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A Duke Energy Company

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February 26, 2003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station, Unit 2
Docket Number 50-414
Response to Request for Additional Information for Unit 2 Reactor Coolant System Cold Leg Elbow Tap Flow Coefficients

References: Catawba Response to Request for Additional Information dated February 7, 2003.

Catawba Proposed License Amendment for Unit 2 Reactor Coolant System Cold Leg Elbow Tap Flow Coefficients dated October 10, 2002.

February 7, 2003 Duke Energy submitted a response to the NRC request for additional information. This response noted that an additional Duke response would be forthcoming with the results of the instrumentation maintenance history review of the elbow tap transmitters. The results of this review are provided in Attachment 1.

During a phone call on February 20, 2003, additional questions were raised by the staff. The responses to those questions are also included in Attachment 1.

This letter contains the following commitment:

> The elbow tap coefficients will be documented in the revision to the Updated Final Safety Analysis Report (UFSAR) or the Core Operating Limits Report. UFSAR revisions are scheduled for 6 months following the completion of Unit 2 refueling outages.

This letter and attachments do not contain any additional commitments.

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The previous conclusions of the No Significant Hazards Consideration as stated in the October 10, 2002 are not affected by this response.

Pursuant to 10 CFR 50.91, a copy of this proposed amendment request is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to G.K. Strickland at (803) 831-3585.

Very truly yours,

Gary R. Peterson

GKS/s
Attachment

Gary R. Peterson, being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this amendment to the Catawba Nuclear Station Facility Operating License Number NPF-52; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

G. R. Peterson

Gary R. Peterson, Site Vice President

Subscribed and sworn to me: 2-26-03
Date

Micky Standridge
Notary Public

My commission expires: 7-10-2012
Date

SEAL

xc (with attachments):

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Attachment 1

Responses to NRC Questions Concerning
Catawba Unit 2 RCS Flow and Flow Instrumentation

On February 7, 2003 Duke Energy submitted a response to the NRC request for additional information concerning Catawba Unit 2 RCS flow and flow instrumentation. The response noted that an additional Duke response would be forthcoming with the results of the instrumentation maintenance history review of the elbow tap flow transmitters. This review is provided below as the response to Question 1. During a phone call on February 20, 2003, additional questions were raised by the staff. Those questions (paraphrased) and the responses to those questions are also provided below.

Question 1:

The Catawba Unit 2 total flowrate (Refer to Figure 1 of the 2/7/2003 submittal) as indicated by the elbow tap instrumentation indicates a decrease in flow in the October 1991 and the February 1993 data, followed by an increase in flow. Explain this data trend relative to the flow prediction of the analytical model, which shows no significant flow change.

Response:

A review of the plant elbow tap flow instrumentation calibration and maintenance history has produced a possible cause for the decrease in indicated RCS flow in the February 1993 time frame. The review did not identify any reason for the flow decrease in 1991. This information is based on a review of vendor data and instrument calibration records. Some of the insights normally available from personnel interviews and equipment testing records are no longer possible due to the passage of time.

Between 1986 and 1993 the installed elbow tap transmitters were manufactured by Tobar and Veritrak. Between 1986 and 1992 the elbow tap and these transmitters were calibrated in accordance with the vendor recommendations. In 1994, the transmitters were replaced with Rosemout transmitters and these transmitters were calibrated in accordance with the vendor recommendations.

In 1993, based on industry experiences, Catawba determined that the transmitter calibration procedure should be revised to include the effect of the system static pressure. A review of vendor documents did not identify any Tobar or Veritrak recommendations for rescaling based on static system pressure. Therefore, the engineering staff concluded that the rescaling would be empirically determined by testing each of the 12 transmitters. The details of the individual transmitter tests are no longer available; however, the tests and subsequent calibration revision resulted in a change in 100% ΔP scaling for the transmitters. A corresponding scaling change for the Operator Aid Computer (OAC) should have been made, but was not. The difference in the scaling between the transmitters and the OAC resulted in an indication of flow which was lower than the flow would have been if the scales had matched. This correction is estimated to result in an approximate 0.74% change in total RCS flow for the 1993 calibration.

The following table shows the transmitter and OAC scaling prior to 1993 (first column) and in 1994 (third column) below. In 1993, the transmitters were rescaled to the values in the second column, but the OAC scaling remained as shown in the first column. The 1994 column represents the new Rosemount transmitter scaling values.

