UNION ELECTRIC COMPANY

CALLAWAY PLANT

STARTUP REPORT

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## 1.0 INTRODUCTION

This report describes the required testing at Callaway Plant from the time the first fuel assembly was loaded into the reactor until the plant was placed in operation. It satisfies the requirement of the Callaway Plant Technical Specifications that a Startup Report be submitted to the NRC after completion of the Startup Test Program.

Callaway Plant uses a 4 loop pressurized water reactor rated at 3411 MW thermal. It is located in Callaway County, Missouri, approximately 80 miles west of the St. Louis metropolitan area, and is owned and operated by the Union Electric Company.

Union Electric joined with four other utilities (Kansas Gas and Electric, Kansas City Power and Light, Rochester Gas and Electric, and Northern States Power) to form the SNUPPS (Standardized Nuclear Unit Power Plant System) Project Organization to manage the design and procurement of 5 standardized nuclear power blocks. Subsequently, Rochester Gas and Electric and Northern States Power cancelled their units and dropped out of the SNUPPS organization. In addition, Union Electric cancelled a second unit at Callaway, so that the only two units remaining are the Callaway Plant and Wolf Creek Generating Station, which is owned jointly by Kansas Gas and Electric and Kansas City Power and Light.

Under the SNUPPS agreement, Bechtel Power Corporation was selected as the architect-engineer, Westinghouse was selected to provide the Nuclear Steam Supply System (NSSS), and General Electric provided the turbine generator. At Callaway, Sverdrup & Parcel and Associates provided architect-engineering services which were not included in the standardized portion of the plant. Daniel International Corporation constructed the plant.

After receipt of a 5% operating license, fuel load was commenced on June 13, 1984 and completed on June 20, 1984. Initial criticality occurred on October 2, 1984, and initial synchronization to the grid on October 24, 1984. Callaway was placed in service on December 19, 1984, 189 days after commencement of fuel load and 48 days after initial criticality. Table 1.1 provides a summary of the Callaway startup program.

TABLE 1.1

CALLAWAY STARTUP SUMMARY

| | |
|---|---|
| Received 5% Operating License | 6/11/84 |
| Commenced Fuel Load | 6/13/84 |
| Completed Fuel Load | 6/20/84 |
| Commenced Precritical Testing | 7/11/84 |
| Initial Criticality | 10/2/84 |
| Initial Synchronization To Grid | 10/24/84 |
| 30% Power | 10/28/84 |
| 50% Power | 11/5/84 |
| 75% Power | 11/23/84 |
| 90% Power | 11/28/84 |
| 100% Power | 12/3/84 |
| Plant Released For Operation | 12/19/84 |
| Number Of Reactor Trips | 16 |
| Burnup Accumulated On Core | 2,112,099 MWHr |
| Number of Hours Reactor Critical | 1615 |
| Net Generation (MWe) | 623,406 MWHr |
| Number of Hours Generator On Line | 910 |

2.0 INITIAL CORE LOADING (S-070001, S-07SJ01)
Procedure ETT-ZZ-07010

OBJECTIVES

The objectives of the subject procedure were:

1. To specify the placement of temporary detectors during the core loading sequence.

2. To specify the sequence of loading fuel assemblies into the core.

3. To specify the Westinghouse fuel assembly identification number, enrichment and type of insert for each core location.

4. To identify the nuclear monitoring channels to be used for each step in the core loading sequence.

5. To provide a check for periodic verification of equipment status and plant conditions to ensure compliance with FSAR and Technical Specifications during initial core loading.

6. To verify uniform boron concentration in the Reactor Coolant System and directly connected auxiliaries prior to fuel load.

SUMMARY OF RESULTS

All Acceptance Criteria for the subject procedure were met. These criteria and discussions of associated results follows:

1. A permanent record of the core configuration is obtained.

   Final locations of the fuel assemblies and inserts were recorded on video tape. In addition, a written inventory was conducted by operations personnel during the video taping.

2. The final as-loaded core configuration is consistent with the fuel assembly core loading plan.

   The video tape and written inventory were reviewed by Reactor Engineering. All fuel assemblies and inserts were in their proper locations according to the core design.

3. Boron concentration is between 2000 ppm and 2150 ppm upon completion of fuel loading.

   The final boron concentration, obtained from the Residual Heat Removal System A sample line, was 2119 ppm at 2000 hrs. on 6-20-84.

4. Prior to initiation of fuel load, the boron concentration of the reactor vessel fluid taken at four equidistant depths in the vessel are within a 30 ppm range of a sample simultaneous-

ly taken from the Residual Heat Removal System.

The boron samples obtained between 1338 and 1355 on 6-13-84 were between 2095 and 2068 ppm for the referenced locations.

DISCUSSION

Prior to the initiation of fuel load plant conditions were established in accordance with the subject procedure. Coolant circulation was established using the Residual Heat Removal (RHR) System. The RHR loop fluid was sampled and analyzed for boron concentration at regular intervals during core loading and was maintained between 2000 ppm and 2150 ppm.

Fuel load was initiated with the insertion of primary source assembly C04 into core position L-15 at 1930 hrs. on June 13, 1984. The final assembly was loaded at 1936 hrs. on June 20, 1984. Total elapsed time was 7 days, 10 minutes. Figures 2.0.1 through 2.0.3 illustrate the final configuration of the core. The sequence of loading operations is presented by Figure 2.0.4.

Initial fuel loading at the Callaway Plant was a semi-dry operation with the reactor vessel fluid level maintained above the centerline of the outlet nozzle (above the fuel) but below the vessel flange. This provided improved visibility of the fuel assemblies during manipulations and accessibility to the vessel flange area when positioning the temporary detectors.

Equipment problems caused significant delays during fuel load. The major problems follow:

1. On June 14 source range channel N32 began spiking and causing spurious High Flux at Shutdown Alarms. The problem was determined to be moisture on connectors in two junction boxes and a damaged connector O-ring. This caused approximately 10 hours delay.

2. On June 15 the Refueling Machine hoist motor was replaced because of damaged slip rings. This caused approximately 12 hours delay.

3. On June 17 a mechanical stop on the Fuel Transfer System bent, creating difficulty in raising the frame on the Fuel Building side. This caused approximately 3 hours delay.

4. On June 18 a shear pin on the Fuel Transfer System was replaced. This caused an approximately 1 hour delay.

5. On June 18, the magnatorque speed controller on the Refueling Machine hoist was replaced due to a malfunction of limit switches. This caused a delay of approximately 15 hours.

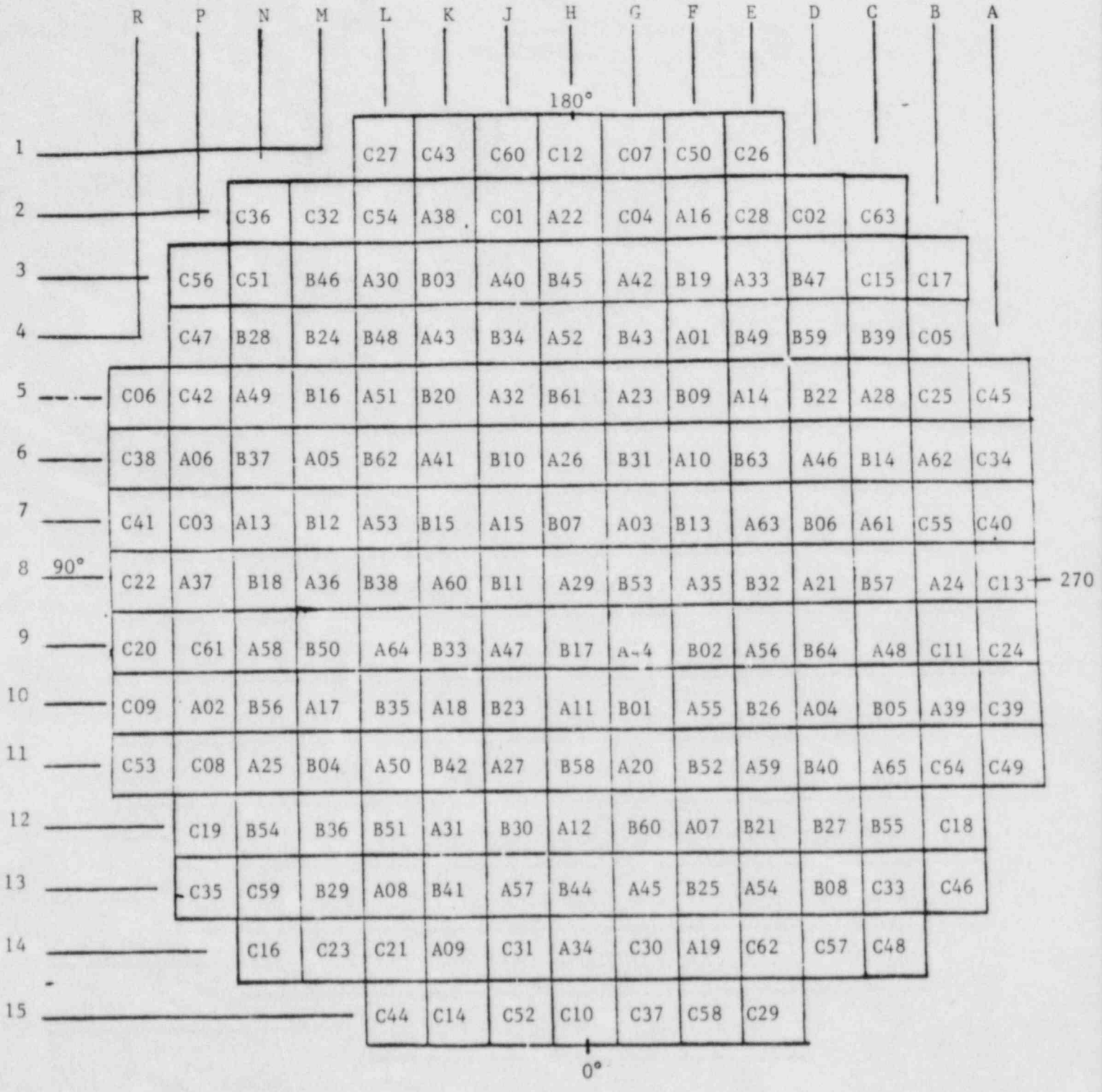
Fuel assembly A47, loaded in Step 38, was picked up for inspection at Step 55E due to visual observations by the reactivity monitoring

crew. A solid piece of grease was removed from the north side of the assembly and the affected area cleaned. The assembly was then reloaded.

Following completion of fuel loading, the reactor vessel and associated equipment were assembled. This was accomplished without any significant problems.

Figure 2.0.1

Callaway Randomized Core
Loading Pattern, Cycle 1



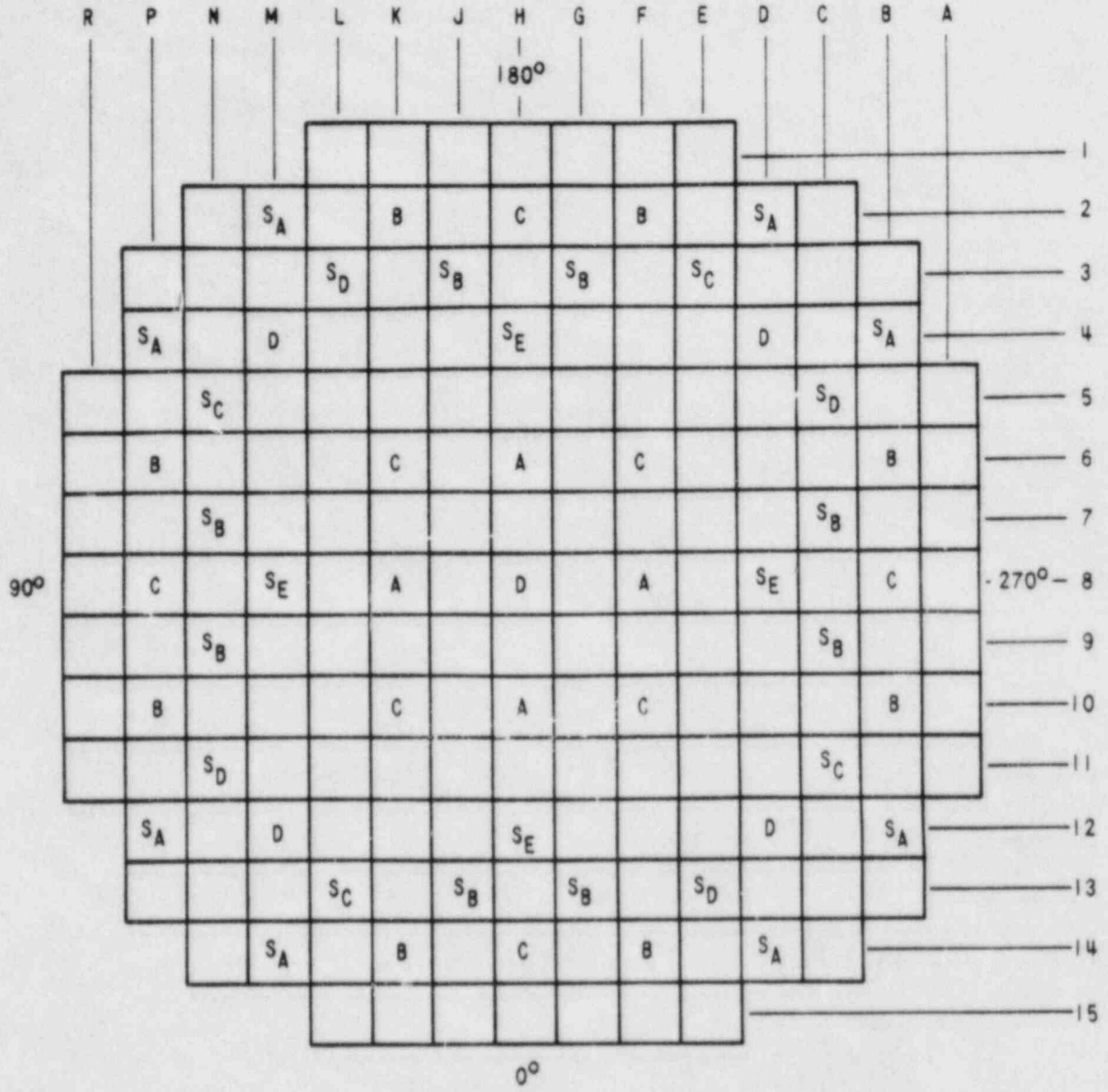

[A] Region 1 (2.1 w/o)

[B] Region 2 (2.6 w/o)

[C] Region 3 (3.1 w/o)

Figure 2.0.2

Control Rod Locations

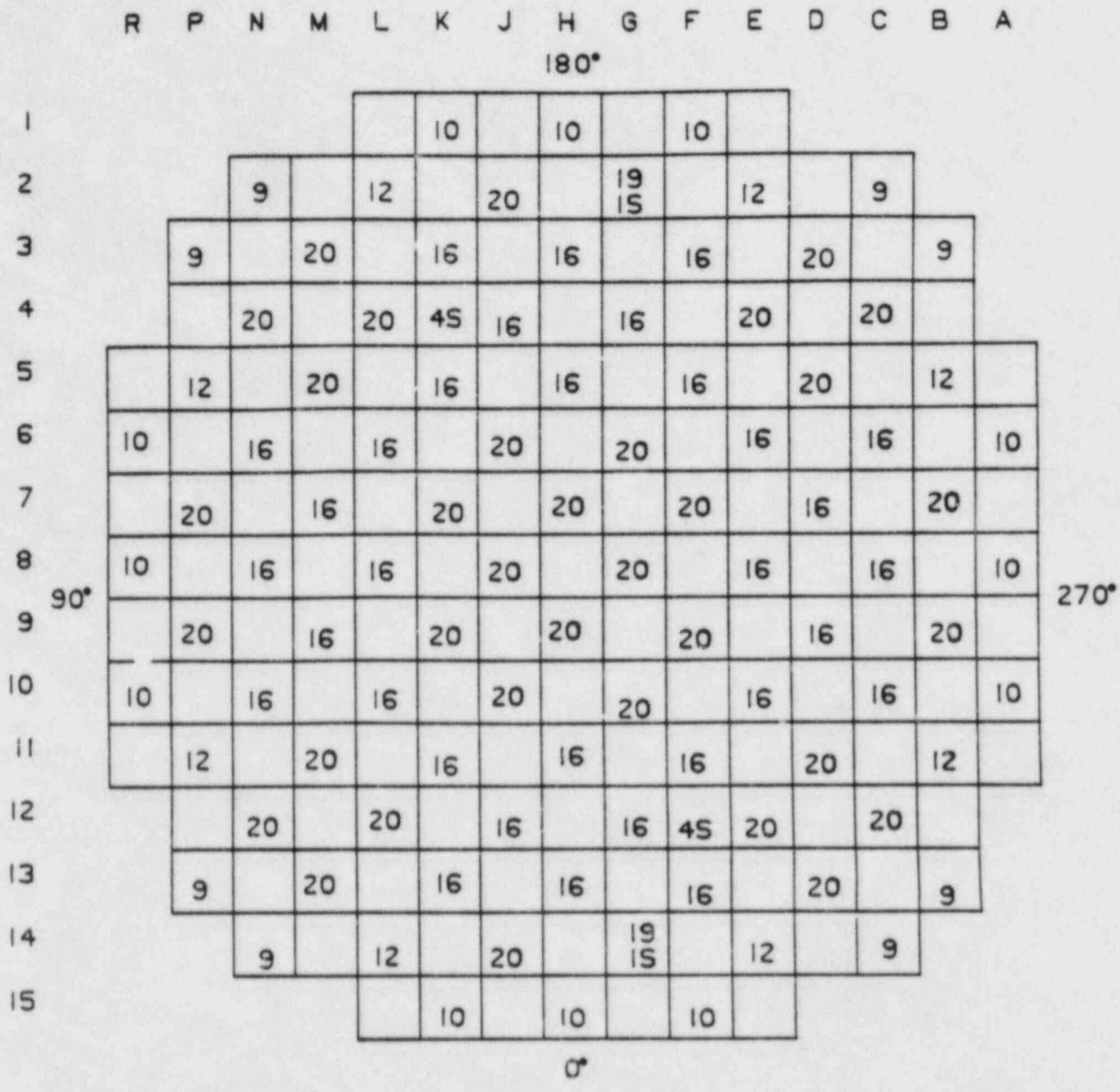



| CONTROL BANK | NUMBER OF RODS |
|---|---|
| A | 4 |
| B | 8 |
| C | 8 |
| D | 5 |
| TOTAL | 25 |

| SHUTDOWN BANK | NUMBER OF RODS |
|---|---|
| $S_A$ | 8 |
| $S_B$ | 8 |
| $S_C$ | 4 |
| $S_D$ | 4 |
| $S_E$ | 4 |
| TOTAL | 28 |

Figure 2.0.3

Burnable Poison And Source Assembly Locations




NUMBER INDICATES NUMBER OF BURNABLE POISON RODS
IS INDICATES PRIMARY SOURCE ROD
4S INDICATES FOUR SECONDARY SOURCE RODS

Figure 2.0.4

CORE LOADING SEQUENCE

Legend For Core Loading Figures



Assembly loaded in permanent position in previous step.



Assembly loaded in temporary position in previous step.




Assembly loaded into position during loading step number N.




Location of Temporary Detector A (B and C).



Assembly with primary source insert.

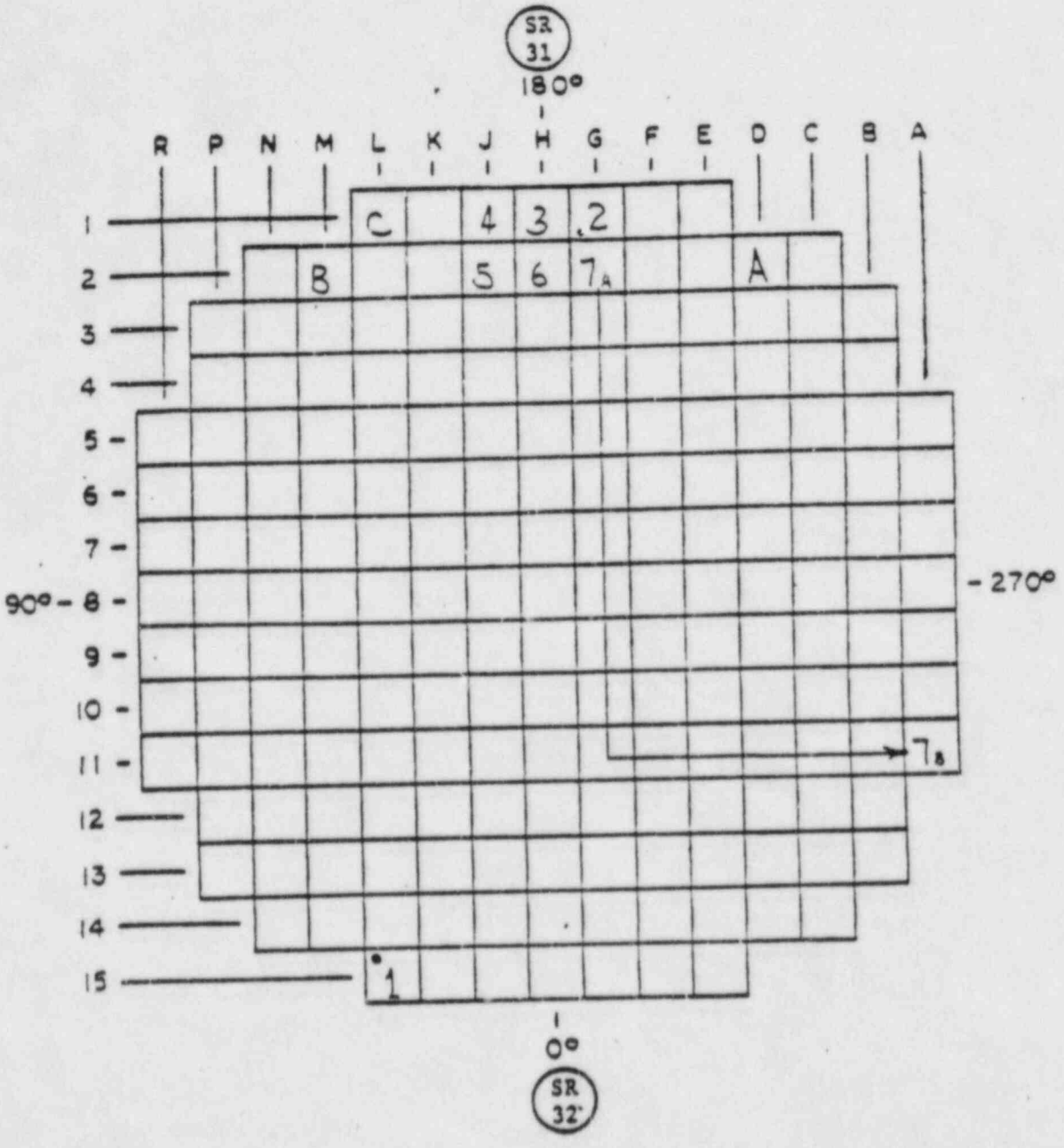


Not as yet loaded.

Note: Arrows indicate detector or fuel movement.

Figure 2.0.4 (cont.)

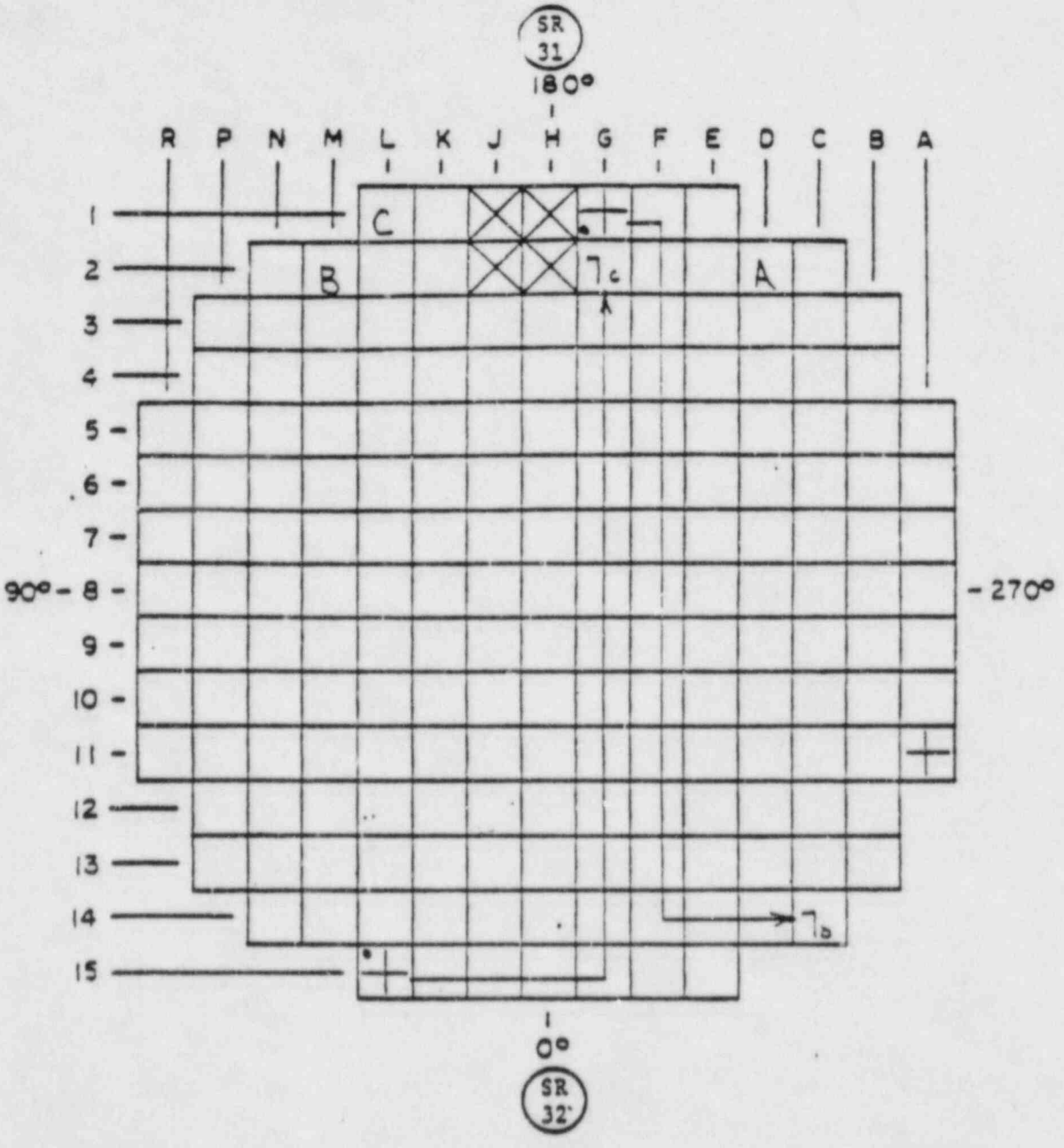
CORE LOADING SEQUENCE




Core Loading Sequence Steps 0A to 7B

Figure 2.0.4(cont.)

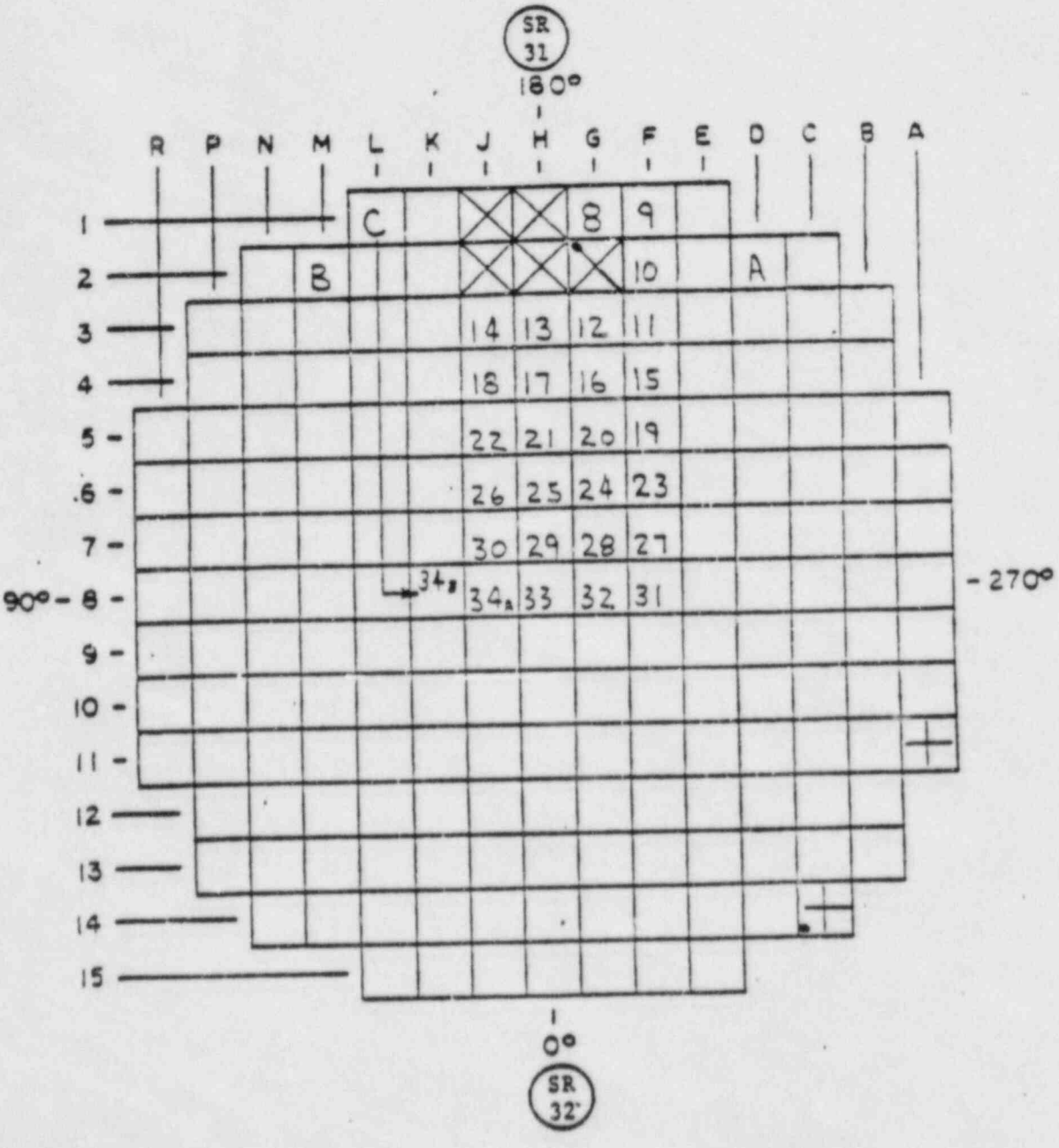
CORE LOADING SEQUENCE




Core Loading Sequence Steps 7c to 7b

Figure 2.0.4(cont.)

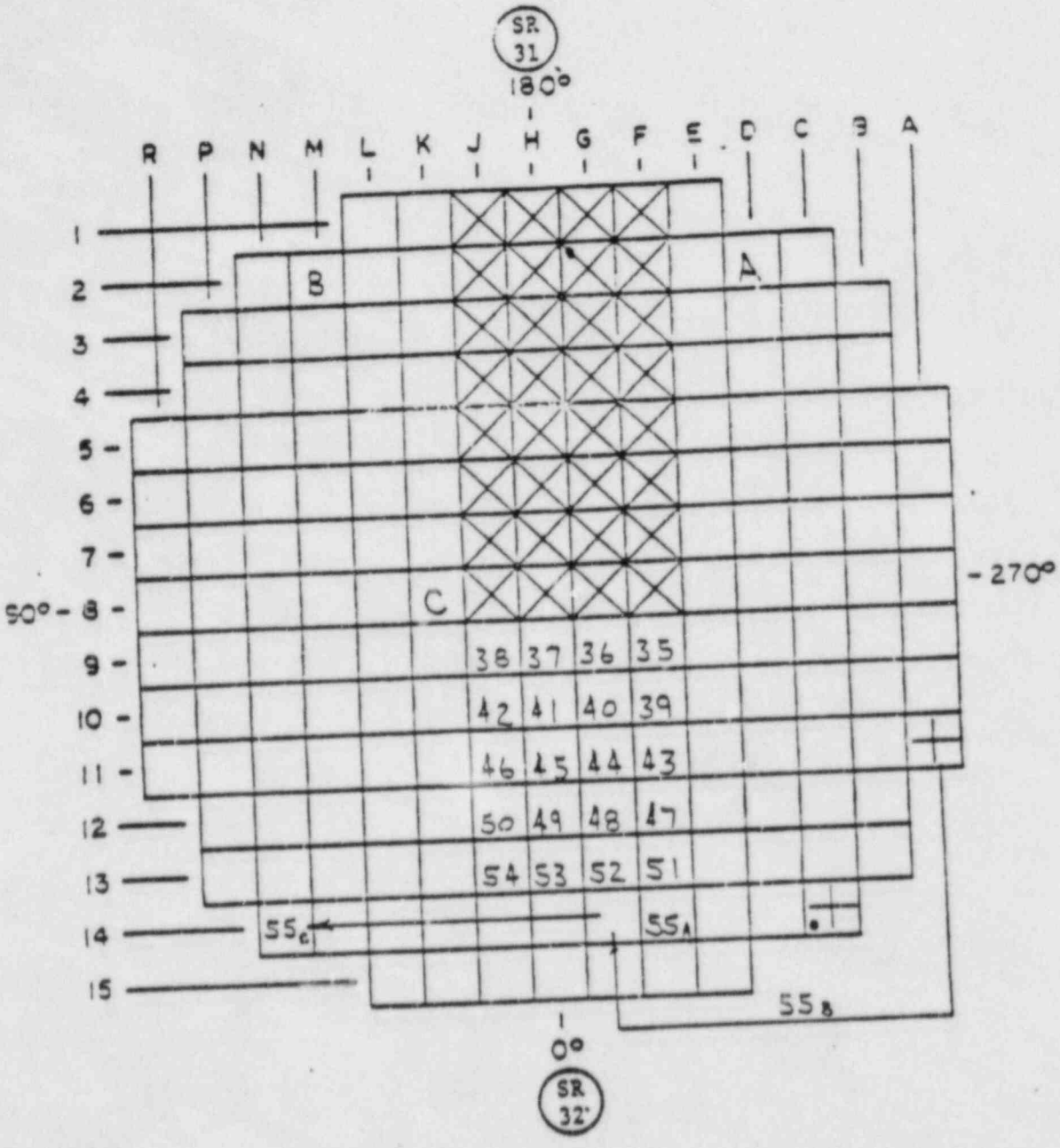
CORE LOADING SEQUENCE




Core Loading Sequence Steps 8 to $34_B$

Figure 2.0.4 (cont.)

