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November 4, 1982
Docket No. 50-336
A02724

Director of Nuclear Reactor Regulation
Attn: Mr. Robert A. Clark, Chief
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: (1) W. G. Counsil letter to R. A. Clark, dated March 4, 1982.

(2) R. A. Clark letter to W. G. Counsil, dated August 19, 1982.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Additional Information, Measurement Uncertainties

At the request of the NRC staff, Northeast Nuclear Energy Company (NNECO) provided justification for the measurement uncertainties utilized in the Millstone Unit No. 2 safety analysis. That information was docketed in Reference (1). Reference (2) requested additional information from Northeast Nuclear Energy Company necessary for the staff to complete their review.

Subsequent to the issuance of Reference (2), Northeast Nuclear Energy Company determined that not all of the information requested would be available within the agreed upon schedule. Realizing that review time must be scheduled with your contractor Batelle Pacific Northwest Laboratories and the Core Performance Branch, my staff arranged a conference call to discuss a revised schedule for submitting the requested information.

The revised schedule agreed to by our respective staffs provides for the following:

A001

Responses to Questions 1,2,3 and 5	November 1, 1982
Response to Question 4	January 1, 1983
Response to Question 6	April, 1983

Additional time is required to respond to Question 4 since Northeast Nuclear Energy Company is installing new process equipment for the measurement channels under review. Currently, the project is approximately fifty percent (50%) complete with the remaining work scheduled to be completed during the 1983 refueling outage. With the process equipment change out, the historical drift data requested by Question 4 would prove to be of little value. Northeast Nuclear Energy Company intends to provide the staff with the appropriate information in order to substantiate the assumptions made in the Reference (1) analysis. This will include historical drift data for equipment which is not replaced. Northeast Nuclear Energy Company does not expect the conclusions of the uncertainty analysis to be affected by the characteristics of the new process equipment being installed at Millstone Unit No. 2.

Question 6 requires input from several different areas including our fuel vendor who must generate additional axial shape information for this purpose. Their time estimates to complete this effort necessitates additional time until April, 1983 for our response to Question 6.

Our responses to Questions 1,2,3 and 5 are complete and are attached. The information provided continues to support the measurement uncertainties utilized in the Millstone Unit No. 2 safety analysis as well as the statistical combination of their equivalent power uncertainties.

We trust you find this information responsive to the Reference (2) requests.

Very truly yours,

Northeast Nuclear Energy Company

W. G. Counsil

W. G. Counsil
Senior Vice President

QUESTION 1

In some cases in your response, uncertainties which are not independent have been combined through the RMS technique. The RMS technique requires that the uncertainty contributions be independent. If they are dependent, their combined effect should be assessed through deterministic methods.

Three cases so identified are the feedwater venturi area expansion factor, feedwater density and feedwater enthalpy. These three factors are each dependent upon the feedwater temperature and are used in determining the core thermal power. Similarly, the feedwater density and steam enthalpy are both dependent upon steam generator pressure and are used in determining the core thermal power.

Correctly combine these dependent uncertainties deterministically to find if the uncertainty in the core power falls within the value used in the safety analyses.

RESPONSE

The response to Question 1 of the Reference provided the basis for determining the individual error contributions associated with the calculation of calorimetric core power. That response assumed all error contributions to be independent and thus were combined using the RMS method to determine the overall core power uncertainty. This response recalculates the overall core power uncertainty after identifying the dependent and independent error contributions and combining these uncertainties in an appropriate statistical manner.

Figure 1 provides a diagram of the measurements and calculations required to determine calorimetric core power. From Figure 1, it is seen that within each steam generator loop the steam enthalpy, h_s, feedwater enthalpy, h_f, and feedwater density, ρ_f, are dependent on the same measurement of steam generator pressure, P_s. Similarly, within each steam generator loop, feedwater enthalpy, h_f, feedwater density, ρ_f, and feedwater venturi area expansion factor, F_a, are dependent on the same measurement of feedwater temperature, T_f. The feedwater temperature measurement, T_f, also provides input to the Loop 2 h_f, ρ_f, and F_a calculations; therefore, the T_f error contributions are not only dependent within a single steam generator loop but also between each steam generator loop. The error contributions that are independent within each steam generator loop are the errors associated with the venturi calibration coefficient, K, the venturi ΔP measurement, and the component of the venturi area expansion factor, F_a, associated with the uncertainty in the linear thermal expansion coefficient.