	Pre-1993 100% Flow ΔP (INWC)	1993 100% Flow ΔP (INWC)	1994 100% Flow ΔP (INWC)
Loop A Ch 1	313.9	316.91	310.2
Loop A Ch 2	338.2	341.75	336.6
Loop A Ch 3	319.8	319.58	315.3
Loop B Ch 1	327.0	327.0	324.4
Loop B Ch 2	367.3	376.0	364.1
Loop B Ch 3	321.8	333.33	324.4
Loop C Ch 1	309.4	335.5	308.2
Loop C Ch 2	350.0	345.0	349.9
Loop C Ch 3	335.5	340.0	335.6
Loop D Ch 1	339.0	342.0	335.6
Loop D Ch 2	340.4	341.5	340.7
Loop D Ch 3	336.7	338.67	335.6

Recalculating the 1993 ΔP data using the 1993 transmitter scaling values results in the following changes to the measured ΔP values:

	Original ΔP Readings With Transmitters / OAC Mismatch (INWC)	Recalculated ΔP Readings Using Correct 1993 Scaling (INWC)
Loop A Ch 1	300.8	303.7
Loop A Ch 2	326.6	330.0
Loop A Ch 3	309.3	309.1
Loop B Ch 1	314.7	314.7
Loop B Ch 2	352.7	361.1
Loop B Ch 3	308.7	319.8
Loop C Ch 1	301.8	327.3
Loop C Ch 2	342.0	337.1
Loop C Ch 3	324.8	329.2
Loop D Ch 1	327.4	330.3
Loop D Ch 2	329.0	330.1
Loop D Ch 3	324.0	325.9

The recalculated 1993 data for 10 of the 12 transmitters is consistent with expected flow results compared to the flow model predictions, a comparison of the 1991 and 1993 data, and by a comparison of the 1993 and 1994 data.

For two of the transmitter 1993 scaling changes, a large and unexplained increase existed for Loop C Channel 1 and Loop B Channel 3. The scaling for these two transmitters increased by

8.4% and 3.6%, whereas the average change for the 12 transmitters was 1.5%. If the two suspect channels are excluded from the average, then the average change of the remaining 10 channels was 0.6%. No reasons could be conclusively identified for the changes in these two transmitter values, but the methodology used to empirically determine the static pressure effect could be a possible source of error. When the transmitters were replaced in 1994, the new transmitter scaling values essentially returned to the 1991 values. The 1994 transmitters were calibrated in accordance with the vendor recommendations which included compensation for static pressure effects. The empirically-determined system static pressure values were unique to the 1993 data only.

Updated graphs of total flow and loop flows using the recalculated ΔP readings are attached as Figures 1 through 5.

Other potential factors which might have an effect on the indicated flow were also examined for their possible contributions to RCS flow changes. Among these were human factors issues (change in technician performing calibrations), replacement of reactor coolant pumps and motors, reactor coolant pump speed and power, and other changes to the RCS during this time period. None of these were found to be significant contributors to the apparent decrease in flow seen in 1991 and 1993.

Duke has found no correlation between the technicians who performed the calibrations in 1991 through 1994 that could explain the 1991 and 1993 data.

With the exception of accounting for the static pressure effects, the calibration process appears to have stayed the same from initial startup until the technical specification amendment in 1995 which changed the elbow tap flow coefficients. A scaling calculation was originated at the time that the new calibration method was implemented.

There is no indication of a generic instrument drift problem for the RCS flow loops over the history of plant operation.

Searches of calibration and maintenance history records and Duke's corrective action process records did not reveal any noted changes or anomalies to RCS flow instrumentation with the exception of replacement of all Unit 2 RCS flow transmitters with Rosemount transmitters in June, 1994. When the transmitters were replaced, the OAC scaling was changed to agree with the new transmitter scaling.

Based on the above research, Duke believes that the decrease in the RCS flow shown in the 1993 data is directly attributable to the discrepancy between the transmitter scaling and the OAC scaling. The methodology used to empirically determine the static pressure effects may have been an additional contributor for the flow decrease but cannot be verified. The anomalous results for two of the twelve transmitters cannot be explained conclusively. None of the other potential causes listed above was found to explain or contribute significantly to the unexpected flow decrease seen in 1991 and April 1993.

Since 1993, additional processes have been implemented to prevent the inadvertent mismatch in the transmitter scaling and OAC scaling. Any change in a transmitter scaling is now controlled by the modification process and with checklists to direct the review of the OAC computer scaling. The modification process Technical Issues Checklist includes the following questions:

- Does the change affect software, firmware, or data?

- Does the change affect any input, output, calculation, constant application, alarm limit or response, database information, graphic display, operating system, or hardware component on the Operator Aid Computer (OAC) or long-term archive?
- Does the change affect any system (e.g. radiation monitoring system, R.G 1.47 bypass monitoring system) that interfaces to the Operator Aid Computer, or long-term archive, or does the change affect the data that an interfacing system transmits to or receives from the OAC?