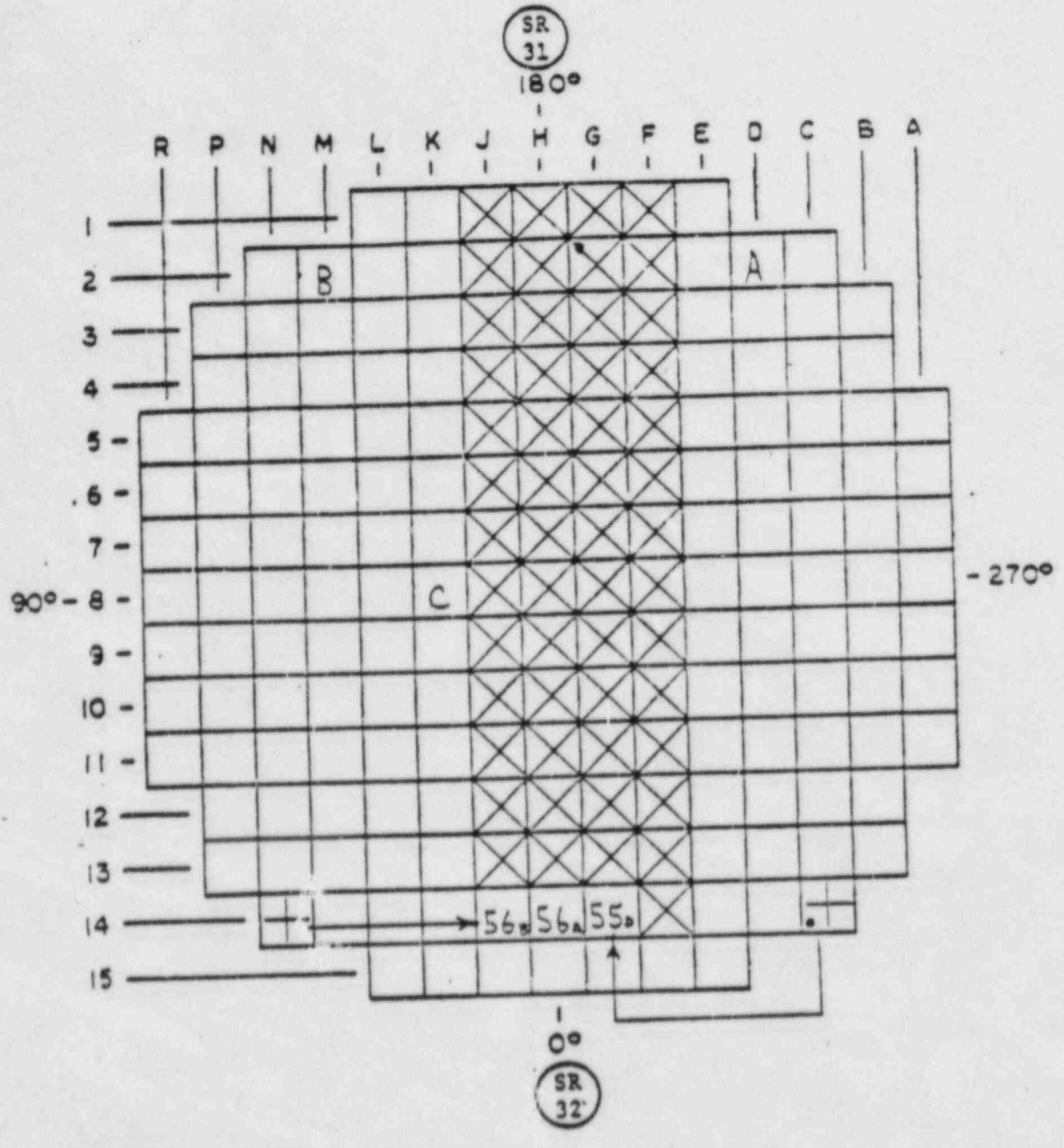
CORE LOADING SEQUENCE




Core Loading Sequence Steps 35 to 55c

Figure 2.0.4 (cont.)

CORE LOADING SEQUENCE

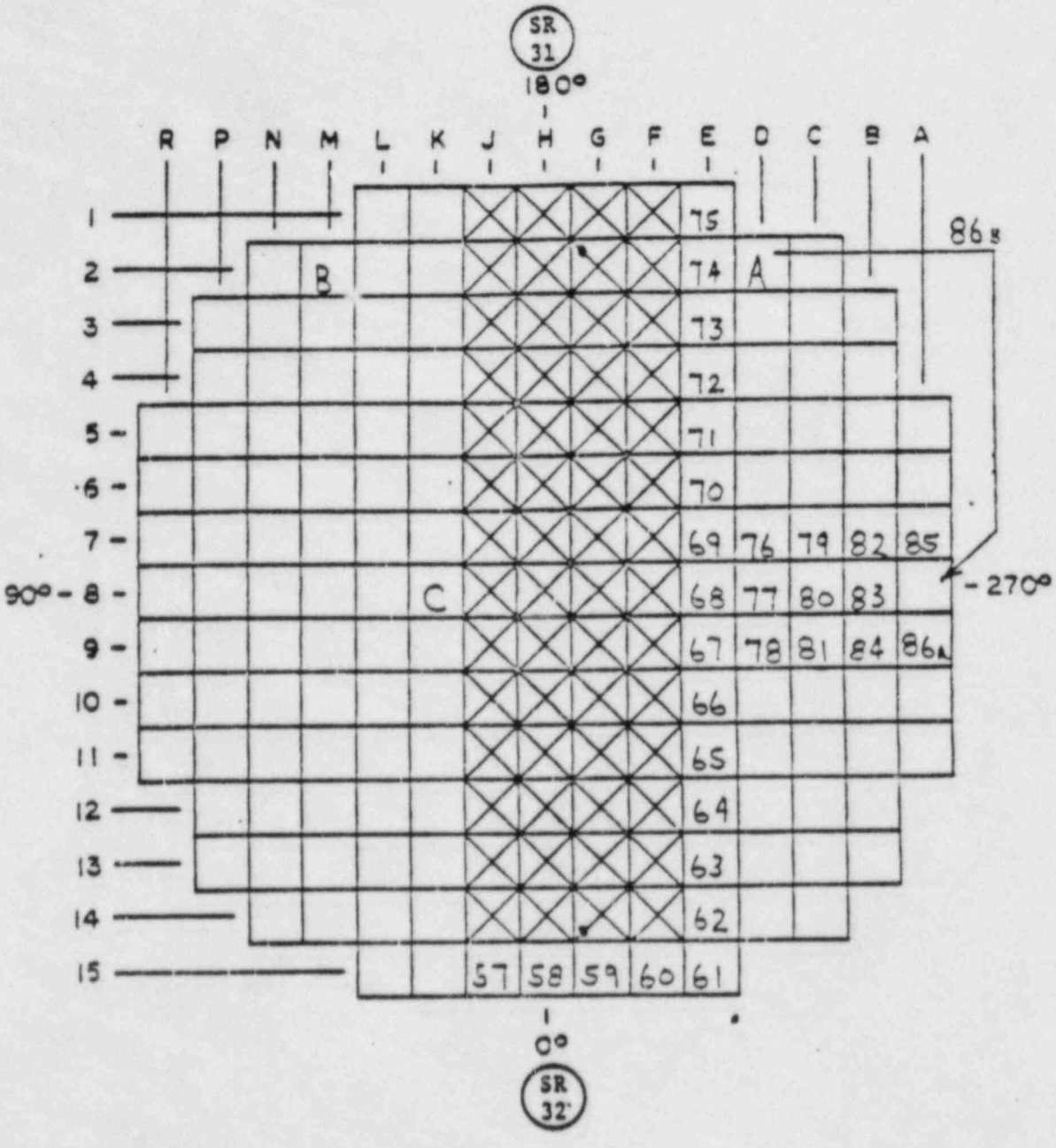



**Core Loading Sequence Steps 55b to 56a**

Step 55E - Raise, inspect and lower assembly A47 in core location J-09.

Figure 2.0.4 (cont.)

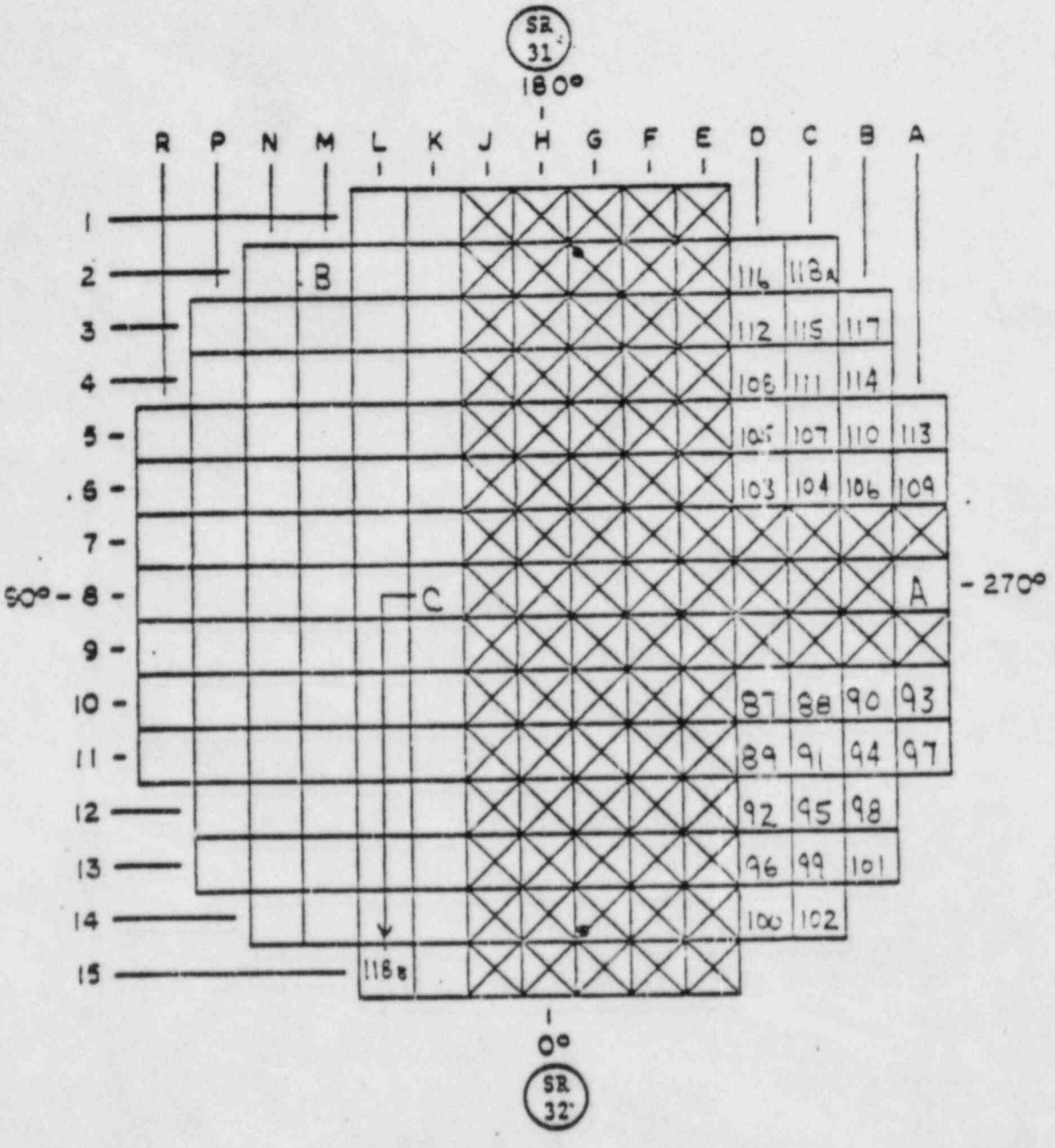
CORE LOADING SEQUENCE




Core Loading Sequence Steps 57 to 86s

Figure 2.0.4 (cont.)

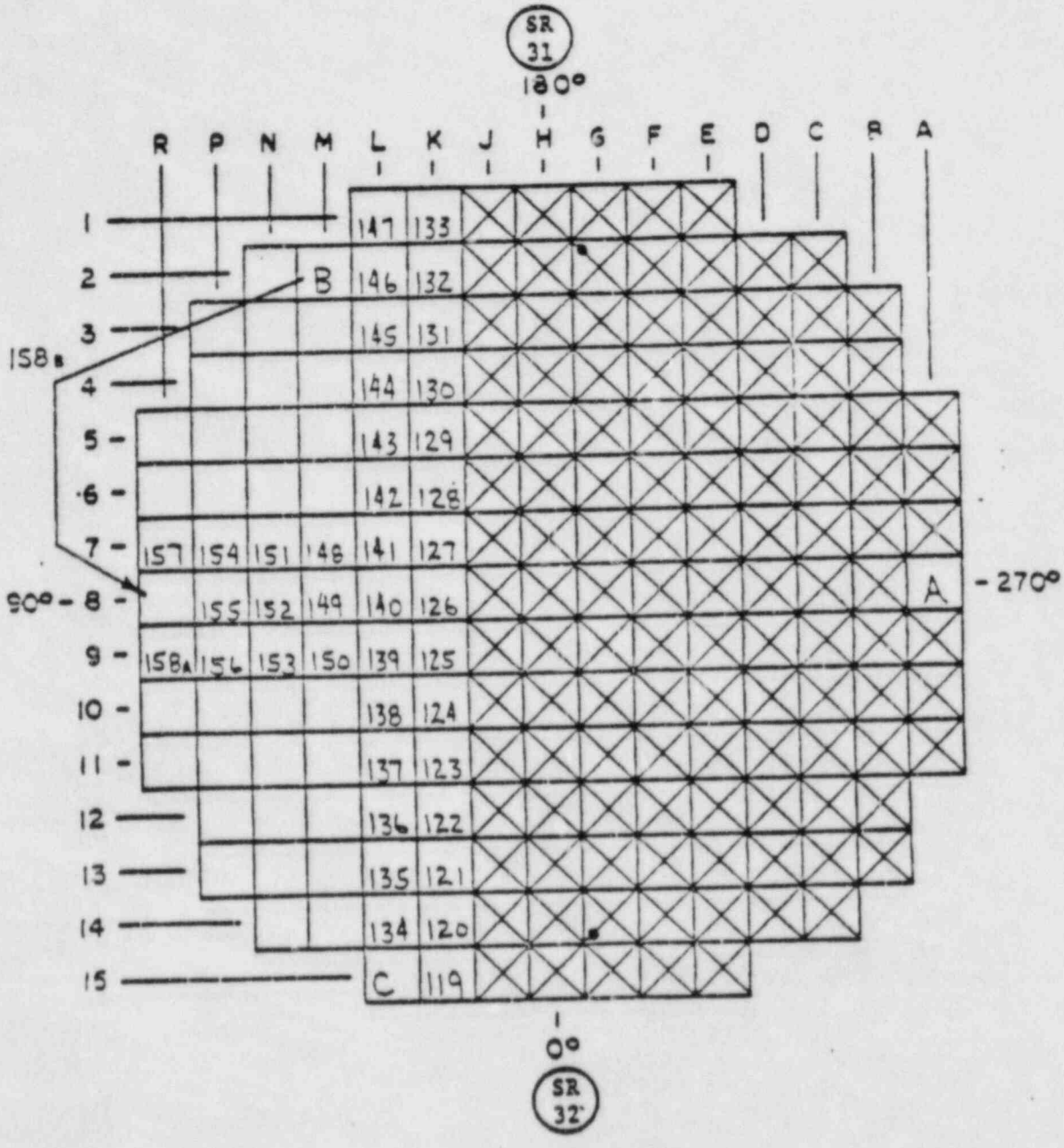
CORE LOADING SEQUENCE




Core Loading Sequence Steps 87 to 118B

Figure 2.0.4 (cont.)

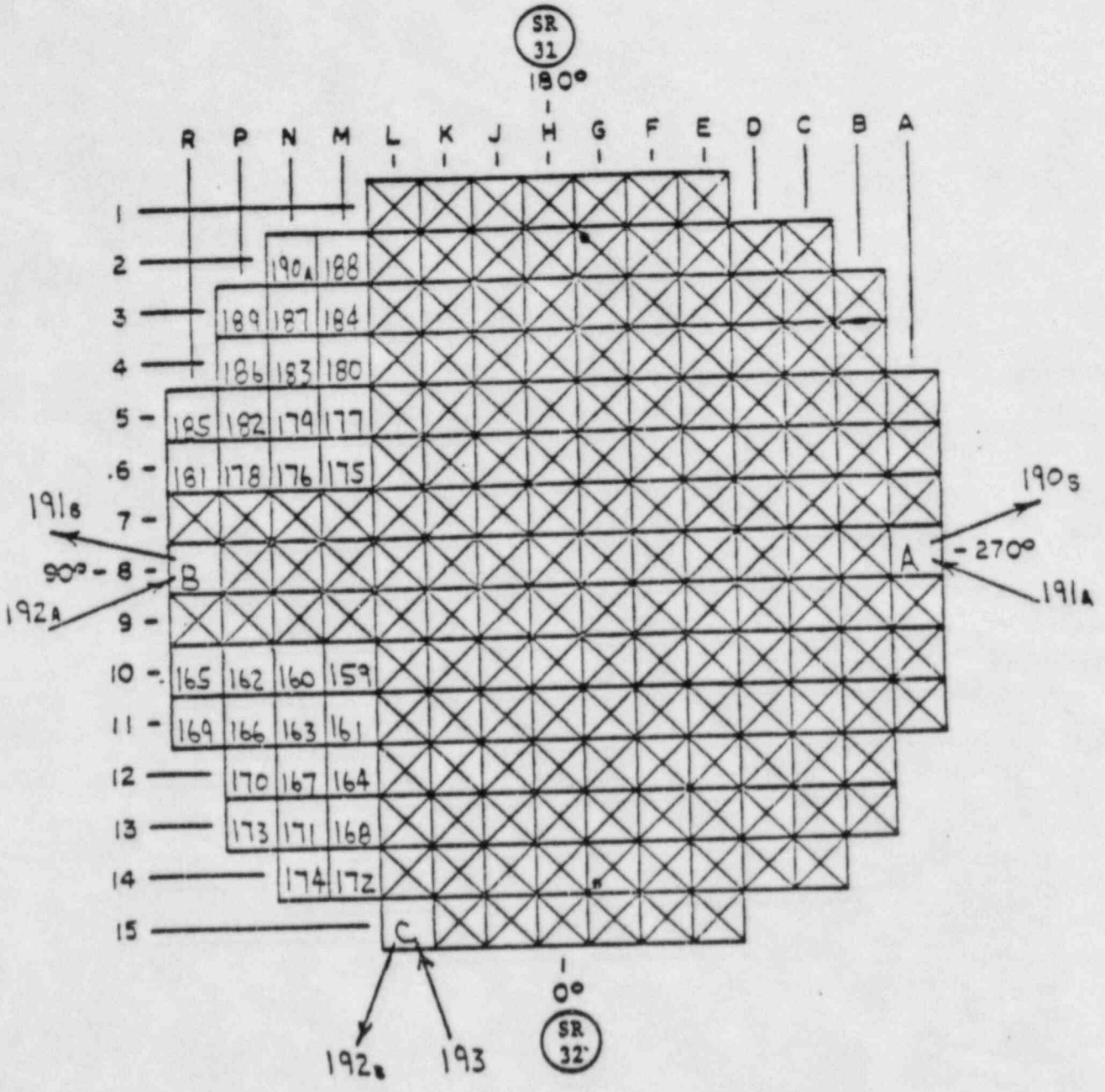
CORE LOADING SEQUENCE




Core Loading Sequence Steps 119 to 158B

Figure 2.0.4(cont.)

CORE LOADING SEQUENCE




Core Loading Sequence Steps 159 to 193

## 2.1 INVERSE COUNT RATE RATIO MONITORING FOR CORE LOADING (S-070002) Procedure ETT-ZZ-07020

### OBJECTIVES

The objectives of the subject procedure were:

1. To describe the methods for obtaining nuclear monitoring data during initial core loading.

2. To describe the sequence, frequency and core conditions for obtaining nuclear monitoring data in order to prevent criticality during core loading.

### SUMMARY OF RESULTS

The Acceptance Criterion for the subject procedure states that the core is loaded without achieving criticality. This criterion was met by observing the trend of core neutron multiplication during the loading using the inverse count rate ratio (ICRR) method. No abnormal changes in the ICRR were noted.

### DISCUSSION

Prior to the initiation of core loading three temporary neutron sensitive detectors ($BF_3$ ion chambers) were suspended in the reactor vessel in locations specified by the core loading sequence. Count rate data from the temporary detectors were obtained at the reactivity monitoring station in containment. Count rate data from source range channels N31 and N32 were obtained in the control room and transmitted to the reactivity monitoring station using headsets. The response from at least one responding detector was displayed continuously on a strip chart recorder during fuel movement within the reactor vessel.

Background readings were obtained for all detectors prior to loading the first assembly. Following step 10 of the core loading sequence, reference values were obtained for the responding detectors. After loading each subsequent fuel assembly, count rate data was collected and the ICRR calculated for each responding detector. The ICRR's were plotted to observe the trend of core neutron multiplication and ensure criticality would not be achieved upon loading the next assembly. The ICRR plots are presented as Figure 2.1.1 through 2.1.5. The count rate and ICRR data are presented as Table 2.1.1.

New reference values were obtained for the affected detector after significant equipment change, after detector position change or after a significant decrease in ICRR due to building a "bridge" between a primary source and a detector. New reference values were obtained prior to resuming core loading after a suspension of four hours or more. A chi-square statistical reliability test was performed on the data used for each reference value calculation.

While obtaining background detector data, it was observed that temporary detector A was spiking. This detector was declared inoperable

and was replaced by temporary detector D at step 22 of the core loading sequence. Temporary detector A was found to have an accumulation of moisture in it.

