Duke Power
A Duke Energy Company

Gary R. Peterson
Vice President

Duke Power
Catawba Nuclear Station
4800 Concord Road
York, SC 29745
(803) 831-4251 OFFICE
(803) 831-3221 FAX

February 7, 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

By letter dated October 10, 2002, Duke Energy Corporation submitted a license amendment request for the Unit 2 Reactor Coolant System Cold Leg Elbow Tap Flow Coefficients. By telecon on January 16, 2003, the Staff requested additional information associated with the submittal. Enclosed is the response to the Staff questions. As discussed in the last paragraph of Attachment 1, an additional response will be provided to include the results of the instrumentation maintenance history review.

This correspondence does not contain any commitments.

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
Attachments:
Attachment 1 - Response to Request for Additional Information
Electronic file as referenced in Attachment 1 [Document Control Desk only]

A001

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.

Gary R. Peterson, Site Vice President

Subscribed and sworn to me: 2-7-03
Date

Notary Public

My commission expires: 7-10-2012
Date

SEAL

xc (with attachment 1):

L.A. Reyes
U.S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

E. F. Guthrie
Catawba NRC Senior Resident
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

R. E. Martin (addressee only)
NRR Senior Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O8-H12
Washington, D.C. 20555-0001

H. Porter
SC DHEC, Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

Attachment 1
Response to NRC RAI Dated December 18, 2002
and Followup Question From Telecon on January 16, 2003

1. A key aspect of the staff's previous review was substantiation that elbow tap coefficients were constant. As described in the Reference 1 Safety Evaluation, the staff believes this should be achieved on a plant-specific basis by a comparison of realistic analysis model predictions and observed flow rate as indicated by the elbow taps. The reasons for any differences should be fully assessed, and corrections should be incorporated into the flow rate determination process to eliminate such differences in the future. Please provide sufficient information for us to understand how you meet these criteria or, if you do not, please explain and justify your approach with respect to the staff's Reference 1 determination. In addition, your response should include the following.

    a. A complete copy of your analytical model, and

    b. A comparison of predicted RCS flow rate in each of the four loops and flow rate determined by the elbow taps with the assumption of constant flow coefficients for past configuration changes such as changes in fuel configuration, SG tube plugging, and reactor coolant pump replacements (if any). The basis for the assumed impact of each configuration change as an input to the analytical model should be described.

Response:

The response to the staff's questions has been subdivided into six topics.

<u>Application of the Analytical Model</u>

NRC SER issues associated with the RCS flow analytical model were addressed in the October 10, 2002 submittal (Attachment 1, p. 5, "RCS Flow Analytical Model"). The analytical model is used to predict the trend of RCS loop flow due to changes in loop hydraulic data. The hydraulic data that are expected to change in the future are mainly core pressure drop due to fuel assembly design changes, and steam generator tube bundle pressure drop due to steam generator tube plugging. Historical design changes for McGuire and Catawba have also included the RTD bypass manifold removal and the reactor vessel internals upflow conversion (McGuire units only). The inputs to the analytical model are best-estimate values for hydraulic resistances and the reactor coolant pump head/flow curves. Some of these input data are obtained from vendors, and some are calculated by Duke. The outputs of the analytical model are individual loop flowrates and total system flowrate. These predicted flowrates are useful in determining a quantitative change in the loop flows for a given design change. The change in analytical model flow is used to determine the expected plant flow based on current plant loop flow data (i.e. current plant flow plus analytical model flow change equals future plant flow). This projection of future plant loop flow can then be compared to assumed flow values used in design analyses to determine if sufficient flow margin exists. If a flow limit is approached due to insufficient flow margin, then the affected design analyses can be reanalyzed, or the design change can be revised or rejected. As an example, the flow decrease from a certain amount of future steam generator tube plugging can be addressed by reanalyzing affected analyses with a lower flowrate, or by sleeving tubes (less pressure drop and less flow reduction) instead of plugging tubes.

The analytical model is not normalized or adjusted based on plant flow data. It is a best-estimate predictor of flow based on input data. The model is revised if better input data or new input data become available. It is not affected by variations in plant data due to normal process variations or instrumentation uncertainty. Therefore flow predictions from the analytical model are more stable and simply trend the change in input values.

The staff's question suggests that the analytical flow model should be corrected to eliminate differences between the flow prediction and plant data. Corrections will be made only when new design data is obtained. However, considering the intended use of the model, there is no use in correcting or normalizing the analytical model to match the plant flow data. The use of the flow change predicted with the analytical model and the actual plant flow data achieves the intended purpose of predicting a future change in plant flow due to a change in the loop hydraulics.

Analytical Model Comparison to Elbow Tap Flow Indication

Figure 1 represents a comparison between the elbow tap flow indication of total RCS flow and the analytical model calculation of the total RCS flow. Two elbow tap indications of total RCS flow are given, one for the old elbow tap coefficients and one for the new proposed coefficients. Figures 2 through 5 contain a similar comparison between the individual loop flows and those calculated by the analytical model.

The following changes, which have the potential to affect RCS flow, have been made to the Catawba Unit 2 RCS since plant startup. These changes have been incorporated in the analytical model using the latest data available.

