Criteria	Assessment of Impact	Conclusion
15.2.1 Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition	Thermal Power <112% RCS Pressure < 110% of design.	Power Operation limits have no impact on startup conditions.	FSAR analysis still applicable.
15.2.2 Uncontrolled Control Rod Assembly Group Withdrawal at Power	Thermal Power <112% RCS Pressure < 110% of design.	For flow ≥ 100% of design and power ≤ 102%, the time to high flux trip will be increased. This is addressed in the sensitivity studies on trip delay time and rod withdrawal rate.	FSAR analysis still applicable.
15.2.3 Control Rod Assembly Misalignment	Thermal Power <112% RCS Pressure < 110% of design.	Thermal power does not exceed initial conditions. This is a depressurization event so peak RCS pressure is not a concern.	FSAR analysis still applicable.
15.2.4 Makeup and purification System Malfunction	Thermal power <112% RCS pressure < 110% of design. 1% ΔK/K Shutdown margin.	For flow ≥ 100% of design and power ≤ 102%, the time to trip will be delayed. This is addressed in the sensitivity study on dilution rates. Power Operation has little impact on post trip shutdown margin.	FSAR analysis still applicable.
15.2.5 Loss of Forced Reactor Coolant Flow.	DNBR >1.3	From FSAR Figure 15.2.5-3 it is seen that MDNBR increases with decreasing power at which coastdown begins.	FSAR analysis still applicable.

Table 1 cont.

Transient	Criteria	Assessment of Impact	Conclusion
15.2.6 Startup of an Inactive Reactor Coolant Loop.	Thermal Power <112% RCS pressure < 110% of design.	Case analyzed in the FSAR is for two pump operation, presently not allowed by Tech Spec. Lower initial power level will increase margin to thermal power limit	FSAR analysis still bounding
15.2.7 Loss of External Load and/or Turbine Trip.	No fuel damage. RCS pressure < 110% of design.	Plant runs back to 15% power. Lower initial power will result in a faster runback and less added heat. Therefore, peak RCS pressure should be lower.	FSAR analysis still bounding
15.2.8 Loss of Normal Feedwater.	No fuel damage. RCS pressure < 110% of design.	Peak RCS pressure can be controlled by the pressurizer safety valves to less than 110% for all allowable power levels.	FSAR analysis is representative.
15.2.9 Loss of all AC Power to the Station Auxiliaries.	No fuel damage. RCS pressure < 110% of design.	Lower initial power level will result in less added heat to the RCS and lower peak RCS pressure.	FSAR analysis still bounding.
15.2.10 Excessive Heat Removal due to Feedwater System Malfunction.	No fuel damage. RCS pressure < 110% of design.	For reduction in Feedwater temperature, MDNBR occurs post trip. Initial power level should have little impact on MDNBR. For increase in Feedwater Flow, the analysis was performed at startup.	FSAR analysis still applicable.
15.2.11 Excessive Load Increase		No analysis.	N/A

Table 1 cont.

Transient	Criteria	Assessment of Impact	Conclusion
15.2.12 Anticipated Variations in the Reactivity of the Reactor		N/A	N/A
15.2.13 Failure of Regulating Instrumentation		No analysis	N/A
15.3.1 Loss of Reactor Coolant from Small Ruptured Pipes or from cracks in Large Pipes which Actuates Emergency Core Cooling.		No analysis	N/A
15.3.2 Unadvertant Loading of a Fuel Assembly into an improper position.		N/A	N/A
15.4.1 Waste Gas Decay Tank Rupture		N/A	N/A
15.4.2 Steam Generator Tube Rupture.	Doses < 10 CFR 100 No additional loss of reactor coolant boundary integrity.	Doses are independent of initial power level.	FSAR analysis still applicable.
15.4.3 Control Rod Ejection Accident.	No additional loss of reactor coolant boundary integrity.	Sensitivity studies are performed at Zero power and full power. This bounds intermediate power levels. The concern in this transient is local peaking effects which are independant of initial power level.	FSAR analysis still applicable.

Table 1 cont.

Transient	Criteria	Assessment of Impact	Conclusion
15.4.4 Steam Line Break	Core shall remain intact. No SGTR induced by the SLB. Doses < 10 CFR 100.	The case analyzed in the FSAR is a double ended break of a 36" steam line. For this case, the reactor trips almost immediately on low RCS pressure. Thus, a lower initial power level will have no impact.	FSAR analysis still applicable.
15.4.5 Break in instrument lines or lines from primary system that penetrate containment.	Doses < 10 CFR 100	A lower initial power level should have little impact on isolation time.	FSAR analysis still applicable.
15.4.6 LOCA	Doses < 10 CFR 100	Lower initial power level will give less limiting clad temperature.	FSAR analysis bounding.
15.4.7 Fuel Handling Accident		N/A	N/A

TOLEDO EDISON

LOWELL E. ROE
Vice President
Facilities Development
(419) 259-5242

Docket No. 50-346

Operating License No. NPF-3

May 26, 1978

Serial No. 436

Director of Nuclear Reactor Regulations
Attention: Mr. John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stolz:

As a result of our review of the uncertainties in the determination of the reactor coolant flow rate at the Davis-Besse Nuclear Power Station Unit No. 1, we have found that the value for the uncertainty given in our letter to you dated April 24, 1978 (Serial No. 428) is incorrect. Attachment 1 to this letter shows that the actual uncertainty value is 2.2%. Attachment 2 is a "Statement Addressing Foreign Material and Deposits in Secondary System."

The uncertainties on the reactor coolant flow rate are incorporated in the technical specification changes given in the attachment to our letter to you dated May 26, 1978 (Serial No. 439).

The reactor coolant flow rate uncertainty also impacts the treatment of the rod bow effect as addressed in our letter to Mr. Roger S. Boyd dated April 10, 1978 (Serial No. 426).

Yours very truly,

Lowell E. Roe

Attachment

jh c/11

8001230653

32-9174 00

Attachment 1 to Toledo Edison Company letter

Dated May 26, 1978; Serial No; 436

DETERMINATION OF TOTAL RC FLOWRATE AND ITS ACCURACY FOR DAVIS-BESSE 1

BY ROBERT W. WINKS
PRINCIPAL ENGINEER
BABCOCK & WILCOX COMPANY
LYNCHBURG, VIRGINIA

MAY 25, 1978

~~8001230657~~ 7

32-9174 00

DETERMINATION OF TOTAL RC FLOWRATE AND ITS ACCURACY FOR DAVIS-BESSE 1

TABLE OF CONTENTS — PAGE NUMBER

TABLE OF CONTENTS	PAGE NUMBER
Introduction	1
Summary	1
Sensor Accuracy	2
Heat Balance Data	5
Calculated Total RC Flowrate	11
Error Analysis	15

DETERMINATION OF TOTAL RC FLOWRATE AND ITS ACCURACY FOR DAVIS-BESSE 1 AT 100% POWER LEVEL

INTRODUCTION

In a B&W nuclear power plant, Gentile flowmeters are used to measure Loop 1 and 2 reactor coolant flowrates. These primary loop flowmeters are not calibrated prior to installation. Loop 1 and 2 feedwater flowrates are measured with calibrated flow meters and B&W utilizes a plant heat balance to set the calibration of the primary loop flowmeters.