FIGURE 2.1.1

ICRR vs. NUMBER OF ASSEMBLIES LOADED

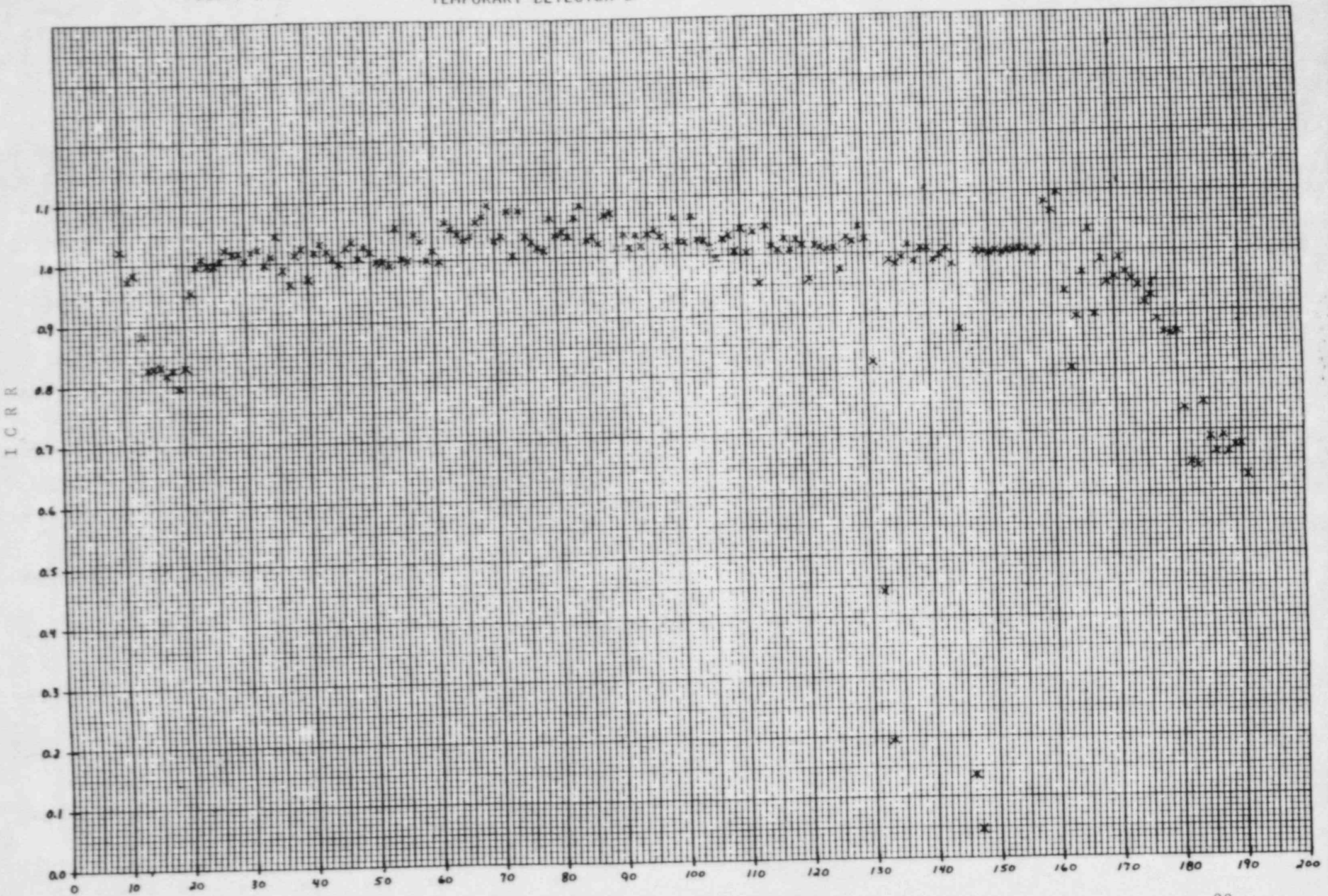
TEMPORARY DETECTOR A



FIGURE 2.1.2

ICRR vs. NUMBER OF ASSEMBLIES LOADED

TEMPORARY DETECTOR B



ICRR vs. NUMBER OF ASSEMBLIES LOADED

FIGURE 2.1.3 TEMPORARY DETECTOR C

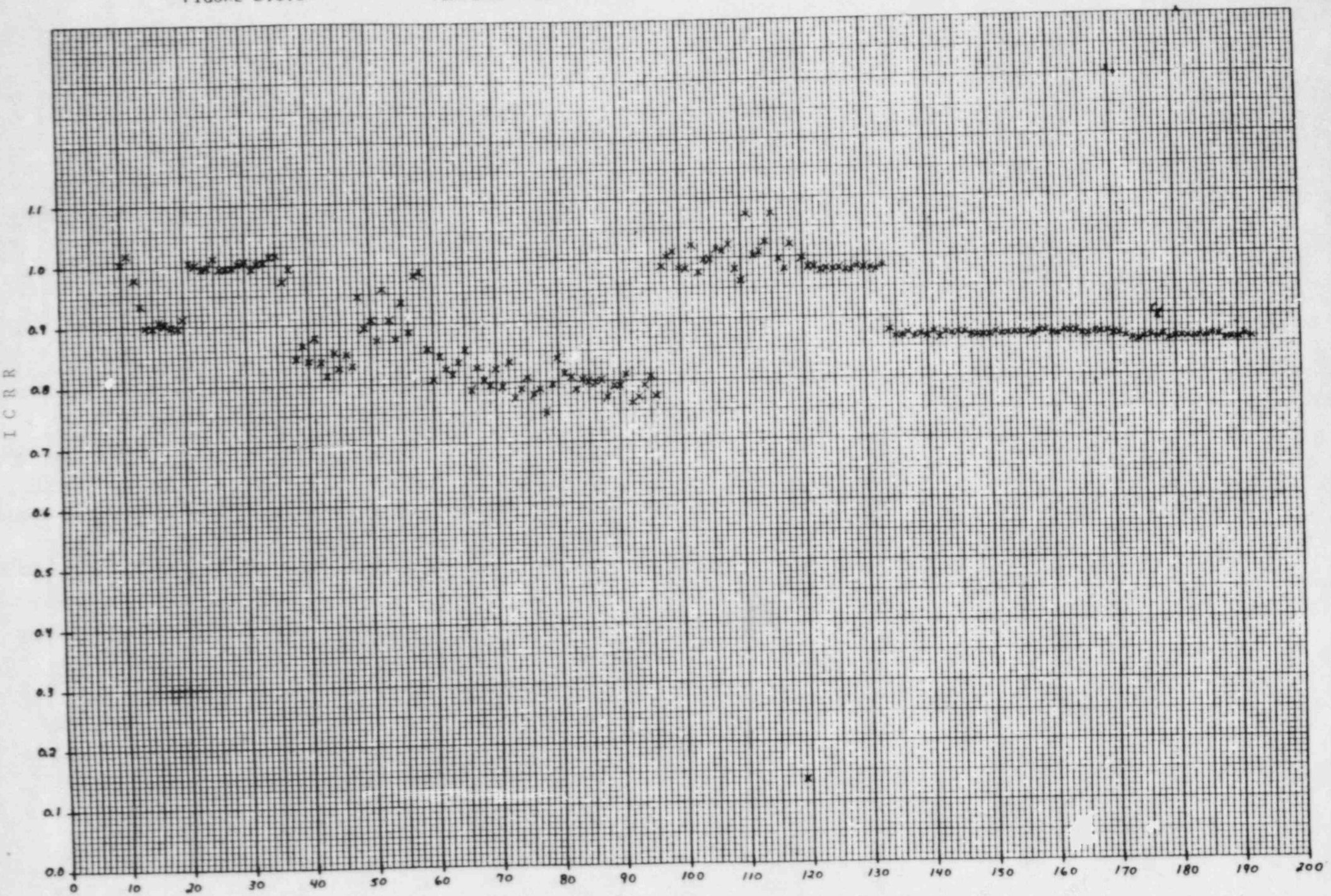


FIGURE 2.1.4

ICRR vs. NUMBER OF ASSEMBLIES LOADED

PLANT CHANNEL N31

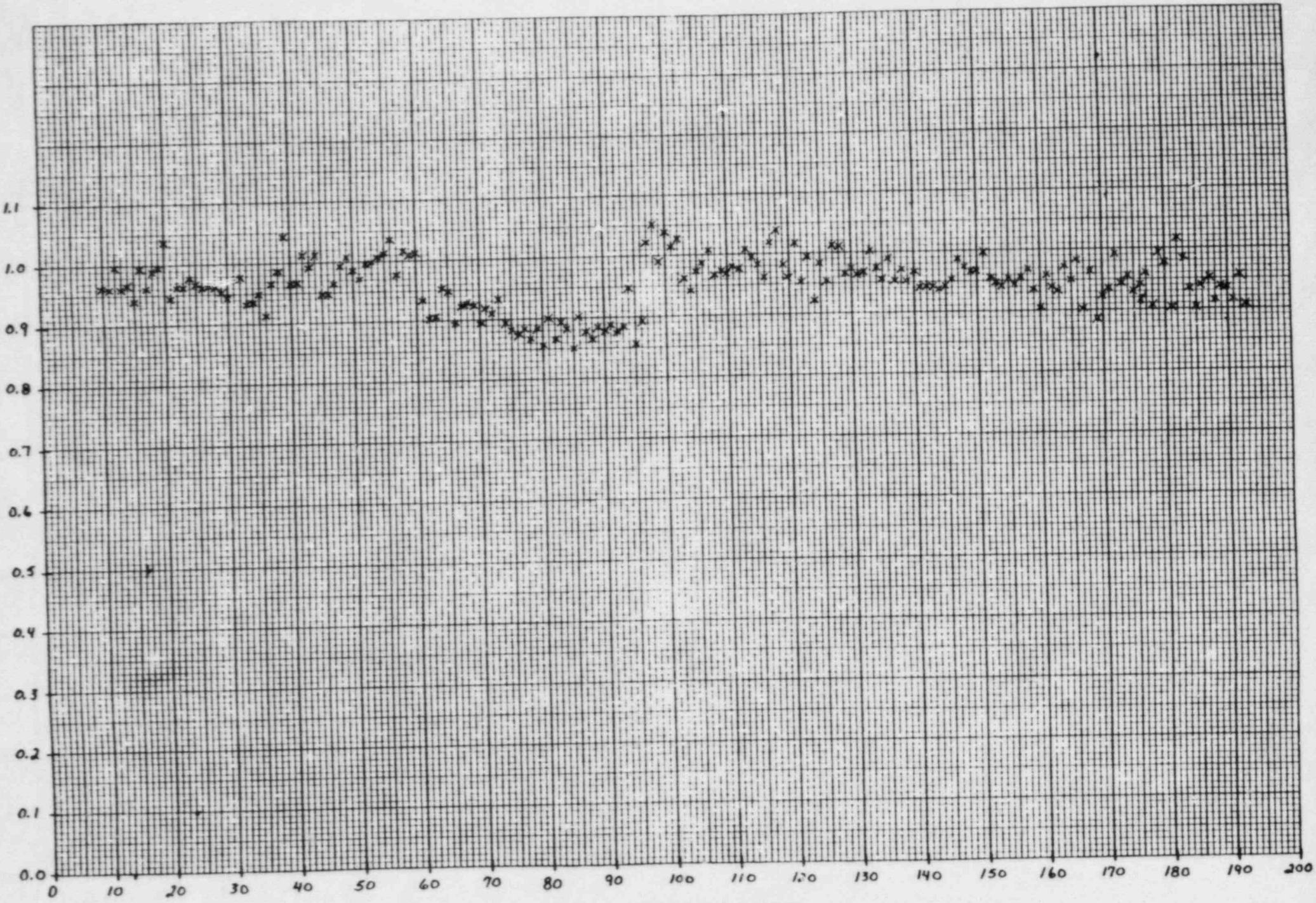


FIGURE 2.1.5

ICRR vs. NUMBER OF ASSEMBLIES LOADED

PLANT CHANNEL N32

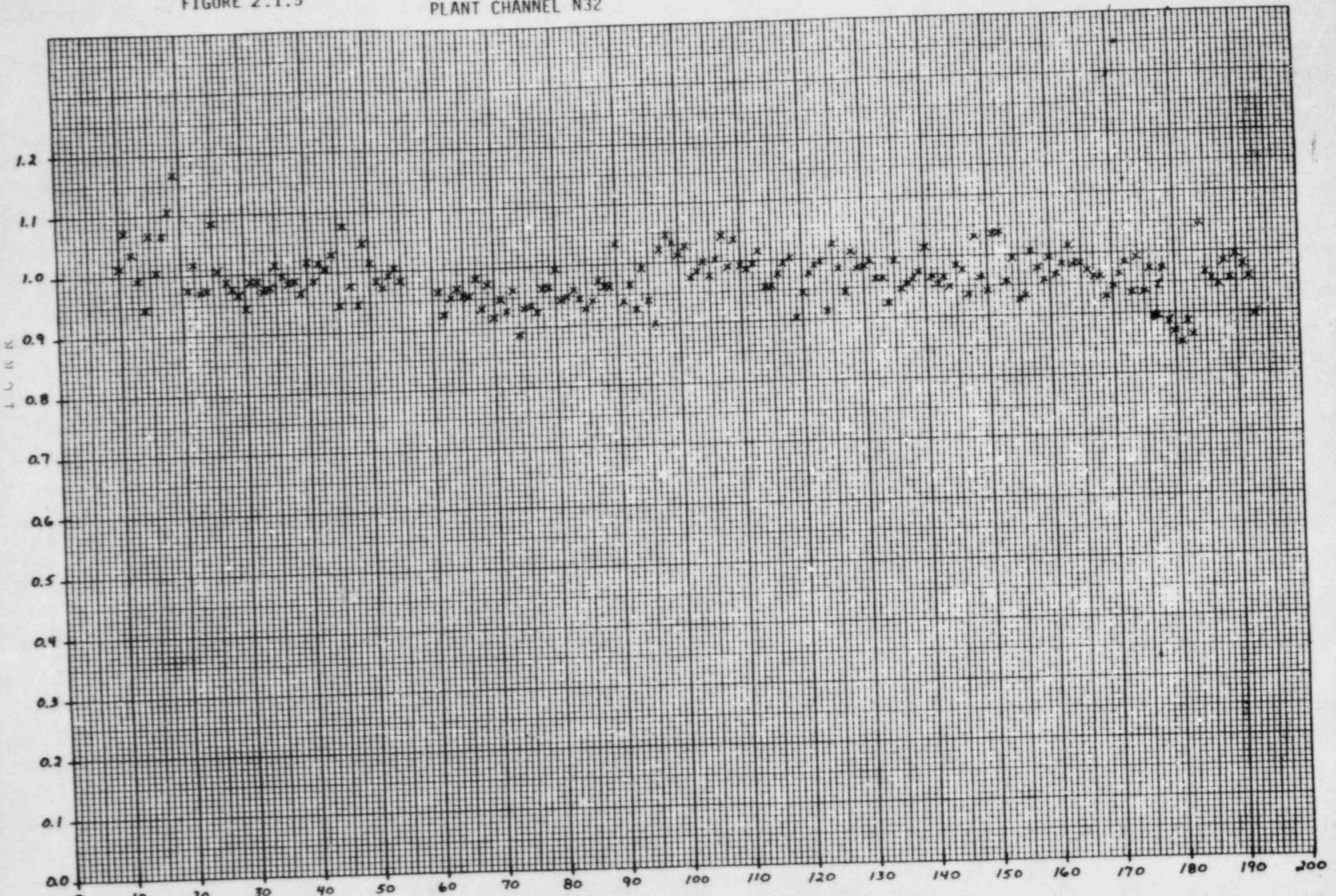


TABLE 2.1.1

SUMMARY OF NUCLEAR MONITORING DATA

| Date | Step | Assm I.D. | Core Pos. | Time | Counts | | | | | | | | | | ICCs | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temporary Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| | | | | | Value | Ref. Val | Value | Ref. Val | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | A | B | C | 31 | 32 |
| 6-13-84 | 0A | - | D02 | 1051 | | | | | | | | | | | | | | | |
| 6-13 | 0B | - | M02 | 1051 | | | | | | | | | | | | | | | |
| 6-13 | 0C | - | L01 | 1051 | | | | | | | | | | | | | | | |
| 6-13 | 1 | C04 | L15 | 1930 | | | | | | | | | | | | | | | |
| 6-13 | 2 | C30 | G01 | 1944 | | | | | | | | | | | | | | | |
| 6-13 | 3 | C12 | H01 | 2207 | | | | | | | | | | | | | | | |
| 6-13 | 4 | C60 | J01 | 2225 | | | | | | | | | | | | | | | |
| 6-13 | 5 | C01 | J02 | 2246 | | | | | | | | | | | | | | | |
| 6-13 | 6 | A22 | H02 | 2308 | | | | | | | | | | | | | | | |
| 6-14 | 7A | C31 | G02 | 0101 | | | | | | | | | | | | | | | |
| 6-14 | 7B | C31 | A11 | 0121 | | | | | | | | | | | | | | | |
| 6-14 | 7C | C04 | G02 | 0147 | | | | | | | | | | | | | | | |
| 6-14 | 7D | C30 | C14 | 0204 | | | | | | | | | | | | | | | |
| 6-14 | 8 | C07 | G01 | 0220 | | | | | | | | | | | | | | | |
| 6-14 | 9 | C50 | F01 | 0239 | | | | | | | | | | | | | | | |
| 6-14 | 10 | A16 | F02 | 0300 | | | 648.0 | 661.4 | 6195.1 | 6220.7 | 472.1 | 455.0 | 433.0 | 438.1 | | 1.021 | 1.004 | 0.964 | 1.012 |
| 6-14 | 11 | B19 | F03 | 0850 | | | 680.0 | | 6098.8 | | 473.8 | | 408.4 | | | 0.973 | 1.020 | 0.960 | 1.073 |
| 6-14 | 12 | A42 | G03 | 0909 | | | 671.7 | | 6355.1 | | 455.8 | | 422.4 | | | 0.985 | 0.979 | 0.998 | 1.037 |
| 6-14 | 13 | B45 | H03 | 0926 | | | 750.0 | | 6648.8 | | 474.1 | | 441.8 | | | 0.882 | 0.936 | 0.960 | 0.992 |
| 6-14 | 14 | A40 | J03 | 0944 | | | 799.0 | | 6912.1 | | 470.4 | | 463.4 | | | 0.828 | 0.900 | 0.967 | 0.945 |
| 6-14 | 15 | A01 | F04 | 1042 | | | 799.0 | | 6938.8 | | 484.1 | | 409.8 | | | 0.828 | 0.897 | 0.940 | 1.069 |
| 6-14 | 16 | B43 | G04 | 1059 | | | 797.0 | | 6857.1 | | 457.4 | | 435.4 | | | 0.830 | 0.907 | 0.995 | 1.006 |
| 6-14 | 17 | A52 | H04 | 1121 | | | 808.4 | | 6884.5 | | 473.8 | | 410.8 | | | 0.818 | 0.904 | 0.960 | 1.067 |
| 6-14 | 18 | B34 | J04 | 1336 | | | 803.0 | | 6915.5 | | 459.8 | | 395.8 | | | 0.824 | 0.900 | 0.990 | 1.107 |
| 6-14 | 19 | B09 | F05 | 1355 | | | 831.0 | | 6916.8 | | 456.8 | | 375.1 | | | 0.796 | 0.899 | 0.996 | 1.168 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| | | | | | Counts | | | | | | | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temporary Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| Date | Step | Assm I.D. | Core Pos. | Time | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | A | B | C | 31 | 32 |
| 6-14-84 | 20 | A23 | G05 | 1821 | | | 797.0 | 661.4 | 6805.5 | 6220.7 | 438.1 | 455.0 | | | | 0.830 | 0.914 | 1.038 | |
| 6-15 | 21 | B61 | H05 | 0214 | | | 854.0 | 813.1 | 6778.1 | 6815.6 | 481.1 | 454.2 | 215.8 | 210.6 | | 0.952 | 1.006 | 0.944 | 0.976 |
| 6-15 | 22 | A32 | J05 | 0231 | | | 816.4 | | 6802.1 | | 472.1 | | 206.8 | | | 0.996 | 1.002 | 0.962 | 1.019 |
| 6-15 | 23 | A10 | F06 | 2209 | 19072.7 | 19163.9 | 803.4 | 810.2 | 6776.1 | 6743.8 | 456.4 | 439.1 | 212.8 | 206.3 | 1.005 | 1.009 | 0.995 | 0.962 | 0.970 |
| 6-15 | 24 | B31 | G06 | 2250 | 19147.3 | | 811.4 | | 6752.5 | | 448.8 | | 211.8 | | 1.001 | 0.999 | 0.999 | 0.978 | 0.974 |
| 6-15 | 25 | A26 | H06 | 2309 | 19019.3 | | 812.0 | | 6666.5 | | 454.1 | | 190.1 | | 1.008 | 0.998 | 1.012 | 0.967 | 1.085 |
| 6-15 | 26 | B10 | J06 | 2331 | 18922.0 | | 807.4 | | 6790.1 | | 456.1 | | 204.4 | | 1.013 | 1.004 | 0.993 | 0.963 | 1.009 |
| 6-15 | 27 | B13 | F07 | 2350 | 19002.7 | | 791.7 | | 6777.1 | | 454.4 | | 209.1 | | 1.008 | 1.023 | 0.995 | 0.966 | 0.987 |
| 6-16 | 28 | A03 | G07 | 0008 | 19062.3 | | 795.7 | | 6772.8 | | 455.8 | | 211.1 | | 1.005 | 1.018 | 0.996 | 0.963 | 0.977 |
| 6-16 | 29 | B07 | H07 | 0028 | 18962.7 | | 797.7 | | 6738.8 | | 459.8 | | 214.1 | | 1.011 | 1.016 | 1.001 | 0.955 | 0.964 |
| 6-16 | 30 | A15 | J07 | 0047 | 19035.7 | | 808.7 | | 6789.5 | | 465.0 | | 218.4 | | 1.007 | 1.002 | 1.008 | 0.943 | 0.944 |
| 6-16 | 31 | A35 | F08 | 0106 | 18907.0 | | 794.7 | | 6792.1 | | 450.8 | | 209.1 | | 1.014 | 1.020 | 0.993 | 0.974 | 0.987 |
| 6-16 | 32 | B53 | G08 | 0124 | 19052.0 | | 792.4 | | 6733.1 | | 449.1 | | 208.8 | | 1.006 | 1.023 | 1.002 | 0.978 | 0.988 |
| 6-16 | 33 | A29 | H08 | 0145 | 18976.0 | | 812.7 | | 6708.5 | | 471.1 | | 212.4 | | 1.010 | 0.997 | 1.005 | 0.932 | 0.971 |
| 6-16 | 34 | B11 | J08 | 0204 | 19018.0 | | 801.0 | | 6649.1 | | 469.8 | | 211.1 | | 1.008 | 1.011 | 1.014 | 0.935 | 0.977 |
| 6-16 | 35 | B02 | F09 | 0825 | 18068 | 18160.3 | 772.4 | 809.0 | 291.1 | 295.9 | 475.4 | 451.6 | 206.1 | 209.5 | 1.005 | 1.047 | 1.016 | 0.950 | 1.016 |
| 6-16 | 36 | A44 | G09 | 0850 | 18058.3 | | 818.0 | | 304.1 | | 494.8 | | 210.4 | | 1.006 | 0.989 | 0.973 | 0.913 | 0.996 |
| 6-16 | 37 | B17 | H09 | 0905 | 18029.0 | | 841.4 | | 297.8 | | 467.4 | | 213.4 | | 1.007 | 0.962 | 0.994 | 0.966 | 0.982 |
| 6-16 | 38 | A47 | J09 | 0934 | 17943.7 | | 799.7 | | 351.5 | | 458.1 | | 212.1 | | 1.012 | 1.012 | 0.842 | 0.986 | 0.988 |
| 6-16 | 39 | A55 | F10 | 0955 | 17797.0 | | 791.2 | | 342.8 | | 433.4 | | 216.8 | | 1.020 | 1.022 | 0.863 | 1.042 | 0.966 |
| 6-16 | 40 | B01 | G10 | 1003 | 18020.0 | | 833.7 | | 352.1 | | 468.1 | | 206.1 | | 1.008 | 0.970 | 0.840 | 0.965 | 1.016 |
| 6-16 | 41 | A11 | H10 | 1020 | 17928.0 | | 795.7 | | 336.8 | | 467.4 | | 212.1 | | 1.013 | 1.017 | 0.879 | 0.966 | 0.988 |
| 6-16 | 42 | B23 | J10 | 1035 | 17841.7 | | 785.7 | | 353.1 | | 446.8 | | 206.4 | | 1.018 | 1.030 | 0.838 | 1.011 | 1.015 |
| 6-16 | 43 | B52 | F11 | 1051 | 17940.0 | | 795.0 | | 363.1 | | 454.8 | | 208.4 | | 1.012 | 1.018 | 0.815 | 0.993 | 1.005 |
| 6-16 | 44 | A20 | G11 | 1109 | 17750.3 | | 803.4 | | 347.8 | | 446.8 | | 203.4 | | 1.023 | 1.007 | 0.851 | 1.011 | 1.030 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| | | | | | Counts | | | | | | | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temporary Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| Date | Step | Assm I.D. | Core Pos. | Time | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | A | B | C | 31 | 32 |
| 6-16-84 | 45 | B58 | H11 | 1309 | 17521.7 | 18160.3 | 810.7 | 809.0 | 352.1 | 295.9 | 475.8 | 451.6 | 222.1 | 209.5 | 1.036 | 0.998 | 0.829 | 0.949 | 0.943 |
| 6-16 | 46 | A27 | J11 | 1336 | 17562.0 | | 790.7 | | 348.1 | | 475.8 | | 194.8 | | 1.034 | 1.023 | 0.850 | 0.949 | 1.076 |
| 6-16 | 47 | A07 | F12 | 1348 | 16209.7 | | 784.4 | | 356.5 | | 467.8 | | 214.1 | | 1.120 | 1.031 | 0.830 | 0.965 | 0.979 |
| 6-16 | 48 | B60 | G12 | 1470 | 15963.0 | | 805.7 | | 312.8 | | 454.8 | | 221.8 | | 1.138 | 1.004 | 0.946 | 0.993 | 0.945 |
| 6-16 | 49 | A12 | H12 | 1440 | 16027.3 | | 790.0 | | 332.1 | | 447.1 | | 200.1 | | 1.130 | 1.024 | 0.891 | 1.009 | 1.047 |
| 6-16 | 50 | B30 | J12 | 1508 | 16043.0 | | 795.7 | | 325.8 | | 458.4 | | 207.1 | | 1.132 | 1.017 | 0.908 | 0.985 | 1.012 |
| 6-16 | 51 | B25 | F13 | 1520 | 16042.3 | | 808.7 | | 339.8 | | 465.4 | | 213.4 | | 1.132 | 1.000 | 0.871 | 0.970 | 0.982 |
| 6-16 | 52 | A45 | G13 | 1539 | 15998.0 | | 808.7 | | 308.5 | | 453.1 | | 215.8 | | 1.135 | 1.000 | 0.959 | 0.997 | 0.971 |
| 6-16 | 53 | B44 | H13 | 1603 | 15966.0 | | 814.7 | | 326.1 | | 452.1 | | 211.1 | | 1.137 | 0.993 | 0.907 | 0.999 | 0.992 |
| 6-16 | 54 | A57 | J13 | 1624 | 16030.7 | | 765.4 | | 338.5 | | 448.8 | | 208.8 | | 1.133 | 1.057 | 0.874 | 1.006 | 1.004 |
| 6-16 | 55A | A19 | F14 | 1642 | 15999.0 | | 807.0 | | 316.1 | | 446.1 | | 213.4 | | 1.135 | 1.002 | 0.936 | 1.012 | 0.982 |
| 6-16 | 55B | C31 | G14 | 1709 | 16087.0 | | 793.0 | | 325.1 | | 454.4 | | 204.8 | | 1.129 | 1.020 | 0.910 | 0.994 | 1.023 |
| | | | | | | | | | | | | | | | | | | | |
| 6-16 | 55D | C30 | G14 | 2027 | 15569.3 | | 816.7 | | 447.5 | | 444.4 | | | | 1.166 | 0.991 | 0.661 | 1.016 | |
| 6-17 | 55E | A47 | J09 | 0021 | 15492.0 | 15606.5 | 797.4 | | 425.8 | 448.4 | 426.4 | | | | 1.007 | 1.015 | 0.942 | 1.059 | |
| 6-17 | 56A | A34 | H14 | 0040 | 15363.3 | | 809.0 | | 502.8 | | 436.4 | | | | 1.016 | 1.000 | 0.883 | 1.035 | |
| 6-17 | 56B | C31 | J14 | 0230 | 15356.3 | | 781.7 | | 490.5 | | 431.1 | | | | 1.016 | 1.035 | 0.914 | 1.048 | |
| 6-17 | 57 | C52 | J15 | 1044 | 15640.0 | 15482.4 | 797.0 | 832.0 | 510.5 | 500.4 | 439.1 | 429.8 | | | 0.990 | 1.044 | 0.980 | 0.979 | |
| 6-17 | 58 | C10 | H15 | 1101 | 15502.0 | | 806.4 | | 507.5 | | 423.4 | | | | 0.999 | 1.032 | 0.986 | 1.015 | |
| 6-17 | 59 | C37 | G15 | 1116 | 15562.3 | | 831.7 | | 584.1 | | 425.4 | | | | 0.995 | 1.000 | 0.857 | 1.010 | |
| 6-17 | 60 | C58 | F15 | 1132 | 15560.7 | | 817.7 | | 619.8 | | 424.1 | | | | 0.995 | 1.017 | 0.807 | 1.013 | |
| 6-17 | 61 | C29 | E15 | 1330 | 14862.3 | | 833.7 | | 592.8 | | 460.8 | | 406.4 | 391.7 | 1.042 | 0.998 | 0.844 | 0.933 | 0.964 |
| 6-17 | 62 | C42 | E14 | 1347 | 14775.7 | | 783.7 | | 609.8 | | 476.1 | | 422.8 | | 1.048 | 1.062 | 0.821 | 0.903 | 0.927 |
| 6-17 | 63 | A54 | E13 | 1411 | 14947.7 | | 791.4 | | 614.1 | | 473.1 | | 411.8 | | 1.036 | 1.051 | 0.815 | 0.908 | 0.951 |
| 6-17 | 64 | B21 | E12 | 1429 | 14870.0 | | 797.7 | | 599.5 | | 450.8 | | 404.4 | | 1.041 | 1.043 | 0.835 | 0.953 | 0.969 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| | | | | | Counts | | | | | | | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temporary Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| Date | Step | Assm I.D. | Core Pos. | Time | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val | A | B | C | 31 | 32 |
| 6-17-84 | 65 | A59 | E11 | 1459 | 14787.0 | 15482.4 | 806.0 | 832.0 | 586.8 | 500.4 | 453.1 | 429.8 | 409.8 | 391.7 | 1.047 | 1.032 | 0.853 | 0.949 | 0.956 |
| 6-17 | 66 | B26 | E10 | 1519 | 14927.0 | | 799.7 | | 635.1 | | 480.4 | | 410.8 | | 1.037 | 1.040 | 0.788 | 0.895 | 0.954 |
| 6-17 | 67 | A56 | E09 | 1539 | 14840.3 | | 783.7 | | 606.5 | | 465.1 | | 398.8 | | 1.043 | 1.062 | 0.825 | 0.924 | 0.982 |
| 6-17 | 68 | B32 | E08 | 1558 | 14862.3 | | 776.7 | | 621.8 | | 462.8 | | 419.8 | | 1.042 | 1.071 | 0.805 | 0.929 | 0.933 |
| 6-17 | 69 | A63 | E07 | 1617 | 14709.3 | | 763.0 | | 629.8 | | 465.8 | | 402.8 | | 1.053 | 1.090 | 0.795 | 0.923 | 0.923 |
| 6-17 | 70 | B63 | E06 | 1636 | 14864.0 | | 808.0 | | 602.8 | | 481.4 | | 426.4 | | 1.042 | 1.030 | 0.823 | 0.893 | 0.919 |
| 6-17 | 71 | A14 | E05 | 1653 | 14838.7 | | 800.7 | | 631.1 | | 467.1 | | 412.8 | | 1.043 | 1.039 | 0.793 | 0.920 | 0.949 |
| 6-17 | 72 | B49 | E04 | 1715 | 15006.7 | | 770.0 | | 601.8 | | 471.1 | | 421.8 | | 1.032 | 1.080 | 0.832 | 0.912 | 0.929 |
| 6-17 | 73 | A33 | E03 | 1738 | 17125.3 | | 828.4 | | 646.5 | | 460.1 | | 407.8 | | 0.904 | 1.004 | 0.774 | 0.934 | 0.961 |
| 6-17 | 74 | C28 | E02 | 1817 | 84090.7 | | 770.0 | | 634.5 | | 480.1 | | 442.1 | | 0.184 | 1.080 | 0.789 | 0.895 | 0.886 |
| 6-17 | 75 | C26 | E01 | 1847 | 234922 | | 802.4 | | 618.5 | | 486.8 | | 420.8 | | 0.066 | 1.037 | 0.809 | 0.883 | 0.931 |
| 6-17 | 76 | B06 | D07 | 1915 | 234911 | 235193 | 810.0 | | 641.5 | | 490.4 | | 418.8 | | 1.001 | 1.027 | 0.780 | 0.876 | 0.935 |
| 6-17 | 77 | A21 | D08 | 1943 | 235005 | | 816.7 | | 634.5 | | 487.1 | | 422.8 | | 1.001 | 1.019 | 0.789 | 0.882 | 0.927 |
| 6-17 | 78 | B64 | D09 | 2002 | 234797 | | 822.0 | | 647.5 | | 495.1 | | 406.1 | | 1.002 | 1.012 | 0.750 | 0.868 | 0.965 |
| 6-17 | 79 | A61 | C07 | 2021 | 234729 | | 780.7 | | 628.8 | | 488.1 | | 406.1 | | 1.002 | 1.066 | 0.796 | 0.881 | 0.965 |
| 6-17 | 80 | B57 | C08 | 2038 | 235064 | | 800.7 | | 595.5 | | 502.8 | | 394.8 | | 1.001 | 1.039 | 0.840 | 0.855 | 0.992 |
| 6-17 | 81 | A48 | C09 | 2057 | 234814 | | 799.4 | | 614.5 | | 477.8 | | 416.1 | | 1.002 | 1.041 | 0.814 | 0.900 | 0.941 |
| 6-17 | 82 | C55 | B07 | 2115 | 235302 | | 803.7 | | 619.5 | | 492.1 | | 412.1 | | 1.000 | 1.035 | 0.808 | 0.865 | 0.950 |
| 6-17 | 83 | A24 | B08 | 2134 | 235181 | | 779.7 | | 635.8 | | 482.4 | | 407.8 | | 1.000 | 1.067 | 0.787 | 0.891 | 0.961 |
| 6-17 | 84 | C11 | B09 | 2153 | 235149 | | 764.7 | | 623.8 | | 487.8 | | 414.4 | | 1.000 | 1.088 | 0.802 | 0.881 | 0.945 |
| 6-17 | 85 | C40 | A07 | 2209 | 234611 | | 809.4 | | 625.8 | | 505.8 | | 421.4 | | 1.002 | 1.028 | 0.800 | 0.850 | 0.929 |
| 6-17 | 86 | C24 | A09 | 2226 | 235166 | | 807.4 | | 625.5 | | 476.8 | | 416.8 | | 1.000 | 1.031 | 0.800 | 0.901 | 0.940 |
| 6-18 | 87 | A04 | D10 | 0014 | 261.3 | 248.0 | 813.7 | | 624.8 | | 490.8 | | 402.4 | | 0.949 | 1.022 | 0.801 | 0.876 | 0.973 |
| 6-18 | 88 | B05 | C10 | 0032 | 279.0 | | 779.4 | | 648.5 | | 492.1 | | 405.8 | | 0.889 | 1.070 | 0.722 | 0.865 | 0.965 |
| 6-18 | 89 | B40 | D11 | 0049 | 250.0 | | 775.7 | | 633.1 | | 487.4 | | 406.4 | | 0.992 | 1.073 | 0.790 | 0.882 | 0.964 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| Date | Step | Assm I.D. | Core Pos. | Time | Counts | | | | | | | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temporary Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| | | | | | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | A | B | C | 31 | 32 |
| 6-18-84 | 90 | A39 | B10 | 0106 | 299.0 | 248.0 | 820.4 | 832.0 | 630.8 | 500.4 | 488.8 | 429.8 | 378.8 | 391.7 | 0.830 | 1.014 | 0.793 | 0.829 | 1.034 |
| 6-18 | 91 | A65 | C11 | 0125 | 286.3 | | 803.0 | | 617.5 | | 483.4 | | 417.1 | | 0.866 | 1.036 | 0.810 | 0.889 | 0.939 |
| 6-18 | 92 | B27 | D12 | 0143 | 302.6 | | 812.7 | | 654.8 | | 491.1 | | 405.4 | | 0.819 | 1.017 | 0.764 | 0.875 | 0.966 |
| 6-18 | 93 | C39 | A10 | 0200 | 369.6 | | 804.0 | | 648.1 | | 486.1 | | 422.4 | | 0.671 | 1.035 | 0.722 | 0.884 | 0.927 |
| 6-18 | 94 | C64 | B11 | 0215 | 365.3 | | 816.4 | | 632.1 | | 453.1 | | 394.8 | | 0.679 | 1.019 | 0.792 | 0.949 | 0.992 |
| 6-18 | 95 | B55 | C12 | 0231 | 362.3 | | 801.0 | | 620.1 | | 504.1 | | 417.1 | | 0.685 | 1.039 | 0.802 | 0.853 | 0.939 |
| 6-18 | 96 | B08 | D13 | 0248 | 388.6 | | 797.7 | | 646.5 | | 482.1 | | 435.1 | | 0.638 | 1.043 | 0.774 | 0.892 | 0.900 |
| 6-19 | 97 | C49 | A11 | 0049 | 415.6 | 380.1 | 793.4 | 818.4 | 648.8 | 639.5 | 493.4 | 503.9 | 399.8 | 408.3 | 0.915 | 1.032 | 0.986 | 1.021 | 1.021 |
| 6-19 | 98 | C18 | B12 | 0108 | 409.0 | | 805.0 | | 637.5 | | 478.8 | | 389.8 | | 0.929 | 1.017 | 1.003 | 1.052 | 1.048 |
| 6-19 | 99 | C33 | C13 | 0125 | 447.6 | | 770.0 | | 632.8 | | 509.1 | | 396.1 | | 0.849 | 1.063 | 1.011 | 0.990 | 1.031 |
| 6-19 | 100 | C57 | D14 | 0146 | 414.3 | | 800.4 | | 649.5 | | 485.1 | | 403.8 | | 0.917 | 1.023 | 0.985 | 1.039 | 1.011 |
| 6-19 | 101 | C46 | B13 | 0213 | 462.0 | | 799.4 | | 650.8 | | 496.4 | | 397.4 | | 0.823 | 1.024 | 0.983 | 1.015 | 1.027 |
| 6-19 | 102 | C48 | C14 | 0231 | 473.6 | | 768.0 | | 626.1 | | 489.8 | | 420.1 | | 0.803 | 1.066 | 1.021 | 1.029 | 0.972 |
| 6-19 | 103 | A46 | D06 | 0830 | 469.3 | 465.2 | 795.4 | 815.7 | 666.8 | 651.1 | 504.1 | 484.7 | 423.8 | 417.3 | 0.991 | 1.026 | 0.976 | 0.962 | 0.985 |
| 6-19 | 104 | B14 | C06 | 0850 | 469.3 | | 794.0 | | 652.8 | | 515.1 | | 417.1 | | 0.991 | 1.027 | 0.997 | 0.941 | 1.000 |
| 6-19 | 105 | B22 | D05 | 0911 | 482.0 | | 805.4 | | 651.5 | | 497.1 | | 427.1 | | 0.955 | 1.013 | 0.999 | 0.925 | 0.977 |
| 6-19 | 106 | A62 | B06 | 0930 | 483.0 | | 818.7 | | 641.8 | | 490.1 | | 416.1 | | 0.963 | 0.996 | 1.014 | 0.989 | 1.003 |
| 6-19 | 107 | A28 | C05 | 0950 | 480.0 | | 793.4 | | 650.1 | | 481.4 | | 400.1 | | 0.969 | 1.028 | 1.001 | 1.007 | 1.043 |
| 6-19 | 108 | B59 | D04 | 1007 | 518.0 | | 790.4 | | 632.1 | | 501.1 | | 421.4 | | 0.898 | 1.032 | 1.022 | 0.967 | 0.990 |
| 6-19 | 109 | C34 | A06 | 1024 | 523.0 | | 809.4 | | 664.1 | | 497.4 | | 402.4 | | 0.890 | 1.008 | 0.980 | 0.974 | 1.037 |
| 6-19 | 110 | C25 | B05 | 1042 | 538.3 | | 780.4 | | 677.5 | | 499.8 | | 420.4 | | 0.864 | 1.045 | 0.961 | 0.970 | 0.993 |
| 6-19 | 111 | B39 | C04 | 1058 | 522.0 | | 813.4 | | 607.5 | | 494.4 | | 422.4 | | 0.891 | 1.003 | 1.072 | 0.980 | 0.988 |
| 6-19 | 112 | B47 | D03 | 1117 | 538.6 | | 785.4 | | 649.1 | | 496.4 | | 419.8 | | 0.864 | 1.039 | 1.003 | 0.976 | 0.994 |
| 6-19 | 113 | C45 | A05 | 1134 | 563.6 | | 856.0 | | 645.1 | | 480.8 | | 411.4 | | 0.825 | 0.953 | 1.009 | 1.008 | 1.014 |
| 6-19 | 114 | C05 | B04 | 1224 | 528.6 | | 778.7 | | 635.1 | | 487.1 | | 435.8 | | 0.880 | 1.048 | 1.025 | 0.995 | 0.958 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| | | | | | Core/Temporary Channels | | | | | | Plant Channels | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| Date | Step | Assem I.D. | Core Pos. | Time | Value | Ref. Val. | Value | Ref. Val | Value | Ref. Val | Value | Ref. Val. | Value | Ref. Val. | A | B | C | 31 | 32 |
| 6-19-84 | 115 | C15 | C03 | 1242 | 534.9 | 465.2 | 803.7 | 815.7 | 607.1 | 651.1 | 492.7 | 484.7 | 435.7 | 417.3 | 0.869 | 1.014 | 1.072 | 0.983 | 0.958 |
| 6-19 | 116 | C02 | D02 | 1303 | 527.3 | | 808.0 | | 653.1 | | 504.1 | | 426.1 | | 0.882 | 1.009 | 0.997 | 0.961 | 0.979 |
| 6-19 | 117 | C17 | B03 | 1322 | 521.0 | | 793.4 | | 664.1 | | 475.1 | | 419.8 | | 0.893 | 1.028 | 0.980 | 1.020 | 0.994 |
| 6-19 | 118 | C63 | C02 | 1340 | 529.0 | | 808.7 | | 638.5 | | 466.8 | | 416.4 | | 0.804 | 1.009 | 1.020 | 1.038 | 1.002 |
| 6-19 | 119 | C14 | K15 | 1538 | 560.3 | | 795.0 | | 53048 | 7161.2 | 494.4 | | 462.8 | | 0.830 | 1.026 | 0.135 | 0.980 | 0.902 |
| 6-19 | 120 | A09 | K14 | 1558 | 528.0 | | 803.0 | | 69420 | 69164.5 | 503.8 | | 441.4 | | 0.805 | 1.016 | 0.996 | 0.962 | 0.945 |
| 6-19 | 121 | B41 | K13 | 1619 | 562.6 | | 849.7 | | 70438 | | 477.1 | | 427.4 | | 0.827 | 0.960 | 0.982 | 1.016 | 0.976 |
| 6-19 | 122 | A31 | K12 | 1710 | 524.3 | | 803.4 | | 70384 | | 508.4 | | 421.4 | | 0.810 | 1.015 | 0.983 | 0.953 | 0.990 |
| 6-19 | 123 | B42 | K11 | 1729 | 562.6 | | 806.0 | | 70673 | | 488.8 | | 419.1 | | 0.827 | 1.012 | 0.979 | 0.992 | 0.996 |
| 6-19 | 124 | A18 | K10 | 1757 | 554.6 | | 808.4 | | 70565 | | 526.8 | | 456.8 | | 0.839 | 1.009 | 0.980 | 0.920 | 0.914 |
| 6-19 | 125 | B33 | K09 | 1816 | 545.0 | | 807.4 | | 70635 | | 493.8 | | 408.1 | | 0.854 | 1.010 | 0.979 | 0.982 | 1.023 |
| 6-19 | 126 | A60 | K08 | 1848 | 560.0 | | 839.4 | | 70598 | | 509.8 | | 424.4 | | 0.831 | 0.972 | 0.980 | 0.951 | 0.983 |
| 6-19 | 127 | B15 | K07 | 1906 | 563.3 | | 794.0 | | 70849 | | 479.4 | | 442.4 | | 0.862 | 1.027 | 0.976 | 1.011 | 0.943 |
| 6-19 | 128 | A41 | K06 | 1930 | 560.3 | | 800.0 | | 70788 | | 479.8 | | 413.1 | | 0.830 | 1.020 | 0.977 | 1.010 | 1.010 |
| 6-19 | 129 | B20 | K05 | 1950 | 562.3 | | 780.7 | | 70414 | | 501.8 | | 424.4 | | 0.827 | 1.045 | 0.982 | 0.966 | 0.983 |
| 6-19 | 130 | A43 | K04 | 2011 | 549.0 | | 797.7 | | 70501 | | 497.1 | | 423.8 | | 0.847 | 1.023 | 0.981 | 0.975 | 0.985 |
| 6-19 | 131 | B03 | K03 | 2028 | 560.5 | | 994.7 | | 70557 | | 504.1 | | 420.4 | | 0.830 | 0.820 | 0.980 | 0.962 | 0.993 |
| 6-19 | 132 | A38 | K02 | 2049 | 573.6 | | 1840.0 | | 70781 | | 500.8 | | 431.8 | | 0.811 | 0.443 | 0.977 | 0.968 | 0.966 |
| 6-19 | 133 | C43 | K01 | 2109 | 578.6 | | 4197.7 | | 70286 | | 484.4 | | 431.8 | | 0.804 | 0.194 | 0.984 | 1.001 | 0.966 |
| 6-19 | 134 | C21 | L14 | 2139 | 536.0 | | 4234.0 | 4186.9 | 78909 | | 498.1 | | 451.8 | | 0.868 | 0.989 | 0.877 | 0.973 | 0.924 |
| 6-19 | 135 | A08 | L13 | 2158 | 562.0 | | 4262.7 | | 79295 | | 505.8 | | 420.4 | | 0.828 | 0.982 | 0.867 | 0.958 | 2993 |
| 6-19 | 136 | B51 | L12 | 2219 | 571.3 | | 4214.0 | | 79620 | | 489.8 | | 440.8 | | 0.814 | 0.994 | 0.869 | 0.990 | 0.947 |
| 6-19 | 137 | A50 | L11 | 2234 | 558.3 | | 4132.0 | | 79383 | | 508.4 | | 436.1 | | 0.833 | 1.013 | 0.871 | 0.953 | 0.957 |
| 6-19 | 138 | B35 | L10 | 2253 | 563.3 | | 4248.0 | | 79743 | | 499.4 | | 434.1 | | 0.826 | 0.986 | 0.867 | 0.970 | 0.961 |
| 6-19 | 139 | A64 | L09 | 2321 | 567.6 | | 4154.7 | | 79499 | | 509.1 | | 428.1 | | 0.820 | 1.008 | 0.870 | 0.952 | 0.975 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| | | | | | Counts | | | | | | | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temperature Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| Date | Step | Assm I.D. | Core Pos. | Time | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | A | B | C | 31 | 32 |
| 6-19 | 140 | B38 | L08 | 2346 | 552.6 | 465.2 | 4155.4 | 41869 | 79828 | 69164.5 | 501.1 | 484.7 | 411.1 | 417.3 | 0.842 | 1.008 | 0.866 | 0.967 | 1.015 |
| 6-20 | 141 | A53 | L07 | 0047 | 565.3 | | 4234.0 | | 79251 | | 515.1 | | 431.4 | | 0.823 | 0.989 | 0.873 | 0.941 | 0.967 |
| 6-20 | 142 | B42 | L06 | 0106 | 586.6 | | 4209.4 | | 79829 | | 514.1 | | 437.8 | | 0.793 | 0.995 | 0.866 | 0.943 | 0.953 |
| 6-20 | 143 | A51 | L05 | 0122 | 572.3 | | 4165.4 | | 79389 | | 515.1 | | 432.1 | | 0.813 | 1.005 | 0.871 | 0.941 | 0.966 |
| 6-20 | 144 | B48 | L04 | 0139 | 574.6 | | 4270.4 | | 79515 | | 516.4 | | 439.4 | | 0.810 | 0.980 | 0.870 | 0.939 | 0.950 |
| 6-20 | 145 | A30 | L03 | 0155 | 577.3 | | 4288.4 | | 79331 | | 514.4 | | 424.1 | | 0.806 | 0.874 | 0.872 | 0.942 | 0.984 |
| 6-20 | 146 | C54 | L02 | 0211 | 574.0 | | 30933 | | 79374 | | 508.1 | | 426.1 | | 0.811 | 0.135 | 0.871 | 0.954 | 0.979 |
| 6-20 | 147 | C27 | L01 | 0230 | 558.6 | | 88986 | | 79638 | | 490.8 | | 445.4 | | 0.833 | 0.047 | 0.868 | 0.988 | 0.937 |
| 6-20 | 148 | B12 | M07 | 0303 | 555.3 | | 89001 | 89086 | 79681 | | 499.1 | | 405.1 | | 0.838 | 1.001 | 0.868 | 0.971 | 1.030 |
| 6-20 | 149 | A36 | M08 | 0324 | 562.0 | | 89121 | | 79752 | | 501.4 | | 433.4 | | 0.828 | 1.000 | 0.867 | 0.967 | 0.963 |
| 6-20 | 150 | B50 | M09 | 0342 | 568.3 | | 89170 | | 79781 | | 501.1 | | 444.1 | | 0.819 | 0.999 | 0.867 | 0.967 | 0.940 |
| 6-20 | 151 | A13 | N07 | 0401 | 547.0 | | 89004 | | 79449 | | 487.1 | | 403.8 | | 0.851 | 1.001 | 0.871 | 0.995 | 1.034 |
| 6-20 | 152 | B18 | N08 | 0420 | 564.6 | | 89131 | | 79558 | | 502.4 | | 402.4 | | 0.824 | 0.999 | 0.869 | 0.955 | 1.037 |
| 6-20 | 153 | A58 | N09 | 0439 | 555.6 | | 89023 | | 79500 | | 512.4 | | 437.1 | | 0.837 | 1.001 | 0.870 | 0.946 | 0.955 |
| 6-20 | 154 | C03 | P07 | 0500 | 559.6 | | 88964 | | 79471 | | 514.4 | | 419.4 | | 0.831 | 1.001 | 0.870 | 0.942 | 0.995 |
| 6-20 | 155 | A37 | P08 | 0518 | 570.0 | | 88654 | | 79457 | | 509.4 | | 450.4 | | 0.816 | 1.005 | 0.870 | 0.951 | 0.926 |
| 6-20 | 156 | C61 | P09 | 0536 | 575.0 | | 89022 | | 79375 | | 512.8 | | 448.1 | | 0.809 | 1.001 | 0.871 | 0.945 | 0.931 |
| 6-20 | 157 | C41 | R07 | 0555 | 566.3 | | 89199 | | 79643 | | 506.8 | | 415.1 | | 0.821 | 0.999 | 0.868 | 0.956 | 1.005 |
| 6-20 | 158 | C20 | R09 | 0612 | 564.3 | | 88916 | | 79169 | | 500.4 | | 426.8 | | 0.824 | 1.002 | 0.874 | 0.969 | 0.978 |
| 6-20 | 159 | A17 | M10 | 0804 | 555.0 | | 358.7 | 387.8 | 79143 | | 518.1 | | 435.4 | | 0.838 | 1.081 | 0.874 | 0.936 | 0.958 |
| 6-20 | 160 | B56 | N10 | 0823 | 547.6 | | 363.4 | | 79488 | | 530.4 | | 419.8 | | 0.849 | 1.067 | 0.870 | 0.914 | 0.994 |
| 6-20 | 161 | B04 | M11 | 0839 | 575.6 | | 353.4 | | 79705 | | 505.1 | | 431.8 | | 0.808 | 1.097 | 0.868 | 0.960 | 0.966 |
| 6-20 | 162 | A02 | P10 | 0856 | 547.0 | | 415.0 | | 79210 | | 515.4 | | 424.8 | | 0.851 | 0.934 | 0.873 | 0.940 | 0.982 |
| 6-20 | 163 | A25 | N11 | 0912 | 556.6 | | 480.0 | | 79333 | | 519.4 | | 412.4 | | 0.836 | 0.808 | 0.872 | 0.933 | 1.012 |
| 6-20 | 164 | B36 | M12 | 0927 | 573.3 | | 434.3 | | 79347 | | 498.8 | | 424.8 | | 0.811 | 0.893 | 0.872 | 0.972 | 0.982 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| | | | | | Counts | | | | | | | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temporary Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| Date | Step | Assm I.D. | Core Pos. | Time | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val. | A | B | C | 31 | 32 |
| 6-20 | 165 | C09 | R10 | 0943 | 569.3 | 465.2 | 402.0 | 387.8 | 79550 | 69164.5 | 510.4 | 484.7 | 424.4 | 417.3 | 0.817 | 0.965 | 0.869 | 0.950 | 0.983 |
| 6-20 | 166 | C08 | P11 | 1051 | 563.3 | | 373.7 | | 79721 | | 493.8 | | 429.8 | | 0.826 | 1.038 | 0.868 | 0.982 | 0.971 |
| 6-20 | 167 | B54 | N12 | 1111 | 554.6 | | 433.0 | | 79355 | | 537.8 | | 433.8 | | 0.839 | 0.896 | 0.872 | 0.901 | 0.962 |
| 6-20 | 168 | B29 | M13 | 1125 | 569.0 | | 394.0 | | 79375 | | 503.4 | | 434.4 | | 0.818 | 0.984 | 0.871 | 0.963 | 0.961 |
| 6-20 | 169 | C53 | R11 | 1143 | 583.3 | | 408.7 | | 79295 | | 547.8 | | 449.4 | | 0.798 | 0.949 | 0.872 | 0.885 | 0.929 |
| 6-20 | 170 | C19 | P12 | 1159 | 584.3 | | 405.7 | | 79649 | | 526.4 | | 442.1 | | 0.796 | 0.956 | 0.868 | 0.921 | 0.944 |
| 6-20 | 171 | C54 | N13 | 1218 | 558.6 | | 392.7 | | 69165 | | 484.7 | | 417.3 | | 0.833 | 0.988 | 0.868 | 0.937 | 0.966 |
| 6-20 | 172 | C23 | M14 | 1242 | 559.0 | | 402.0 | | 80235 | | 489.4 | | 424.4 | | 0.832 | 0.965 | 0.862 | 0.990 | 0.983 |
| 6-20 | 173 | C35 | P13 | 1259 | 544.6 | | 406.7 | | 80537 | | 513.8 | | 447.8 | | 0.854 | 0.954 | 0.859 | 0.943 | 0.932 |
| 6-20 | 174 | C16 | N14 | 1318 | 552.5 | | 411.4 | | 80816 | | 506.1 | | 420.4 | | 0.842 | 0.943 | 0.856 | 0.958 | 0.993 |
| 6-20 | 175 | A05 | M06 | 1336 | 559.3 | | 425.7 | | 80468 | | 521.4 | | 446.4 | | 0.832 | 0.911 | 0.860 | 0.930 | 0.935 |
| 6-20 | 176 | B37 | N06 | 1400 | 584.6 | | 418.0 | | 80222 | | 515.4 | | 429.4 | | 0.796 | 0.928 | 0.862 | 0.940 | 0.972 |
| 6-20 | 177 | B16 | M05 | 1416 | 568.0 | | 438.0 | | 80392 | | 505.1 | | 467.1 | | 0.819 | 0.885 | 0.860 | 0.960 | 0.893 |
| 6-20 | 178 | A06 | P06 | 1434 | 566.0 | | 448.4 | | 79985 | | 533.8 | | 429.8 | | 0.822 | 0.865 | 0.865 | 0.908 | 0.971 |
| 6-20 | 179 | A49 | N05 | 1517 | 551.6 | | 450.0 | | 80608 | | 487.1 | | 470.8 | | 0.843 | 0.862 | 0.858 | 0.995 | 0.886 |
| 6-20 | 180 | B24 | M04 | 1537 | 539.0 | | 448.4 | | 80393 | | 495.4 | | 479.8 | | 0.863 | 0.865 | 0.860 | 0.978 | 0.870 |
| 6-20 | 181 | C38 | R06 | 1554 | 549.3 | | 525.0 | | 80389 | | 537.1 | | 490.8 | | 0.847 | 0.739 | 0.860 | 0.902 | 0.850 |
| 6-20 | 182 | C42 | P05 | 1611 | 565.6 | | 597.2 | | 80400 | | 476.1 | | 470.1 | | 0.822 | 0.649 | 0.860 | 1.018 | 0.888 |
| 6-20 | 183 | B28 | N04 | 1627 | 541.3 | | 602.0 | | 80384 | | 493.1 | | 484.4 | | 0.859 | 0.644 | 0.860 | 0.983 | 0.861 |
| 6-20 | 184 | B46 | M03 | 1643 | 552.0 | | 518.4 | | 80257 | | 518.8 | | 399.4 | | 0.843 | 0.748 | 0.862 | 0.934 | 1.045 |
| 6-20 | 185 | C06 | R05 | 1659 | 545.3 | | 565.7 | | 80419 | | 538.1 | | 432.4 | | 0.853 | 0.686 | 0.860 | 0.901 | 0.965 |
| 6-20 | 186 | C47 | P04 | 1714 | 541.0 | | 583.0 | | 80000 | | 515.8 | | 437.1 | | 0.860 | 0.665 | 0.865 | 0.940 | 0.955 |
| 6-20 | 187 | C51 | N03 | 1731 | 536.3 | | 562.0 | | 80261 | | 510.1 | | 441.1 | | 0.867 | 0.690 | 0.862 | 0.950 | 0.946 |
| 6-20 | 188 | C32 | M02 | 1753 | 542.3 | | 583.7 | | 80588 | | 531.1 | | 425.4 | | 0.858 | 0.664 | 0.858 | 0.913 | 0.981 |
| 6-20 | 189 | C56 | P03 | 1812 | 538.3 | | 574.0 | | 80500 | | 516.4 | | 437.8 | | 0.364 | 0.676 | 0.859 | 0.939 | 0.953 |