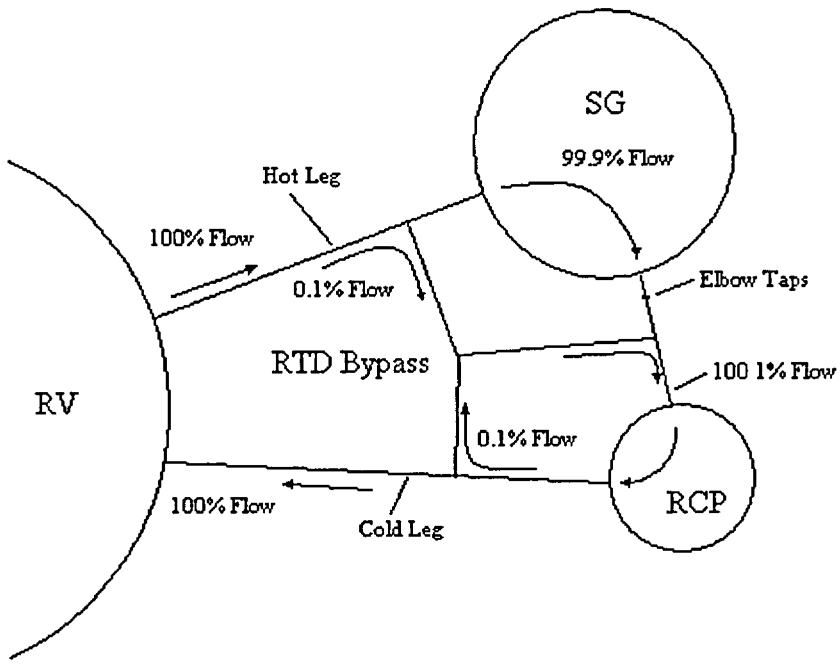
Additionally, in 1995 when the license amendment was approved to use the elbow tap coefficients for measuring RCS flow, a formal scaling calculation was developed, reviewed, and approved. Changes to the transmitter scaling will be evaluated and documented within the calculation

Question 2:

Provide a numerical example of the modeling of the RTD bypass manifold in the analytical flow model.

Response:

The analytical flow model is designed to produce a total RCS flow prediction based on the total system pressure drop. With the RTD bypass manifold in place several components of the system pressure drop needed to be revised to account for the flows redirected to the bypass manifold. The drawing below displays the relevant areas of one of the RCS loops and the associated flows:



As can be seen from the above drawing the diversion of 111 gpm (≈ 0.11% flow) to the RTD bypass from the hot leg will result in a decreased flow of 99.9% flow through the steam generator and elbow tap portion of the crossover leg. Therefore, the elbow tap measured loop flow will be low by the amount of the diverted portion of flow. In addition to the hot leg diversion of flow, a small amount of flow, 172 gpm (≈ 0.18% flow), is also diverted from the cold leg to the RTD bypass manifold. The combined diverted hot and cold leg flows are then directed back to the RCS loop in the crossover leg downstream from the elbow taps. This results in an increase in loop flow (100.18% flow) through the reactor coolant pump to the cold leg diversion point. The loop flow once again returns to 100% flow in the cold leg as the flow returns to the reactor vessel.

In order to model the RTD bypass manifold in the analytical flow model the flow used to adjust the pressure drops through the components affected are increased or decreased as appropriate to produce the correct pressure drops. In this case, the flows through half the hot leg, the steam generator, and half the crossover leg are decreased by 111 gpm to reflect the diverted RTD bypass flow from the hot leg. This results in the 100% flow in the hot leg being reduced to approximately 99.9% flow through the steam generators. Likewise, the flows through the reactor coolant pump suction, weir plate and half the cold leg are increased by the returned RTD bypass flows of 111 and 172 gpm. This results in the 99.9% flow from the crossover leg to the pump suction being increased to approximately 100.18% flow through the reactor coolant pump to the cold leg. Once the flow passes the cold leg diversion to the RTD bypass manifold then the flow is decreased by 172 gpm to reduce the flow back to 100% flow. For example, when the changes described above are made to the analytical model for the August 1986 data. The following ΔP changes result:

	With RTD Bypass Manifold	Without RTD Bypass Manifold	ΔP Difference
Hot leg =	2.1203	2 1220	0.0017
Steam generator =	33.2889	33.3518	0.0629
Elbow tap 90° elbow =	1.2971	1.2996	0.0024
Pump suction =	2.8229	2 8124	-0.0106
Weir plate =	1.9896	1.9821	-0 0074
Cold leg =	2.5841	2 5788	-0.0053
Inlet nozzle =	10.1664	10.1631	-0.0033
Downcomer =	0.3678	0.3677	-0.0001
Downcomer exit =	2.7383	2.7374	-0.0009
Core support =	2.2478	2.2471	-0.0007
Lower core plate =	7.0493	7.0470	-0.0023
Bottom nozzle =	2.7763	2.7754	-0.0009
Core =	15.0095	15.0046	-0.0049
Top nozzle =	0.9544	0.9540	-0.0003
Upper core plate =	3.8849	3.8837	-0.0013
Outlet nozzle =	2.3664	2.3656	-0.0008
Thermal driving head =	-1.3000	-1.3000	0.0000
Total Pressure Drop =	**90.3640**	**90.3923**	**0.0283**
RCS Flow =	**401,818**	**401,752**	**-66.0**

The table above shows that removing the RTD bypass manifold will result in a slight increase in the total loop ΔP, mainly due to the increased flow through the steam generators, and produce a small decrease in flow (≈ 66 gpm). This represents the change only for the removal of the RTD bypass manifold and does not represent other changes such as steam generator tube plugging. Also, this does not represent the change in flow which will be reflected by the elbow taps. The elbow taps would be expected to show an increase in flow of approximately 378 gpm (4*111 gpm – 66 gpm).