An error analysis on the equations used to determine the total reactor core flowrate (Loop 1 plus Loop 2) has revealed that the errors in reactor coolant temperatures and feedwater flowmeter differential pressure are the most significant terms in calculating accurate values of total reactor core flowrate

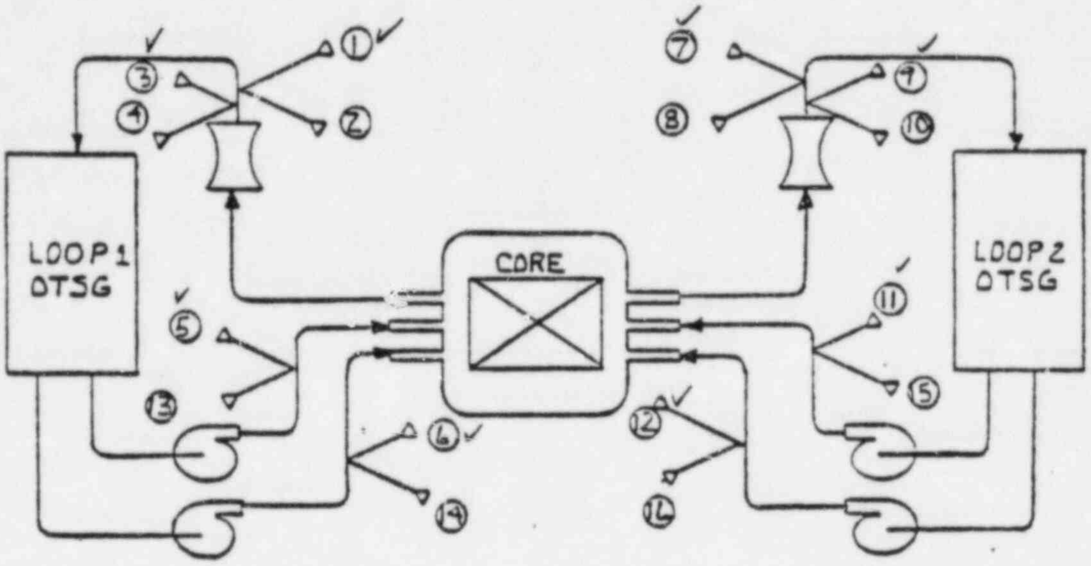
SUMMARY

The calculated RC flowrate for Davis-Besse 1 at 100% power is 113.2% times the design flowrate of 352,000 gpm. The accuracy is ±2.2%. This was determined with RC temperature instrument string errors equal to ±0.79F and feedwater flowmeter ΔP errors equal to ±1.25% and steam temperature errors equal to ±4.2F.

Reactor coolant system temperatures are measured with ± 1/4% accurate pre-calibrated RTD's over a range of 520 to 620F. Similarly, the Bailey Meter Company differential pressure transmitters are calibrated to ± 1/2% at time of installation. The two feedwater flowmeters are calibrated to ± 1/2% prior to installation.

For normal everyday conditions in the instrumentation area of the plant, B&W has determined the accuracy of all input measurements used in this error analysis. (Refer to page 4 .)

ACCURACY OF MAJOR INSTRUMENTATION USED FOR PLANT HEAT BALANCE CALCULATIONS

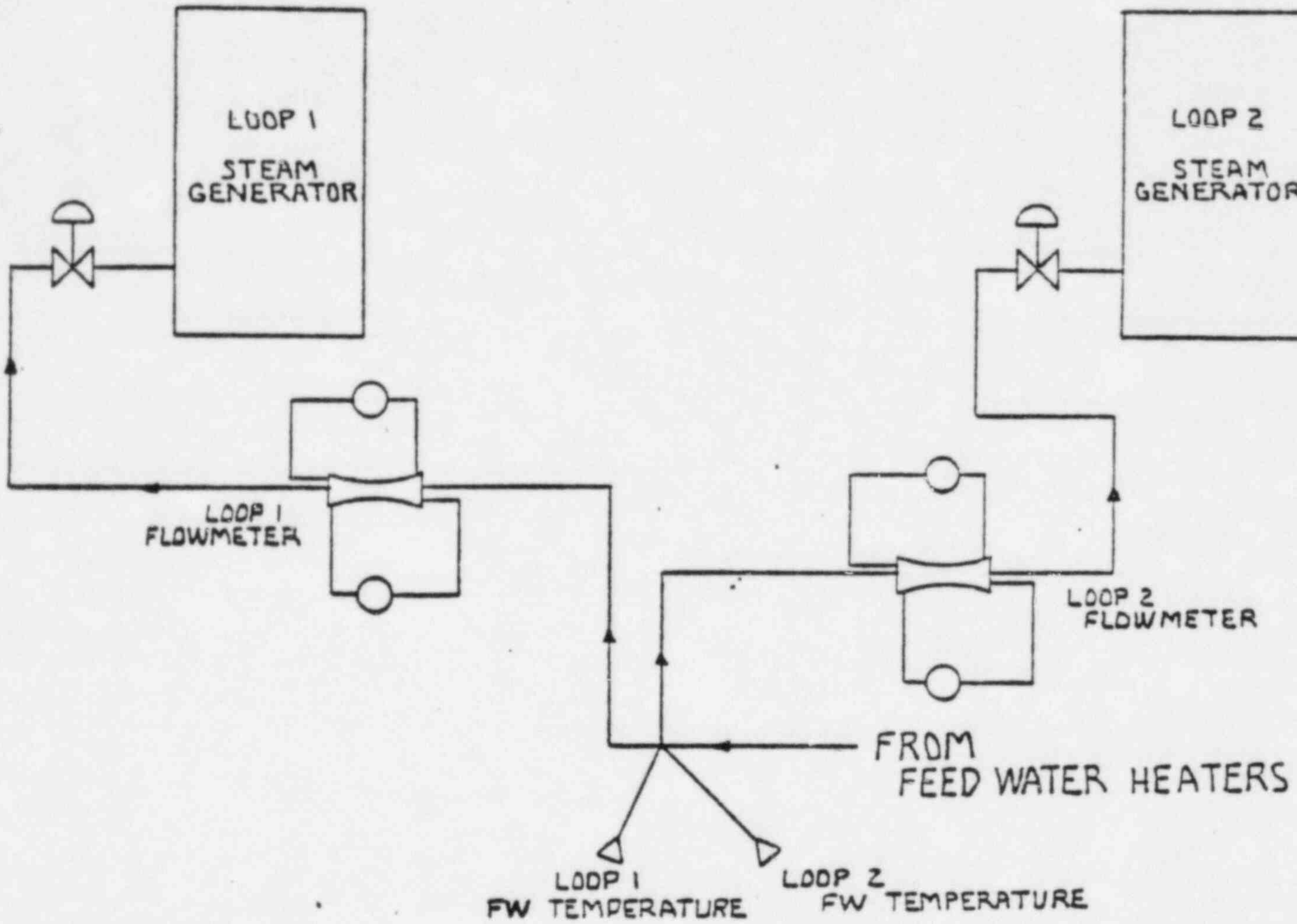


The following RTD's, each calibrated to ± 1/4°F, and tables prepared and sent with the sensor, were used for plant heat balance data for calculating total RC flowrate:

No.	Description
1	Loop 1 hot leg temp. Narrow Range, 520 to 620F
3	Loop 1 hot leg temp. Narrow Range, 520 to 620F
7	Loop 2 hot leg temp. Narrow Range, 520 to 620F
9	Loop 2 Hot leg temp. Narrow Range, 520 to 620F
5	Loop 1 cold leg temp. Narrow Range, 520 to 620 F
6	Loop 1 cold leg temp. Narrow Range, 520 to 620 F
11	Loop 2 cold leg temp. Narrow Range, 520 to 620 F
12	Loop 2 cold leg temp. Narrow Range, 520 to 620 F

RTD's 13, 14, 15 and 16 are wide range (50 to 650°F) sensors and were not used.
RTD's 2,4,8 and 10 are inputs to the RPS and were not used for heat balance.

FEEDWATER FLOWRATE MEASUREMENT ACCURACY



Each feedwater flowmeter is calibrated with 455°F water (rated flow) and the flow coefficient for each set of taps (two on each flowmeter) is supplied by the vendor.

The required accuracy is ± 1/2%.

Each ΔP transmitter is calibrated to the range specified by the measured flow coefficient within an accuracy of ± 0.25%.

The accuracies for the different parameters are shown in the Table on the next page. The accuracy for each parameter was conservatively calculated by summing the string errors. The environmental errors from changes in temperature and the errors from the computer were included in the values shown in the Table.

ACCURACY OF PRIMARY AND SECONDARY SIDE MEASUREMENTS USED FOR CALCULATION OF TOTAL RC FLOWRATE

PARAMETER	MEASUREMENT ACCURACY %	SPAN	ACCURACY UNITS
RC hot leg temp.	± 0.79 ✓ Calc 32-9208, p.4	520 to 620F	✓ ±0.79 F
deleted RPS thots			
RC cold leg temp.	± 0.79 ✓	520 to 620F	✓ ±0.79 F
Steam temp.	± 0.60 ✓	0 to 700F	±4.2 F
Feedwater temp.	± 1.13	0 to 600F	±6.8 F
Feedwater pressure	±1.0% ✓	0 to 1500 psig	± 15 psi
Steam pressure	±1.89% ✓	0 to 1200 psig	± 23 psi
RC pressure	± 0.77% ✓	0 to 2500 psig	± 19 psi
Feedwater Flow	± 1.25% see 32-9209 p13	0 to 960 inches (Std. H_2O)	± 12. inches
RC Flowrate	± 1.046 32-9209 p5	0 to 910 inches (Std. H_2O)	± 9.5 inches

32-9174 00

COMPARISON OF HOT LEG TEMPERATURES (T_H)

FOR DAVIS-BESSE 1 AT 100% POWER ON

APRIL 5, 1978

TIME	LOOP 1 T719 (°F)	LOOP 1 T720 (°F)	LOOP 1 T721 (°F)	LOOP 1 T722 (°F)
14:39	605.6	605.8	605.9	606.1
14:44	606.0	606.1	606.2	606.1
14:50	606.0	606.2	606.1	606.2
14:54	605.9	606.1	606.4	606.4
14:59	605.6	606.1	605.9	605.9
15:04	605.8	606.1	606.1	606.1
15:09	605.9	605.9	605.9	606.0
15:14	605.9	606.1	606.2	606.4
Midpoint	605.8	606.0	606.15	606.15
Span	± 0.2	± 0.2	± 0.25	± 0.25 (during minute

Average T_H = 606.03 F

TIME	LOOP 2 T729 (°F)	LOOP 2 T730 (°F)	LOOP 2 T728 (°F)	LOOP 2 T731 (°F)
14:39	604.9	605.1	604.3	604.9
14:44	605.3	605.3	604.8	605.0
14:50	605.4	605.4	604.9	605.2
14:54	605.1	605.3	604.4	605.2
14:59	605.1	605.0	604.6	604.8
15:04	605.1	605.1	604.4	604.8
15:09	605.0	604.9	604.6	604.7
15:14	605.2	605.4	604.7	605.2
Midpoint	605.15	605.15	604.6	604.95
Span	± 0.25	± 0.25	+ 0.3	± 0.25 (during minute

Average T_H = 604.96 F

(This uses RPS That values)

COMPARISON OF COLD LEG TEMPERATURES (Tc) FOR

DAVIS-BESSE I AT 100% POWER ON APRIL 5, 1978

TIME	RCP 1-1 TEMP(°F)	RCP 1-2 TEMP(°F)
14:39	559.2	558.6
14:44	559.1	558.4
14:50	558.9	558.3
14:54	558.9	558.3
14:59	558.8	558.1
15:04	559.0	558.4
15:09	558.9	558.3
15:14	559.2	558.4
Midpoint	559.0	558.35
Span	± 0.2	± 0.25 (during 35 minut

Loop 1 Avg. Tc = 558.7 F

TIME	RCP 2-1 TEMP(°F)	RCP 2-2 TEMP(°F)
14:39	559.2	559.1
14:44	558.7	559.4
14:50	558.7	559.3
14:54	558.7	558.7
14:59	558.7	558.6
15:04	558.3	558.8
15:09	558.4	558.8
15:14	559.1	558.9
Midpoint	558.75	559.0
Span	± 0.45	± 0.40 (during 35 minute

Loop 2 Avg. Tc = 558.9 F

COMPARISON OF STEAM TEMPERATURES (Ts) AND PRESSURES (Ps)

FOR DAVIS-BESSE 1 AT 100% POWER ON APRIL 5, 1978

TIME	LOOP 1 STEAM TEMP (°F)	LOOP 2 STEAM TEMP(°F)
14:39	595.6	596.2
14:44	595.6	596.7
14:50	595.5	596.7
14:54	595.5	596.4
14:59	595.4	596.6
15:04	595.5	596.4
15:09	595.5	596.4
15:14	595.6	596.4
Midpoint	595.5	596.45
Span	± 0.1	± 0.25 (during 35 minutes

TIME	LOOP 1 STEAM PRESS. (psig)	LOOP 2 STEAM PRESS. (psig)
14:39	905.8	881.2
14:44	907.8	879.4
14:50	905.2	882.0
14:54	905.4	881.1
14:59	906.1	885.9
15:04	905.1	881.1
15:09	903.7	879.9
15:14	905.1	884.7
Midpoint	905.75	882.65
Span	± 2.05	± 3.25 (during 35 minutes)