1) RTD bypass manifold removal (February 1988 outage)

2) SG tube plugging (July 1986 – Present)

3) Transition to Mark-BW fuel from Westinghouse OFA fuel (February 1993 – October 1995)

4) Transition to Westinghouse RFA fuel from Mark-BW fuel (April 2000 – Present)

RTD Bypass Manifold Removal

The RTD bypass manifold was removed at Catawba Unit 2 during the February 1988 refueling outage after Cycle 1. Prior to this outage the RCS flow as indicated by the elbow taps would be slightly lower since the RTD bypass manifold diverts a small amount of reactor coolant from the hot leg into the crossover leg downstream of the elbow taps. Also, a small amount of loop flow is diverted from the cold leg and is recirculated back to the crossover leg. An examination of the RTD bypass manifold flow verification test, documented in the Catawba Unit 2 Startup Report, indicates that approximately 111 gpm/loop and 173 gpm/loop are diverted from the hot and cold legs respectively. Therefore, the analytical model pressure drops in the hot leg, crossover leg and cold leg are now adjusted to reflect the diverted flows for Cycle 1. This is a new change to the analytical model that was not present in prior submittals. The pressure drops in the portion of the hot leg downstream of the bypass line along with the steam generator and crossover leg upstream of the RTD bypass return are adjusted by decreasing the loop flows by 111 gpm/loop. The pressure drops in the crossover leg downstream of the bypass system return line along with the reactor coolant pump and cold leg upstream of the bypass system diversion are adjusted by increasing the loop flow by 173 gpm/loop. As stated in the October 10, 2002 submittal, the expected increase in the total elbow tap flow as the result of RTD bypass manifold removal would be approximately 379 gpm. The elbow tap indication of flow resulted in an increase in flow of about 650 gpm. With these changes to the analytical model the flow following RTD bypass manifold removal is slightly less than the previous cycle, including the small change due to steam generator tube plugging (7 tubes).

The analytical flow model now predicts the change in loop hydraulics due to the RTD bypass manifold removal. The elbow tap measured flow shows the flow increase since the flow in the section of the RCS where the flow indication is measured is directly affected by the RTD bypass removal. This difference between the analytical model and the elbow tap indication of flow is expected and appropriate.

Steam Generator Tube Plugging

Steam generator tube plugging has occurred at Catawba Unit 2 for almost all the cycles since plant startup. The following table shows the loop specific steam generator tube plugging that has occurred to date.

	Loop A # Tubes	Loop B # Tubes	Loop C # Tubes	Loop D # Tubes	Total # Tubes	Percent
Jul-86	8	8	8	8	32	0.17
Aug-86	1	7	1	5	14/46	0.25
Feb-88	4	1	0	2	7/53	0.29
Feb-89	2	5	1	0	8/61	0.33
Jun-90	9	1	2	7	19/80	0.44
Oct-91	7	0	4	1	12/92	0.50
Feb-93	14	6	13	10	43/135	0.74
May-94	7	11	5	9	32/167	0.91
Nov-95	9	2	5	6	22/189	1.03
Apr-97	1	5	0	4	10/199	1.09
Oct-98	1	5	1	2	9/208	1.14
May-00	0	4	2	1	7/215	1.18
Oct-01	0	0	0	0	0/215	1.18
Total	63	55	42	55	215	
Percent	1.38	1.20	0.92	1.20	1.18	

The analytical model includes these changes by reducing the number of steam generator tubes and thus the available flow area in the individual loops. The analytical model shows the impact of the tube plugging, but the effect is very small since the number of tubes plugged since Catawba Unit 2 startup has been small. The analytical model has shown a much larger flow reduction for the other McGuire and Catawba units due to higher levels of steam generator tube plugging prior to steam generator replacement, as would be expected.

Transition to FANP Mark-BW Fuel

The transition to FANP Mark-BW fuel began with the February 1993 outage. The first core containing Mark-BW fuel consisted of 39% Mark-BW fuel and 61% Westinghouse OFA fuel. The second transition core was installed during the May 1994 refueling outage. The second core consisted of 85% Mark-BW fuel and 15% Westinghouse OFA fuel. The final transition core was installed during the November 1995 refueling outage and consisted of a full core of Mark-BW fuel. The vendors supplied the following pressure drop information for the fuel assemblies. Duke considers these data to be accurate.

	Full Core Westinghouse OFA Fuel ΔP (psi)	Full Core FANP Mark-BW Fuel ΔP (psi)
Bottom Nozzle	3.2	3.12
Core	17.3	16.88
Top Nozzle	1.1	1.07
Total	21.6	21.07

Using the information above, the fuel pressure drops in the analytical model were weighted to produce pressure drops which reflect the number of fuel assemblies of each type in the core. Since the fuel pressure drop information indicates that the FANP Mark-BW fuel has less pressure drop than the Westinghouse OFA fuel it is expected that a slight increase in the RCS flow would occur following each of the three refuelings. The analytical model reflects the increasing flow in each of transition core reloads. Since the increases in flow are not large, due to the small decrease in fuel ΔP and concurrent steam generator tube plugging during this period, the graphical depiction of the changes in the figures shows only small increases in flow. This prediction is in contrast to the elbow tap flow which initially decreases then recovers to reflect an increase in RCS flow during the transition. While increased steam generator tube plugging occurred during this time frame the small increase in tube plugging would not result in the flow decrease exhibited by the elbow tap indication of flow. A change in crud on the fuel or instrument drift is the likely cause of the decreases shown in the elbow tap flow data during this period. The flow decreases, each about 0.2%, are not unusual variations to see between cycle flow surveillances.

Transition to Westinghouse RFA Fuel

The transition to Westinghouse RFA fuel began with the April 2000 outage. The first core consisted of 37% RFA fuel and 63% FANP Mark-BW fuel. The second transition core was installed during the October 2001 refueling outage. The second core consisted of 79% RFA fuel and 21% Mark-BW fuel. The third transition core will be installed during the March 2003 refueling outage and will consist of 90% RFA fuel and 10% Mark-BW fuel. Westinghouse supplied the following pressure drop information for the RFA fuel. Duke considers these data to be accurate.