Duke believes this modeling to be an accurate accounting for the effect of the removal of the RTD bypass manifold on the total RCS flow based on the total RCS pressure drop. It is realized the RCS flow as indicated by the elbow tap coefficients will realize a flow increase following the removal of the RTD bypass manifold. A comparison of the elbow tap flow to the analytical model can be made as described above where the expected increase in RCS flow, as indicated by the elbow taps, is expected to be approximately 378 gpm. This effect was previously not included in the analytical flow model.

Question 3:

Confirm that the fuel assembly ΔP values used in the analytical model have been normalized to the correct flowrate.

Response:

Fuel assembly flow testing is performed by the fuel vendors to determine the form loss coefficient values and the pressure drop along the fuel assembly. These tests are usually performed at conditions significantly different from the expected plant full power conditions. Following testing the results are analytically scaled to provide the ΔP values at conditions corresponding to full power operating conditions. Duke obtains these data from the vendors and they are used in the analytical model. The Westinghouse OFA fuel assembly ΔP is 21.6 psi at a reference flow of 406,800 gpm, and these values are used in the analytical model. The Mk-BW fuel assembly ΔP is 2.4% lower than the OFA fuel assembly at a reference flow of 403,600 gpm, but was incorrectly modeled at a reference flow of 406,800 gpm in the analytical model. Using form loss coefficients provided by Westinghouse, the Westinghouse RFA fuel assembly ΔP is 25.1 psi at a reference flow of 390,000 gpm, but was incorrectly modeled at a reference flow of 406,800 gpm. These changes in addition to the changes described in the response to Question 4 result in a decrease in the predicted March 2003 analytical model flow of 829 gpm or 0.2%. Figures 1-5 have been updated to reflect these changes in the analytical model.

Question 4:

The analytical model assumed that the reactor vessel internal flow distribution was not affected by a change in fuel assembly ΔP. Determine the change in loop flow when this effect is taken into account.

Response:

The analytical flow model has assumed a constant reactor vessel internal flow distribution based on calculations performed by FANP for Mk-BW fuel at the time that the analytical model was developed. The value used for Catawba Unit 2 is a total core bypass flowrate of 5.7%. This data was not changed in the analytical model for Westinghouse fuel. Duke has now evaluated the change in the RCS loop flow using 7.02% core bypass flow based on Westinghouse calculations for Westinghouse RFA fuel in Catawba Unit 2. To make this comparison consistent, a spreadsheet was developed to approximate the change from the 7.02% value for Westinghouse fuel by substituting the 25.1 psi core pressure drop for RFA fuel at a reference flowrate of 390,000 gpm to 20.75 psi for FANP Mk-BW fuel at a reference flowrate of 403,600 gpm. The core bypass flow then decreases to 6.17% for FANP Mk-BW fuel. A similar change for Westinghouse OFA fuel based on 21.6 psi at a reference flow of 406,800 gpm resulted in a core bypass flow value of 6.24%. These changes in the core bypass flow as the fuel design transitions from OFA to Mk-BW to RFA are then input to the analytical model. These changes in addition to the changes described in the response to Question 3 result in a decrease in the predicted March 2003 analytical model flow of 829 gpm or 0.2%. Figures 1-5 have been updated to reflect these changes in the analytical model.

Question 5:

Confirm that the fuel assembly ΔP data used in the analytical model accounts for the effect of subcooled boiling at full power conditions.

Response:

Duke analyzes core reload designs to quantify the extent of subcooled boiling in the McGuire and Catawba cores due to the industry issue known as axial offset anomaly (AOA). The core designs are constrained by the AOA effect in that the core radial peaking factors must be less than would risk the occurrence of AOA. For that reason the McGuire and Catawba cores do not operate with any bulk boiling in the fuel assembly subchannels. Therefore, the Catawba Unit 2 plant flow data is not affected by any two-phase pressure drop in the core, since there is no bulk boiling. A VIPRE analysis has been performed to show the effect of increasing the radial peaking from 1.2 to 1.8 in eight fuel assemblies to obtain a fuel assembly exit void fraction of 2% in those assemblies. With this unrealistic increase in the power distribution the core pressure drop increased by only 0.03 psi. Such a change in core pressure drop will cause no measurable change in loop flow. The fuel assembly pressure drop data used in the analytical model does not include two-phase effects since there is no bulk boiling in the fuel assembly subchannels at nominal full power conditions. This is consistent with plant operation.

Question 6:

Provide plant data showing the change in RCS flow with power level.

Response:

See Figure 6 from the startup of Catawba Unit 2 Cycle 12. Flow decreases by approximately 1% as power increases from zero to full power. This effect is caused by the enthalpy rise across the core increasing the volumetric flowrate in the high temperature part of the primary loop. The increase in volumetric flow in half of the loop causes an increase in loop ΔP and a corresponding decrease in flow as the reactor coolant pumps shift to a higher head / lower flow point on the pump head/flow curve.