COMPARISON OF FEEDWATER TEMPERATURES (T_F) AND

PRESSURES (P_F) FOR DAVIS-BESSE 1 AT 100% POWER

ON APRIL 5, 1978

TIME	LOOP 1 FEEDWATER TEMP (°F)	LOOP 2 FEEDWATER TEMP (°F)
14:39	459.8	459.8
14:44	459.9	459.9
14:50	460.1	460.2
14:54	459.8	460.1
14:59	460.2	460.2
15:04	459.9	459.9
15:09	460.0	459.9
15:14	459.9	460.0
Midpoint	460.0	460.0
Span	±0.2	±0.2 (during 35 minutes)

T_f = 460.0 °F for Loops 1 and 2

TIME	LOOP 1 FEEDWATER PRESSURE (psig)	LOOP 2 FEEDWATER PRESSURE (psig)
14:39	942.4	955.9
14:44	942.8	956.1
14:50	944.6	957.2
14:54	943.2	956.7
14:59	944.9	958.8
15:04	943.2	956.1
15:09	943.2	956.7
15:14	943.2	957.0
Midpoint	943.65	957.35
Span	±1.25	±1.45 (during 35 minutes)

COMPARISON OF MEASURED FEEDWATER FLOWRATES (W_F)

FOR DAVIS-BESSE 1 AT 100% POWER ON APRIL 5, 1978

TIME	LOOP 1 FEEDWATER FLOWRATE (MPPH)	LOOP 2 FEEDWATER FLOWRATE (MPPH)
14:39	5.806	5.784
14:44	5.787	5.753
14:50	5.841	5.811
14:54	5.812	5.780
14:59	5.808	5.787
15:04	5.820	5.814
15:09	5.798	5.762
15:14	5.822	5.785
Average value:	5.812	5.785
Midpoint:	5.814	5.784
Span:	± 0.027	± 0.030 (during 35 minutes

PRIMARY SIDE ENTHALPY CALCULATION: 32-9174 00

NOMENCLATURE

H_H = Reactor Coolant hot leg enthalpy

H_C = Reactor Coolant cold leg enthalpy

H_S = Enthalpy of steam at steam generator outlet

H_F = Enthalpy of feedwater to the steam generator

ΔH = Change in enthalpy, an added subscript: 'pri' or 'sec' implies the primary and secondary loops respectively.

LOOP 1:

H_H @ 606.0°F and 2159 psia = 622.41 Btu/lb

H_C @ 558.7°F and 2220 psia = 558.14 Btu/lb

ΔH_{pri} = 64.27 Btu/lb

LOOP 2:

H_H @ 605.0°F and 2159 psia = 620.94 Btu/lb

H_C @ 558.9°F and 2220 psia = 558.39 Btu/lb

ΔH pri = 62.55 Btu/lb

SECONDARY SIDE ENTHALPY CALCULATION:

LOOP 1:

H_S @ 595.5°F and 920 psia = 1254.63 Btu/lb

H_F @ 460°F and 958 psia = 441.73 Btu/lb

ΔH_{sec} = 812.90 Btu/lb

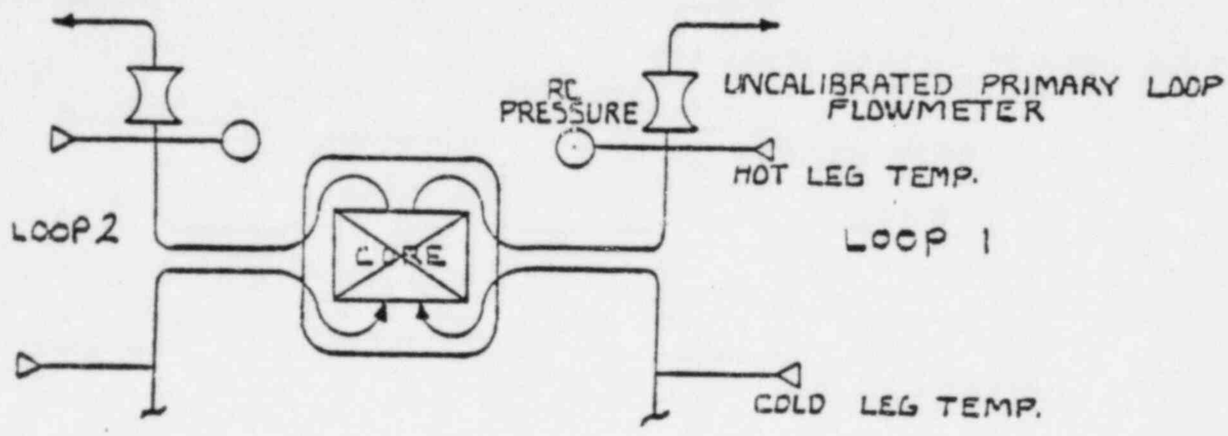
LOOP 2:

H_S @ 596.5°F and 897 psia = 1258.03 Btu/lb

H_F @ 460°F and 972 psia = 441.74 Btu/lb

ΔH_{sec} = 816.29 Btu/lb

CALCULATED TOTAL RC FLOWRATE



PRIMARY LOOP CONFIGURATION

For the entire primary-secondary system, the heat balance equation is

$$W_{RC1}(\Delta H_1)_{PRIMARY} + W_{RC2}(\Delta H_2)_{PRIMARY} + Q_{PUMPS}$$

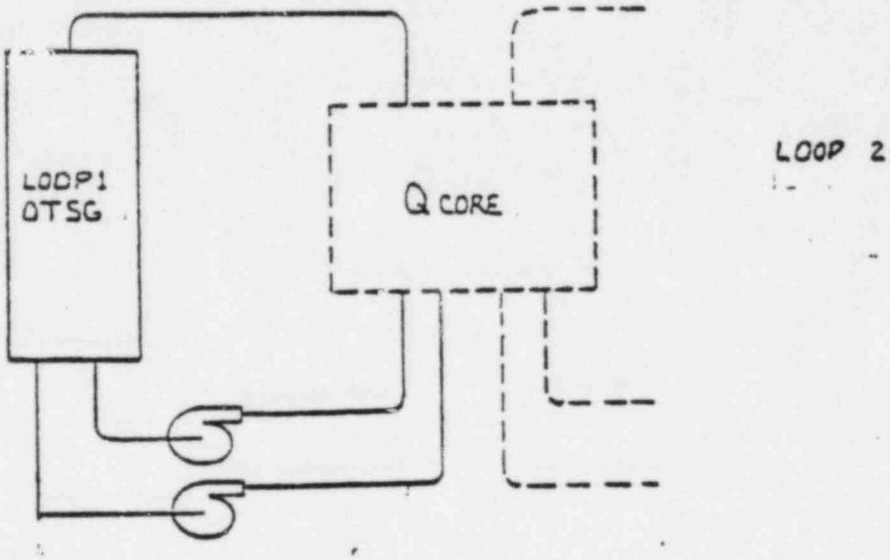
$$= Q_{SG} + Q_{RADIATION\ LOSSES}$$

where W_{RC1} = Reactor Coolant Flow Rate for loop 1

and. W_{RC2} = Reactor Coolant Flow Rate for loop 2

Q_{PUMPS} = Heat input by the Reactor Coolant Pumps

$= 75.3 \times 10^6$ Btu/hr



SECONDARY LOOP CONFIGURATION

$$Q_{SG} = (W_F \, \Delta H_{sec})_{LOOP1} + (W_F \, \Delta H_{sec})_{LOOP2}$$

$$Q_{\substack{RADIATION \\ LOSSES}} = 0.54 \times 10^6 \text{ Btu/hr}$$

substituting the above in the heat balance equation

$$W_{RC1} (\Delta H_1)_{PRIMARY} + W_{RC2} (\Delta H_2)_{PRIMARY} + 75.3 \times 10^6 \frac{\text{Btu}}{\text{hr}}$$