TABLE 2.1.1 (cont.)

SUMMARY OF NUCLEAR MONITORING DATA

| Date | Step | Assm I.D. | Core Pos. | Time | Counts | | | | | | | | | | ICRR | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | Temporary Channels | | | | | | Plant Channels | | | | | | | | |
| | | | | | A | | B | | C | | 31 | | 32 | | | | | | |
| | | | | | Value | Ref. Val. | Value | Ref. Val. | Value | Ref. Val | Value | Ref. Val. | Value | Ref. Val | A | B | C | 31 | 32 |
| 6-20-84 | 190 | C36 | N02 | 1842 | 578.3 | 465.2 | 572.4 | 382.8 | 80581 | 69164.5 | 517.8 | 484.7 | 419.8 | 417.3 | 0.804 | 0.678 | 0.858 | 0.936 | 0.994 |
| 6-20 | 191 | C13 | A08 | 1901 | | | 617.7 | | 80291 | | 530.4 | | 426.1 | | | 0.628 | 0.861 | 0.914 | 0.979 |
| 6-20 | 192 | C22 | R08 | 1918 | | | | | 80557 | | 508.4 | | 435.1 | | | | 0.859 | 0.953 | 0.959 |
| 6-20 | 193 | C44 | L15 | 1936 | | | | | | | 533.4 | | 465.4 | | | | | 0.909 | 0.897 |

## 2.2 NUCLEAR INSTRUMENTATION SYSTEM (S-07SE01)
Procedure ETT-SE-07010

### OBJECTIVES

The objectives of this test were to verify that the Source and Intermediate Range Nuclear Instrumentation System performed the required indication and control functions prior to core loading and that the Power Range Nuclear Instrumentation System performed the required indication and control functions prior to Initial Criticality.

### SUMMARY OF RESULTS

It was determined that the source range level trip occurred at $1x10^{5}$ cps, the Intermediate Range Trip occurred at $2.5x10^{-5}$ amps, the Intermediate Range P-6 setpoint was determined to be $1x10^{-10}$ amps and the Intermediate Range High Level Rod Stop was determined to be $2x10^{-5}$ amps. The acceptance criteria for each occurrence were: Source Range Trip $\leq 1.6x10^{+5}$ cps, Intermediate Range Trip $\leq 3.53x10^{-5}$ amps, Intermediate Range P-6 setpoint $\geq 1x10^{-10}$ amps and the Intermediate Range High Level Rod Stop setpoint $\leq 2.1x10^{-5}$ amps. Therefore, all the acceptance criteria were met.

For each Power Range channel, the appropriate Instrumentation and Control Surveillance procedure was utilized to verify that the following setpoints were correct: Overpower Trip Low Range, Overpower Trip High Range, Positive and Negative Rate Trips, Power Above Permissives P-8, P-9 and P-10 and Overpower Rod Stop. The acceptance criteria associated with this portion of the test was that the setpoints were set as specified by the appropriate I&C procedure, therefore all acceptance criteria were met.

### DISCUSSION

Detector voltage readings were taken. Source Range Detector voltages were taken from the detector volts meter on each drawer and the Intermediate and Power Range detector voltages were obtained by measuring the voltage at various test points.

Testing of the Source and Intermediate Range Channels was conducted by placing the channel in the "Test" mode and introducing a test signal, to simulate various count rates. A check of the High Flux At Shutdown Alarm and the Audio and Scaler Timers was satisfactorily performed. During this test, it was found that the Bistable reset levels on each Source Range Channel was greater than the given acceptable level. I&C corrected this problem by resetting the levels. Also, it was found that during the test, the channels did not indicate an acceptable count rate. This was found to be due to a wrong thumbwheel setting and was corrected. The test was conducted again with satisfactory results.

Testing was performed on the Power Range Channels in accordance with the appropriate I&C controlling procedure.

The Source and Intermediate Range rate circuits were tested by utilizing the rate test switch. The circuits provided test inputs of 1 and 5 dpm and the rates are then read from the appropriate instrument.

A test of the Source Range Flux Doubling Instrumentation was performed. This was accomplished by applying a voltage to the detector circuitry and then doubling the applied voltage ten minutes later. After satisfactory completion, the voltage was returned to its normal value.

## 2.3 CORE LOADING INSTRUMENTATION AND NEUTRON SOURCE REQUIREMENTS (S-07SC01)

Procedure ETT-SC-07010

### OBJECTIVES

The objectives of this procedure were:

1. To verify proper alignment and calibration of the temporary core loading instrumentation prior to the start of fuel loading,

2. To verify the signal to noise ratio is greater than 2, and

3. To check the neutron response of the temporary core loading instrumentation package and the plant Nuclear Instrumentation System (NIS) Source Range channels,

   a. Prior to the start of fuel loading, and

   b. Prior to the resumption of fuel loading following any delay of eight hours or more.

### SUMMARY OF RESULTS

Prior to the start of fuel loading, the temporary core loading instrumentation was aligned and calibrated using a 4.9Ci Am-Be neutron source. An operating voltage of 1950 volts and a discriminator setting of 2.5 volts were determined for Detectors A, B, C, and the spare detector. These values were identical to the pre-shipment values furnished by Westinghouse. The signal to noise ratio was satisfactory for all channels. The neutron response of the temporary core loading instrumentation and the plant NIS Source Range Channels was verified by a positive change in count rate as the neutron level was increased before the start of fuel loading. The neutron response was verified by performance of a statistical evaluation due to ALARA considerations prior to the resumption of fuel loading following all delays of eight hours or more.

### DISCUSSION

The neutron source was positioned near the temporary instrumentation detectors on the operating floor of containment to give a count rate of at least 10,000 CPM on each of the three channels. Starting at 1800 volts, the high voltage was varied in 50 volt increments for each channel up to 2500 volts. Voltage vs. CPM curves were then plotted to determine the operating voltage. Starting at zero volts, the discriminator voltage was varied in 0.5 volt increments for each channel up to 4.0 volts. Discriminator Voltage vs. CPM curves were then plotted to determine the discriminator setting.

Prior to the start of fuel load, the neutron response checks were performed by moving the source to a position as close as possible to each of the detectors. Following all delays of eight hours or more, 10 count rate observations of 400 to 5000 counts per observation were

obtained for each channel. A statistical evaluation was then performed by calculating a Reliability Factor to verify that the count rate measurements did not exhibit a greater deviation from the Poisson Distribution than was expected from chance alone.

From the 6/1/84 initial checkout to the 6/13/84 start of fuel loading, high humidity inside containment required dessicant and plastic covering to be placed over the preamps and around all connections to prevent moisture accumulation. This solved the moisture problem caused by high humidity. On 6/15/84 Detector A was replaced by the spare detector due to water leakage into Detector A. No additional problems were encountered during this test.

## 2.4 REACTOR SYSTEMS SAMPLING FOR CORE LOAD (S-07SJ01)

(This procedure was combined with ETT-ZZ-07010, Initial Core Loading.)

3.1 INCORE INSTRUMENTATION OPERABILITY TEST (S-07SR03 and S-07SR04)
Procedure ETT-SR-07040

OBJECTIVES

The objective of this procedure was to demonstrate the operability of the Flux Mapping System under hot and cold conditions, prior to physics testing.

SUMMARY OF RESULTS

The following items were verified during this test:

1. The Flux Mapping System (FMS) operates as designed for the System Initialization Mode.

2. The FMS operates as designed for the Manual Flux Map Mode.

3. The FMS operates as designed for the Set/Modify Limits Mode.

4. The FMS operates as designed for the Set/Detector Voltage Mode.

5. The FMS operates as designed in the Automatic Flux Map Mode by providing a full core flux map when the start action is entered.

6. The FMS operates as designed in the Special Flux Map Mode.

7. All thimbles are clear of obstruction with the exception of Thimble J01. With this path inoperable, the Flux Mapping System still satisfies Technical Specification requirements.

8. Each drive was capable of its detector going through the full length of each thimble with the exception of Drives A & B not being able to go through thimble H15 (and no detectors through J01 - See #7 above.) Auto Flux Maps normally use the C Drive for Thimble H15 and thus does not impact Technical Specification operability.

9. The FMS operated as designed in the Data Transfer Mode with the exception of the data link between the NSSS computer and the FMS not functioning due to NSSS computer problems. This function was proven operable during the performance of S-07SR01, Initial Startup Flux Mapping.

This test went through all eight (8) functions that are associated with the FMS and verified that they operated correctly and that the required alarms and interlocks functioned properly.

DISCUSSION

This test consisted of ten separate test sections which verified all of the FMS capabilities: 1) System Initialization; 2) Set/Modify Limits Mode Operational Checkout; 3) Manual Mode Operation Checkout; 4) Set Detector Voltage Function Checkout; 5) Thimble Checkout; 6) Data Transfer Mode Operational Checkout; 7) Peripheral Device Failure Checkout; 8) Automatic Flux Map Mode Operational Checkout; 9) Special Flux Map Mode Operational Checkout; and 10) Thimble Verification. Section 10 was performed at cold shutdown and sections 8 and 9 at hot standby.

After performance of this test, a large number of deficiencies were identified, for which Westinghouse assistance was requested for resolution. Westinghouse determined that the majority of the problems were procedural errors carried over from the original Westinghouse test procedures. These changes had not been incorporated by Westinghouse since the system software was still being modified. These procedural errors do not affect test acceptance. The problems encountered are summarized in the following paragraphs.

The System Initialization test checked the ability of the FMS to properly initialize the system for use from either CRT, transfer control between the CRT's, and signal appropriate error messages. This section was performed satisfactorily. The only problem encountered in this section was an inadvertent "15-Path Transfer Failure - A and B" which was due to power being off to the A and B drives.

The Set/Modify Limits Mode Operational Checkout section tested the ability of the FMS to enter new drive path limits, obtain a printout of the values and load the new values onto a cassette tape. A problem was encountered with the printouts not matching up to the entered values, which was caused by the write protect tabs not being in the correct position on the cassette tapes. This section was performed satisfactorily after this problem was resolved.

The Manual Mode Operational Checkout section tested the ability of the FMS to perform manual manipulations of the drives/detectors to preset locations and verify that the system performed as designed and/or signalled appropriate error messages. Several problems were encountered during performance of this section, which were resolved by Westinghouse. Final detector positions were required to be within ±.1 "of the input position - nonconformance was determined to be a procedural error since slow speed tolerance is ±.5" and fast speed tolerance is 12". Other problems were encountered due to unexpected or incorrect error messages, which were due to procedural errors. This section was completed successfully after correction of the procedural deficiencies.

The Set Detector Voltage Function Checkout section tested the ability of the FMS to set detector voltages and obtain a printout of the voltage plot. The detector voltages are set prior to running any auto or special flux map. This section was performed satisfactorily with the only problem being a procedural deficiency due to an error message not being received.

The Thimble Checkout section tested the ability of each detector to be inserted into every thimble. The major problem encountered in this section was the message "NO DETECTOR MOTION" caused by the detector running into the end of the thimble. This was due to the difference in the number of turns on the drive cable which drives the resolver and the inaccuracy in the way the first path insertion lengths were measured with the manual control box. Revised insertion limits were developed and input each time this problem was observed. Final insertion limits were eventually verified in a separate procedure. Thimble J01 was found to be blocked during performance of this section which caused several deficiencies since the thimble must be deleted to run an auto flux map. Generic problems were encountered in this section and others with the overtorque function of the drive units. The overtorques could not be set properly, however this is a redundant function since a "NO DETECTOR MOTION" message will be received if the drives become stalled. Another overtorque problem was traced to a faulty triggering card in the motor control box. A final problem was encountered with Drive B not being able to enter Thimble H15.

The Data Transfer Mode Operational Checkout section tested the ability of the FMS to transfer data from a floppy disk to the printer, a second floppy disk, or to the NSSS computer. This section was performed satisfactorily with the exception of the data link between the FMS and the NSSS computer not functioning due to NSSS computer problems. This problem was resolved and the link verified during later testing (S-07SR01).

The Peripheral Device Failure Checkout section tested the ability of the FMS to signal appropriate warnings to the console operator due to any equipment failures, e.g. printer, floppy disk unit, etc. This section was performed satisfactorily following resolution of some procedural problems (conflicting error messages received).

The Automatic Flux Map Mode Operational Checkout section tested the ability of the FMS to perform an automatic flux map using all or any combination of drives and signal appropriate errors to the console operator. In addition to miscellaneous procedural problems, the problem with "NO DETECTOR MOTION", as discussed previously, was encountered. Drive A also would not go through Thimble H15. With Drives A and B not being able to go through H15 and no drives through J01, the system is still capable of meeting Technical Specification operability requirements.

The Special Flux Map Mode Operational Checkout section tested the ability of the FMS to perform a special flux map using selected thimbles and drives and signal appropriate error messages. No additional deficiencies were discovered during this section but several similar problems (procedural errors, Thimble H15 stuck for Drives A and B, "NO DETECTOR MOTION") were encountered.

The Thimble Verification section (performed at cold shutdown) tested each thimble using a dummy detector. This section was performed satisfactorily after resolving the initial problem with path insertion limits.

## 3.2 CONTROL ROD DRIVE MECHANISM (CRDM) OPERATIONAL TEST (S-07SF01) Procedures ETT-SF-07014 and ETT-SF-07015

### OBJECTIVES

The objectives of this test were to verify proper slave cycler timing, to demonstrate the proper operation of each CRDM under both cold shutdown and hot standby conditions, and to demonstrate that all Digital Rod Position Indicators (DRPI) operate satisfactorily over the entire range of rod travel.

### SUMMARY OF RESULTS

The operability of each CRDM was verified by a demonstration of drive line insertion and withdrawal during which CRDM coil timing traces were obtained. These traces showed conformance to the requirements for proper CRDM operation and timing. This criteria was verified during both cold and hot conditions. The DRPI System control board display indication and NSSS computer readout of demand position (not necessarily the actual position) for each rod agreed with its associated group step counter within $\pm 4$ steps. All rod bottom LEDs actuated within 4 steps of the zero indication of their associated group step counters. For control banks, the pulse-to-analog converter bank position readout agreed with its associated group step counter within $\pm 1$ step. All rods operated satisfactorily when being withdrawn and inserted over their entire range of travel. All traces were normal during both cold and hot mechanism timing tests.

### DISCUSSION

These tests were performed by pulling control rods and stopping at various points to record data. This data verified DRPI display and gray code (at Data Cabinets A and B in containment) data versus control room step counters. For control banks, pulse-to-analog converter data was verified versus the step counters. At 48 steps, rods were withdrawn and reinserted 10 steps while operating a visicorder that was connected to the stationary gripper, movable gripper and lift coil of each mechanism to obtain a trace of the associated coil currents. In addition, sound signals from a carbon microphone affixed to the top of each rod travel housing were selected for the appropriate rod and recorded as a fourth visicorder trace. The resultant traces were examined by comparison with a correct trace using a clear plastic overlay. A normal CRDM stepping trace is illustrated in Figure 3.2.1.

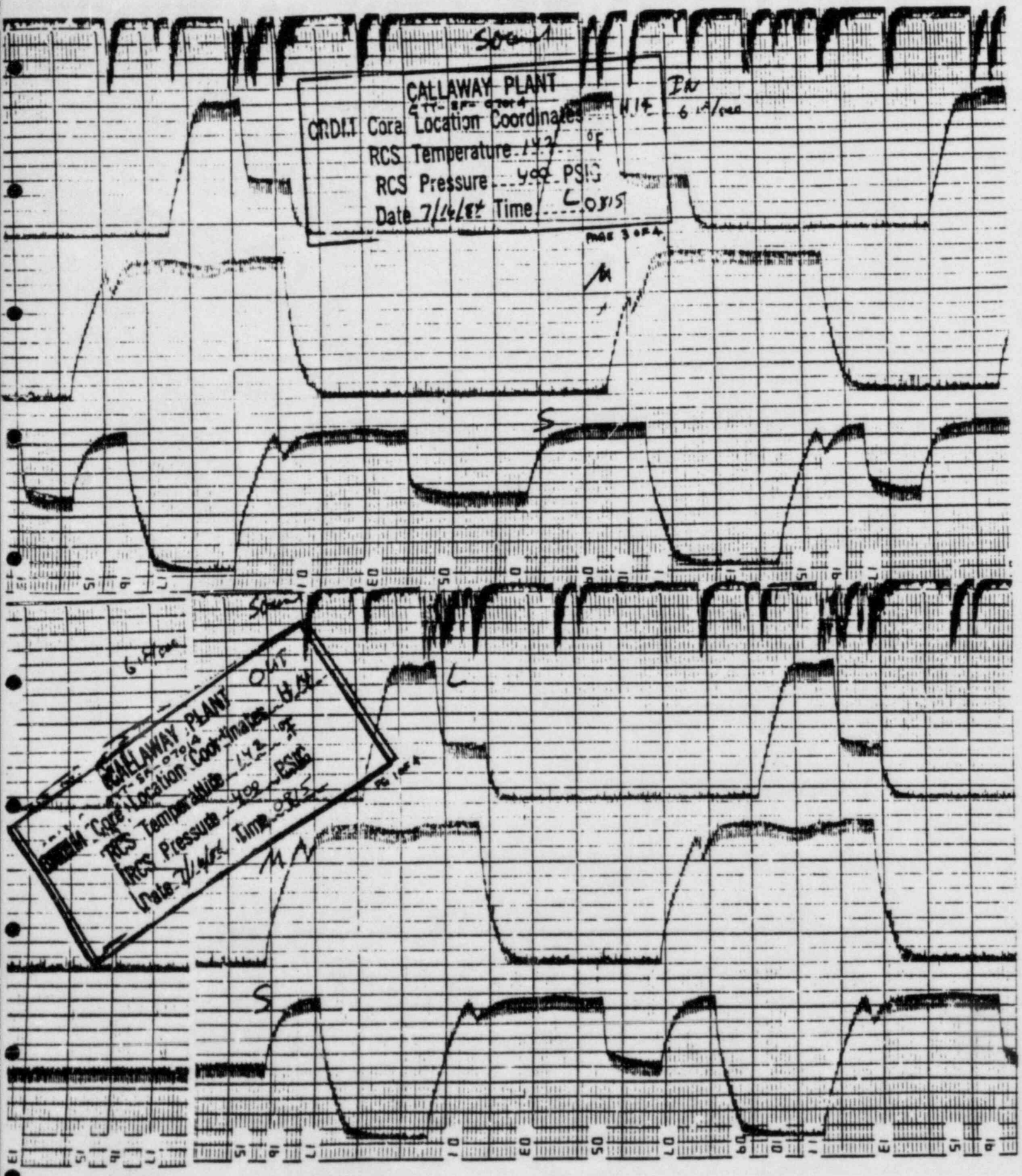
Several problems were encountered during the performance of the test under cold conditions. NSSS computer readout of demand position was unavailable due to a design discrepancy between the DRPI wiring and NSSS computer. This discrepancy was resolved and testing was successfully completed. During the testing of Control Bank B (CBB), Rod F-14 DRPI indicated 66 steps while all other rods in CBB and the step counters for CBB read 72 steps. (This was still within $\pm 12$ steps as required by Tech Specs.) Testing for CBB was continued until at 76 steps (on step counters) F-14 showed General Warning (GW) and Rod Bottom

(RB) LEDs. All other rods indicated normal. CBB was reinserted. Upon investigation it was discovered that two pins were touching in the DRPI connector for rod F-14. The pins were straightened out and CBB was successfully retested. During Control Bank D (CBD) testing, Rod D-4 DRPI indicated 72 while the step counters for CBD indicated 79 steps. Rods were reinserted after some troubleshooting was performed. The problem was a bad DRPI connector for rod D-4. The connector was replaced and CBD was successfully retested. During Control Bank A (CBA), testing rod K-8 would not withdraw. Upon investigation it was discovered that the blown fuse indicator (BFI) was indicating that the stationary gripper coil for K-8 had a blown fuse. The fuse and BFI were replaced and CBA was successfully tested. During CBC testing the timing trace of rod B-8 did not show a sound trace. Upon investigation it was found that a wiring mismatch was present. A jumper was installed to allow completion of testing. The wiring problem was corrected subsequently.

During testing at hot standby conditions several problems were encountered. An attempt was made to withdraw Shutdown Bank A. There was no movement due to a Power Range Overpower Rod Stop. Power Range Channel N44 Rod Stop was placed in bypass and testing resumed. While withdrawing Shutdown Bank A DRPI showed no indication of rod movement with the step counters showing 38 steps. Shutdown Bank A was inserted to 0 steps and testing suspended. Troubleshooting revealed that DRPI Data Cabinet B had been placed in "Test". DRPI Data Cabinet B was returned to normal. While attempting to withdraw Shutdown Bank C, a Rod Control Urgent Failure Alarm was received. The same alarm was received while attempting to withdraw Shutdown Bank D. Following replacement of a card in Power Cabinet SCDE, an attempt was made to withdraw Shutdown Bank C and the Rod Control Urgent Failure Alarm occurred. An attempt to withdraw Shutdown Bank E was made with the same result. Testing was suspended to determine the problem, which turned out to be a slave cycler card in the Logic Cabinet. Testing of Shutdown Banks C, D, and E was then completed. During hot testing of CBC a sound signal was not obtained from microphone B-8. Troubleshooting revealed incorrect wiring inside the head area junction box. It was determined a retest was not necessary because all coil traces were satisfactory, and satisfactory sound and coil traces were obtained under cold conditions.