Figure 1
Analytic Model vs. Elbow Tap Flow Comparison, RCS Total Flow

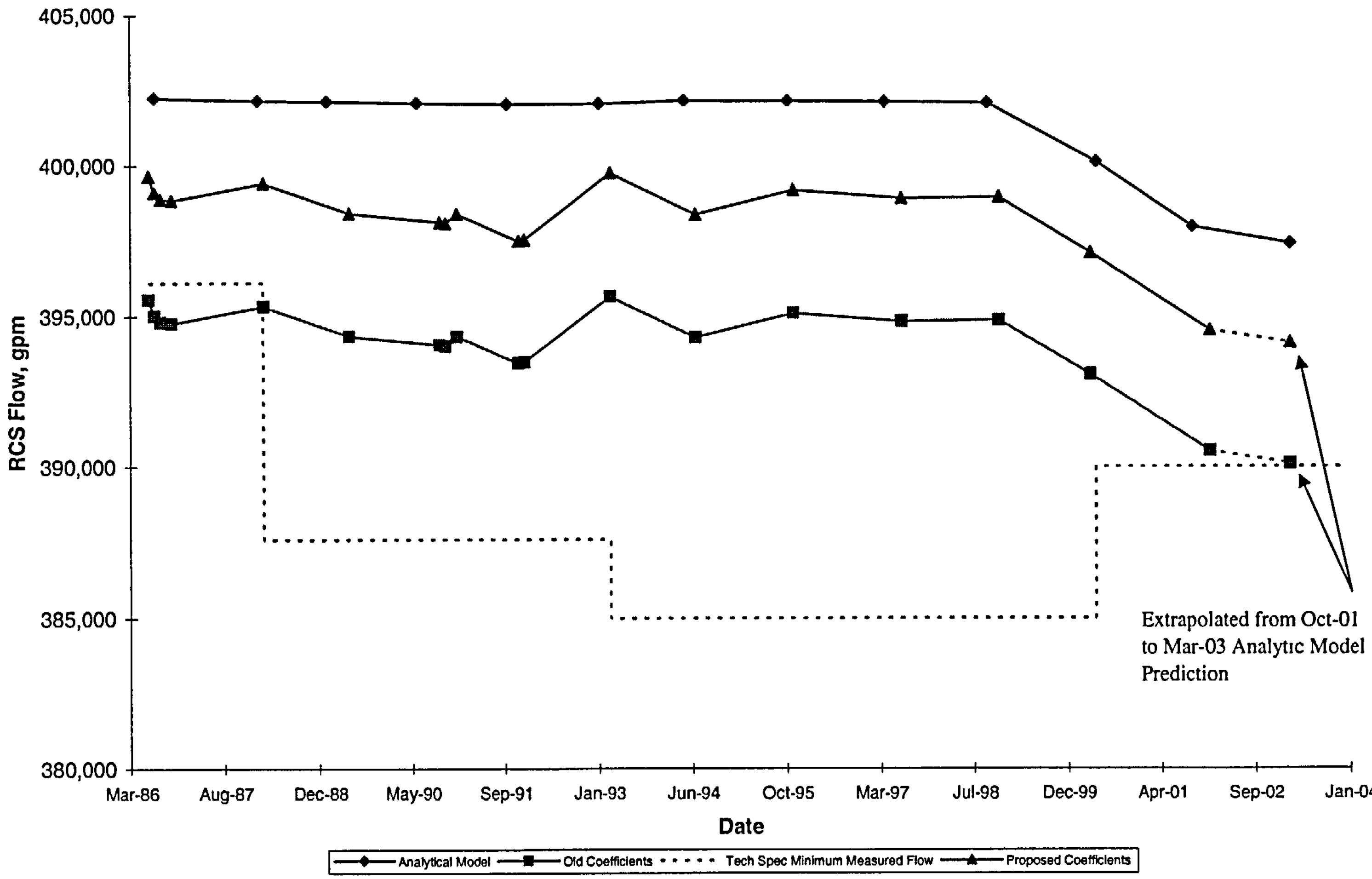


Figure 2
Analytic Model vs. Elbow Tap