$$= (W_F \Delta H_{sec})_{LOOP1} + (W_F \Delta H_{sec})_{LOOP2} + 0.54 \times 10^6 \frac{\text{Btu}}{\text{hr}}$$

or $W_{RC1} (\Delta H_1)_{PRIMARY} + W_{RC2} (\Delta H_2)_{PRIMARY} = (W_F \Delta H_{sec})_{LOOP2}$

$$+ (W_F \Delta H_{sec})'_{LOOP1} - 74.76 \times 10^6 \frac{\text{Btu}}{\text{hr}}$$

or $W_{RC1}(\Delta H_1)_{PRIMARY} = (W_F \Delta H_{sec})_{LOOP1} - \frac{74.76 \times 10^6}{2} \frac{Btu}{hr}$

and $W_{RC2}(\Delta H_2)_{PRIMARY} = (W_F \Delta H_{sec})_{LOOP2} - \frac{74.76 \times 10^6}{2} \frac{Btu}{hr}$

Solving for W_{RC1} and W_{RC2}:

$$W_{RC1} = \frac{(W_F \Delta H_{sec})_{LOOP1} - 37.38 \times 10^6 \text{ Btu/hr}}{(\Delta H_1)_{PRIMARY}}$$

and

$$W_{RC2} = \frac{(W_F \Delta H_{sec})_{LOOP2} - 37.38 \times 10^6 \text{ Btu/hr}}{(\Delta H_2)_{PRIMARY}}$$

From page 10

$(\Delta H_1)_{PRI} = 64.27$ Btu/lb

$(\Delta H_2)_{PRI} = 62.55$ Btu/lb

$(\Delta H_{sec})_{LOOP\,1} = 812.90$ Btu/lb

$(\Delta H_{sec})_{LOOP\,2} = 816.29$ Btu/lb

Substituting the above values and noting that

$W_{F\,LOOP1} = 5.812 \times 10^6$ lb/hr

$W_{F\,LOOP2} = 5.785 \times 10^6$ lb/hr,

$C \Rightarrow 21.28 \times 10^6$

$$W_{RC1} = \frac{(5.812\times10^6)(812.90) - 37.38\times10^6 \text{ Btu/hr}}{(64.27)}$$

$$= 72.93 \times 10^6 \text{ lb/hr}$$

$$W_{RC2} = \frac{(5.785\times10^6)(816.29) - 37.38\times10^6 \text{ Btu/hr}}{(62.55)}$$

$$= 74.90 \times 10^6 \text{ lb/hr}$$

Total RC flow rate $= W_{RCT} = W_{RC1} + W_{RC2}$

$$= 147.83 \times 10^6 \text{ lb/hr}$$ vs 148.34×10^6

The total RC flowrate in gpm may be calculated as

$$W_{RCT}(\text{gpm}) = \frac{W_{RCT}\ (\text{lb/hr})}{\frac{60}{7.4805} \times \rho_{\text{cold leg}}}$$

$\rho_{\text{cold leg}} = 46.26$ lb/ft^3 at 558.8 °F and 2220 psia

$$W_{RCT}(\text{gpm}) = 3.984 \times 10^5 \text{ gpm}$$

The design flowrate for RC system $= 4 \times 88000$ gpm/pump
$= 352,000$ gpm

Therefore the ratio to design flowrate is : $\dfrac{398400}{352000}$
$= 113.2\%$

ERROR ANALYSIS OF CALCULATED RC FLOWRATE

The basic equation for Loop 1 or 2 RC flowrate is:

$$W_{RC} = \frac{W_F\left[\Delta H_{sec}\right] - \frac{C}{2}}{\left[\Delta H_{PRI}\right]}$$

where $C = Q_{pumps} - Q_{rad.\ losses} = 74.76 \times 10^6$ Btu/hr

If one assumes a extreme case of a 10% vs 50% error in C, then C would be equal to 8.2236×10^7 Btu/hr rather than 74.76×10^6 Btu/hr.

Then total RC Flowrate would become:

$$W_{RC1} = \frac{5.812 \times 10^6 \times 812.90 - 37.38 \times 10^6 \text{ Btu/hr}}{64.27} = 72.87 \times 10^6 \text{ lb/hr}$$

$$W_{RC2} = \frac{5.785 \times 10^6 \times 816.29 - 37.38 \times 10^6 \text{ Btu/hr}}{62.55} = 74.84 \times 10^6 \text{ lb/hr}$$

Total RC flow rate = 147.7×10^6 lb/hr and the

Change in total RC flowrate is 0.08% vs 0.23%

For this small change we shall assume C is a constant and is 74.76×10^6 Btu/hr. If it is in error, its influence on the final value is insignificant.

ERROR ANALYSIS OF CALCULATED RC FLOWS

For either Loop A or B the equation for RC flow is:

$$W_{RC} = W_F \frac{(H_S - H_F)}{(H_H - H_C)}$$

$$= C\sqrt{\rho \Delta P}\,\frac{\left[\phi(T_S, P_S) - \phi(T_F, P_F)\right]}{\left[\phi(T_H, P_H) - \phi(T_C, P_C)\right]}$$

Note: Introducing a constant in the expression will not change the following work.

$$\& \quad W_{RC_T}' = W_{RC_1} + W_{RC_2} .$$

Determine the expression for the error in W_{RC} for a normal distribution of small errors in each of the measured parameters

$$W_F = C\sqrt{\rho \cdot \Delta P}$$

Where ρ is a function of T_F and P_F; and C, the flow coefficient is assumed to be a constant.

$$dW_F = \left[\left(\frac{\partial W_F}{\partial \rho}\, d\rho\right)^2 + \left(\frac{\partial W_F}{\partial \Delta P}\, d\Delta P\right)^2\right]^{1/2}$$

$$d\rho = \frac{\partial \rho}{\partial T_F}\, dT_F + \frac{\partial \rho}{\partial P_F}\, dP_F \text{ ; But } \underline{\frac{\partial \rho}{\partial P_F} \approx 0}.$$

$$\frac{\partial W_F}{\partial \rho} = \frac{C}{2}\sqrt{\frac{\Delta P}{\rho}} \quad ; \quad \frac{\partial W_F}{\partial \Delta P} = \frac{C}{2}\sqrt{\frac{\rho}{\Delta P}}$$

The value for each enthalpy in the first equation given on the previous page was determined from temperature and pressure measurements. The value of W_F was obtained from the feedwater temperature, and from feedwater differential pressure measurements.