	Full Core Westinghouse RFA Fuel ΔP, psi
Bottom Nozzle	3.72
Core	20.10

Top Nozzle	1.28
Total	25.10

Using the information above, the fuel pressure drops in the analytical model were weighted to produce pressure drops which reflect the current number of fuel assemblies of each type in the core. Since the fuel pressure drop information indicates that the Westinghouse RFA fuel has a much greater pressure drop than the Mark-BW fuel it is expected that a significant decrease in the RCS flow would occur following each of the three refuelings during the transition. The analytical model reflects the decrease in flow in each of transition core reloads which have occurred. The decreases in flow are fairly large reflecting the relatively large change in the fuel ΔP. Comparing the flow changes predicted by the analytical model with the RCS flow changes measured by the elbow tap indication of flow show that the changes in flow between the model and the elbow taps is in good agreement. Any slight differences can be attributed to instrument drift and uncertainties. Therefore, the flow changes modeled for the transition to Westinghouse RFA fuel are considered to be in reasonable agreement with the elbow tap ΔP indications.

Summary

Duke uses the RCS flow analytical model to provide quantitative values of RCS flow changes for planning purposes. This model assumes that the true cold leg elbow tap flow coefficients remain constant over time due to no known or observed degradation processes. Duke has proposed a method for determining the values of the flow coefficients that retains a sufficient amount of conservatism. All known hydraulic losses are modeled with the best available information. The vendor reactor coolant pump head/flow curves are used. The analytical model has been revised to include the presence of the RTD bypass manifold during Cycle 1. This modeling approach is intended to be a best-estimate methodology. Corrections to the analytical model will be made whenever better hydraulic data is obtained. Correcting or normalizing the analytical model using plant flow data, as suggested by the NRC staff, is inconsistent with its intended application. Rather, a predicted change in flow is used with the latest plant flow data to project future flow values. The flow margin that exists is then used for decision making regarding a need to change the flow assumed in design analyses and the technical specification flow limit, or the impact of a proposed design change or maintenance activity. The analytical model has proven to be sufficiently accurate for the intended purposes.

An electronic copy of the analytical model is included with this letter. Loop-specific flow predictions and plant data are included in the electronic file.

2. In Reference 2, you identify data obtained from several plants using N-16 instrumentation to measure RCS flow rate for the purpose of comparison with data obtained using the cold leg elbow tap instrumentation and to support your elbow tap instrumentation conclusions. Please provide additional information regarding the plants and the data. Include:

   a. A description of the N-16 instrumentation, the tests, and the comparisons,

   b. Sufficient test data to fully support your conclusions, and

   c. References covering the application of N-16 instrumentation for the mentioned plants, such as a WCAP.

Response:

The data obtained from plants with N-16 RCS flow instrumentation that was provided in the October 10, 2002 submittal was intended to illustrate general agreement between RCS measured flow based on a diverse flow instrument and RCS measured flow based on the cold leg elbow tap instrumentation. These data are obviously limited in scope, but represent additional information that adds some qualitative technical justification to support the position that the elbow tap flow instrumentation is a reliable indication of RCS flow over a period of time. These data were obtained from other organizations with the intent of adequately addressing the staff's concerns. We do not see significant value in obtaining the additional detailed information requested by the staff, since the data is of limited usefulness. We do not believe that the effort required to provide the requested information is justified, and some of the requested data is likely to be unavailable for submittal by Duke due to vendor proprietary restrictions. For these reasons, the requested information will not be provided. We believe that the data provided in the October 10, 2002 submittal remains of value to the resolution of the staff's concerns absent the additional requested information.

3. The Catawba Unit 2 total flowrate (Refer to Figure 1) 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, in particular relative to the possible effects of the core power distribution.

Response:

During the time period of the decrease in flow of concern, four types of potentially relevant design changes occurred in Catawba Unit 2. The response to Question #1 includes the steam generator tube plugging information and the Westinghouse OFA to FANP Mk-BW fuel assembly design transition information that occurred in this time period. Not included in the response to Question #1 was a decrease in the primary temperature program of -3.3°F for the purposes of slowing the rate of steam generator tube corrosion. This occurred in November 1995. Also not included in the response to Question #1 was the change in the core loading pattern during each refueling outage. Both of these changes were considered to have minimal effect on the flow, and so they were not mentioned. Neither of these effects (primary loop temperature reduction or core power distribution) accounted for the changes in measured flow in 1991 and 1993 but is included in the discussion below for completeness of the historical flow trend review.

Additional details on these two changes follows.

<u>Primary Loop Temperature Reduction</u>

The primary coolant temperature did not change for 1991 and 1993.

The -3.3°F reduction in primary coolant temperature in 1995 would cause the following associated changes. Since the reactor coolant pumps are constant volume flowrate pumps, the volumetric flow would not change. The mass flowrate would increase due to the higher density following the temperature reduction. The effect on loop pressure drop would be insignificant due to a minimal change in fluid velocity. The flow data for the time at which this change was implemented, November 1995, shows a small increase in flow. This measurement also was affected by a small amount of steam generator tube plugging (flow decrease) and the third and final cycle of the fuel transition to FANP Mk-BW fuel (flow increase). The analytical model predicted no change in flow between 1994 and 1995. Duke's interpretation is that the change in measured flow is small and is due to process measurement drift and expected variation in flow.