Figure 3.2.1

Normal CRDM Timing Trace



3.3 ROD CONTROL SYSTEM (S-07SF02)
Procedure ETT-SF-07020

OBJECTIVES

To demonstrate and document that the Rod Control System satisfactorily performs the required control, inhibit and indication functions in order to verify it is ready for use prior to initial criticality.

SUMMARY OF RESULTS

The control, inhibit and indicating functions of the Rod Control System were satisfactorily demonstrated and verified, with the exception of speed indication for Shutdown Banks C, D and E.

DISCUSSION

The urgent and non-urgent alarm functions were tested in the Power cabinets. The urgent alarm resulted for each test case where a rod control inhibit function was tested. The rod inhibit functions which were tested are listed below:

A. Regulation Failure (Stationary and Movable)
B. Phase Failure
C. Multiplexing Error
D. Logic Error

For each inhibit function listed above, a failure symptom was physically simulated for each power cabinet and the Reactor Operator was requested to attempt to withdraw rods. The net result was that rods didn't withdraw and the urgent alarm was received at the Main Control Board (MCB) and locally at the power cabinet being tested, thus satisfying the objective of the test. The non-urgent alarm circuits were also tested at each power cabinet by removing fuses and thus simulating a Power Supply (P/S) failure. The non-urgent alarm was received at the MCB and the power supply failure was received at the power cabinet, thus satisfying the objective of the test.

The urgent and non-urgent alarms were tested in the logic cabinet in the same manner as the power cabinets. An actual logic circuit defect was inserted into the logic cabinet and the Reactor Operator was requested to attempt to withdraw control rods. The urgent alarm was verified at the MCB and locally at the logic cabinet and there was no outward motion of the control rods. Both the logic circuits and the pulse failure test were items tested for the urgent alarm inhibit function. The non-urgent alarm function was physically tested by de-energizing a Power Supply (P/S) by pulling fuses in the logic cabinet and verifying the non-urgent alarm was received at the MCB and at the logic cabinet. Thus, all objectives were met.

The control and indication functions were tested by actual control rod withdrawal, insertion, DRPI reading, bank position indication readings, rod speed verification, and rod group transfer to and from the

DC Hold Bus. Both the shutdown and Control Rod banks underwent this form of testing. The Control Bank rods also underwent a sequenced bank overlap check by control rod withdrawal and insertion. The control system was also checked by inserting a manual rod withdrawal block condition and then having the Reactor Operator attempt to withdrawn control rods. The rods didn't withdrawn and the test was satisfactory. Thus, all objectives concerning control functions were satisfied.

A Design Review Criterion was not met for Shutdown Banks C, D, and E speed indication. Rod speed was as expected when measured independently, but an indication problem exists. Westinghouse has been requested to review the design for this indication and initiate the necessary design change to correct the problem. This deficiency has no impact on nuclear safety and was evaluated by the Onsite Review Committee as posing no restraint to continued power operation.

## 3.4 CONTROL ROD DROP TIME MEASUREMENTS (S-07SF03)
Procedures ETT-SF-07033, ETT-SF-07034

### OBJECTIVES

The objectives of the subject procedures were as follows:

1. To measure the drop time of each Rod Cluster Control Assembly (RCCA) with the Reactor Coolant System (RCS) in a hot, no flow condition (HNF).

2. To measure the drop time of each RCCA with the RCS in a hot, full flow condition (HFF).

3. To retest those RCCA's whose drop time deviated from the mean drop time by more than two standard deviations a sufficient number of times ($\geq 3$) to reasonably ensure proper performance during subsequent plant operations.

4. To monitor the effectiveness of the thimbles dashpot region for decelerating the RCCA's during each rod drop time measurement.

### SUMMARY OF RESULTS

All acceptance criteria for the subject procedures were met. These criteria and discussions of associated results follows:

1. The drop time for each RCCA is less than or equal to 2.2 seconds (HFF).

   The longest drop time as measured from the beginning of decay of stationary gripper coil voltage to dashpot entry (HFF) was 1.40 seconds at core location J-03.

2. The rod drop traces are consistent and exhibit free fall with no abnormalities or evidence of binding.

   Review of the rod drop traces indicated satisfactory results.

3. The rod deceleration through the dashpot region must be similar for all rods being dropped.

   The deceleration time (time from dashpot entry to bottom of dashpot) for the rods ranged from 0.39 to 0.45 seconds for the HNF case and from 0.45 to 0.50 seconds for the HFF case.

4. Each RCCA whose drop time deviated from the mean drop time by more than two standard deviations, when retested, exhibits a time consistency with 0.02 seconds.

   All rod drop times were within the allowed band. No rods were required to be retested.

DISCUSSION

The Callaway Plant utilizes a Digital Rod Position Indication (DRPI) System. A Rod Position Detector is supplied for each rod; this includes the rods for the control banks and the shutdown banks. Each standard detector is in essence a tube with 42 assemblies of coiled wire slipped over the tube and spaced out along its length at 3.75 inch intervals. Each shutdown detector has a reduced number of coil assemblies (20) and a space without coils in the middle section of the detector. The tube fits over the rod travel housing. AC power from a constant voltage source is applied to each coil of wire on the tube. A low frequency is used so that the electromagnetic flux generated by the coils penetrates the thickness of the rod travel housing. Since the rod within the housing is ferromagnetic, a concentration of flux is realized as the rod moves through a coil. The concentration of flux is reflected as an increased impedance to the AC flow.

To prevent total loss of information due to a single coil failure, the outputs of every other coil on the tube are connected as inputs to one Data Cabinet, while the outputs of the remaining coils are connected to the other Data Cabinet. Within each Data Cabinet there is a detector/encoder card for each Rod Position Detector. A detector/encoder card samples the voltages across terminating resistors, each of which is effectively in series with one of the detector coils. When a rod is out of a coil, the voltage sample level is high; when the rod moves into a coil, the voltage sample level drops. At the detector/encoder card, samples from adjacent coils are monitored by differential amplifiers. When the rod end is between two adjacent coils, the corresponding differential amplifier output is high; otherwise, the output is low. Thus, the position of a particular rod is reflected in the output states of the associated detector/encoder card differential amplifiers.

A test point on each detector/encoder card in each Data Cabinet provides a convenient means of recording the profile of a rod drop test. Under normal operations, the test point measures the ac excitation applied to the particular A or B channel of coils. When this voltage is removed, the residual magnetism of the rod causes a voltage to be momentarily induced in each coil as the rod passes through. These voltages sustain a pulse of current in the common line, which can be recorded using a visicorder.

Figure 3.4.1 is an example of a recording of such a current pulse monitored at the test point for a standard detector. The notches in the pulse indicate the rod top passing through a coil. Therefore, if a particular coil is shorted or open, the profile identifies it. Also, if the rod temporarily binds at any point, the approximate location of the bind can be identified in the profile. For shutdown detectors, since rod travel is not monitored to the same accuracy in the mid-range as for a standard detector, visicorder trace sensitivity is not as sharp in that area. However, all critical parameters, such as initiation of drop, dashpot entry, and snubber contact are present at full sensitivity; and the rod drop for shutdown detectors can be measured, as with the standard detectors.

A block diagram of the test arrangement is presented as Figure 3.4.2. Note in Figure 3.4.1 that the Data A and Data B traces are out of phase, due to the alternating arrangement of the A and B coils. By summing the A and B signals a smooth trace is output which allows precise definition of dashpot entry. A 60-Hz trace was recorded for use as a time standard. Sound traces were obtained from carbon microphones mounted in housings attached to each CRDM rod travel housing.

Figures 3.4.3 and 3.4.4 presents the rod drop times.

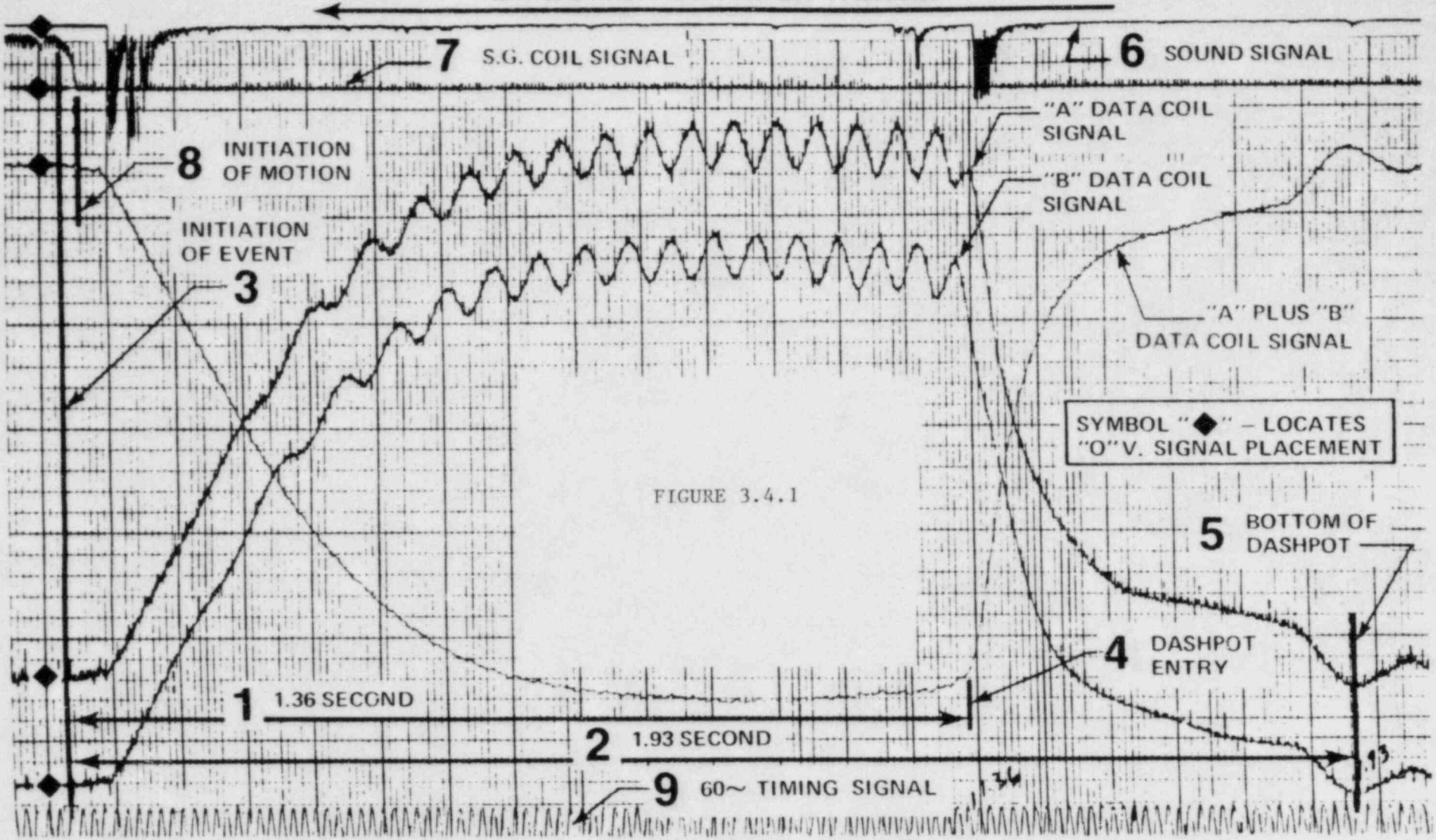



FIGURE 3.4.1

**THIS TRACE FORMAT IDENTIFIES:**

- 1- DROP TIME VALUE, TO DASHPOT
- 2- DROP TIME VALUE, TOTAL
- 3- INITIATION OF EVENT
- 4- DASHPOT ENTRY
- 5- BOTTOM OF DASHPOT
- 6- SOUND SIGNAL
- 7- S.G. COIL SIGNAL
- 8- INITIATION OF MOTION
- 9- 60~ TIMING SIGNAL

## RECOMMENDED OPTIMUM "ROD DROP" TEST TRACE FORMAT ARRANGEMENT

FIGURE 3.4.2

ROD DROP TESTS BLOCK DIAGRAM

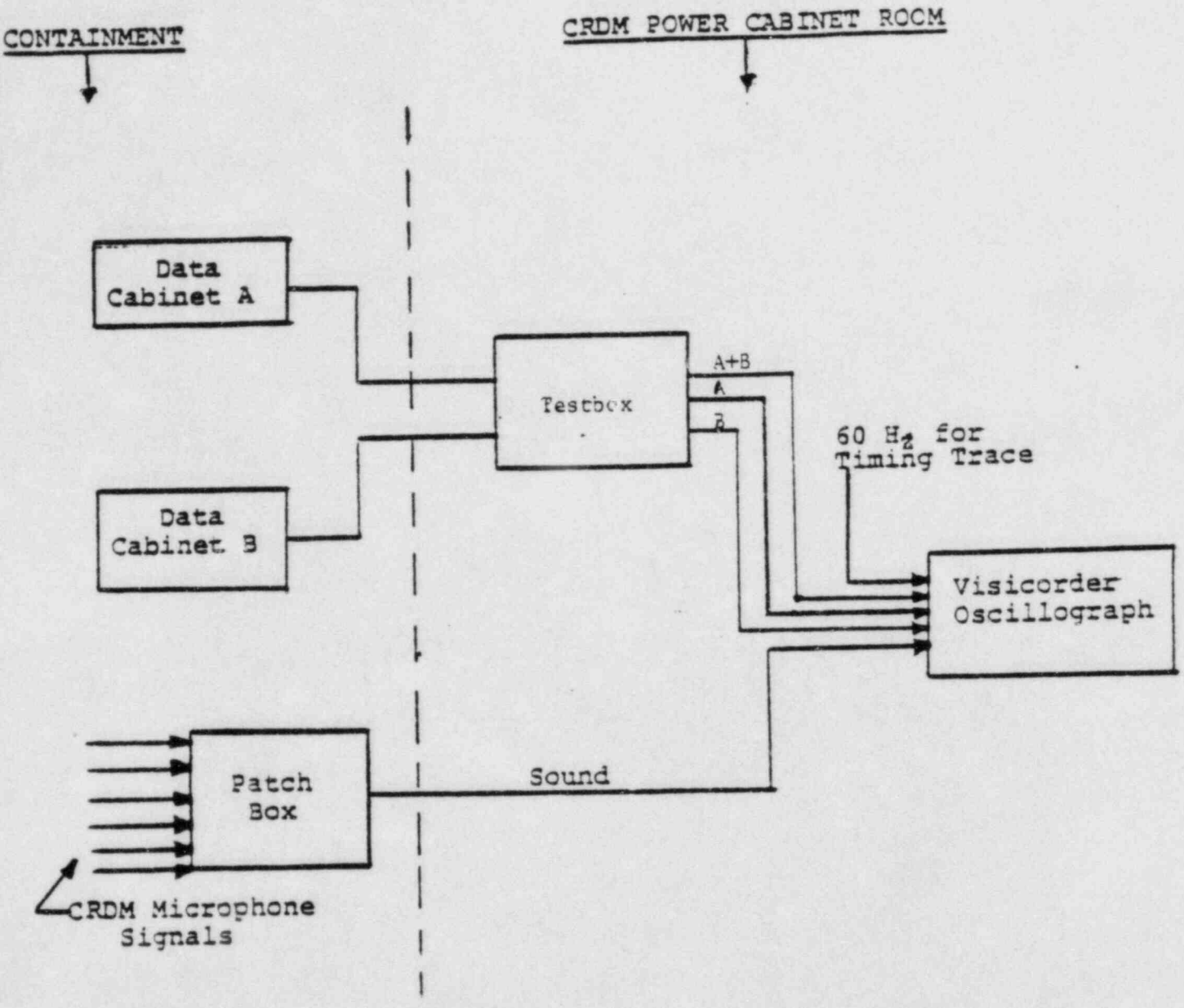
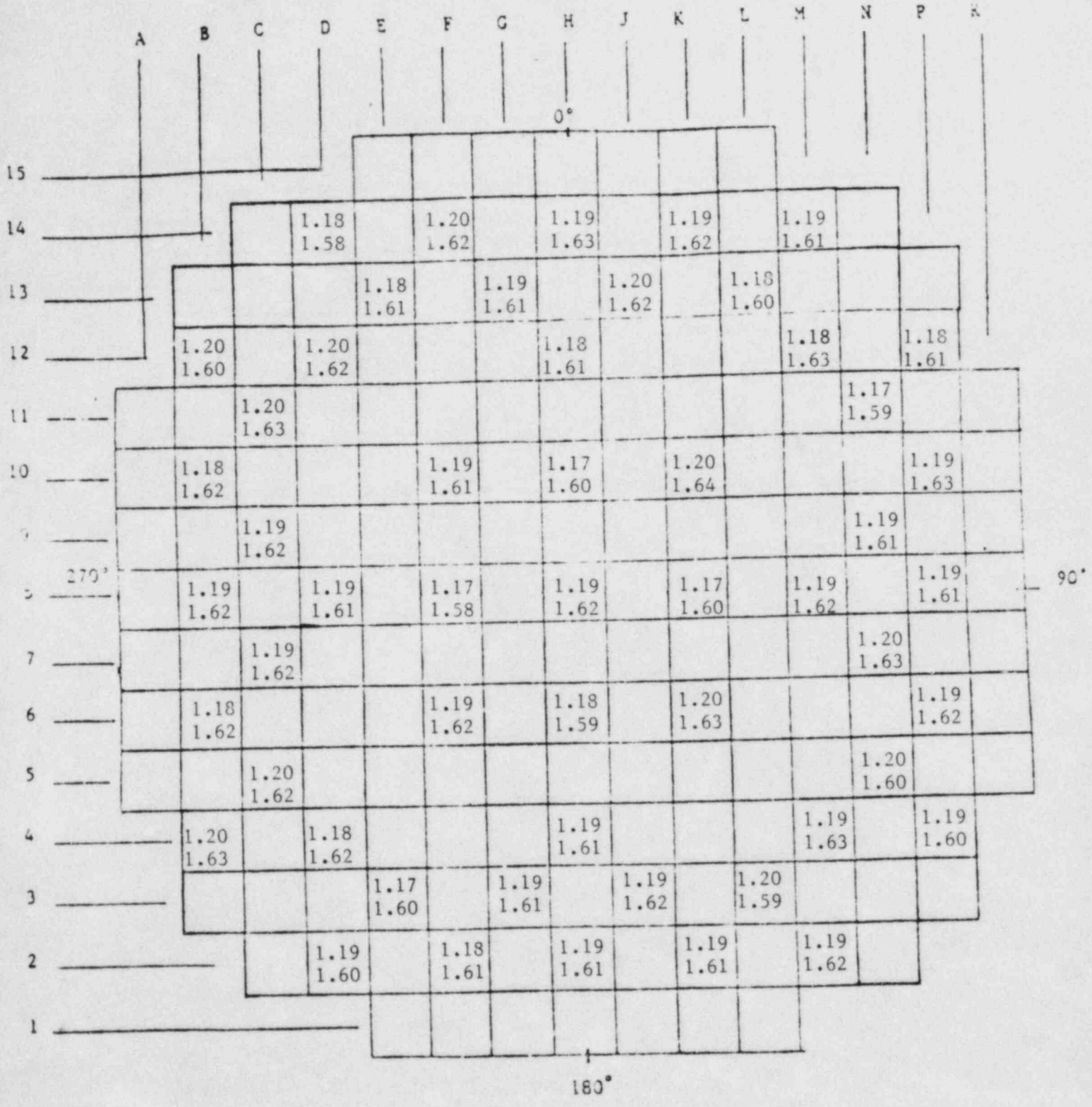


FIGURE 3.4.3

ROD DROP TIME TABULATION

HOT, NO FLOW




| |
|---|
| X.XX |
| X.XX |

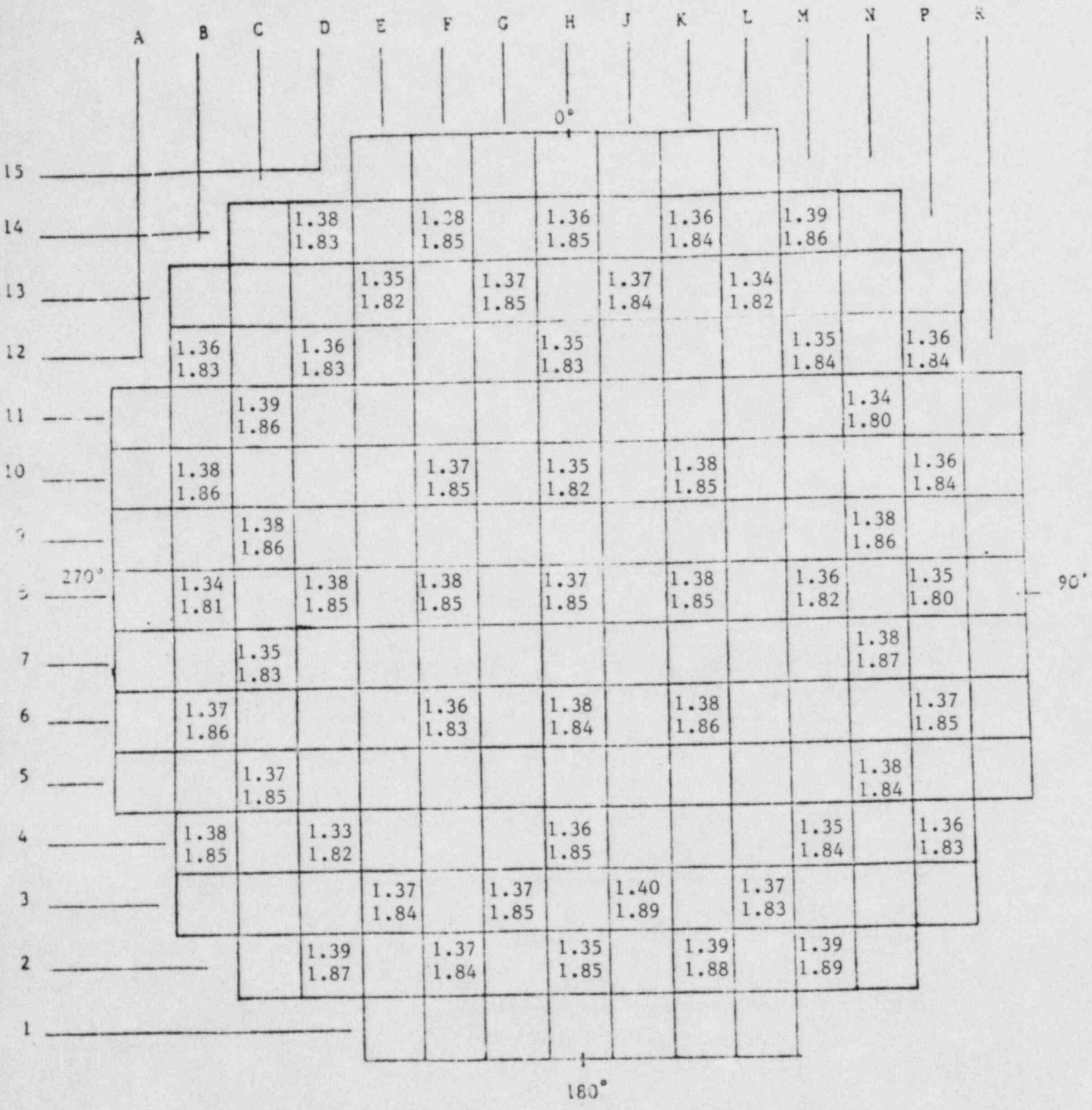
TIME TO DASHPOT ENTRY (SECONDS) - AVERAGE = 1.19

TIME TO DASHPOT BOTTOM (SECONDS) - AVERAGE = 1.61

FIGURE 3.4.4

ROD DROP TIME TABULATION

HOT, FULL FLOW




| | |
|---|---|
| X.XX | TIME TO DASHPOT ENTRY {SECONDS} - AVERAGE = 1.37 |
| X.XX | TIME TO DASHPOT BOTTOM {SECONDS} - AVERAGE = 1.84 |

## 3.5 DIGITAL ROD POSITION INDICATION SYSTEM (S-07SF04)
Procedure ETT-SF-07040

### OBJECTIVES

The object of this test was to verify that the Digital Rod Position Indication System (DRPI) satisfactorily performs the required indication and alarm functions prior to the control rods being operated.

### SUMMARY OF RESULTS

This test was performed satisfactorily with all alarm and indication functions working as required. Acceptance criteria for verifying control rod position vs. DRPI output during rod movement was transferred to procedure ETT-SF-07014, CRDM Timing Test (ambient).

### DISCUSSION

Performance of this test involved five (5) separate test sections; (1) Half Accuracy Test; (2) Rod Bottom Test; (3) Rod Error Test; (4) Simulated Data Transmission (Normal Codes); and (5) Simulated Data Transmission (Error Codes) Test.

The Half Accuracy Test checks the ability of DRPI to signal applicable warnings when either A data or B data from a Data Cabinet is rejected (simulating Failure). This is performed by using the Accuracy Mode switch on the rear of the DRPI display in the control room. This section was performed satisfactorily.

The Rod Bottom Test checks the ability of the DRPI to signal the applicable warnings when rod bottom codes are manually inputted to the DRPI display by means of the RBT-1 and RBT-2 test push-buttons. A problem was encountered with the RPI Rod DEV annunciator not working which was due to an annunciator input wire being lifted per a temporary modification. The temporary modification was restored and the annunciator worked satisfactorily after which this section was performed satisfactorily. In addition, it was discovered at the end of this section that the RPI ROD DEV annunciator would not clear; however, this was determined to be a normal condition whenever the shutdown banks are less than 210 steps and the control banks are not greater than 12 steps.

The Rod Error Test checks the ability of the DRPI to signal appropriate warnings when rod error codes are manually inputted to the DRPI display by means of the ETA and ETB test push-buttons. This test section was performed satisfactorily.

The Simulated Data Transmission (Normal Codes) Test checks the ability of the DRPI to display the appropriate rod positions by setting the switches on the test/monitor cards in the Data Cabinets to the appropriate position. The results are viewed at the DRPI display in the control room to verify the correct codes have been transmitted without error and the correct rod positions are displayed. This test section was performed satisfactorily.

The Simulated Data Transmission (Error Codes) Test check the ability of the DRPI to signal the appropriate warnings when rod position codes contain errors or differ by an excessive amount. This is done by inputting test signals to the test/monitor cards and checking the DRPI display to verify that the error have been correctly identified and the proper alarms displayed. This test section was performed satisfactorily.

## 3.6 RTD BYPASS FLOW MEASUREMENT (S-07BB01)
Procedure ETT-BB-07010

### OBJECTIVES

The objectives of this procedure were to determine the flow rate necessary to achieve the design reactor coolant transport time in each resistance temperature detector (RTD) bypass loop, to measure the flow rate in each RTD bypass loop to ensure that the transport times are acceptable, and to set and verify the RTD bypass loop low flow alarm setpoint.

### SUMMARY OF RESULTS

The flow rate in the hot leg RTD bypass loop of reactor coolant loops 1, 2, 3 and 4, with flow in both the hot and cold bypass loops was greater than or equal to the minimum required flow for the design transport time of one second.

The results are tabulated below:

| LOOP | MINIMUM REQUIRED BYPASS FLOW (GPM) | ACTUAL BYPASS FLOW (GPM) |
|---|---|---|
| 1 | 100.22 | 115 |
| 2 | 99.72 | 153 |
| 3 | 100.47 | 157 |
| 4 | 111.03 | 129 |

The flow rate in the cold leg RTD bypass loop of reactor coolant loops 1, 2, 3 and 4 with flow in both the hot and cold leg bypass loops was greater than or equal to the minimum required flow for the design transport time of one second.

The results are tabulated below:

| LOOP | MINIMUM REQUIRED BYPASS FLOW (GPM) | ACTUAL BYPASS FLOW (GPM) |
|---|---|---|
| 1 | 64.70 | 147 |
| 2 | 63.38 | 109 |
| 3 | 64.98 | 137 |
| 4 | 65.47 | 133 |

Low flow alarm setpoints were determined and the instruments were adjusted within tolerance. The low flow alarms were verified.

### DISCUSSION

Flow in the hot leg bypass loops was measured with no flow in the cold leg bypass loops; likewise, flow in the cold leg bypass loops was measured with no flow in the hot leg bypass loops. To verify that Acceptance Criteria were met, the following equations were used.

$$F_h^1 = \frac{F_t}{(1+ F_c/F_h)}$$

AND

$$F_c^1 = F_t - F_h^1$$

WHERE: $F_h^1$ = Calculated hot leg RTD bypass flow with flow in both the hot and cold leg bypass loops.

$F_c^1$ = Calculated cold leg RTD bypass flow with flow in both the hot and cold leg bypass loops.

$F_h$ = Measured hot leg bypass flow.

$F_c$ = Measured cold leg bypass flow.

$F_t$ = Measured total bypass flow.

## 3.7 PRESSURIZER HEATER AND SPRAY CAPABILITY TEST (S-07BB02)
Procedure ETT-BB-07020

### OBJECTIVES

The objectives of this procedure were to determine the rate of pressure reduction caused by fully opening the pressurizer spray valves and the rate of pressure increase from the operation of all pressurizer heaters.

### SUMMARY OF RESULTS

The pressurizer was depressurized to 1985 psig in 124 seconds. It was also pressurized to 2285 psig in 156 seconds. This met the acceptance criteria. The depressurization and pressurization are depicted in Figures 3.7.1 and 3.7.2, respectively.

### DISCUSSION

The depressurization was performed by placing the pressurizer level controller in manual, de-energizing the pressurizer heaters and then opening spray valves PCV-455B and PCV-455C. The time that it took the Reactor Coolant System (RCS) to depressurize to 1985 psig was then recorded.

The pressurization was performed by placing the pressurizer level controller in manual, placing the spray valves in manual, placing a power signal to the control group heaters so they wouldn't cycle off, placing the PORV's in the closed position, and then energizing the pressurizer heaters. The time that it took the RCS to pressurize to 2285 psig was then recorded.

PRESSURE RESPONSE TO ACTUATION
OF ALL PRESSURIZER HEATERS

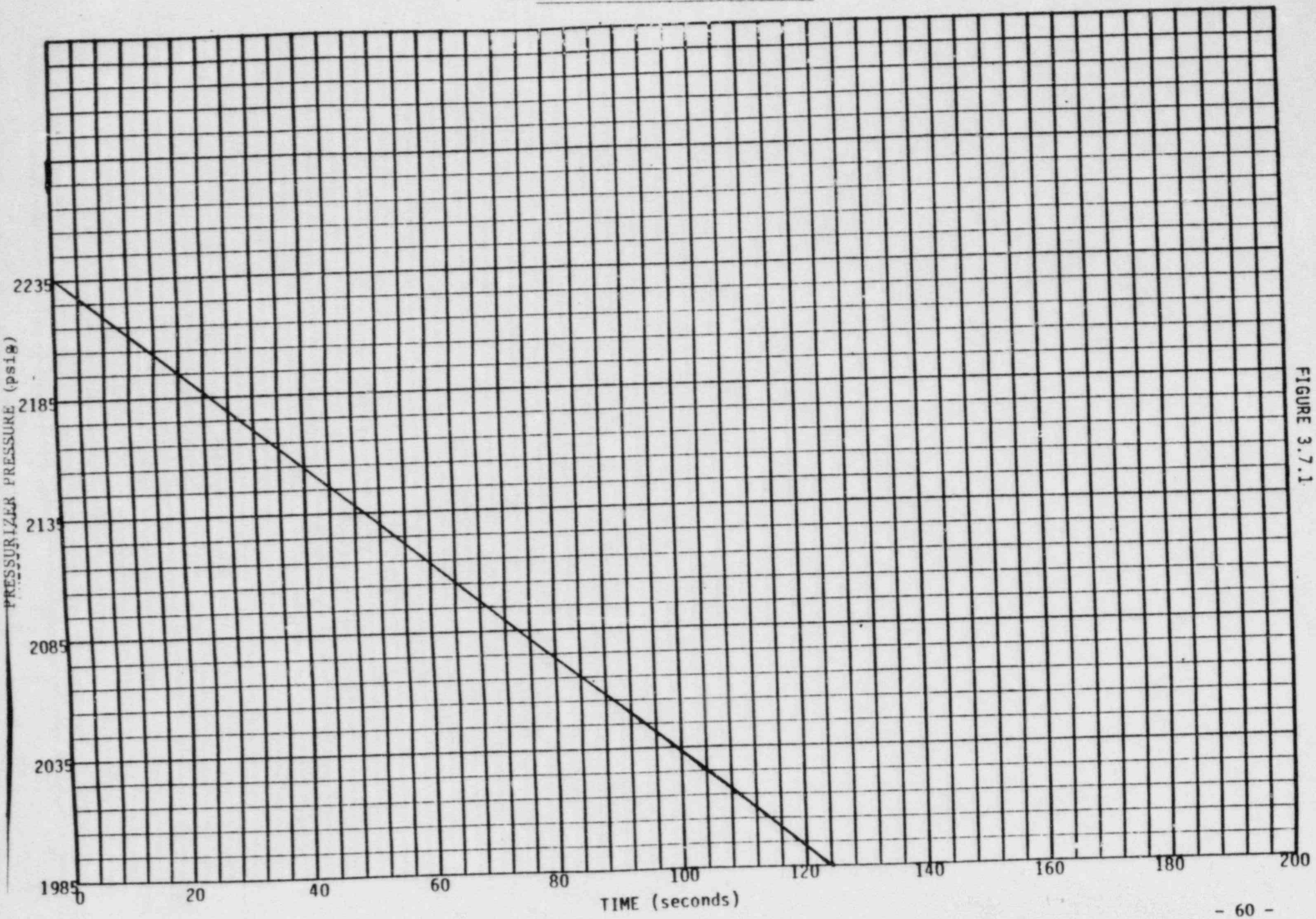



FIGURE 3.7.1

PRESSURE RESPONSE TO ACTUATION
OF ALL PRESSURIZER HEATERS

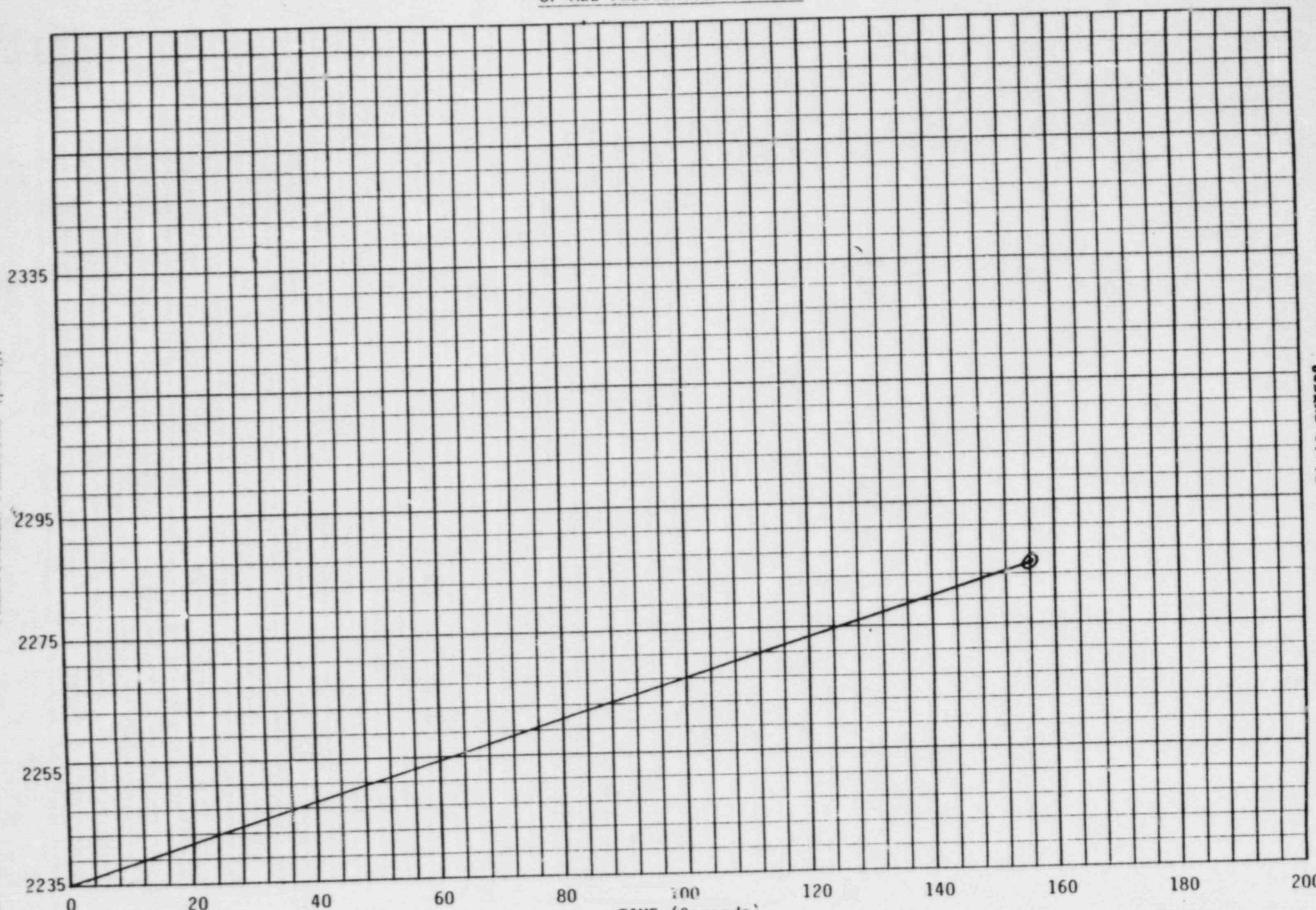



FIGURE 3.7.2

3.8 REACTOR COOLANT SYSTEM FLOW MEASUREMENT (S-07BB03)
Procedure ETT-BB-07030

OBJECTIVES

The objectives of this test were:

1. To confirm that Reactor Coolant System (RCS) Flow with the core installed and at hot standby conditions, but before initial criticality, is equal to or greater than 90% of the thermal design value by measuring loop elbow differential pressures and using them to compute RCS Flow.