In the first equation given on the previous page, the only parameters that depend on a common measurement are H_F and W_F. Specifically, H_F and W_F both depend on T_F. Consequently, in the error analysis, a correlation term is developed to account for the dependence of H_F and W_F on T_F.

The correlation term is:

$$2\left(\frac{\partial W_{RC}}{\partial W_F}\right)\left(\frac{\partial W_F}{\partial T_F}\right)\left(\frac{\partial W_{RC}}{\partial \frac{H_S-H_F}{H_H-H_C}}\right)\left(\frac{\partial \frac{H_S-H_F}{H_H-H_C}}{\partial T_F}\right) dT_F^{\,2}$$

$$\frac{\partial W_{RC}}{\partial W_F} = \left(\frac{H_S-H_F}{H_H-H_C}\right)$$

$$\frac{\partial W_F}{\partial T_F} = \left(\frac{\partial W_F}{\partial \rho_F}\right)\left(\frac{\partial \rho_F}{\partial T_F}\right) = \left(\frac{C}{2}\sqrt{\frac{\Delta P}{\rho_F}}\right)\left(\frac{\partial \rho_F}{\partial T_F}\right)$$

$$\frac{\partial W_{RC}}{\partial \frac{H_S-H_F}{H_H-H_C}} = W_F = C\sqrt{\rho \Delta P}$$

$$\frac{\partial \frac{H_S-H_F}{H_H-H_C}}{\partial T_F} = \left(\frac{\partial \frac{H_S-H_F}{H_H-H_C}}{\partial H_F}\right)\left(\frac{\partial H_F}{\partial T_F}\right) = \left(\frac{-1}{H_H-H_C}\right)\left(\frac{\partial H_F}{\partial T_F}\right)$$

The correlation term becomes:

$$2\left(\frac{H_S-H_F}{H_H-H_C}\right)\left(\frac{C}{2}\sqrt{\frac{\Delta P}{\rho}}\right)\left(\frac{\partial \rho}{\partial T_F}\right)\left(C\sqrt{\rho \Delta P}\right)\left(\frac{-1}{H_H-H_C}\right)\left(\frac{\partial H_F}{\partial T_F}\right) dT_F^{\,2}$$

$$\text{or} \; -C^2 \Delta P \, \frac{(H_S-H_F)}{(H_H-H_C)^2}\left(\frac{\partial \rho}{\partial T_F}\right)\left(\frac{\partial H_F}{\partial T_F}\right) dT_F^{\,2}$$

For the Bailey Meter Company - supplied feedwater flowmeters which are calibrated prior to shipment to the site, we have the following typical equation:

$$W_F = C\sqrt{\rho\,\Delta P}$$

and

$$C = A\,d^2\,C_a\,C_d\,F_a$$
$$= (A\,C_a\,F_a) \times d_o^2\,C_d \quad \text{where}$$

A = conversion factor
C_a = approach factor (dependent on the beta ratio)
F_a = effect of thermal expansion of flowmeter throat diameter
d_o = measured throat diamter - inches

and C_d is the flow coefficient derived from the calibration data for each set of pressure taps.

$$\text{Loop 1} \quad W_F = \frac{C_1\sqrt{\rho_1\,\Delta P_1} + C_2\sqrt{\rho_1\,\Delta P_2}}{2},$$

$$= \frac{K d_o^2}{2}\left[C_{d_1}\sqrt{\rho_1\,\Delta P_1} + C_{d_2}\sqrt{\rho_1\,\Delta P_2}\right]$$

Typically d_o = 9.121 inches measured to ± 0.001 inches thus $\delta d_o / d_o \approx$ 0.011%, which is negligible

Also, at or near full power flow, the test data values of C_d are 0.9909, 0.9916, 0.9928 and 0.9878. The uncertainty in the above values is negligible as compared to other uncertainties.

Based on the above description, C will be assumed to be constant for this error analysis.

The expression for dW_F may then be written as:

$$dW_F = \left[\left(\frac{C}{2}\sqrt{\frac{\Delta P}{\rho}}\,d\rho\right)^2 + \left(\frac{C}{2}\sqrt{\frac{\rho}{\Delta P}}\,d\Delta P\right)^2\right]^{1/2}$$

$$dW_F = \left[\left(\frac{C}{2}\sqrt{\frac{\Delta P}{\rho}}\,\frac{\partial \rho}{\partial T_F}\,dT_F\right)^2 + \left(\frac{C}{2}\sqrt{\frac{\rho}{\Delta P}}\,d\Delta P\right)^2\right]^{1/2}$$

Since T_F determines feedwater enthalpy as well as feedwater flowrate, an error in T_F can result in non-independent errors in H_F and W_F.

Continuing with the development of dW_{RC};

$$H_x = \phi\,(T_x, P_x)$$

and

$$dH_x = \left[\left(\frac{\partial H_x}{\partial T_x}\,dT_x\right)^2 + \left(\frac{\partial H_x}{\partial P_x}\,dP_x\right)^2\right]^{1/2}$$

From $W_{RC} = W_F \dfrac{(H_S - H_F)}{(H_H - H_C)}$ a general expression is developed for the accuracy of either Loop 1 or 2 RC flowrate

$$dW_{RC} = \left[\left(\frac{\partial W_{RC}}{\partial W_F}\,dW_F\right)^2 + \left(\frac{\partial W_{RC}}{\partial H_S}\,dH_S\right)^2 + \left(\frac{\partial W_{RC}}{\partial H_F}\,dH_F\right)^2 + \left(\frac{\partial W_{RC}}{\partial H_H}\,dH_H\right)^2 + \left(\frac{\partial W_{RC}}{\partial H_C}\,dH_C\right)^2 - C^2 \Delta P \frac{(H_S - H_F)}{(H_H - H_C)^2}\left(\frac{\partial \rho}{\partial T_F}\right)\left(\frac{\partial H_F}{\partial T_F}\right) dT_F^{\,2}\right]^{1/2}$$

Calculate coefficient C for the feedwater flowmeter:

$$W_F\ (\text{lbs/hour}) = C\sqrt{\rho\left(\frac{\text{lbs}}{\text{ft}^3}\right)\Delta P\,(\text{inches std. water})}$$

From Bailey Meter Co. Calibration sheets

$$C = \frac{7.0\times10^6\ \text{lbs/hr}}{\sqrt{51.49\times956.5}} = 0.3154\times10^5.$$

ρ_F at 455 F and 1065 psia = 51.49 lbs/ft^3

and ΔP (full scale) = 956.5 inches of standard water.

Calculate typical value of ΔP at full power:

$$\Delta P = \left(\frac{W_F}{C}\right)^2\frac{1}{\rho} = \left(\frac{5.812\times10^6}{0.03154\times10^6}\right)^2\left(\frac{1}{51.26}\right) = 662 \text{ INCHES}$$

ρ at 460F and 1050 psia = 51.26 lbs/ft^3

Calculate $\frac{H_S - H_F}{(H_H - H_C)^2}$:

Use Loop 2 values to obtain $\frac{816.29}{(62.55)^2} = 0.2086$.