The uncertainties in the measured plant parameters were previously calculated in the response to Question 1 of the Reference as follows:

Steam Pressure, P_s	$\pm$ 16.7 psi
Feedwater Temperature, T_f	$\pm$ 9.9 ^oF
Feedwater Venturi ΔP	$\pm$ 1.24% ΔP span
	($\pm$ .703% nominal flow)

The errors in steam generator thermal output associated with independent error contributions (K, ΔP, F_a errors) were previously calculated in the response to Question 1 of the Reference. In summary, the uncertainty associated with the venturi calibration coefficient, K, results in a $\pm$ .37% error in steam generator thermal output; the uncertainty in the venturi area expansion factor, F_a, attributed to the uncertainty in the linear thermal expansion coefficient results in a $\pm$ .034% steam generator power uncertainty; and the venturi ΔP measurement error results in a $\pm$ .703% uncertainty in steam generator thermal output. Since these three error contributions are independent within each steam generator loop, the errors can be combined statistically using the RMS method to give a $\pm$ .795% error in steam generator thermal output. These results are summarized on Table 1.

Within each loop, steam and feedwater enthalpy and feedwater density are dependent on the same steam generator pressure measurement, P_s. However, since each steam generator loop has an independent pressure measurement, the overall enthalpy and density errors associated with steam generator pressure measurement errors are independent between loops.

Based on the ASME Steam Tables, a steam generator pressure uncertainty of +16.7 psi results in enthalpy and density uncertainties equivalent to the following steam generator thermal power uncertainties:

h_s (P_s)	-.075%
h_f (P_s)	-.0017%
ρ_f (P_s)	+.0066%

Since these three error contributions are all dependent on P_s, the overall uncertainty is determined by addition to give a -.0701% uncertainty in steam generator thermal output. The signs of the above uncertainties indicate the bias in steam generators thermal output uncertainty. For a P_s error biased in the negative direction (-16.7 psi), the signs are reversed. Since the P_s measurement is independent of the T_f and ΔP measurements, the overall pressure uncertainty can be expressed as $\mp$ .0701% for a $\pm$ 16.7 psi uncertainty in steam generator pressure. These results are summarized on Table 1.

Within each loop, feedwater enthalpy and density and the venturi area expansion factor, F_a, are dependent on the same feedwater measurement, T_f. Since the same feedwater temperature measurement is utilized for both loops, the T_f error contributions are not only dependent within a loop but also between each steam generator loop.

Based on the ASME Steam Tables and the F_a dependence on T_f, a feedwater temperature uncertainty of +9.9 $^{\circ}F$ results in enthalpy, density and F_a uncertainties equivalent to the following steam generator thermal power uncertainties:

$h_f\,(T_f)$	- 1.4%
$\rho_f\,(T_f)$	- .415%
$F_a\,(T_f)$	+ .02%

Since these three error contributions are all dependent on T_f, the overall uncertainty is determined by addition to give a -1.795% uncertainty in steam generator thermal output. The signs of the above uncertainties indicate the bias in steam generator thermal output. For a T_f error biased in the negative direction, (- 9.9 $^{\circ}F$), the signs are reversed. Since the T_f measurement is independent of the P_s and ΔP measurements within a steam generator loop, the overall T_f uncertainty within a loop can be expressed as $\mp$ 1.795% for a $\pm$ 9.9 $^{\circ}F$ uncertainty in feedwater temperature. These results are summarized on Table 1.

The overall uncertainty in core thermal power is obtained by first converting the individual loop error contributions to a percent of core thermal power and then combining the individual error contributions from each loop in an appropriate statistical manner.

Since the independent error contributions are not only independent within each steam generator loop but also between loops, the core power uncertainty attributed to the independent errors can be calculated as follows:

$$\left[\left(\frac{.795}{2}\right)^2 + \left(\frac{.795}{2}\right)^2\right]^{1/2} = \pm\ .562\%\ \text{core thermal power}$$

Since the P_s error contributions are independent between each steam generator loop and the P_s dependence within each loop has already been taken into account, the core power uncertainty attributed to the P_s errors can be calculated as follows:

$$\left[\left(\frac{.0701}{2}\right)^2 + \left(\frac{.0701}{2}\right)^2\right]^{1/2} = \pm\ .0496\%\ \text{core thermal power}$$

Since the T_f error contributions are dependent between steam generator loops, the core power uncertainty attributed to the T_f errors is determined by direct addition as follows:

$$\left[\frac{1.795}{2} + \frac{1.795}{2}\right] = \pm\ 1.795\%\ \text{core thermal power}$$

Since there are two steam generators, the factor of two in the denominator of each of the individual error contributions is used to convert the errors from percent of steam generator thermal output to percent of core thermal power.