2. To confirm using calorimetric data taken at 50, 75, 90 and 100% power, that RCS Flow is equal to or greater than the thermal design value.

SUMMARY OF RESULTS

Based on calculations performed at the various plant conditions - hot standby and 50, 75, 90 and 100% power - all acceptance criteria were met and the test was performed satisfactorily. A compilation of the required flow rates versus the actual flow rates for each power level is provided below:

| Power Level | Required Flow (GPM) | Actual Flow (GPM) |
|---|---|---|
| Hot Standby | 344,520 | 432,682 |
| 50% | 382,800 | 412,823 |
| 75% | 390,838 | 397,100 |
| 90% | 390,838 | 411,862 |
| 100% | 390,838 | 411,100 |

DISCUSSION

The method for determining RCS Flow during precritical testing (at hot standby) was by measuring loop elbow differential pressure ($\Delta P$). This was done by measuring the output voltage from each loop elbow tap differential pressure transmitter (3 per loop), correcting the readings for instrument drift and then converting the voltages to $\Delta P$ (in. $H_2O$). The three transmitter flows were then averaged to determine each loop flow rate. The total RCS Flow was determined by adding up the flow for each loop and was calculated to be 432,682 GPM which met the acceptance criteria of 344,520 GPM. This flow calculation is not as precise as using calorimetric data obtained during power operations, but provides sufficient accuracy for plant startup prior to power ascension.

For determining the RCS Flow during power ascension, hot leg temperature ($T_{Hot}$), cold leg temperature ($T_{Cold}$) and steam generator thermal power ($Q_1$) data for each loop was obtained from the respective Thermal Power Measurement and Statepoint Data Collection procedure for

each power level. From this data and using values obtained from the ASME Steam Tables, flow was calculated by the following formula:

$$\text{FLOW (GPM)} = \frac{(V_c \ (Q_i - 12 \times 10^6) \ )}{(h_{Hot} - h_{Cold}) \ 8.02}$$

where h is the enthalpy for each loop $T_{Hot}$ and $T_{Cold}$ and $V_c$ is the specific volume for each cold leg. Flows for each loop were than added together to obtain a total RCS Flow Rate. The calculated flows for each power level met the specified acceptance criteria as delineated in the previous section.

3.9 REACTOR COOLANT SYSTEM FLOW COASTDOWN (S-07BB04)
Procedure ETT-BB-07040

OBJECTIVES

The objectives of this procedure were to measure the rate of which reactor coolant flow changes subsequent to various reactor coolant pump trips, to compare the measured flow coastdown profiles assumed in the FSAR Safety Analysis, to verify that the reactor coolant system low-flow trip delay times are less than or equal to the low-flow trip delay times assumed in the FSAR Safety Analysis, and to verify that the reactor coolant pump undervoltage trip and underfrequency trip delay times are within design specifications.

SUMMARY OF RESULTS

The slope of the inverse of flow versus time curve, for the 4/4 case Figure 3.9.1 was 0.090, compared with the slope of the inverse of Figure 3.9.2 of 0.078. Because the measured slope was greater than that assumed in the FSAR Safety Analysis, Westinghouse was requested to evaluate the impact. In ensuing response, Westinghouse stated that following a re-analysis of FSAR Chapter 15.3.2, all safety criteria were met. The re-analysis will be included in the next revision of the FSAR.

Figure 3.9.2 shows the data that was taken during the performance of this procedure superimposed on FSAR Figure 15.3-9. The best estimate actual core flow is not greater than or equal to that assumed in the FSAR Safety Analysis (FSAR Figure 15.3-9). Westinghouse re-analyzed FSAR Chapter 15.3-2, and determined that all safety criteria were met.

Figure 3.9.3 shows that data that was taken during the performance of this procedure superimposed on FSAR Figure 15.3-1. The best estimate actual core flow for the 2/4 case was not greater than that assumed in the FSAR Safety Analysis and hence, assistance was requested from Westinghouse to evaluate the impact. The Westinghouse response states that following a re-analysis of FSAR Chapter 15.3.2, all safety criteria were met.

Table 3.9.1 shows the data that was used to verify that the trip delay times were acceptable. At the time the test was performed, it was assumed that the gripper response times were the same as the acceptance criteria of procedure ESP-SA-02415, (Response Time Test For Steam Generator Level High-High Feedwater Isolation and Turbine Trip) namely 0.15 seconds. Since that time, successful completion of procedure ESP-SA-02415 indicates that this was a conservative assumption and hence, all safety criteria and design specifications were met.

DISCUSSION

The procedure for this test involved tripping two or four reactor coolant pumps (depending upon the portion of the test being conducted) from the full-flow condition to produce a flow coastdown. A strip chart recorder was employed to record loop flows and reactor coolant pump

breaker position. The Reactor Trip Time and Longest Flow Sensor Response times were obtained from procedure CS-03SB01 (Reactor Protection Logic System Pre-Operational Test). The gripper release times used were the acceptance criteria of ESP-SA-02415, as the gripper release times had not been measured at the time of the test.

The loop flow values were normalized to the loop flows existing with all reactor coolant pumps in operation and were adjusted for sensor time delay. These normalized flows were used to obtain a second order least squares fit for each curve and these fitted curves averaged to obtain the total core flow ratio for each scenario. These corrected total core flow ratio curves were compared to the FSAR curves.

TABLE 3.9.1

TRIP DELAY TIMES

| | Reactor Trip[2] Breaker Trip Time | Longest Sensor[2] Response Time | Gripper[3] Release Time | Total Time | Acceptance Criteria |
|---|---|---|---|---|---|
| Reactor Coolant Pump Undervoltage | 0.08 | 1.150 | 0.15 | 1.245 | $< 1.5$ |
| Reactor Coolant Pump Underfrequency | 0.08 | 0.155 | 0.15 | 0.385 | $< 0.6$ |
| Low Flow | 0.08 | 0.085 | 0.15 | 0.315 | $\leq 1.0$ |

(1) All times in seconds.

(2) From Pre-Operational Test CS-03SB02, Data Sheet 8.4.1, and Supplemental Record (JTG Punchlist Item #181).

(3) Using Acceptance Criteria from ESP-SA-02415.

4/4 INVERSE TOTAL CORE FLOW
VERSUS TIME

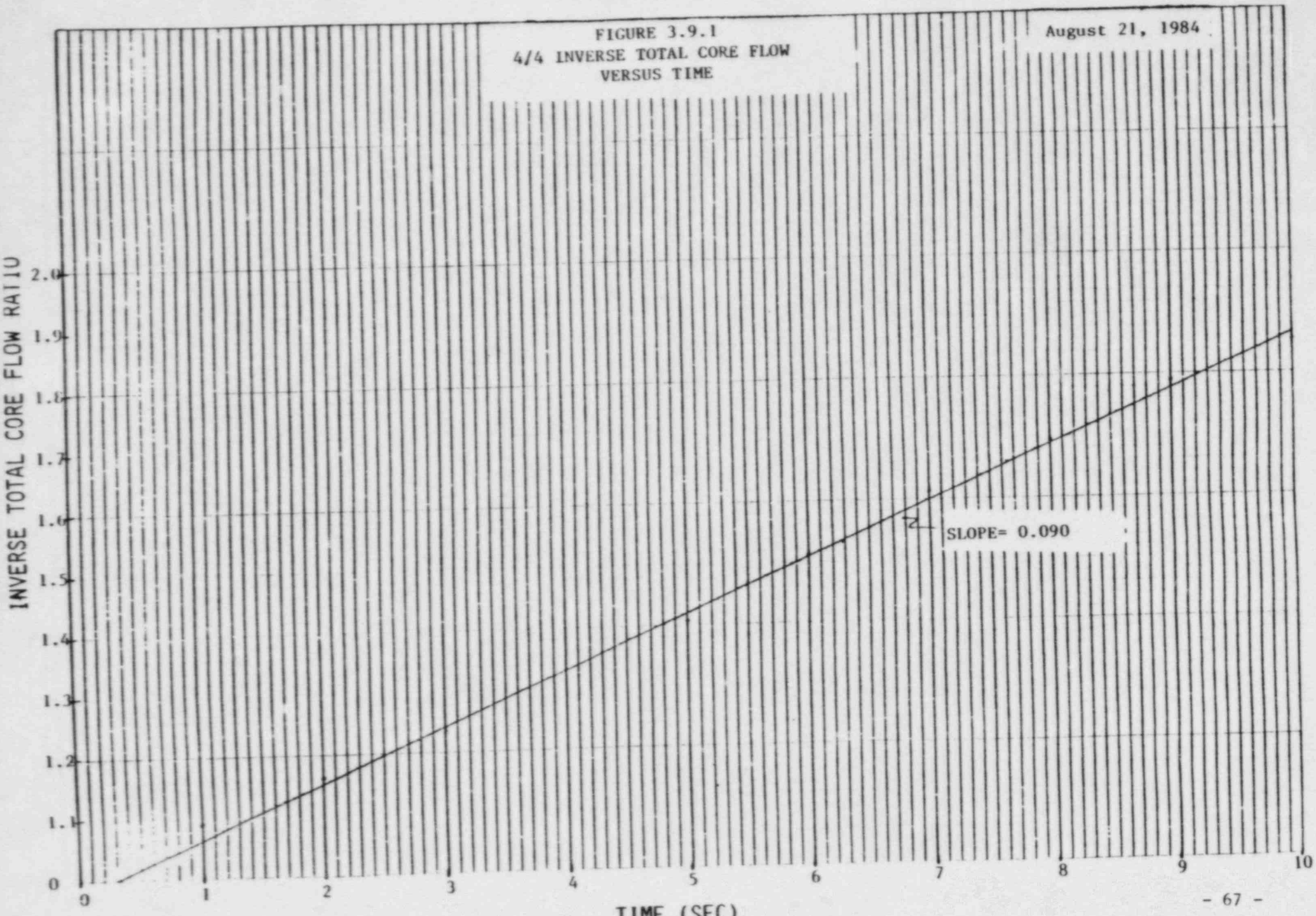
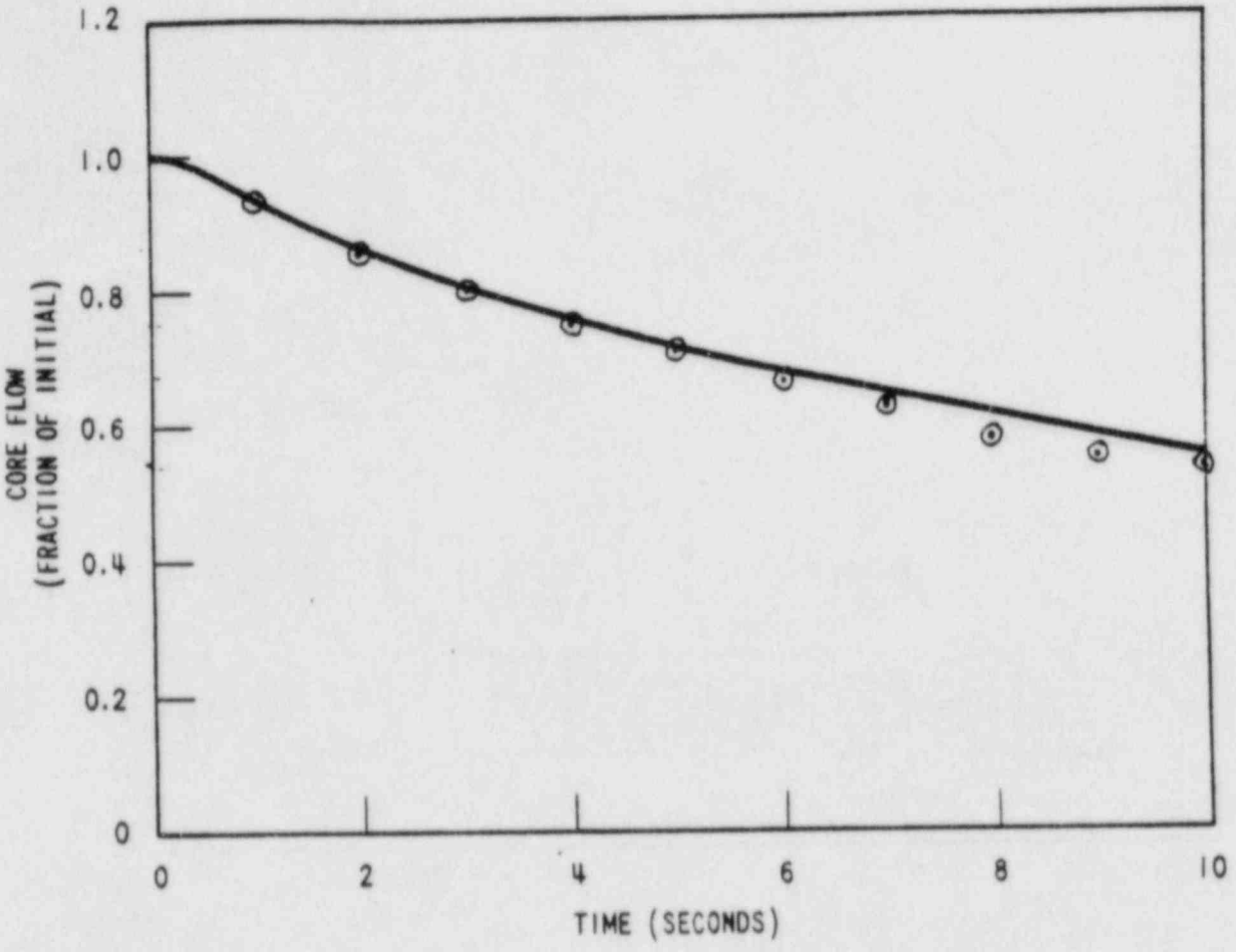


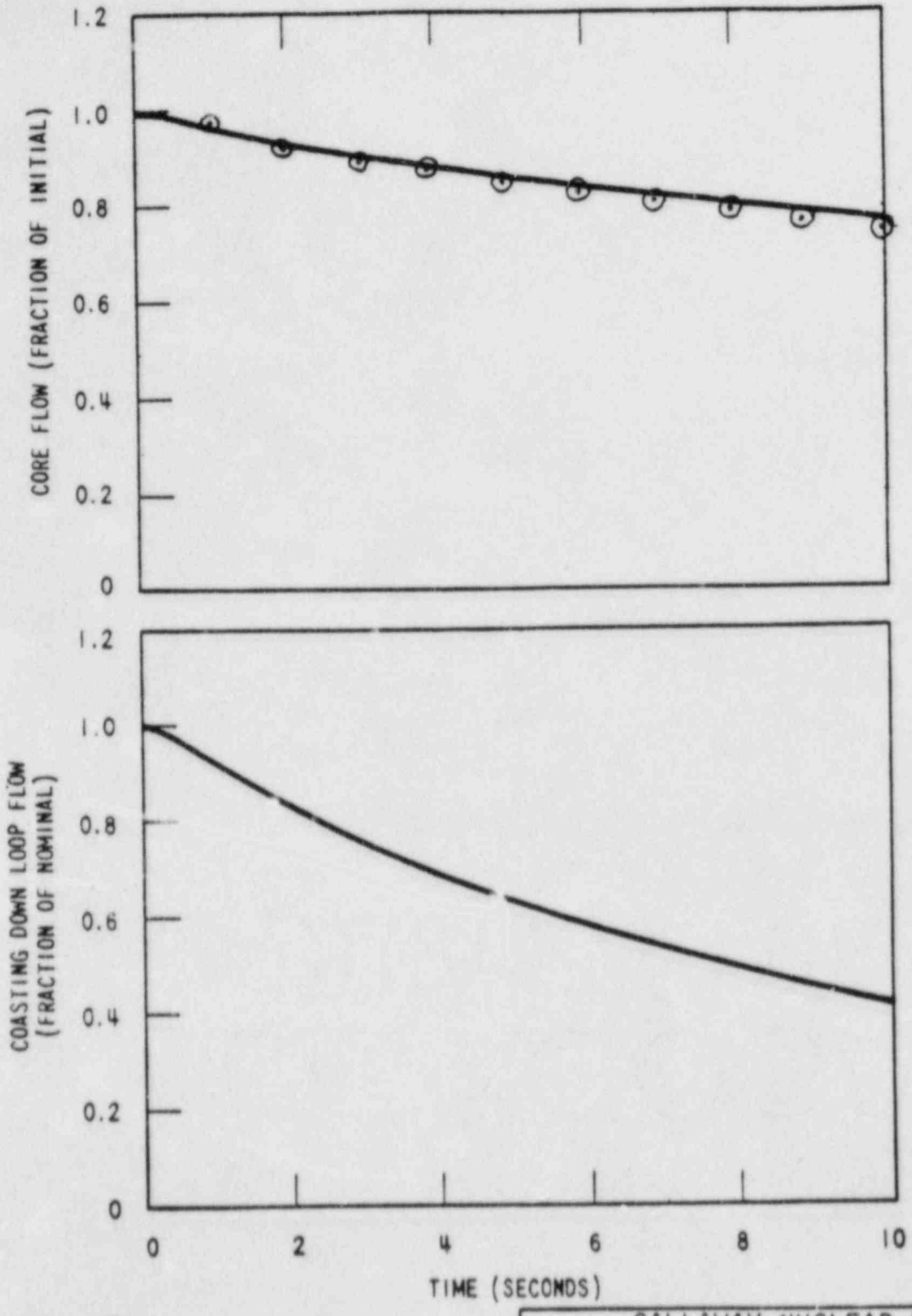
FIGURE 3.9.2




⊙ DATA POINT

| CALLAWAY NUCLEAR PLANT |
| --- |
| FSAR FIGURE 15.3-9 |
| **CORE FLOW COASTDOWN FOR FOUR LOOPS IN OPERATION, FOUR PUMPS COASTING DOWN** |

FIGURE 3.9.3




⊙ DATA POINT

CALLAWAY NUCLEAR PLANT

FSAR FIGURE 15.3-1

**FLOW TRANSIENTS FOR FOUR LOOPS IN OPERATION, TWO PUMPS COASTING DOWN**

## 3.10 PRESSURIZER CONTINUOUS SPRAY FLOW VERIFICATION (S-07BB05)
Procedure ETT-BB-07050

### OBJECTIVES

The objectives of this test were to establish a setting for the pressurizer continuous spray throttle valves to obtain an optimum continuous spray flow and to establish the setpoint for the pressurizer spray line low temperature alarms.

### SUMMARY OF RESULTS

The continuous spray flow valves BB-V-082 and BB-V-083 were throttled to 5½ and 5 turns open, respectively. This established a break-point temperature of 539°F for both pressurizer spray loops. The setpoints for the pressurizer spray line low temperature alarms were established at 504°F.

### DISCUSSION

This test throttled pressurizer continuous spray flow valves BB-V-0082 and BB-V-0083 and established a "break-point". The "break-point" is defined as the point where a change in valve position produces an abrupt change in spray line equilibrium temperature. The valve is left in this position in order to maintain an optimum continuous spray flow. This optimum spray flow is intended to avoid thermal shock to the spray lines in the event of initiation of spray flow. This test also establishes the setpoint for the pressurizer spray line low temperature alarms as the equilibrium temperature at the "break-point" minus 35±15°F. The data obtained by throttling these valves are presented in Figures 3.10.1 and 3.10.2.

Spray Line Equilibrium Temperature
Vs. BB-V-0083 Position

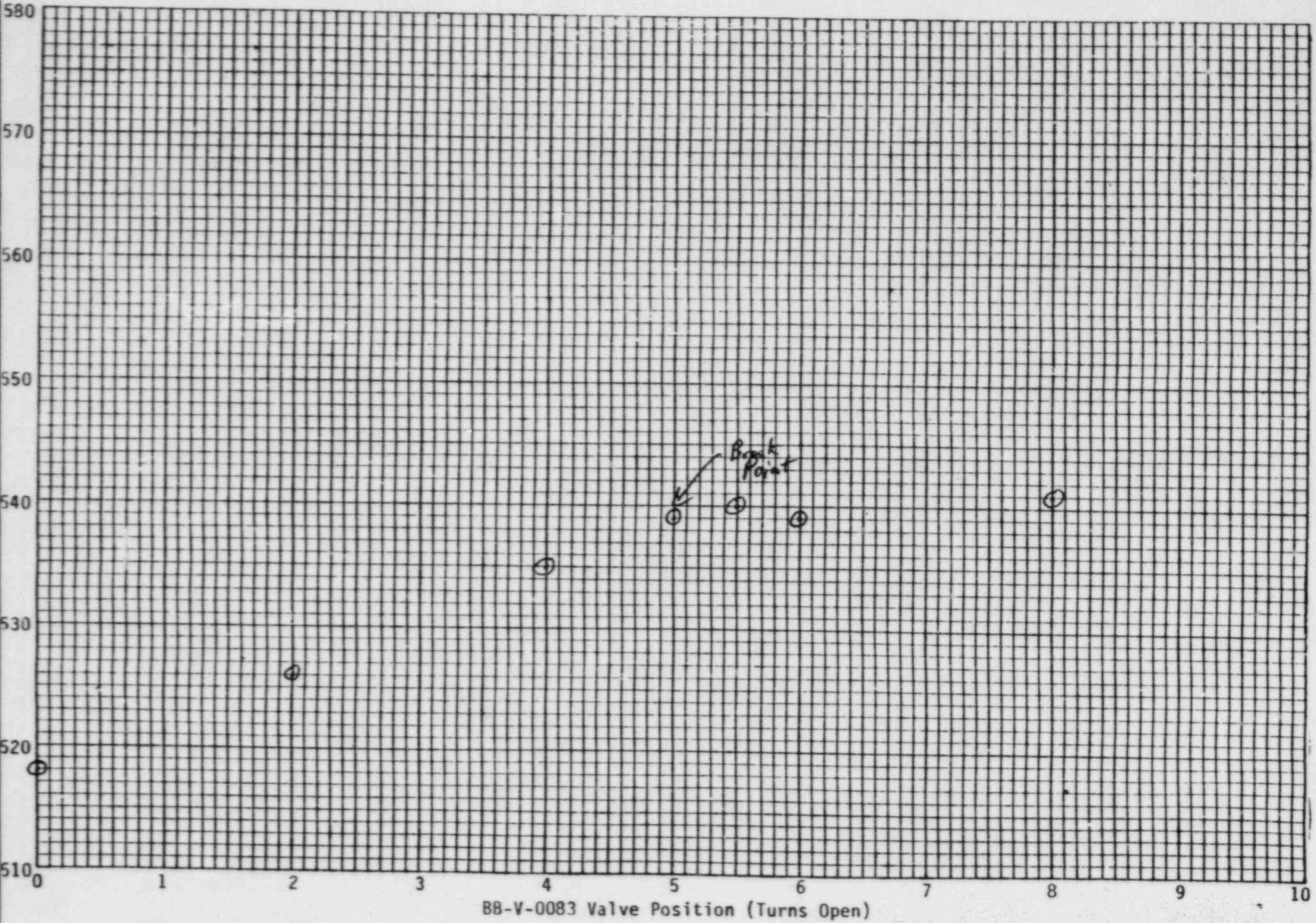



FIGURE 3.10.1

Spray Line Equilibrium Temperature
Vs. BB-V-0082 Position

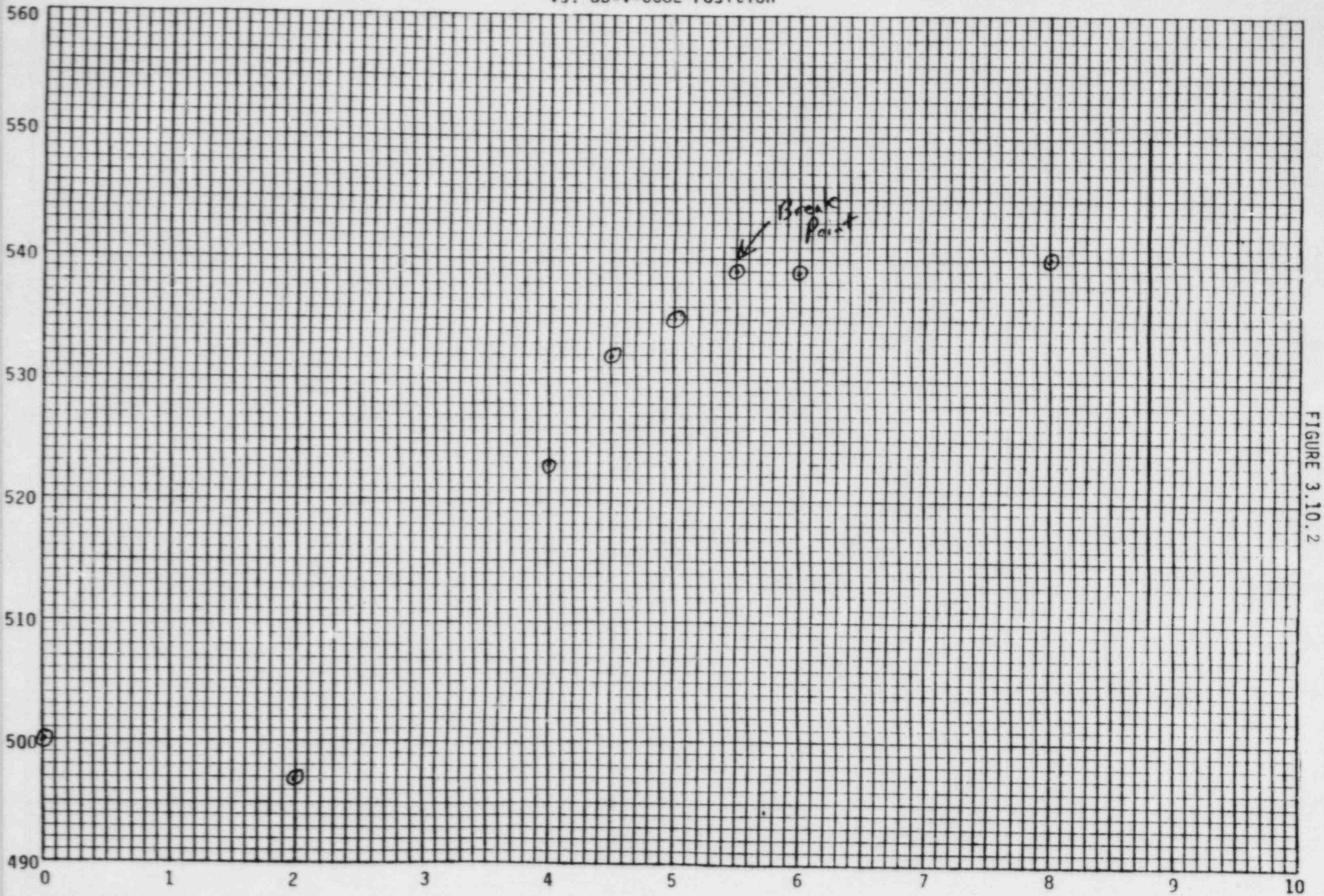



FIGURE 3.10.2

## 3.11 RTD/TC CROSS CALIBRATION (S-07BB06)
Procedure ETP-ZZ-00008

### OBJECTIVES

The objectives of this test were to verify the expected resistance versus temperature characteristics of installed RTD's at the 435°F and 557°F plateaus, to verify the incore thermocouple response, and to determine installation corrections for individual RTD's, as necessary.

### SUMMARY OF RESULTS

During the test, it was determined that RTD TE-411B had the leads crossed and RTD TE-433A had a bad card. All other RTD's exhibited a behavior such that the measured resistance corresponded to temperature that was within ±2.0°F of the average indicated reactor coolant system temperature.

No acceptance criteria was associated with the data taken at this plateau.

After correcting the problems found with RTD's TE-411B and TE-433A, the test was performed at the 557°F plateau. The resistance measured for each RTD corresponded to a temperature within ±2°F of the average indicated RCS temperature. It was determined that 10 RTD's indicated a temperature that was greater than 0.3°F different than the average of the RTD readings. For the RTD's that indicated a temperature greater than 0.3°F different than the RCS average indicated temperature, the difference was forwarded to I&C for input as a correction to the appropriate loop calibration procedure.

Because the acceptance criteria for this procedure was that at the 557°F plateau, all RTD's must indicate a temperature within ±2°F of the RCS calculated average indicated temperature and that any RTD indicating a temperature different from the average of the RTD temperature indications by more than ±0.3°F must be corrected, all acceptance criteria for this test were met.

At the 435°F plateau, the steam pressure saturation temperature was not within ±2°F of each of the thermocouple readings. This was attributed to steam line pressure losses at a lower steam pressure.

At the 557°F plateau, the thermocouple map average temperature was not within ±2°F of each of the individual thermocouple indications. This was due to three thermocouples that were giving an indication approximately 30°F below the average. When the three thermocouple readings that are questionable are removed from the average, each individual thermocouple reading is within ±2°F of the average.

### DISCUSSION

This procedure was performed at two temperature plateaus during the heatup phase, 435°F and 557°F. The data obtained at the 557°F plateau was used for development of the individual RTD correction factors.

Individual RTD readings were obtained by pulling the appropriate card from the Reactor Protection cabinets and reading the RTD resistance utilizing a card edge connector and a digital voltmeter. Two RTD's were read off each card edge connector, the second being a spare. Each RTD was read four different times, reversing the data taking sequence each time. The four RTD resistance readings were then averaged and converted to temperature using the manufacturer's data. Each indication was then compared with the average indicated RCS temperature to determine the installation correction factor. Before and after all the data was collected at each plateau, a thermocouple map was taken. The average RCS temperature was compared with the average of the temperatures obtained from the incore thermocouple map and the temperature calculated from the steam pressures recorded for each loop.

Testing at the 557°F plateau was performed satisfactorily after some difficulty was encountered in establishing stable RCS temperature. Review of the data determined that the mean temperature deviation was 0.50°F, excluding RTD TE-421B, having a deviation of 2.35°F. An analysis performed by I&C indicates that erroneous data was obtained for TE-421B during the test performed at the 557°F plateau. This was further substantiated in that data taken at the 435°F plateau did not indicate a problem and additional data taken at a later date did not indicate a problem. Hence, no corrective action was taken for TE-421B.