<u>Core Power Distribution</u>

The predicted core power distributions for Catawba Unit 2, Cycle 4 through 8, are shown in the attached Figures 6 and 7. These cycles span the time period of the flow decrease of concern. Also shown is a histogram (Figure 8) for these five power distributions showing the number of fuel assemblies as a function of assembly radial peaking factor. The histogram provides an indication of the flatness of the power distribution. It is conceivable that different power distributions could have an impact on the core pressure drop and consequently the loop flowrate. Duke has not considered flow to be sensitive to the core power distribution for the types of core power distributions that characterize McGuire and Catawba reload cores. To quantify this effect

Duke has analyzed Cycles 6 and 7, which are the most different relative to the core power distribution, with the VIPRE-01 code to calculate the core pressure drop. A one-eighth core VIPRE model with one channel per fuel assembly was used. The results of this analysis shows that there was no difference in the core pressure drop to the second decimal point. This was expected since there is no bulk boiling in the core. To confirm that this VIPRE model is capable of calculating a change in core pressure drop due to a change in power distribution, the same model was run with an arbitrary increase in the peaking factor in one fuel assembly (8 fuel assemblies out of 193 in full core). The peaking factor was increased from 1.2 to 1.8 to obtain a channel exit void fraction of 2%. With this unrealistic increase in the power distribution the core pressure drop increased by 0.03 psi. Such a change in core pressure drop will cause no measurable change in loop flow. These results support a conclusion that the core power distribution changes in Catawba Unit 2 during the time period of interest are not associated with the indicated flow trend.

The plant elbow tap flow instrumentation maintenance history is being reviewed for the time period of interest. Should this review identify any contributing causes to the indicated flow trend, that information will be forwarded to the staff. The known design changes cannot account for the observed trend, and no other explanations have been identified.