Since the independent errors, the P_s errors and the T_f errors, are all independent error contributions, these error contributions can be statistically combined using the RMS method to give an overall core power uncertainty of $\pm$ 1.88%. The $\pm$ 1.88% uncertainty is the error in core thermal power expressed as a percent of the nominal measured core thermal power of 2700 MW. These calculations are also summarized on Table 1.

Conclusion

The response to Question 1 of the Reference provided the results of a calculation which indicated a core power uncertainty of $\pm$ 1.18 percent. This result was obtained by assuming all error contributions were statistically independent. This response recalculated the overall core power uncertainty to be $\pm$ 1.88% by taking into account the dependent error contributions resulting from feedwater temperature and steam generator pressure measurement uncertainties. This change is due primarily to the large sensitivity in the core power uncertainty due to uncertainties in feedwater temperature. The results of this calculation verifies that the $\pm$ 2% core power uncertainty assumed in the Millstone Point Unit 2 Reload analyses is conservative.

NNECO is currently evaluating the feedwater temperature measurement system. The intent of this evaluation progam is to identify possible areas of improvement in order to provide a more accurate indication of feedwater temperature and thus a smaller core power uncertainty.

REFERENCE - W. G. Counsil to R. A. Clark, "Millstone Unit 2 Measurement Uncertainties," March 4, 1982.

TABLE 1

Errors in Steam Generator Thermal Output and Core Thermal Power

Errors in Steam Generator Thermal Output

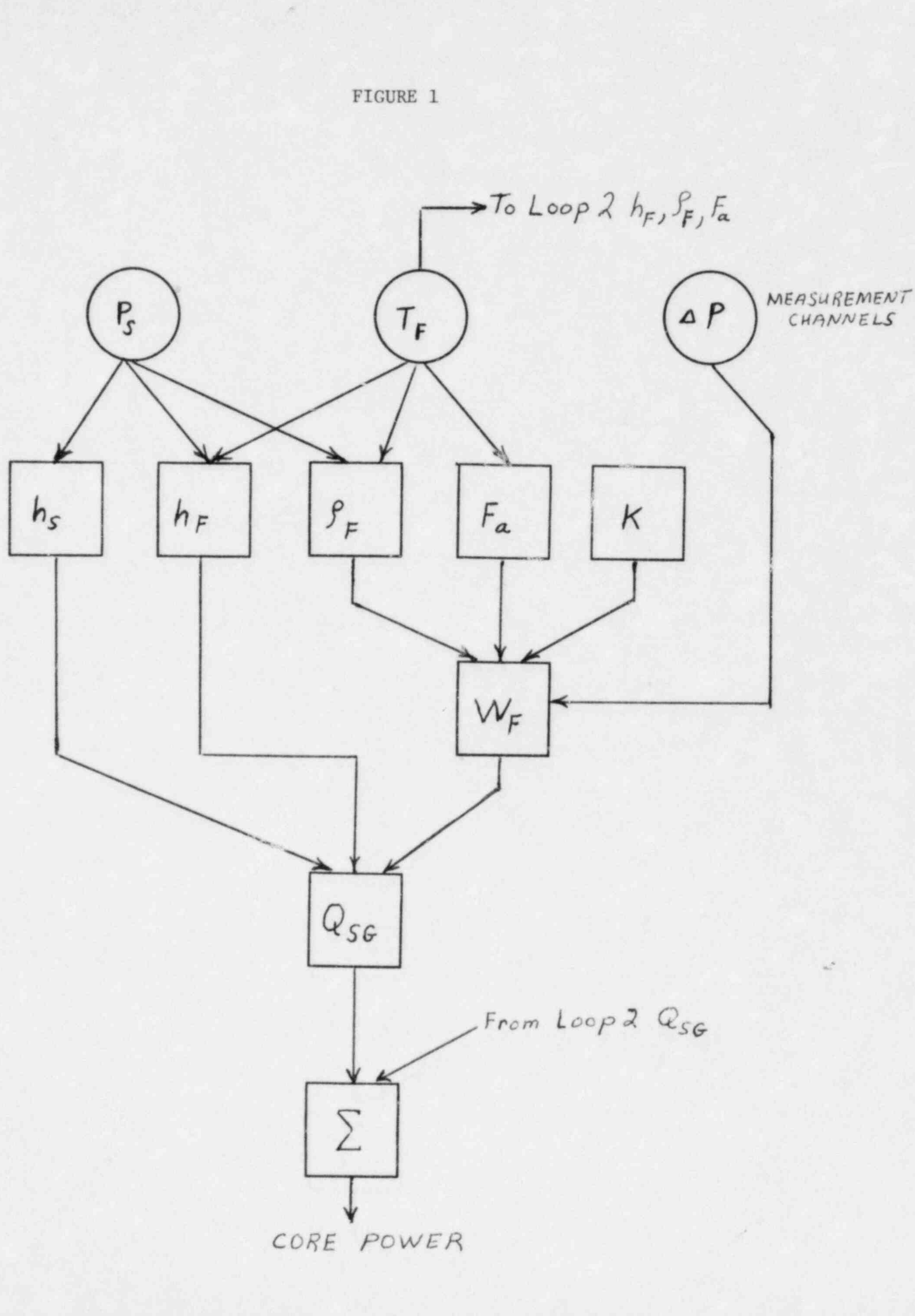
Error Component	Error (% SG Thermal Output)
1. Independent Errors	
-Due to venturi calibration coefficient (K)	$\pm$ .37%
-Due to venturi area expansion factor (F_a) (linear thermal expansion coefficient uncertainty)	$\pm$ .034%
-Due to Δ P measurement	$\pm$.703%
Subtotal of Independent Errors (RMS)	$\pm$ .795%
2. Errors due to steam Pressure (P_s)	
-Error in steam enthalpy (h_s)	-.075%
-Error in feedwater enthalpy (h_f)	-.0017%
-Error in feedwater density (ρ_f)	+.0066%
Total of P_s Errors	$\pm$ .0701%
3. Errors Due to Feedwater Temperature (T_f)	
-Error in feedwater enthalpy (h_f)	-1.4%
-Error in feedwater density (ρ_f)	- .415%
-Error in F_a	+ .02%
Total of T_f Errors	$\pm$1.795%