During the comparison of the calculated RCS average temperature and the incore thermocouple map, two problems were encountered. At the 435°F plateau, the steam pressure saturation temperature was not within ±2°F of each thermocouple readings. It was determined that this is due to line losses at a lower steam pressure. At the 557°F plateau, the thermocouple map average temperature was not within ±2°F of each of the thermocouple readings. This was because three thermocouples, (E-02, G-12, and L-08) were reading approximately 30°F lower than the other thermocouples. When the three erroneous thermocouples were eliminated from the average, the average thermocouple temperature was within ±2°F of each of the thermocouple readings.

## 4.1 INITIAL CRITICALITY (S-070004)
Procedure ETT-ZZ-07040

### OBJECTIVES

The objective of this procedure was to achieve initial reactor criticality in a cautious and controlled manner.

### SUMMARY OF RESULTS

Initial criticality was achieved at 0422 on October 2, 1984. After power was leveled out above the source range, Control Bank D was at 147 steps with a boron concentration of 1311 ppm. The design critical position was Control Bank D at 160 steps with a boron concentration of 1292 ppm.

### DISCUSSION

This procedure provided for a slow approach to initial criticality. The approach to criticality was initiated by withdrawing Shutdown Banks A, B, C, D and E, stopping at 50 step intervals to obtain inverse count rate ratio (ICRR) data. Control Banks were then withdrawn in 50 step increments to a final position with Bank D at 160 steps. The reactor coolant system was then diluted at 60 gpm to reduce the boron concentration. During the dilution, the reactor coolant system was sampled every 15 minutes and the ICRR plotted. When the ICRR reached .1, the dilution rate was reduced to 30 gpm. When criticality was achieved, reactor power was leveled off above the source range and critical data was obtained.

## 4.2 INVERSE COUNT RATE RATIO MONITORING FOR APPROACH TO INITIAL CRITICALITY (S-070003)

Procedure ETT-ZZ-07030

### OBJECTIVES

This procedure was used in conjunction with the initial criticality procedure to predict when criticality would occur by plotting Inverse Count Rate Ratios (ICRR).

### SUMMARY OF RESULTS

This procedure was completed satisfactorily.

### DISCUSSION

During rod withdrawal, the ICRR was computed at 50 step increments. During Reactor Coolant System (RCS) dilution, the ICRR was calculated every 15 minutes (concurrent with boron concentration sampling) and plotted as a function of time, boron concentration, and gallons of water added to the RCS. When the ICRR reached 0.1, the ICRR's were renormalized to 1.0, the dilution rate was reduced, and the ICRR's were plotted at 5 minute intervals. This was continued until the reactor was critical. The ICRR plots are shown in Figures 4.2.1 to 4.2.18.

# ICRR vs. RCC BANK POSITION

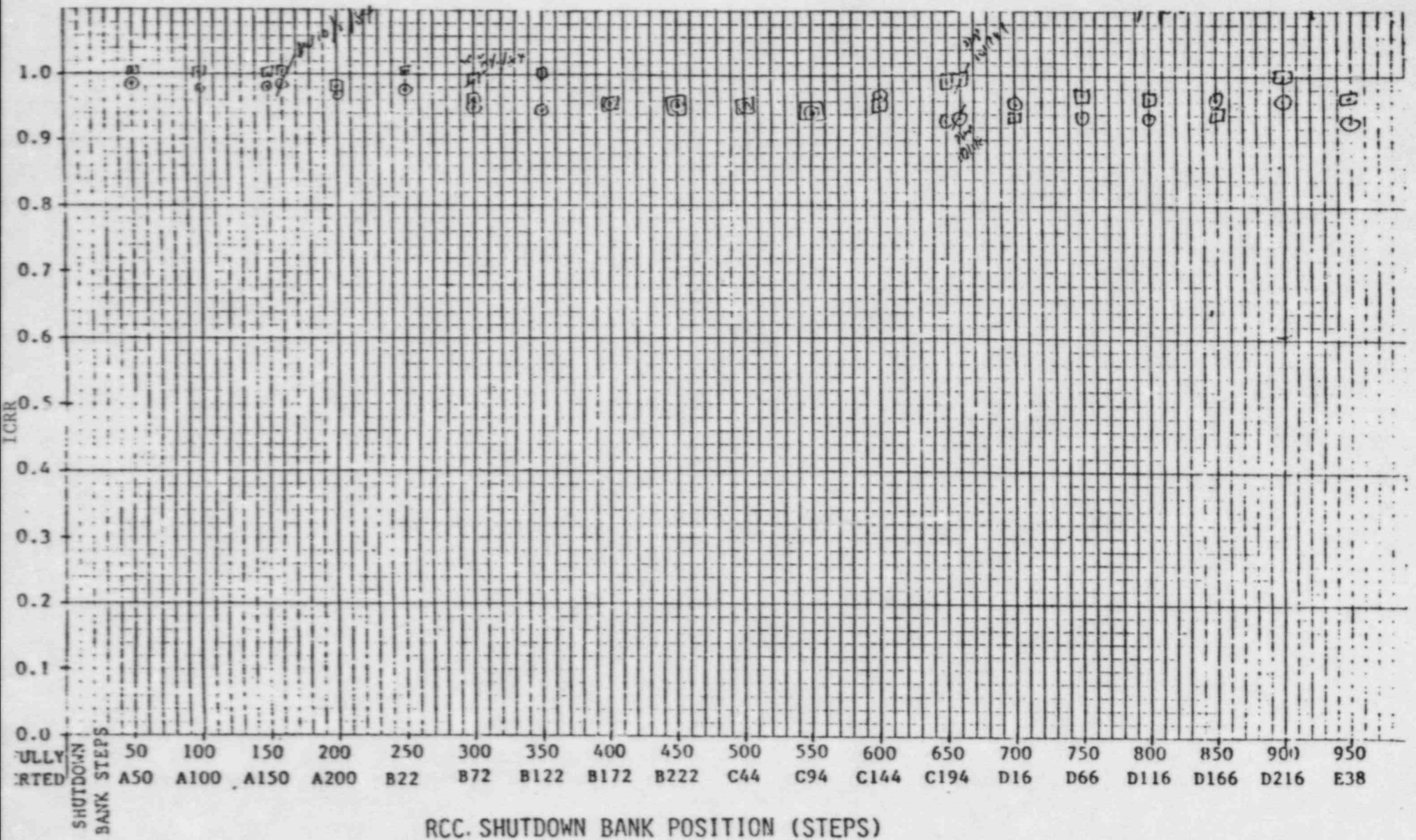



Figure 4.2.1

ICRR vs. RCC BANK POSITION

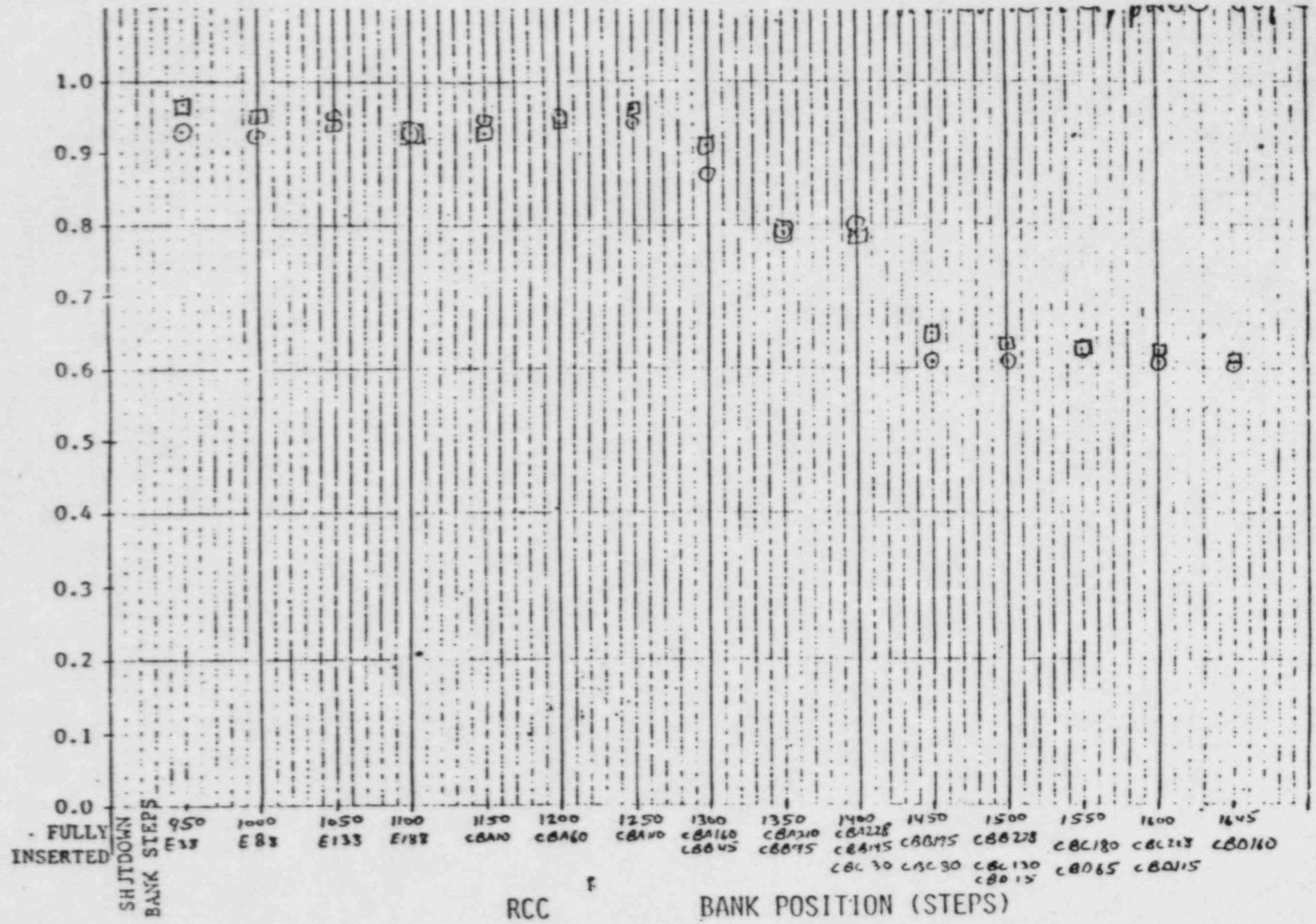



Figure 4.2.2

ICRR vs. TIME OF RCS BOREON DILUTION

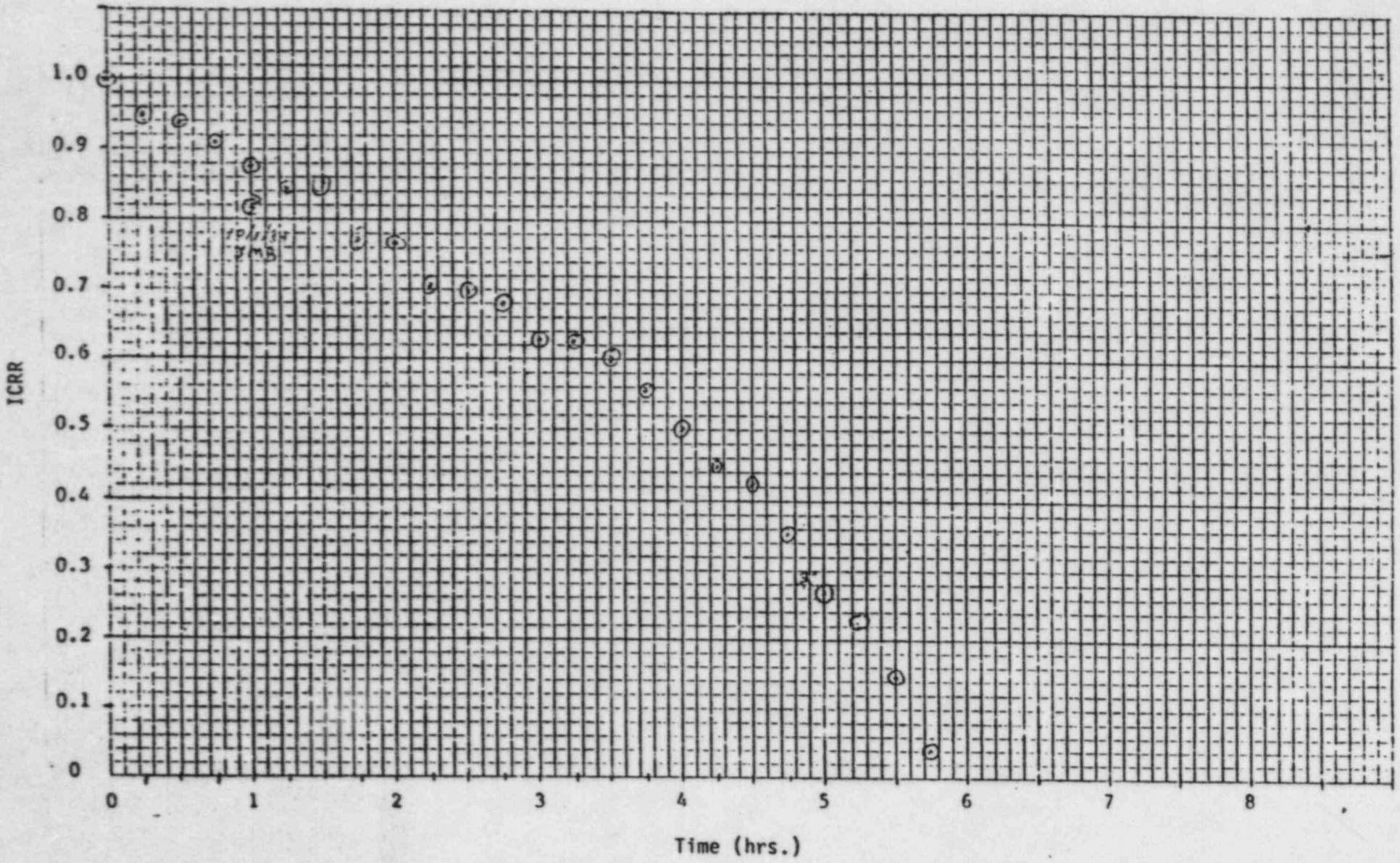



Figure 4.2.3

ICRR vs.TIME OF RCS BORON DILUTION, N31

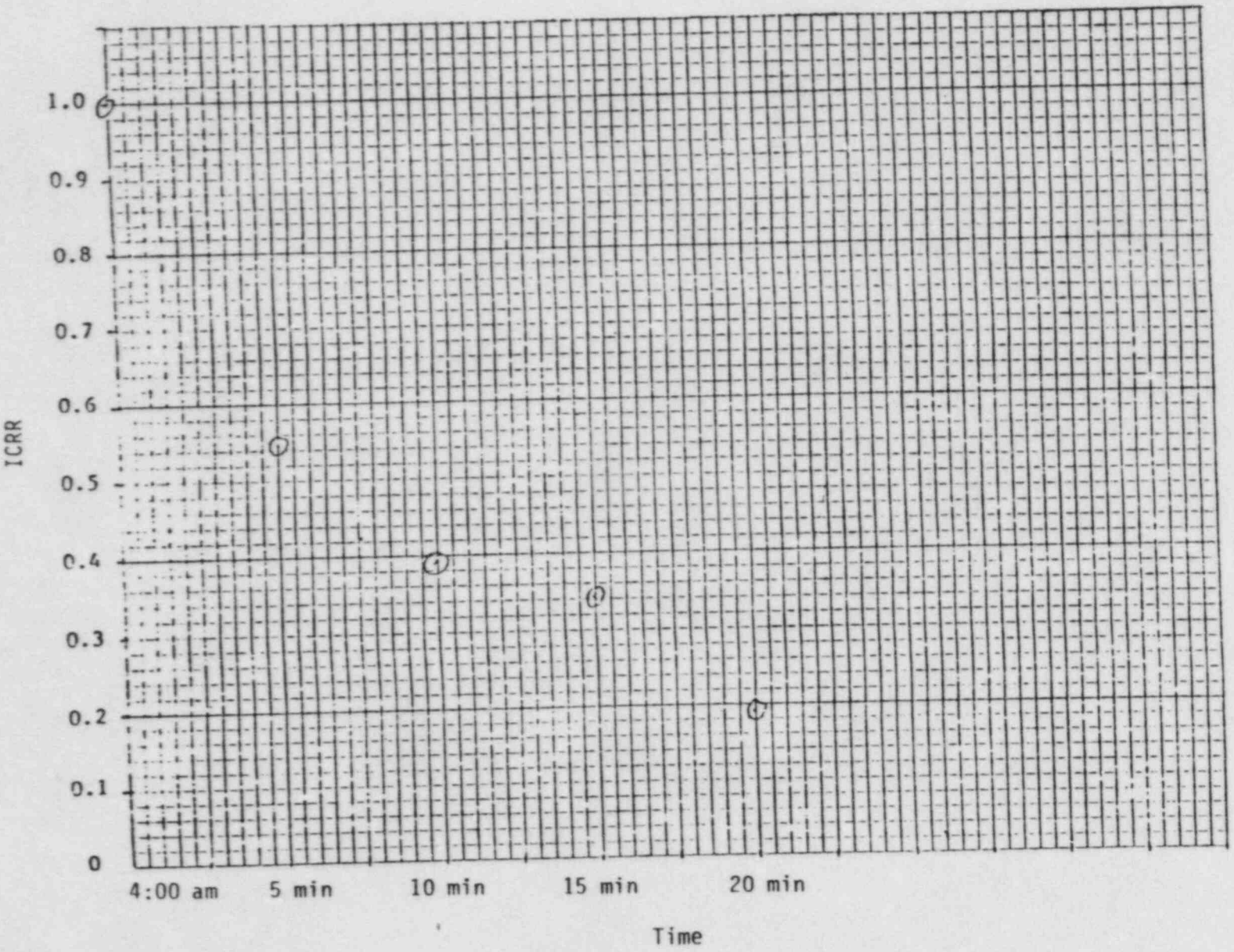



Figure 4.2.4

ICRR vs. TIME OF RCS BORON DILUTION, N32

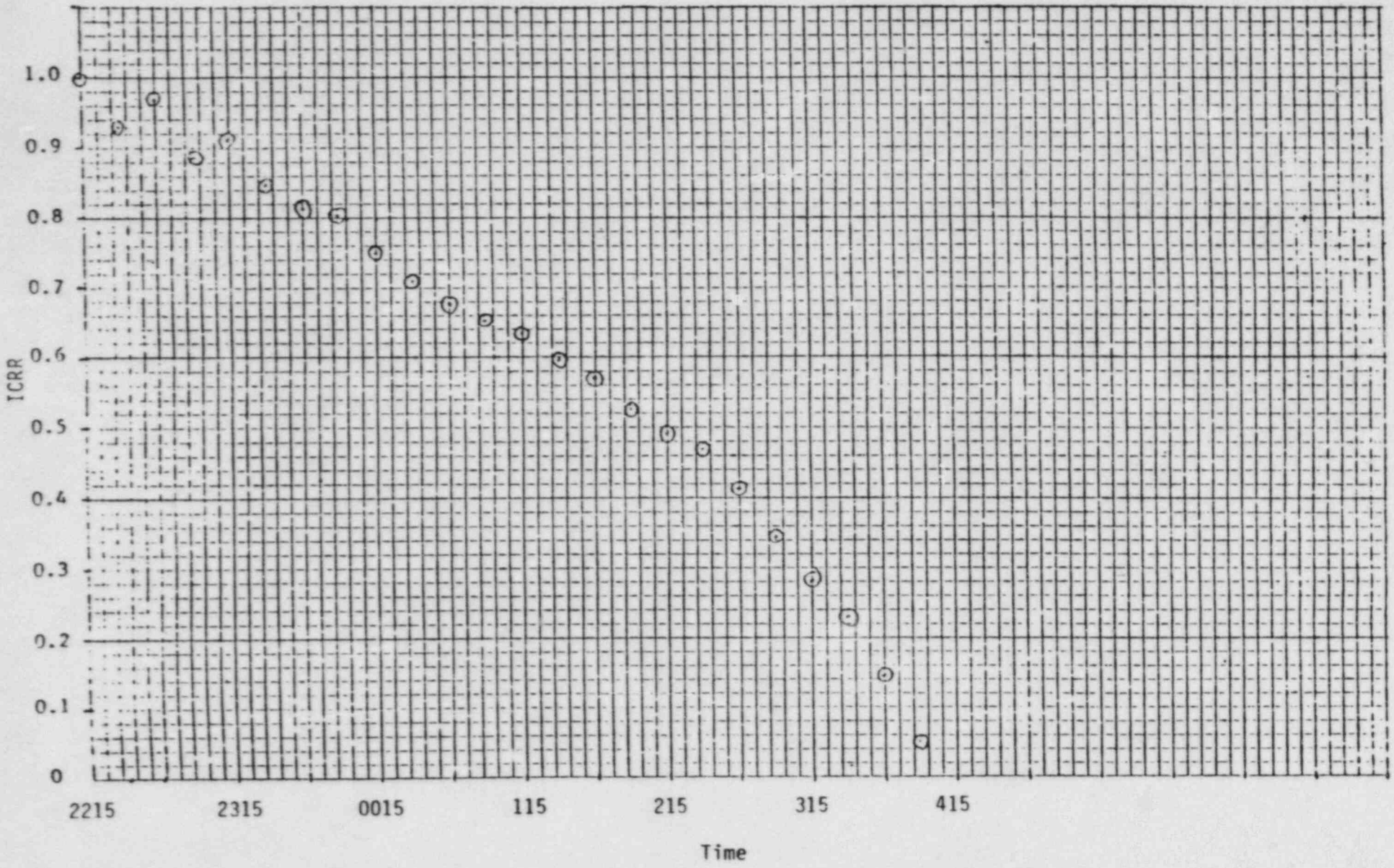
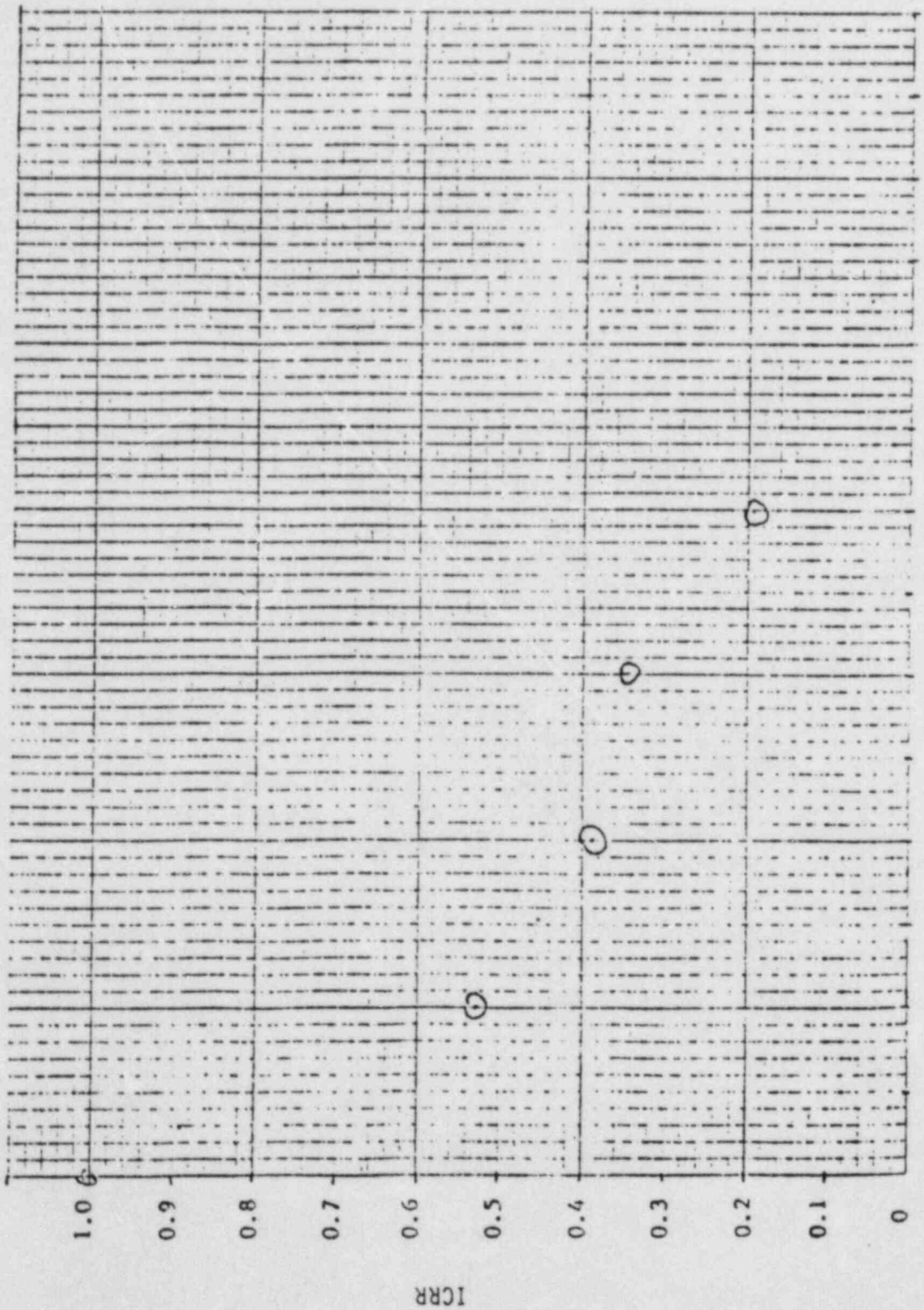



Figure 4.2.5

Figure 4.2.6


ICRR vs. TIME OF RCS BORON DILUTION, N32
ICRR
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
4:00 am
5 min
10 min
15 min
20 min
20 min
Time

ICRR vs. TIME OF RCS BORON DILUTION, N35

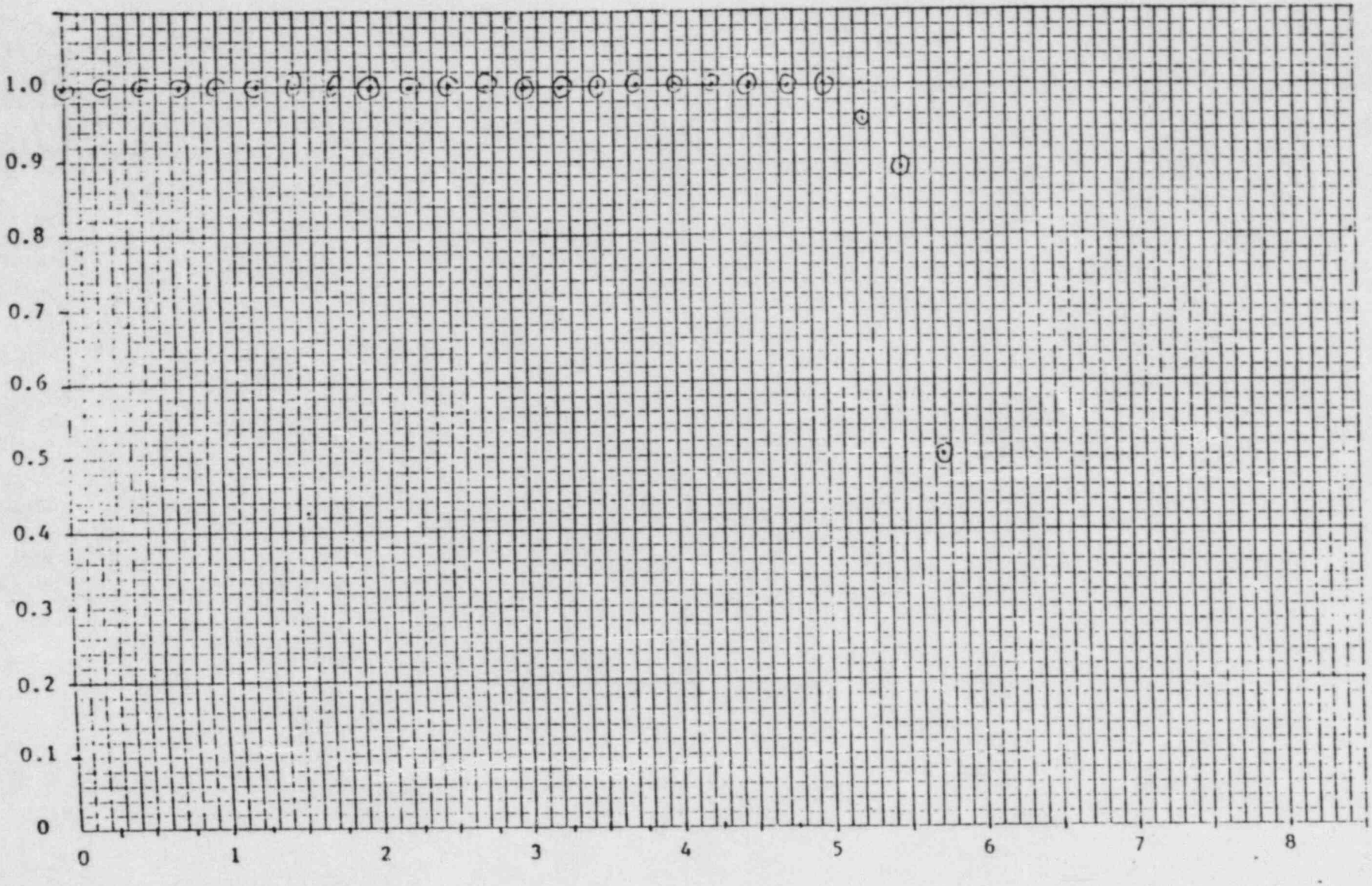



Figure 4.2.7

Figure 4.2.8

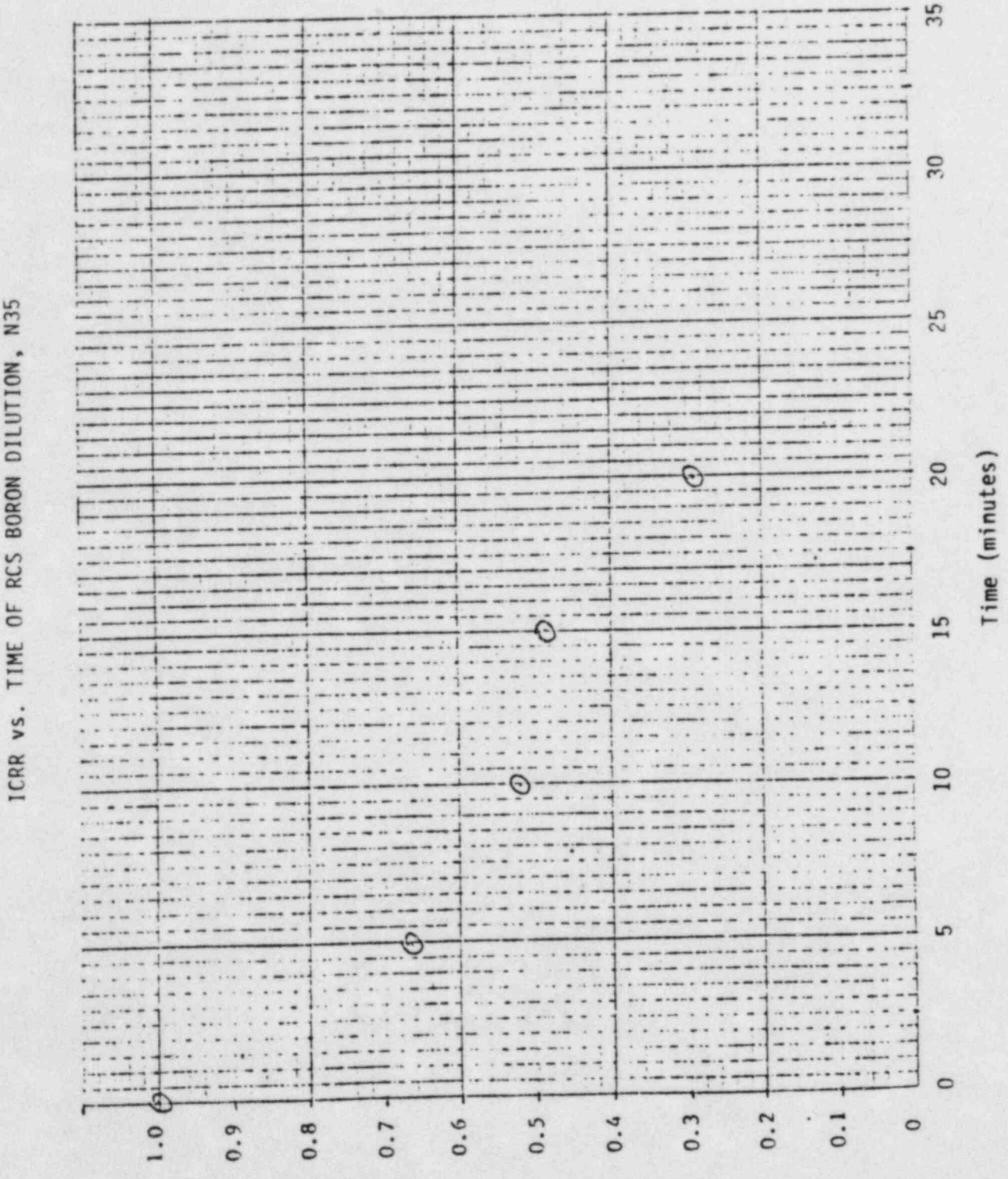

ICRR vs. TIME OF RCS BORON DILUTION, N35
ICRR
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
5
10
15
20
25
30
35
Time (minutes)