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

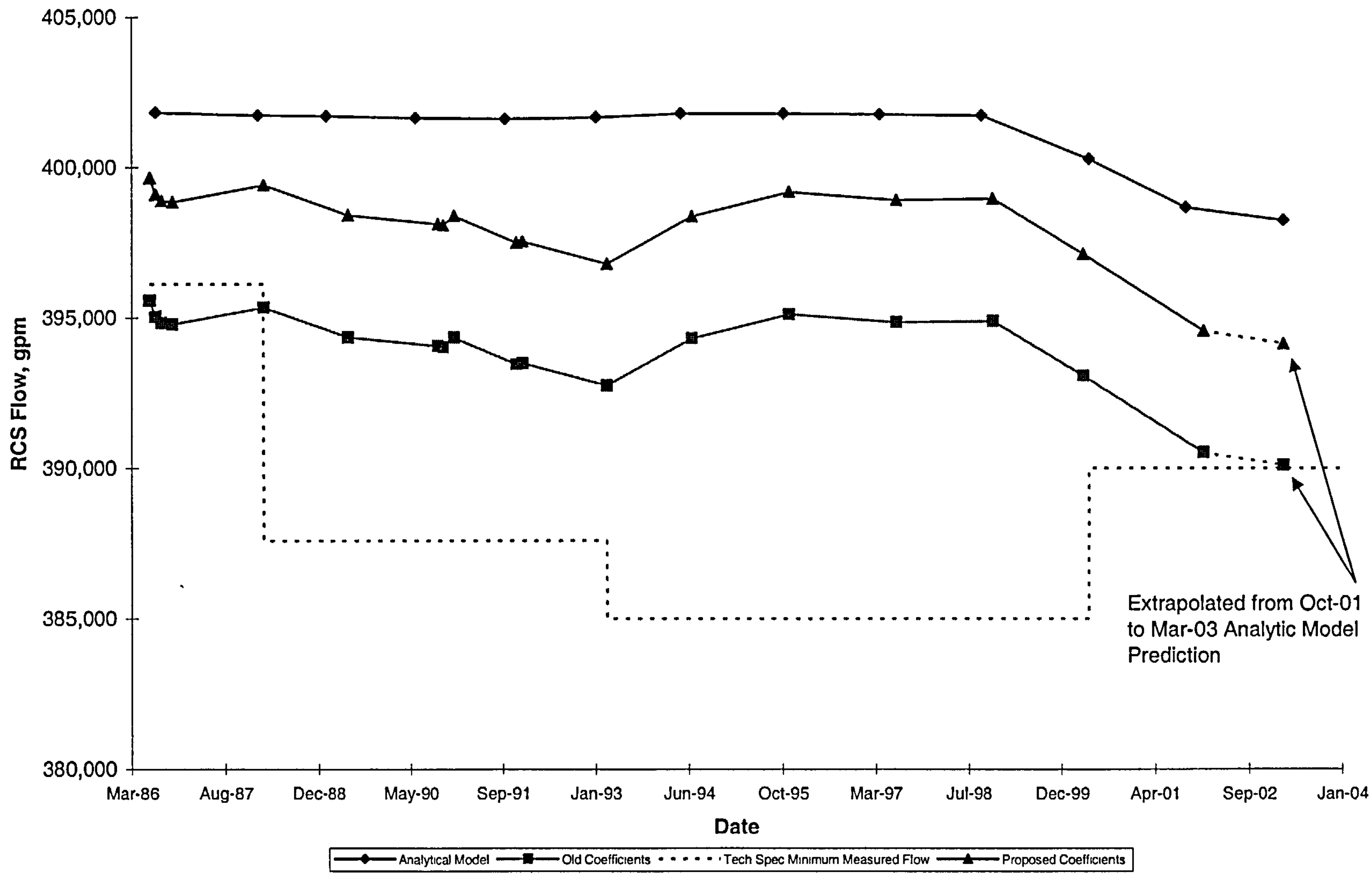


Figure 2
Analytic Model vs. Elbow Tap Flow Comparison, Loop A

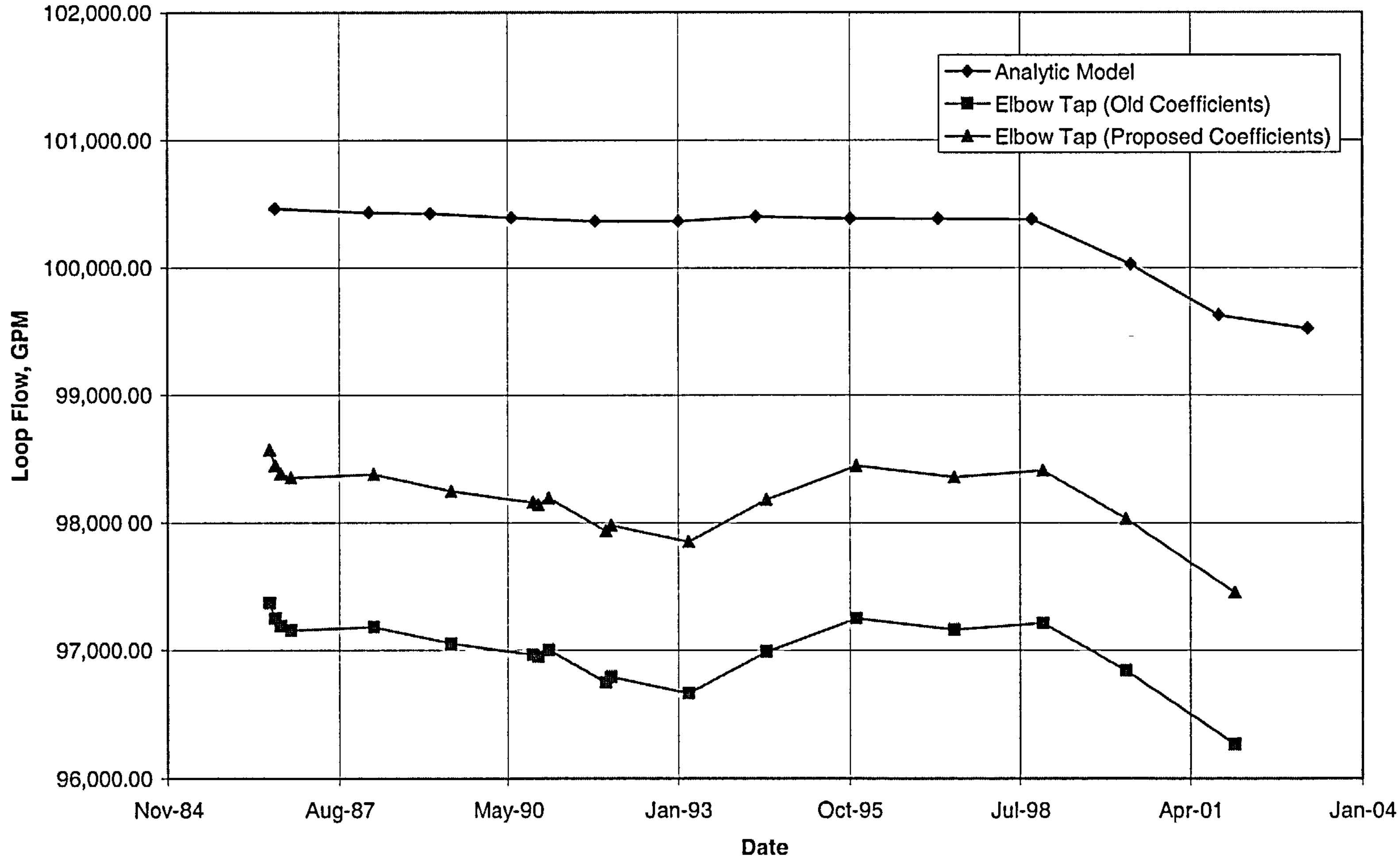


Figure 3
Analytic Model vs. Elbow Tap Flow Comparison, Loop B

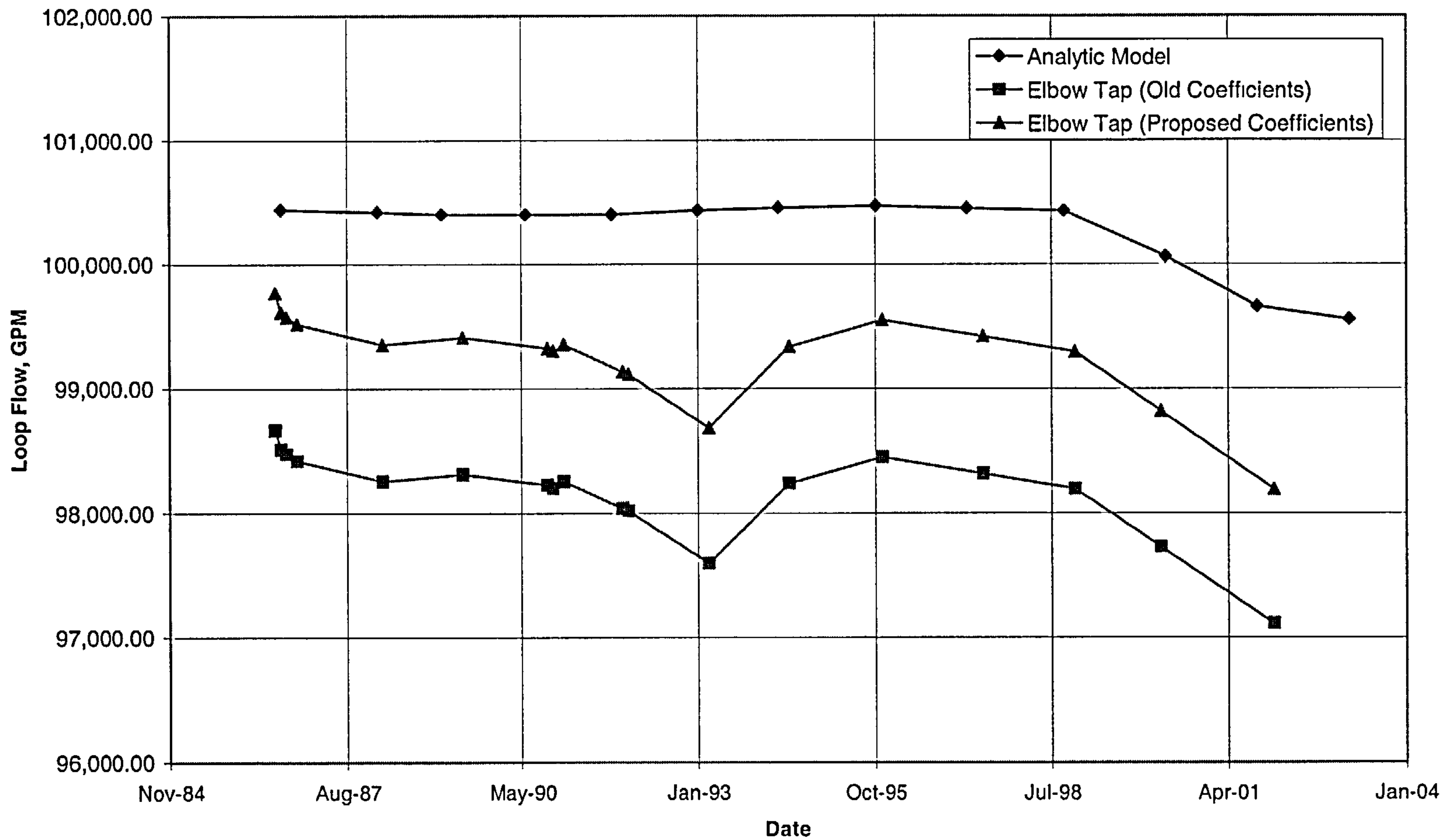


Figure 4