$$\text{\% Core Thermal Power Uncertainty} = \left[2\left(\frac{.795}{2}\right)^2 + 2\left(\frac{.0701}{2}\right)^2 + \left(\frac{1.795}{2} + \frac{1.795}{2}\right)^2\right]^{1/2}$$

$$= \pm\ 1.88\%$$

The $\pm$ 1.88% uncertainty is the error in core thermal power expressed as a percent of the nominal measured core thermal power of 2700 MW.

FIGURE 1


To Loop 2 h_F, ρ_F, F_a
P_S
T_F
ΔP
MEASUREMENT CHANNELS
h_S
h_F
ρ_F
F_a
K
W_F
Q_{SG}
From Loop 2 Q_{SG}
Σ
CORE POWER

QUESTION 2

The reactor coolant flow calculation uses the core thermal power. What is the impact of the correct use of independent variables in determining core power uncertainty on the reactor coolant flow uncertainty?

RESPONSE

The response to Question 1 of the Reference provided the results of an analysis of the uncertainties associated with determining reactor coolant flowrate based on a plant calorimetric. As discussed in that response, the uncertainties in the determination of reactor coolant flowrate are based on the uncertainties associated with core thermal power, average hot leg enthalpy, and average cold leg enthalpy. The hot and cold leg enthalpies are based on the measurements of hot leg temperature (THOT), cold leg temperature (TCOLD), and pressurizer pressure, (P_p). Both the hot and cold leg enthalpies are based on the same pressurizer pressure measurement (P_p), therefore the pressure contribution of the enthalpy uncertainties will be dependent error contributions. The response to Question 1 of the Reference assumed that all errors associated with the core thermal power calculation as well as the enthalpy uncertainties associated with pressure to be independent error contributions. This response will recalculate the reactor coolant flow uncertainty to determine the impact of the use of dependent and independent error contributions.

The overall uncertainty in core thermal power was determined in Question 1 to be ± 1.88 percent of the nominal measured core power of 2700 MW. This result is based on the use of dependent and independent error contributions. Since reactor coolant flowrate determination is directly proportional to the measured core thermal power, the core thermal power uncertainty is expressed as ± 1.88 percent of nominal flow.

The THOT and TCOLD measurement uncertainties were previously calculated in the response to Question 1 of the Reference. Based on the ASME Steam Tables and plant conditions at 100% power, the temperature measurement uncertainties result in enthalpy uncertainties equivalent to the following reactor coolant flow uncertainties:

h_h (THOT)	± 1.44% nominal flow
h_c (TCOLD)	± 1.27% nominal flow

As discussed in the response to Question 1 of the Reference, an additional hot leg enthalpy uncertainty attributed to hot leg temperature gradient effects resulted in an equivalent reactor coolant flow uncertainty of ± 1.14 percent of nominal flow.