Evaluate $\frac{\partial \rho}{\partial T_F}$ for variation around 460 F:

ρ at 457.4 F = 51.335 lbs/ft^3
ρ at 462.2 F = 51.099 lbs/ft^3

$$\frac{\partial \rho}{\partial T_F} = \frac{0.236}{-4.8} = -0.0492 \ \frac{\text{lb}}{\text{ft}^3\text{-F}}$$

Evaluate $\frac{\partial H_F}{\partial T_F}$ for variation around 460 F:

h_F at 462F = 443.99 Btu/lb

h_F at 458F = 439.48 Btu/lb

$$\frac{\partial H_F}{\partial T_F} = 1.128 \text{ BTU/lb-F}$$

Using the above values, the complete correlation term (P) may be calculated as:

$$P = -(3.154)^2 \times 10^8 \times 0.662 \times 10^3 \times 0.2086 \times (-4.92 \times 10^{-2})\ (1.128)\ (dT_f)^2$$

or

$$P = 7.624 \times 10^9 \times (dT_f)^2$$

Evaluate the following terms:

$$\frac{\partial W_{RC}}{\partial W_F}, \quad \frac{\partial W_{RC}}{\partial H_S}, \quad \frac{\partial W_{RC}}{\partial H_F}, \quad \frac{\partial W_{RC}}{\partial H_H}, \quad \frac{\partial W_{RC}}{\partial H_C}$$

$$\frac{\partial W_{RC}}{\partial W_F} = \frac{\partial}{\partial W_F}\left[W_F \frac{(H_S - H_F)}{H_H - H_C}\right] = \frac{H_S - H_F}{H_H - H_C} \equiv A$$

$$\frac{\partial W_{RC}}{\partial H_S} = \frac{\partial}{\partial H_S}\left[\frac{W_F H_S - W_F H_F}{H_H - H_C}\right] = \frac{W_F}{H_H - H_C} \equiv B$$

$$\frac{\partial W_{RC}}{\partial H_F} = \frac{\partial}{\partial H_F}\left[\frac{W_F H_S - W_F H_F}{H_H - H_C}\right] = \frac{- W_F}{H_H - H_C} \equiv - B$$

$$\frac{\partial W_{RC}}{\partial H_H} = \frac{\partial}{\partial H_H}\left[\frac{W_F (H_S - H_F)}{(H_H - H_C)}\right] = \frac{- W_F (H_S - H_F)}{(H_H - H_C)^2} = \frac{- W_{RC}}{H_H - H_C} \equiv -D$$

$$\frac{\partial W_{RC}}{\partial H_C} = \frac{\partial}{\partial H_C}\left[\frac{W_F (H_S - H_F)}{(H_H - H_C)}\right] = \frac{+ W_{RC}}{H_H - H_C} \equiv D$$

$$A = \frac{\frac{812.90}{64.27} + \frac{816.29}{62.55}}{2} = 12.85$$

$$B = \frac{\frac{5.812 \times 10^6}{64.27} + \frac{5.785 \times 10^6}{62.55}}{2} = 9.146 \times 10^4 \ \frac{lb^2}{BTU\text{-}hr}$$

$$D = \frac{\frac{73.18 \times 10^6}{64.27} = \frac{75.16 \times 10^6}{62.55}}{2} = 1.170 \times 10^6 \ \frac{lb^2}{BTU\text{-}hr}$$

Rewriting the expression for $d\,\dot{W}_{RC}$:

$$d\,W_{RC} = \left[A^2\,(d\,W_F)^2 + B^2(dH_S)^2 + B^2(dH_F)^2 + D^2(dH_H)^2 + D^2(dH_C)^2 + P\right]^{\frac{1}{2}}$$

OR

$$d\,W_{RC} = \left[A^2\left(\frac{C}{2}\sqrt{\frac{\Delta P}{\rho}}\,\frac{\partial \rho}{\partial T_F}\,dT_F\right)^2 + A^2\left(\frac{C}{2}\sqrt{\frac{\rho}{\Delta P}}\,d\Delta P\right)^2\right.$$
$$+ B^2\left(\frac{\partial H_S}{\partial T_S}\,dT_S\right)^2 + B^2\left(\frac{\partial H_S}{\partial P_S}\,dP_S\right)^2 + B^2\left(\frac{\partial H_F}{\partial T_F}\,dT_F\right)^2 + B^2\left(\frac{\partial H_F}{\partial P_F}\,dP_F\right)^2$$
$$\left. + D^2\left(\frac{\partial H_H}{\partial T_H}\,dT_H\right)^2 + D^2\left(\frac{\partial H_H}{\partial P_H}\,dP_H\right)^2 + D^2\left(\frac{\partial H_C}{\partial T_C}\,dT_C\right)^2 + D^2\left(\frac{\partial H_C}{\partial P_C}\,dP_C\right)^2 + P\right]^{\frac{1}{2}}$$

Where P is the correlation term developed on Page 16.

Evaluate all the remaining coefficients:

$$\frac{C}{2}\sqrt{\frac{\Delta P}{\rho}}\ \frac{\partial\rho}{\partial T_F} = \frac{.3154\times10^5}{2}\sqrt{\frac{662}{51.26}}\times(-0.0492)$$

$$= -2788\ ^{\text{lbs}}/_{\text{Hr.}-{}^\circ\text{F}}$$

$$\frac{C}{2}\sqrt{\frac{\rho}{\Delta P}} = 15770\sqrt{\frac{51.26}{662}} = 4388\ ^{\text{lbs}}/_{\text{Hr.}-\text{inches std. water}}$$

$\frac{\partial H_S}{\partial T_S}$ (around 596F and 910 psia) 592F to 600F $=$ 0.835 Btu/lb $-{}^\circ$F

$\frac{\partial H_S}{\partial P_S}$ (around 596F and 910 psia) 890 to 930 psia $=$ -0.112 Btu/lb $-{}^\circ$F

$\frac{\partial H_F}{\partial T_F}$ (around 460F and 960 psia) 453 to 467 F $=$ 1.126 Btu/lb $-{}^\circ$F

$\frac{\partial H_F}{\partial P_F}$ (around 460F and 960 psia) 900 to 1020 psia $= -$ 0.00042 Btu/lb - psi

$\frac{\partial H_H}{\partial T_H}$ (around 605.5F and 2160 psia) 604.5 to 606.5F $=$ 1.47 Btu/lb $-{}^\circ$F

$\frac{\partial H_H}{\partial P_H}$ (around 605.5 F and 2160 psia) 2140 to 2180 psia $=$ -0.0056 Btu/lb - psi

$\frac{\partial H_C}{\partial T_C}$ (around 558.8F and 2220 psia) 557.8 to 559.8 F $=$ 1.260 Btu/lb $-{}^\circ$F

$\frac{\partial H_C}{\partial P_C}$ (around 558.8F and 2220 psia) 2200 to 2240 psia $= -$ 0.0020 Btu/lb - psi