Analytic Model vs. Elbow Tap Flow Comparison, Loop C

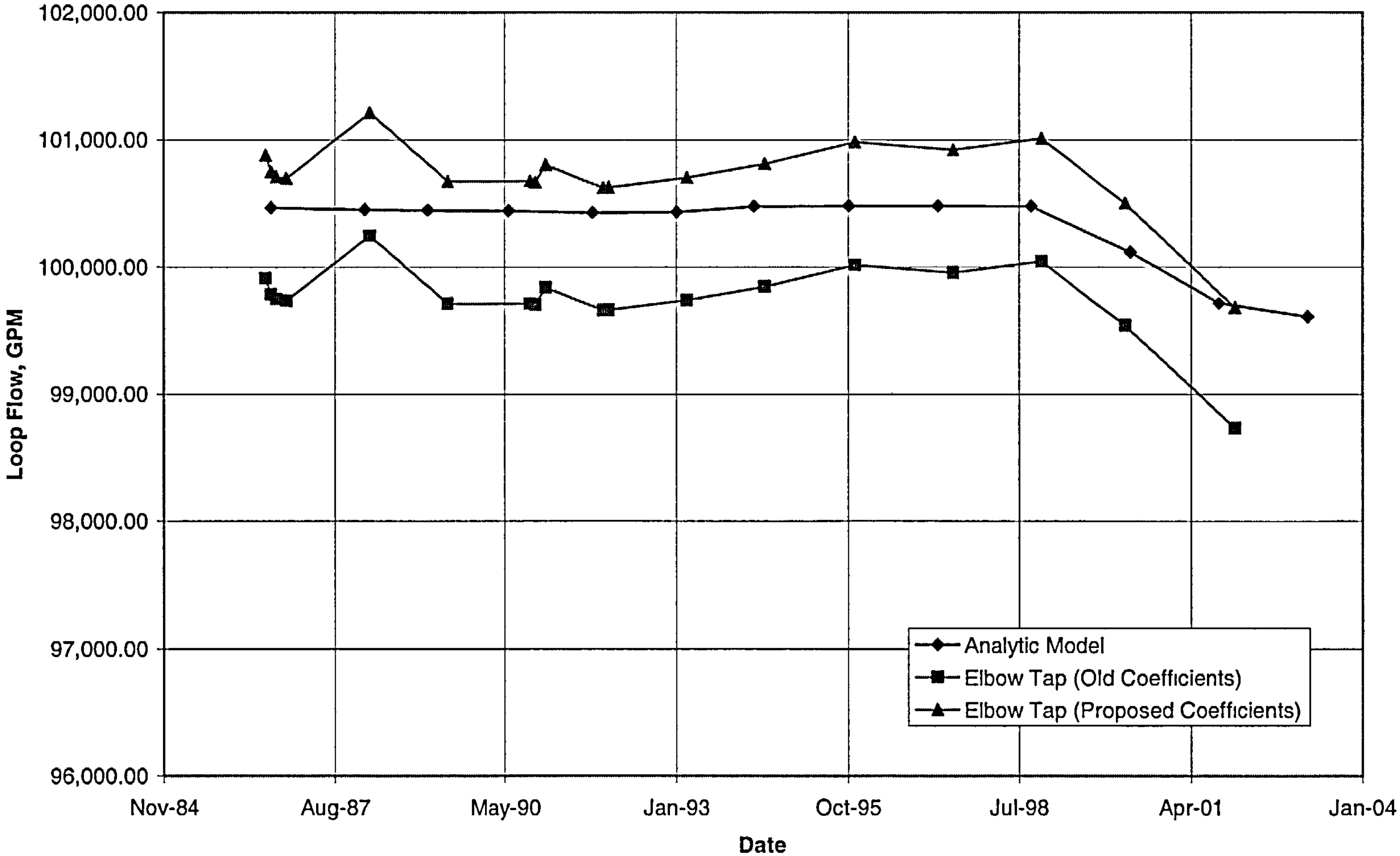
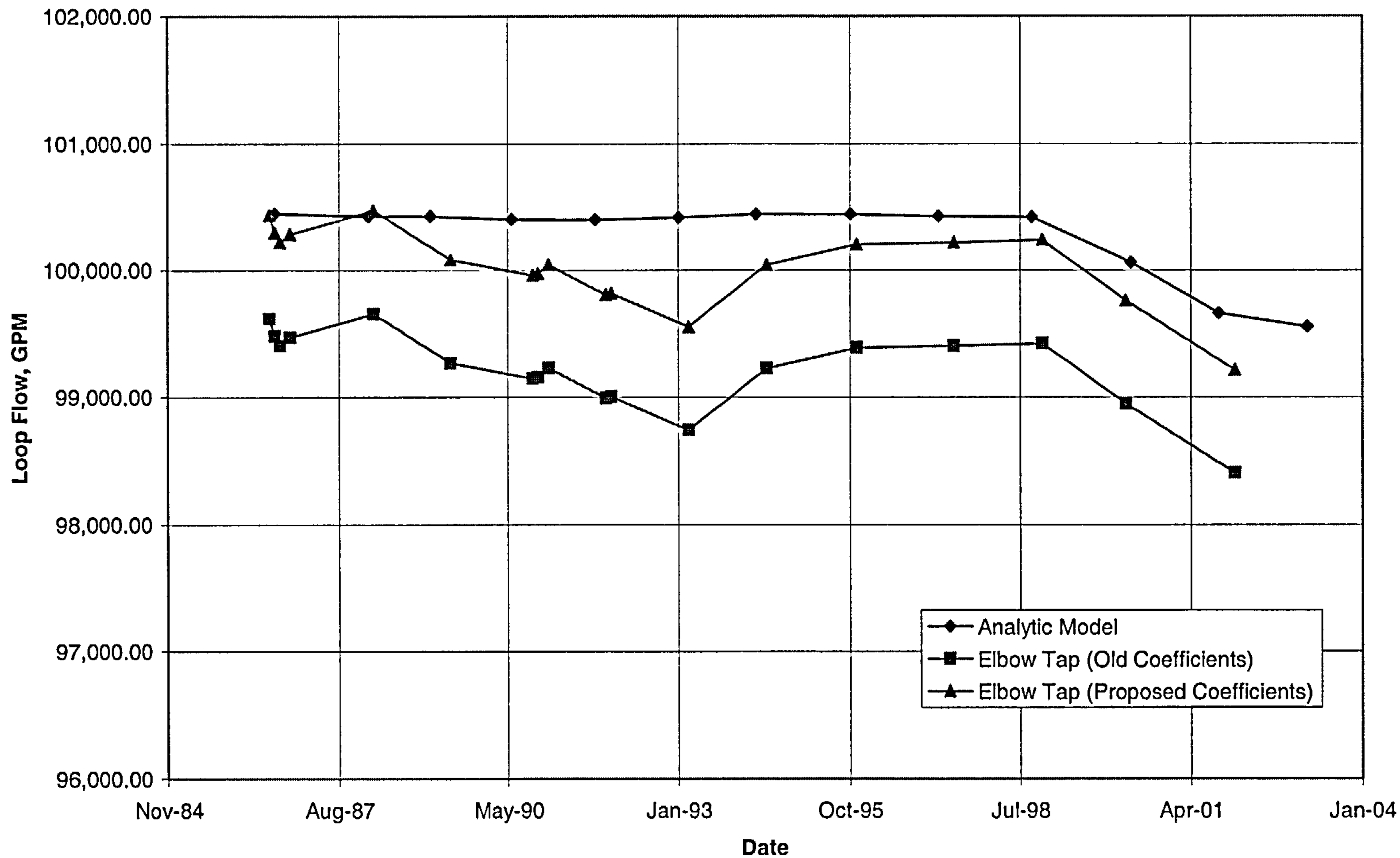


Figure 5
Analytic Model vs. Elbow Tap Flow Comparison, Loop D



C2C4 (Jun-90)

0.9499					
1.2443	1.1808				
1.1764	1.1461	1.1696			
1.2486	0 9992	1.2683	0.9686		
0.9578	1.2483	1.0678	1.2605	1.2323	
1.0973	1.0728	1.2516	1.1930	1.1576	0.5475
1.1340	1.1569	1.0168	1.1108	0 6248	0.2229
0.8058	0 6692	0.7188	0 4396		

C2C5 (Oct-91)

0.9383					
1.2569	1.0211				
1.1842	1.2290	0 9415			
1.2444	1.1442	1.1844	0 9925		
0.9506	1.2204	1.0702	1 2682	0 9861	
1.2024	1.1462	1.2536	1.1610	1.1372	0.5206
1.0632	1.1879	1 0835	1.1003	0 5842	0.2436
0.8528	0 6734	0.7937	0 4185		