Based on the ASME Steam Tables and plant conditions at 100% power, the pressurizer pressure uncertainty results in enthalpy uncertainties equivalent to the following reactor coolant flow uncertainties:

$h_h(P_p)$ + .12% nominal flow
$h_c(P_p)$ - .04% nominal flow

The signs of the above uncertainties indicate the reactor coolant flow bias for a pressure measurement uncertainty biased in the positive direction. For pressure measurement uncertainties biased in the negative direction, the signs are reversed. Since these two enthalpy uncertainties are dependent on the same pressurizer pressure measurement, the overall uncertainty is found by addition to give an equivalent reactor coolant flow uncertainty of ± .08% nominal flow.

It can be seen that the pressure uncertainties are negligible when compared with the temperature and core power uncertainty contributions.

Since the core power, temperature, and pressure uncertainties are independent of each other, these error contributions can be combined statistically using the RMS method to yield an overall reactor coolant flow uncertainty of ± 2.92% of nominal measured flow. (See Table 1)

It is preferable to express the reactor coolant flow uncertainty as a percent of the design volumetric flowrate of 324800 GPM. Typical flow calculations performed at 100% power indicate a nominal measured flow of 123.6 percent of design; therefore the reactor coolant flow uncertainty can be expressed as follows:

Reactor Coolant Flow Uncertainty = 1.236 x 2.92%
= ± 3.61% of design flow

CONCLUSIONS

The response to Question 1 of the Reference provided the results of an analysis which determined the reactor coolant flow uncertainty to be ± 3.13% of design flow. This analysis was based on the assumption that all error contributions were independent.

The present analysis recalculated the reactor coolant flow uncertainty to determine the impact of the correct use of dependent and independent variables. The results indicate that the reactor coolant flow uncertainty increased to a value of ± 3.61% of design flow.

This analysis verifies that the ± 4% reactor coolant flow uncertainty utilized in the Millstone Point Unit 2 Reload Analyses is conservative.

REFERENCE - W. G. Counsil to R. A. Clark, "Millstone Unit 2 Measurement Uncertainties," March 4, 1982.

TABLE 1

Errors in Reactor Coolant Flow Determination

Error Component	Error (% nominal flow)
1. Core Thermal Power Uncertainty ($\pm$ 1.88%)	$\pm$ 1.88%
2. Error Due to Average THOT Error in hot leg enthalpy (h_h)	$\pm$ 1.44%
3. Error Due to Temperature Gradient Effect Error in hot leg enthalpy (h_h)	$\pm$ 1.14%
4. Error Due to Average TCOLD Error in cold leg enthalpy (h_c)	$\pm$ 1.27%
5. Error Due to Pressurizer Pressure (P_p)	
Error in hot leg enthalpy (h_h)	+ .12%
Error in cold leg enthalpy (h_c)	- .04%
Total of P_p errors	$\pm$ .08%
TOTAL ERROR (RMS)	$\pm$ 2.92%

Typical measured flow is 123.6% of the design flow of 324800 GPM; therefore:

% Reactor Coolant Flow Uncertainty = 1.236 x 2.92%
= $\pm$ 3.61% of design flow

QUESTION 3

The feedwater flow element fouling causes the calculated reactor power to be somewhat low at all times except immediately subsequent to an element cleaning. The use of this calculated power is conservative. However, the core flow determination also uses the calculated power. What is the effect of accounting for the fact that the calculated flow is low at all times except just after an element cleaning? Also, explain how the feedwater flow element is cleaned and how effective the cleaning method is.

RESPONSE

Feedwater flow element fouling will cause the calculated or indicated reactor power to be higher than the actual power. Thus, an indicated power of 100% may exist when the actual power is slightly less than 100%. The flow element fouling is normally within the range of a one (1) percent bias in the calculated feedwater flow and not expected to exceed two (2) percent. The use of this indicated power is conservative.

The reactor coolant system (RCS) flow determination also uses the calculated power as an input and thus will be biased in the same direction by an equal amount. That is, the actual flow may be slightly lower than the indicated flow. The effect of accounting for the slightly lower actual flow would be to bias the flow uncertainty in the negative direction. There are two reasons from a safety viewpoint why a flow bias is not required.