Flow Comparison, Loop A

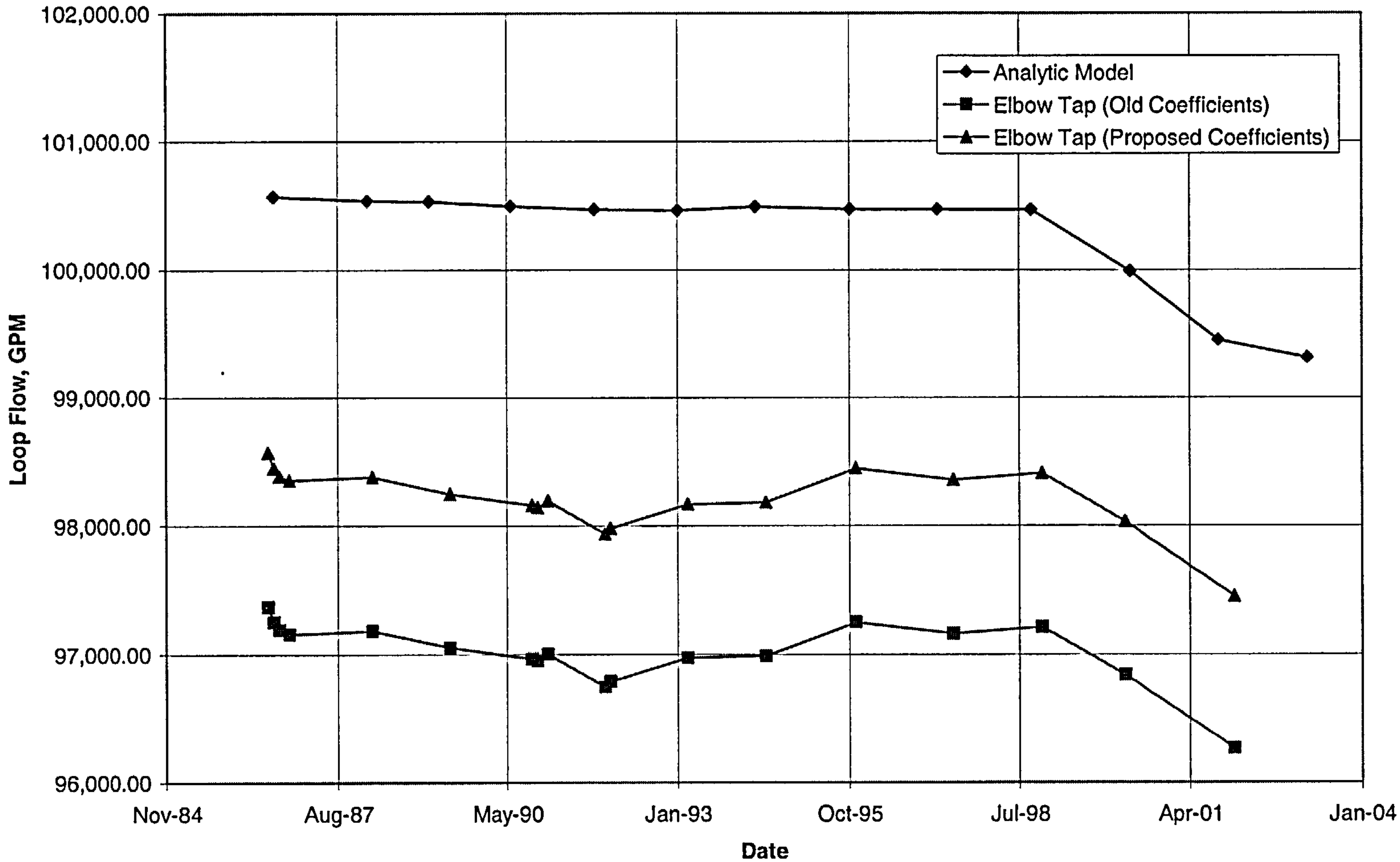


Figure 3
Analytic Model vs. Elbow Tap Flow Comparison, Loop B