Substituting each of the calculated coefficients into the equation for d W_{RC} we obtain:

$$
\begin{aligned}
d\,W_{RC} = \Big[& (12.85)^2\,(-2788)^2\,(dT_F)^2 + (12.85)^2\,(4388)^2\,(d\Delta P)^2 \\
& + (91460)^2\,(0.835)^2\,(dT_S)^2 + (91460)^2\,(-0.112)^2\,(dPs)^2 \\
& + (91460)^2\,(1.126)^2\,(dT_F)^2 + (91460)^2(.00042)^2\,(dP_F)^2 \\
& + (1.170\text{X}10^6)^2\,(1.47)^2(dT_H)^2 + (1.170\text{X}10^6)^2(-0.0056)^2(dP_H)^2 \\
& + (1.170\,\text{X}10^6)^2(1.26)^2(dT_C)^2 + (1.170\text{X}10^6)^2(-0.0020)^2(dP_C)^2 \\
& + 7.624 \text{ X } 10^9\,(dT_F)^2 \Big]^{1/2}
\end{aligned}
$$

$$
\begin{aligned}
d\,W_{RC} = \Big[& 3.179 \text{ X } 10^9\,(d\Delta P)^2 + 2.953 \text{ X } 10^{12}\,(dT_H)^2 + 2.173 \text{ X } 10^{12}\,(dT_C)^2 \\
& + 1.952 \text{ X } 10^{10}\,(dT_F)^2 + 0.764 \text{ X } 10^{10}\,(dT_S)^2 + 0.429 \text{ X } 10^8\,(dP_H)^2 \\
& + 0.055 \text{ X } 10^8\,(dP_C)^2 + 1.049 \text{ X } 10^8\,(dP_S)^2 + .148 \text{ X } 10^4\,(dP_F)^2 \Big]^{1/2}
\end{aligned}
$$

$$
\begin{aligned}
(d\Delta P)^2 &= (12)^2 = 144. \\
(d\,T_H)^2 &= (0.79)^2 = 0.62 \\
(d\,T_C)^2 &= (0.79)^2 = 0.62 \\
(d\,T_F)^2 &= (6.8)^2 = 46. \\
(d\,T_S)^2 &= (4.2)^2 = 18. \\
(d\,P_H)^2 &= (19)^2 = 361 \\
(d\,P_C)^2 &= (19)^2 = 361 \\
(d\,P_S)^2 &= (23)^2 = 529 \\
(d\,P_F)^2 &= (15)^2 = 225
\end{aligned}
$$

$$
d\,W_{RC} = \left[\begin{array}{l}
4.578 \text{ X } 10^{11} + \\
1.834 \text{ X } 10^{12} + \\
1.347 \text{ X } 10^{12} + \\
0.898 \text{ X } 10^{12} + \\
0.138 \text{ X } 10^{12} + \\
1.549 \text{ X } 10^{10} + \\
1.986 \text{ X } 10^{9} + \\
0.555 \text{ X } 10^{11} + \\
0.333 \text{ X } 10^{6}
\end{array}\right]^{1/2}
\quad
\begin{array}{l}
\text{for } (d\Delta P) \\
\text{for } (d\,T_H) \\
\text{for } (d\,T_C) \\
\text{for } (d\,T_F) \\
\text{for } (d\,T_S) \\
\text{for } (d\,P_H) \\
\text{for } (d\,P_C) \\
\text{for } (d\,P_S) \\
\text{for } (d\,P_F)
\end{array}
$$

$$d\dot{W}_{RC} = \sqrt{474.8 \times 10^5} = \pm 2.18 \times 10^6 \text{ lbs/hour}$$

or $\pm 2.95\%$ for either loop.

Since the flowrates for the two RC loops were calculated from measurements taken with two completely different sets of sensors, the total RC flowrate percentage error from the heat balance measurements is:

$$* \quad \pm 0.707\ (2.95) = \pm 2.09\%$$

As shown on p.4, the RC flowrate string error for the Gentile flowrates is 1.046%. After the RC flowmeter ΔP transmitters are calibrated against the results obtained from the heat balance determination, then the total fractional error in each loop flowrate will be:

$$= \sqrt{(.0295)^2 + (.01046)^2}$$

$$= \sqrt{.000980}$$

$$\text{or } = \pm 3.1\ \%$$

The total RC flowrate is simply the sum of both loop flowrates, but the extremes of erroneous signals will not occur simultaneously; therefore, we can say: that the percentage error in the total RC flowrate from the Gentile flowrate measurements, as calibrated by the heat balance results, is $\left(\frac{3.1\%}{\sqrt{2}}\right) = 2.2\%$.

$$* \quad dW_{TOT} = \left\{ \left(\frac{\partial W_{TOT}}{\partial W_1} dW_1\right)^2 + \left(\frac{\partial W_{TOT}}{\partial W_2} dW_2\right)^2 \right\}^{1/2}$$

$$\frac{\partial W_{TOT}}{\partial W_1} = \frac{\partial W_{TOT}}{\partial W_2} = 1$$

$$\therefore \quad dW_{TOT} = \left\{ (dW_1)^2 + (dW_2)^2 \right\}^{1/2}$$

subst for dW_1 & dW_2:

$$dW_{TOT} = \left\{ (2.95\%\, W_1)^2 + (2.95\, W_2)^2 \right\}^{1/2}$$

if $W_1 \approx W_2$, then $\frac{W_1}{W_T} = \frac{W_2}{W_T} = \frac{1}{2}$

divide by W_T and substitute $W_1 = W_2 = \frac{W_{TOT}}{2}$

$$\text{then} \quad \frac{dW_{TOT}}{W_{TOT}} = \left\{ \frac{(2.95\%)^2 \left(\frac{W_{TOT}}{2}\right)^2 (2)}{W_{TOT}^2} \right\}^2$$

$$\therefore \quad \frac{dW_{TOT}}{W_{TOT}} = \left\{ \frac{(2.95\%)^2}{2} \right\}^2 = 0.707\,(2.95)^* \text{ in } \%$$

32-9174 00

STATEMENT ADDRESSING FOREIGN
MATERIAL AND DEPOSITS IN SECONDARY
SYSTEM

The OTSG Feedwater Chemistry control is designed to minimize the ingress of contaminants to the units. These contaminants include both suspended and dissolved solids. Chemistry control utilizes the all-volatile chemicals ammonia and hydrazine which will not deposit or form insoluble solids which could deposit on critical surfaces. The use of these chemicals is designed to minimize the corrosion of feedwater train materials and thus the input of corrosion product oxides into the steam generators. Water purity is further maintained through the use of full flow condensate polishers (powdered resin) which, in addition to removing dissolved solids, also provide excellent filters. These filters are located such that they process all the water coming from the condenser. As a result, damage to orifices in the feedwater train is highly improbable due to the aforementioned feedwater chemistry and purity control.

Shot blasting for the feedwater piping was done in the shop before installation of the piping. Thus, no shot blasting was done with the feedwater flow elements installed.