First, as noted above, any changes in calculated reactor power are matched by equal changes in calculated flow. The actual power and flow may be slightly lower than the indicated values. For all safety analysis applications, the credit due to the lower power offsets the penalty associated with a lower flow. In fact, in the DNB area, the credit from a reduction in power of one (1) percent is approximately double the penalty associated with a reduction in primary flow of one (1) percent (see reference). Secondly, the low flow trip setpoint (LFTS) is calibrated to an absolute flow at the beginning of each fuel cycle. This calibration is based upon the measured RCS flow rate as determined during power ascension physics testing at full power. This measured flow is not biased since the feedwater flow element is cleaned prior to each cycle startup. If the measured flow does increase due to feedwater flow element fouling, the LFTS will not be adjusted to take credit for this change in flow and thus the LFTS will not be biased in the non-conservative direction. That is, all inputs to the calculation of primary flow are reviewed for accuracy and consistency before the LFTS is changed.

The feedwater flow element is cleaned using a device called a hydro laser which is widely used for cleaning the interior surfaces of piping. The hydro laser consists of a multiple jet spray head attached to a long length of flexible hose. Water under very high pressure (approximately 3000 psi) is supplied to the spray head which creates high velocity water jets which are directed radially out from the spray head onto the pipe surfaces. The device is fed into the feedwater piping through an access point in close proximity to the feedwater elements. The position of the device is carefully controlled to assure that the feedwater flow element and adjacent piping are cleaned. Multiple passes are also made to assure that all the deposits are removed. The feedwater elements have been visually inspected before and after cleaning on several occasions utilizing a boroscope. Inspection results indicate that the hydro laser cleaning procedure is very effective. Virtually all of the deposits present are removed. The comparison of plant performance data taken before and after feedwater element cleaning also confirms the effectiveness of the cleaning.

CONCLUSION

In summary, small differences between indicated and actual reactor coolant system flow rates need not be accounted for because they are offset by the conservative value calculated for reactor power. The measured flow used to calibrate the LFTS is determined at the beginning of the fuel cycle when the feedwater flow element is clean. The LFTS is not recalculated due to changes in indicated RCS flow which result from feedwater flow element fouling. Effective methods do exist and are employed at Millstone 2 to clean the feedwater flow elements.

Reference: W. G. Counsil to R. A. Clark, "Reload Safety Evaluation, Millstone 2 Cycle 4", June 3, 1980.

QUESTION 5

In general, uncertainties in computer A-D conversions, resistor values and calibration uncertainties are given with no substantiation. Presumably these are derived from calibration procedures and/or design specifications. What are the Quality Control and Quality Assurance procedures used to confirm the given values?

RESPONSE

Computer A-D conversion uncertainties are derived from calibration procedures at the plant. The I & C calibration procedures (primarily the IC-2400 and SP-2400 series procedures at Millstone 2) are used to verify the computer indication for those instruments providing an input to the computer. Any instrument loop found outside of specifications are referred to the Millstone Computer Operations Group via a maintenance request (MR). Upon receiving an MR indicating that an instrument loop is out of specification, the Computer Operations Group will use procedure COP-2102, "Process Computer Analog Input Calibration", to calibrate computer analog inputs to within 0.2% of full range of the input signal. This value is consistent with that used in the uncertainty analyses.

The error allowance associated with the resistor values is obtained from the specifications defining the uncertainty in the 100 ohm precision resistors used in the measurement channels to convert current signals to voltage signals. In addition, the calibration procedures used at Millstone Point Unit 2 provide for a direct check on the resistor values by comparing the calibrated transmitter current output with the computer indicated voltage signal. This comparison is performed to ensure that no faulty resistance values exist in the measurement channel loops.

The calibration uncertainties assumed in the Reference uncertainty analysis are based on criteria specified in plant calibration procedures. Each measurement channel has a specific calibration procedure which defines the method, range and frequency of channel calibration. In addition, these procedures provide a criterion for the "as left" calibration allowance. The calibration uncertainties utilized in the Reference analyses are based on these criteria. The calibration data provided in the response to Question 1 of the Reference verifies that the actual "as left" calibration accuracy is well within the criteria specified in the calibration procedures. An example of a typical calibration procedure is Procedure No. SP2402C, "Steam Generator Pressure Calibration", which specifies the criteria for calibrating the eight steam generator safety grade pressure channels.

References: W. G. Counsil to R. A. Clark, "Millstone Unit 2 Measurement Uncertainties", March 4, 1982.