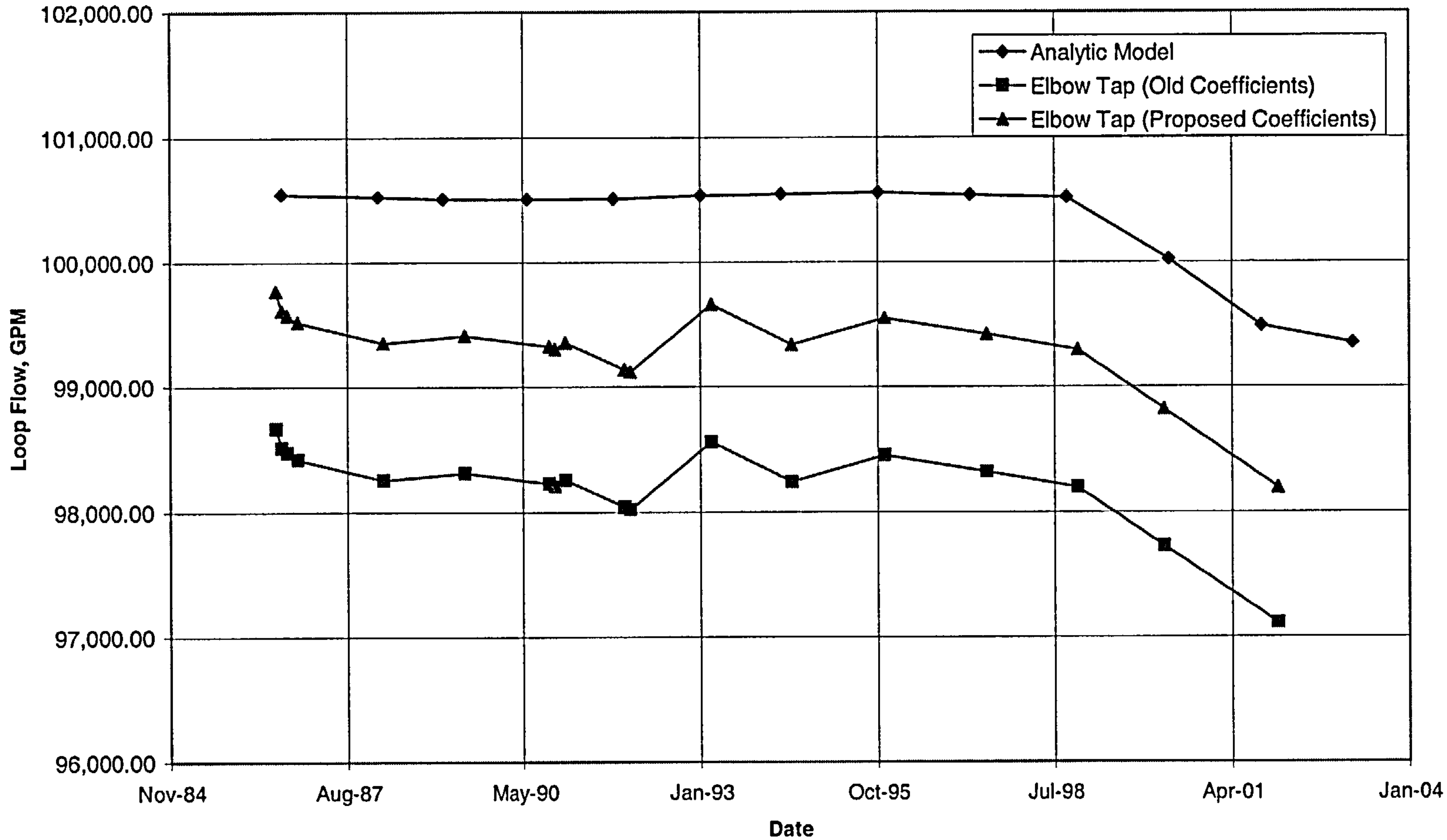


Figure 4
Analytic Model vs. Elbow Tap Flow Comparison, Loop C

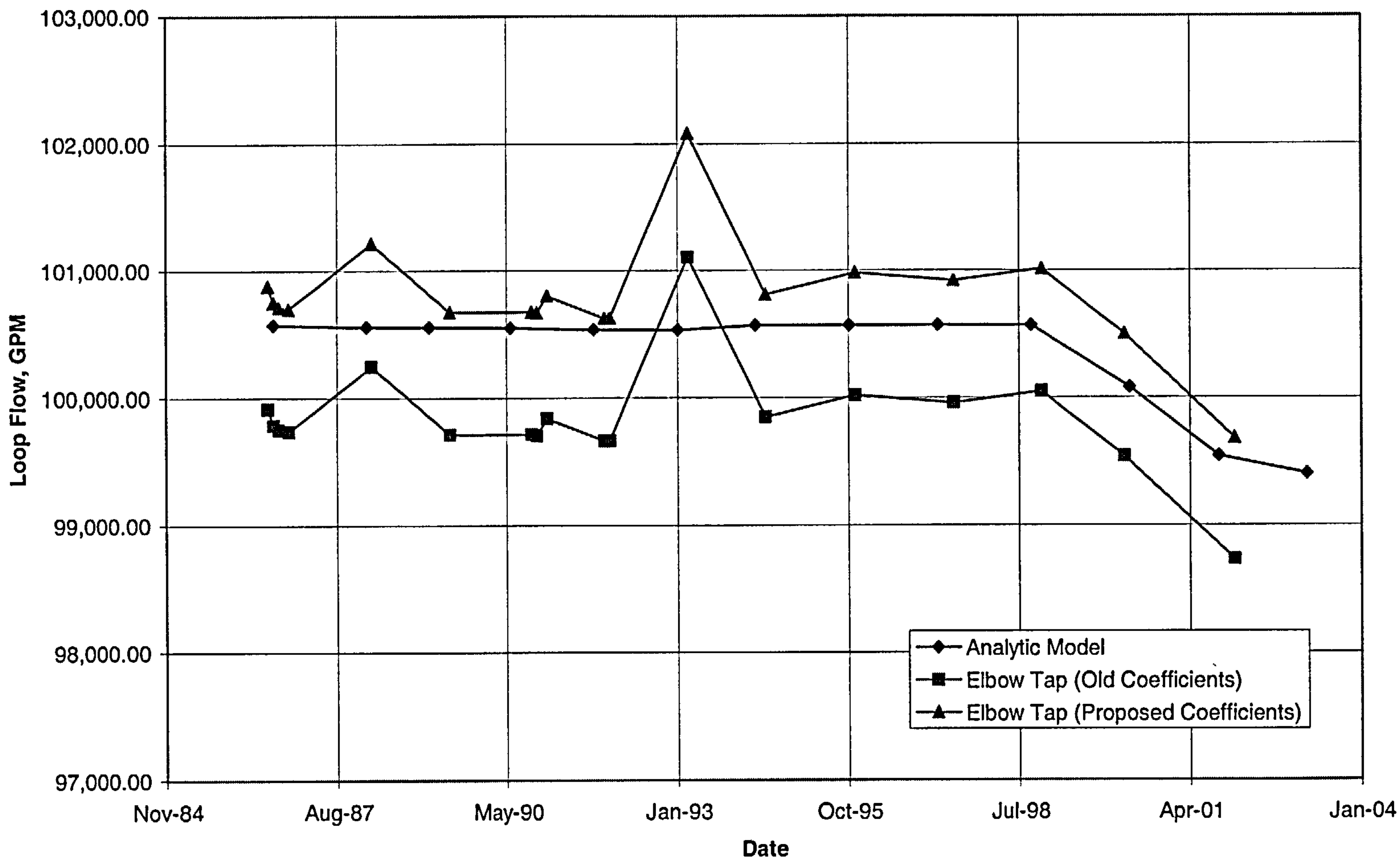


Figure 5