C2C6 (Feb-93)

0 9329					
1.2658	1.0063				
0.9602	1.2778	1 0178			
1.1420	0 9751	1.2759	0 9577		
0.9419	1.2329	0.9876	1.2251	0.9121	
1.2488	1.2135	1.2580	1.2040	1.2017	0.7361
1.0444	1.1550	1.0889	1.1410	0 6499	0.3372
0.6427	0.6180	0 7740	0.4549		

C2C7 (May-94)

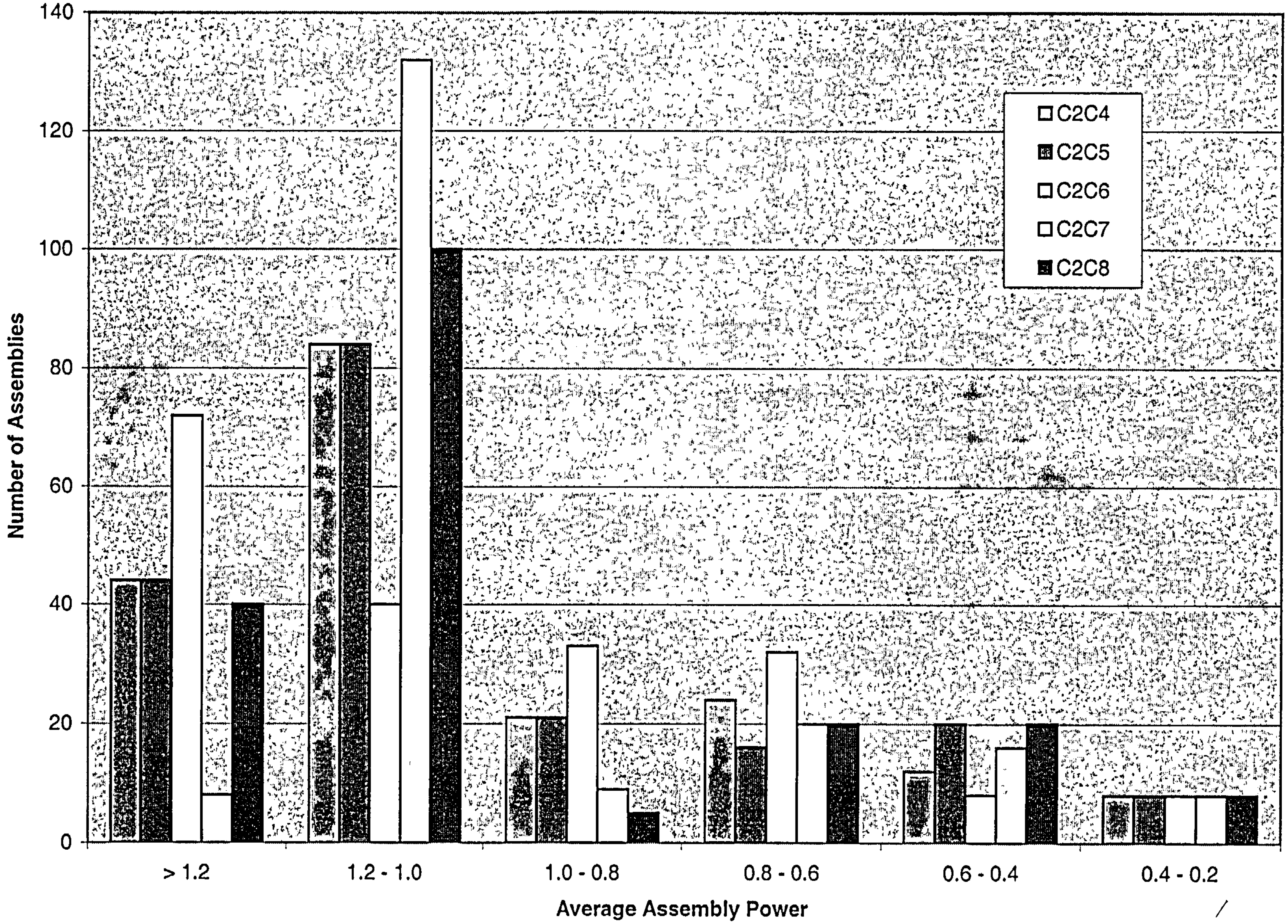
0 8193					
1.0066	0 9392				
1.1221	1 1412	1.1802			
1.1657	1.1888	1.1999	1.1154		
1.0265	1.1839	1.2082	1.1983	1.1909	
1.1259	1.1999	1.1734	1.1657	1.1455	0 8576
1.1643	1.1085	1.0516	1.0189	0.6905	0.3428
0.7867	0 5435	0.7051	0.4491		

Figure 6

C2C8 (Nov-95)

0.8975					
1.1177	1.1068				
1.1340	1.1782	1.0630			
1.2413	1.1694	1.2038	1.1933		
1.2018	1.2222	1.1864	1.2188	1.1726	
1.1937	1.1432	1.2059	1.1515	1.1436	0 7051
0.9325	1.1057	1.0951	1.0932	0 6270	0 2948
0.4299	0 5461	0 7738	0 4157		

Figure 7



140
120
100
80
60
40
20
0
Number of Assemblies
C2C4
C2C5
C2C6
C2C7
C2C8
> 1.2
1.2 - 1.0
1.0 - 0.8
0.8 - 0.6
0.6 - 0.4
0.4 - 0.2
Average Assembly Power

Figure 8