Analytic Model vs. Elbow Tap Flow Comparison, Loop D

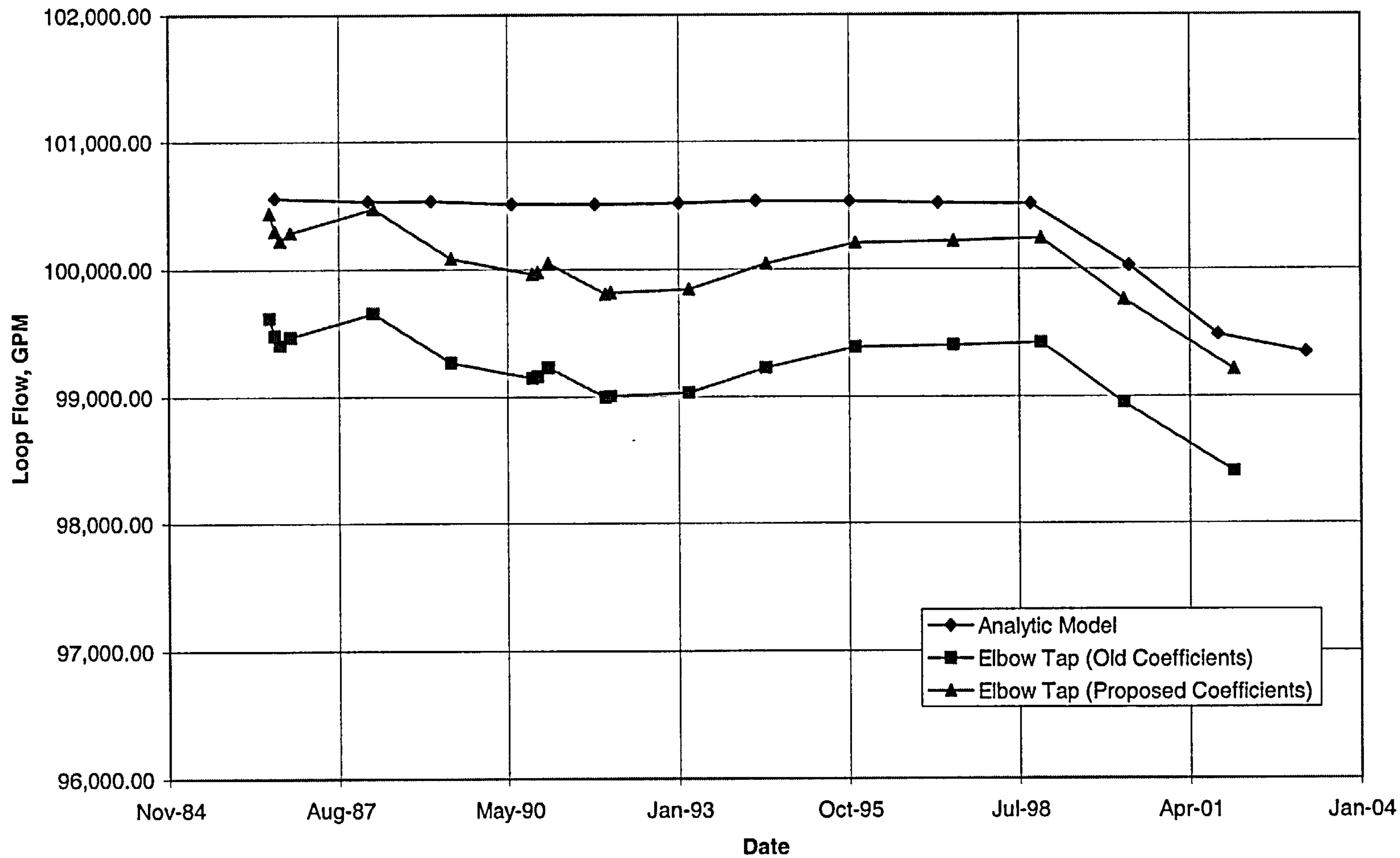


Figure 6
Reactor Coolant Flow vs. Power
C2C12 Initial Power Escalation