Figure 4.2.9

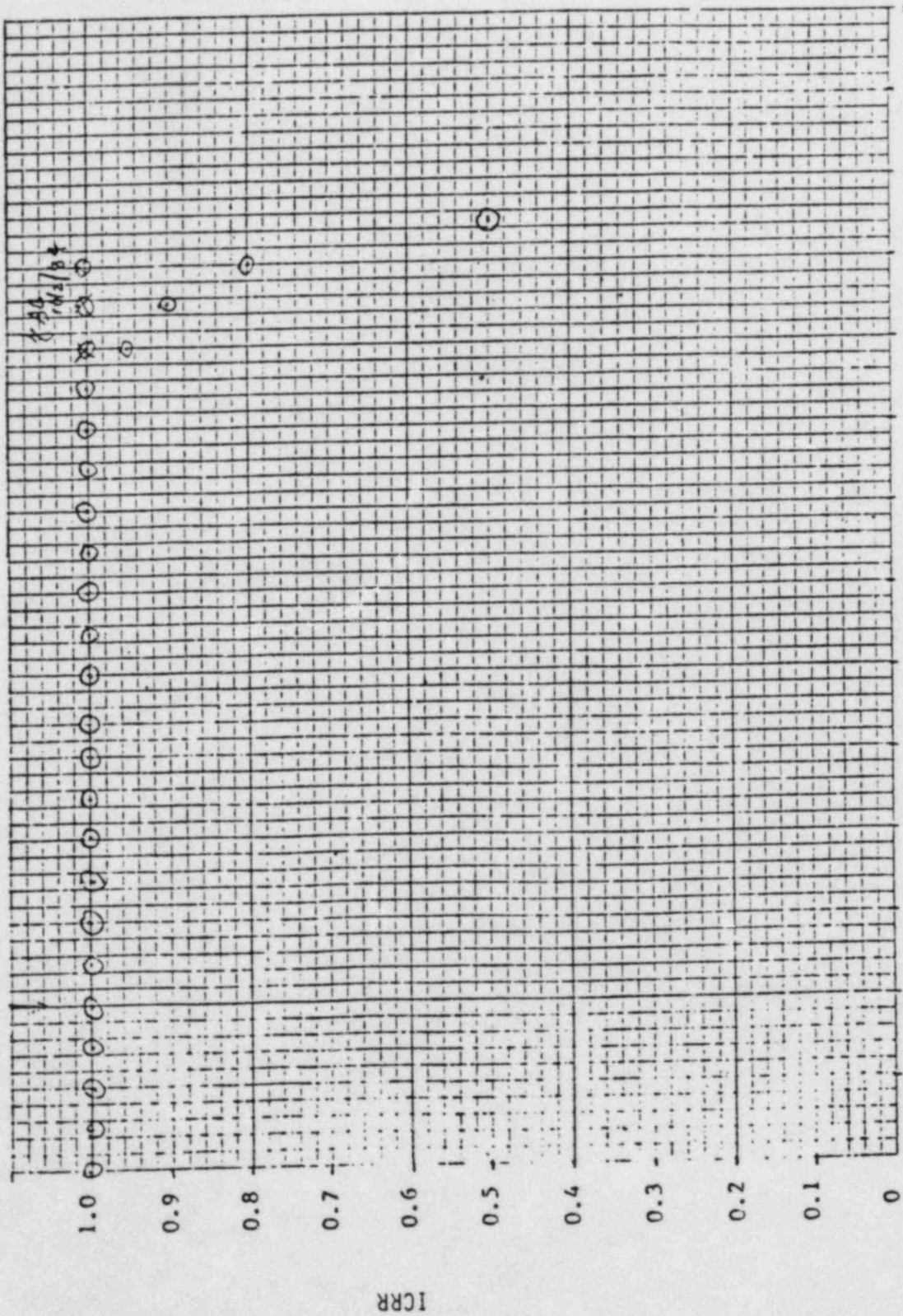
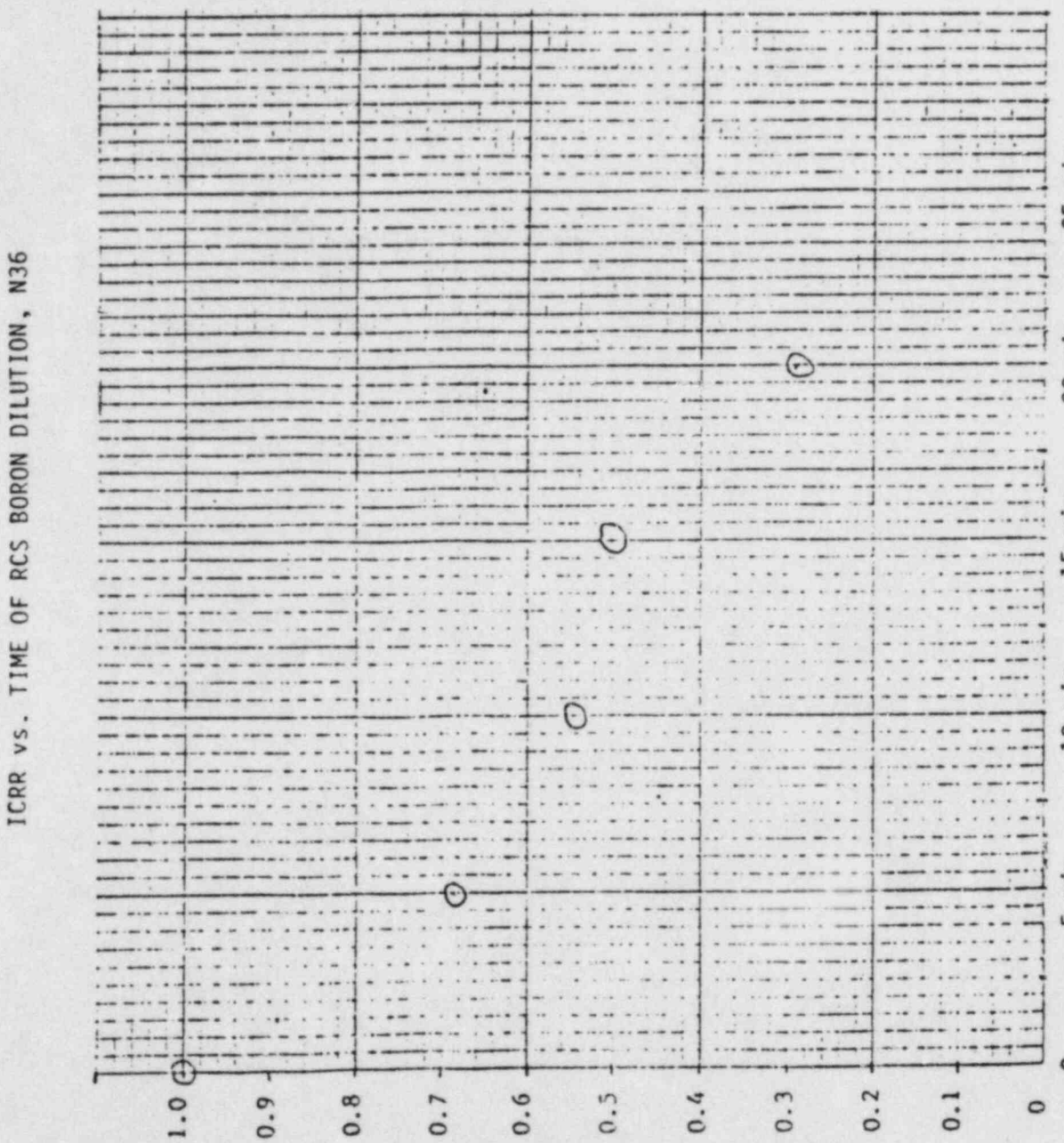

ICRR vs. TIME OF RCS BORON DILUTION, N36
ICRR
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
1
2
3
4
5
6
7
Time (nrs.)

Figure 4.2.10


ICRR vs. TIME OF RCS BORON DILUTION, N36
ICRR
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
5 min
10 min
15 min
20 min
25 min
Time (minutes)

ICRR vs. REACTOR MAKEUP WATER ADDITION, N31

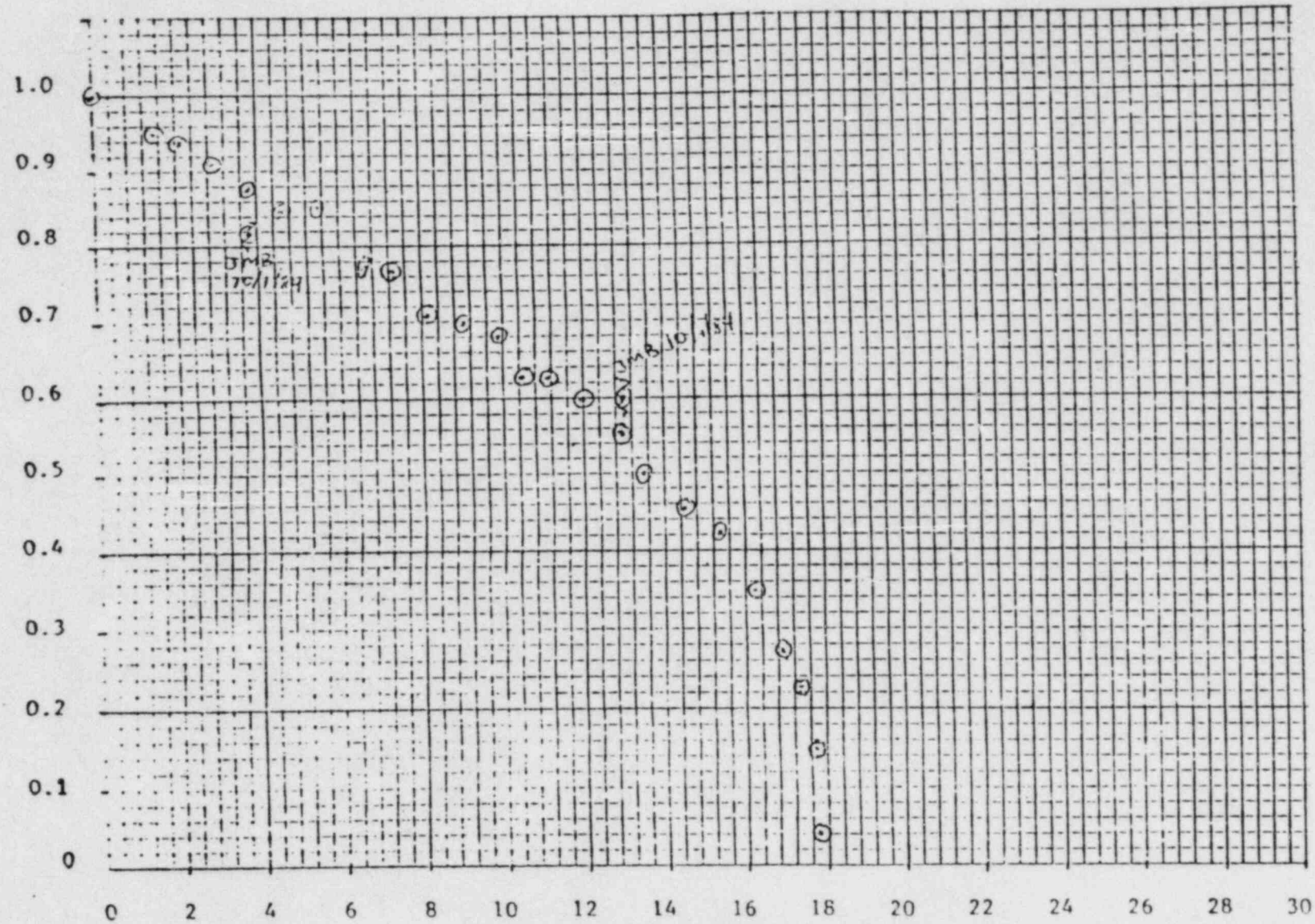



Figure 4.2.11

ICRR vs. REACTOR MAKEUP WATER ADDITION, N32

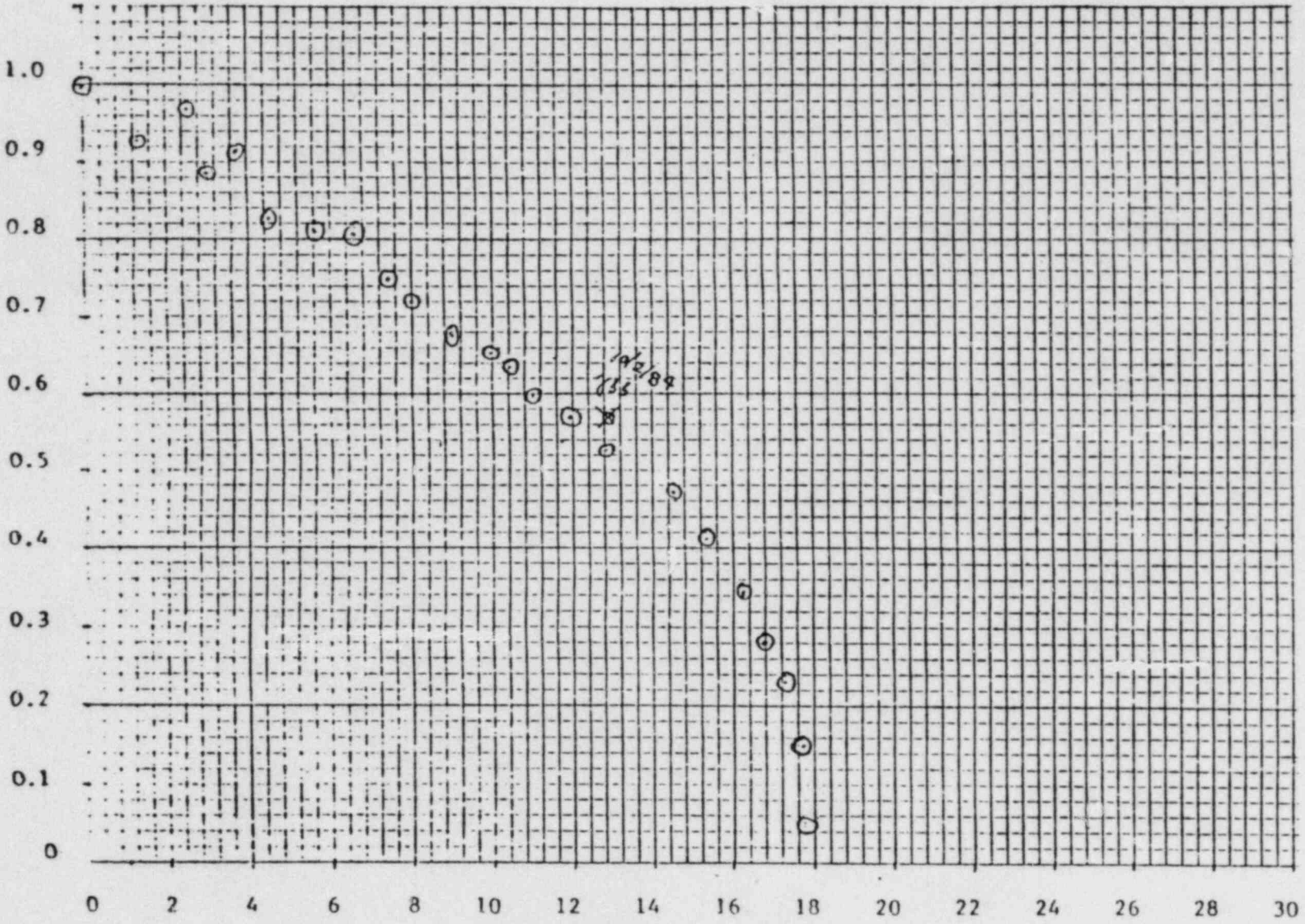



Figure 4.2.12

ICRR vs. REACTOR MAKEUP WATER ADDITION, N35

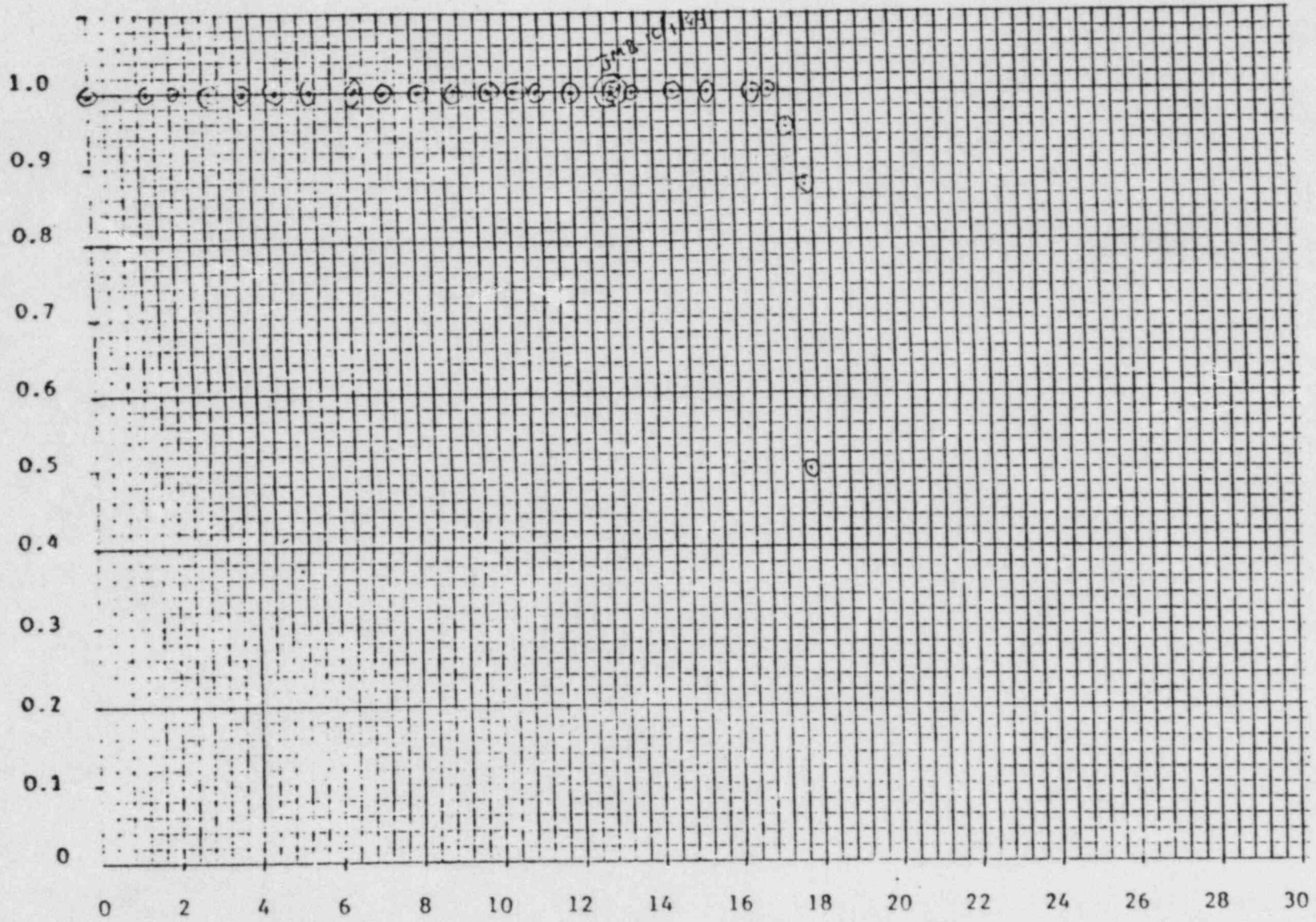



Figure 4.2.13

ICRR vs. REACTOR MAKEUP WATER ADDITION, N36

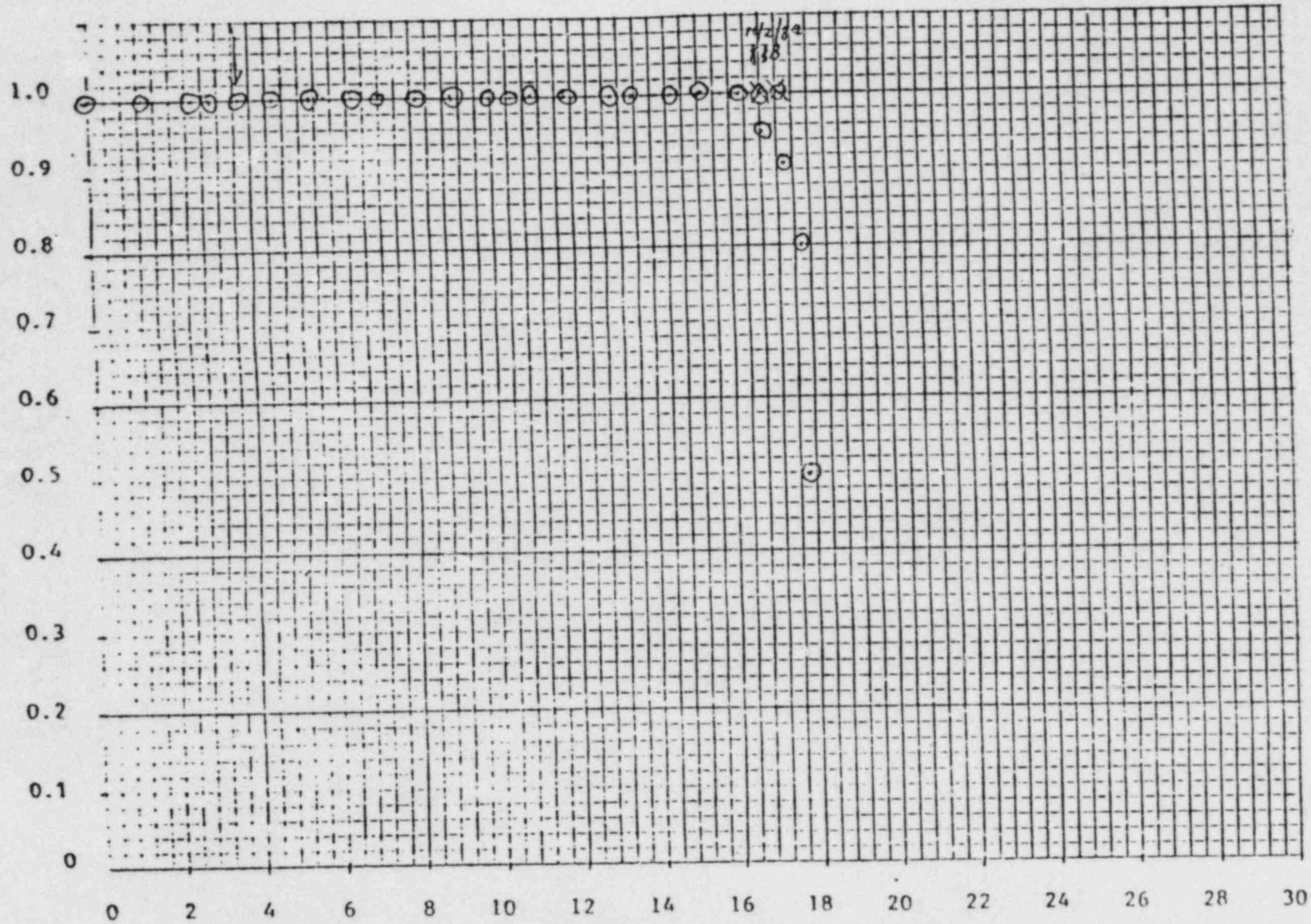



Figure 4.2.14

ICRR vs. RCS BORON CONCENTRATION, N31

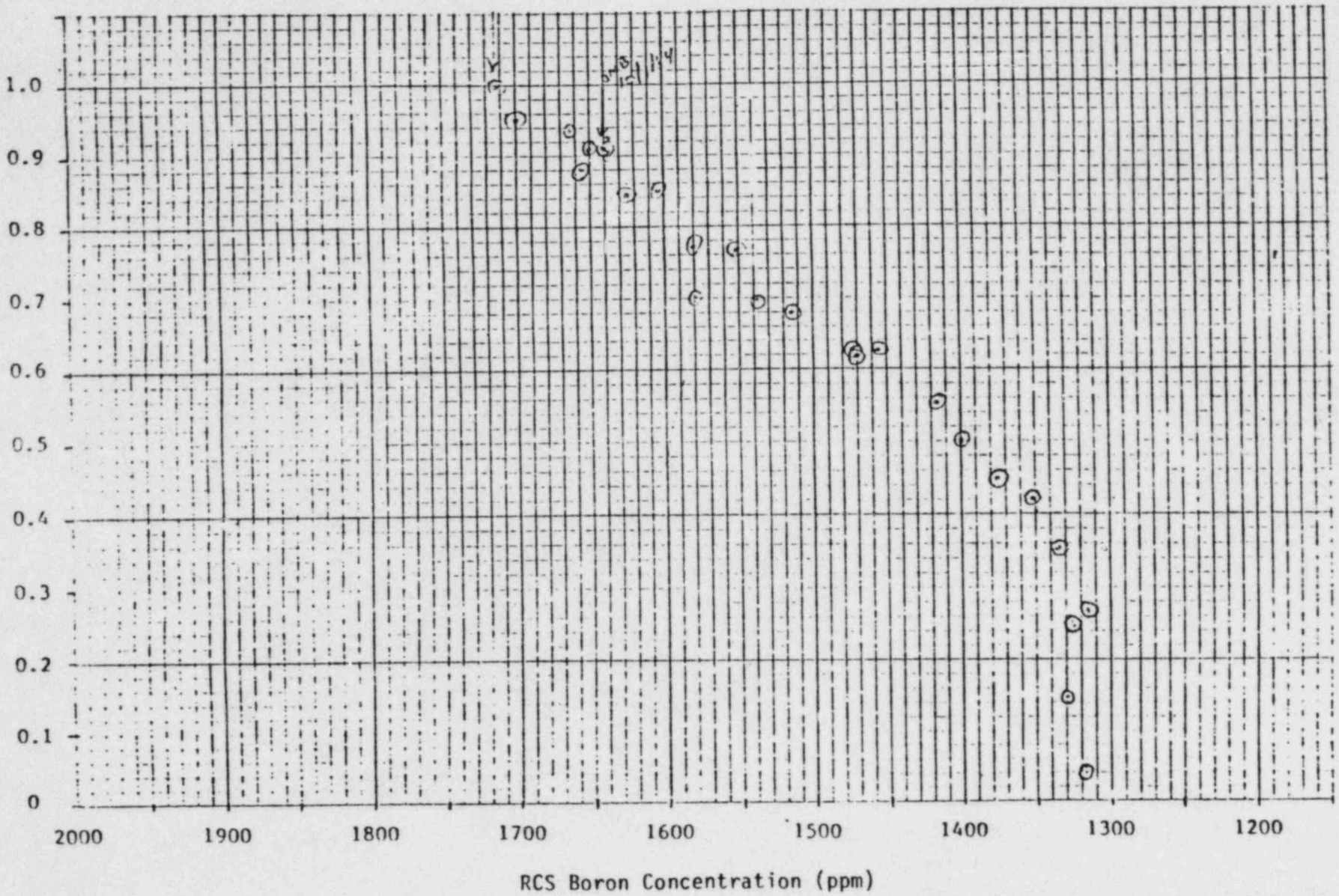



Figure 4.2.15

ICRR vs. RCS BORON CONCENTRATION, N32

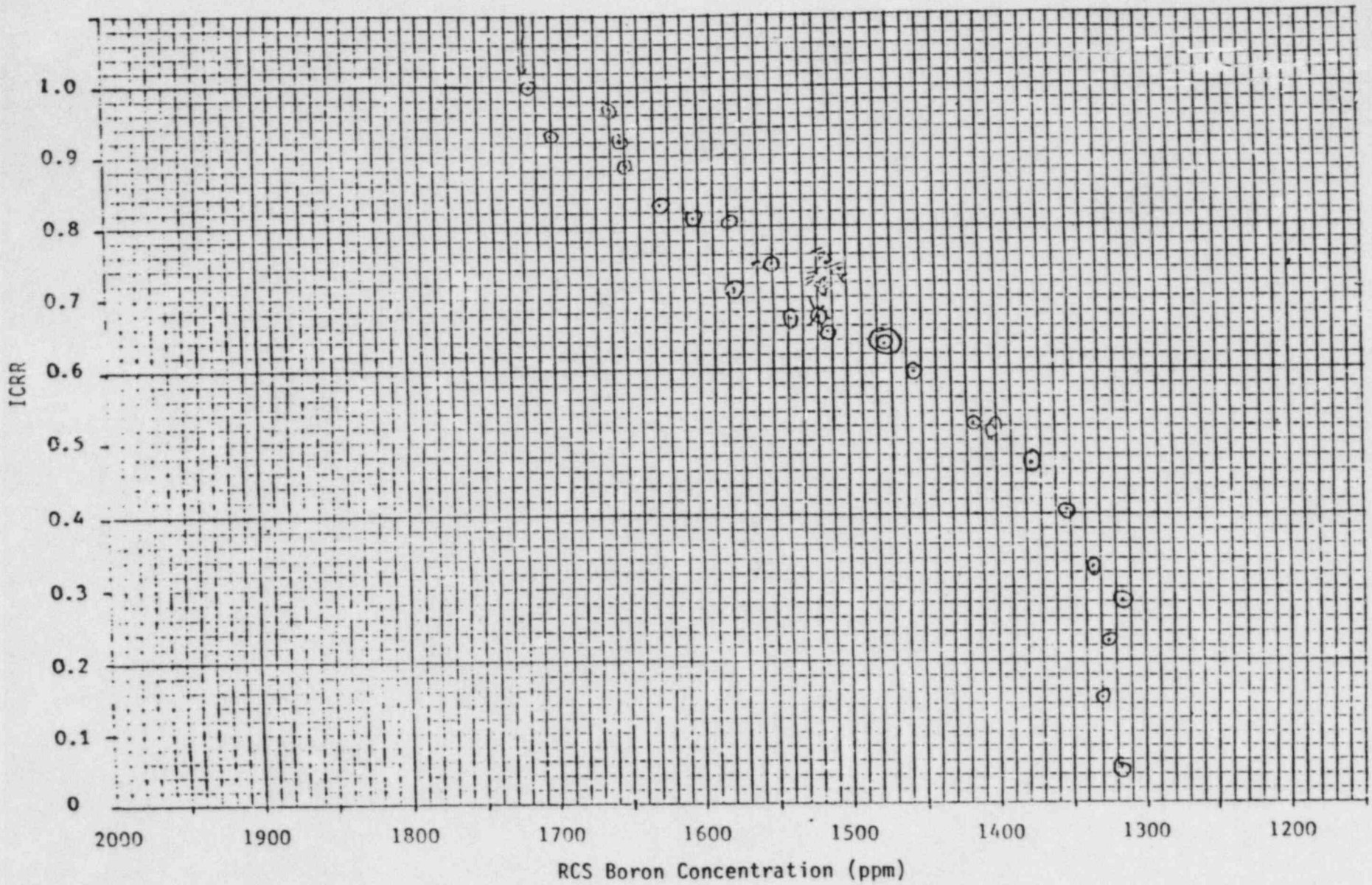



Figure 4.2.16

ICRR vs. RCS BORON CONCENTRATION, N35

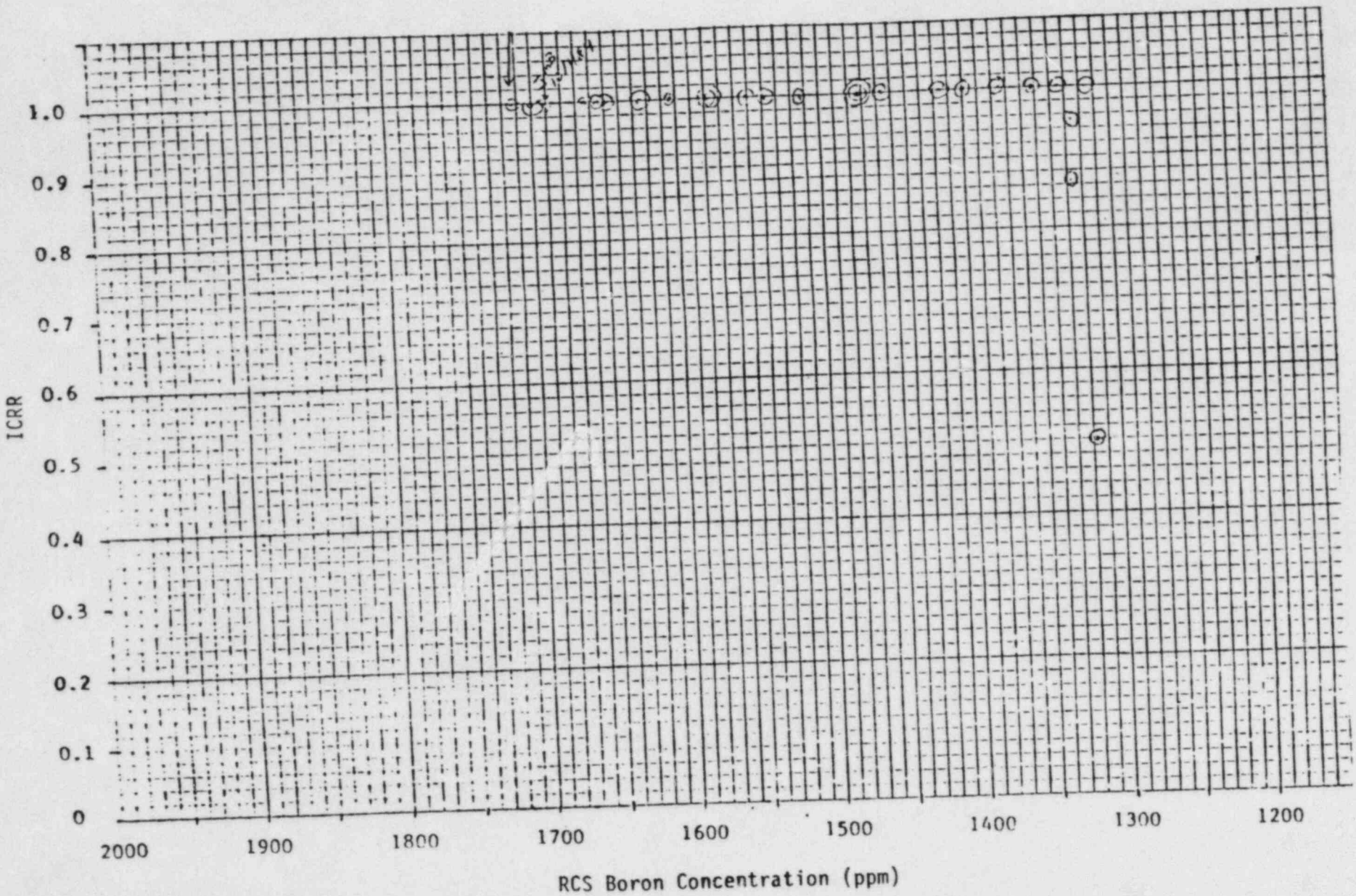



Figure 4.2.17

Figure 4.2.18


ICRR vs. RCS BORON CONCENTRATION, N36
ICRR
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
2000
1900
1800
1700
1600
1500
1400
1300
1200
RCS Boron Concentration (ppm)

## 4.3 DETERMINATION OF CORE POWER RANGE FOR PHYSICS TESTING (S-070005) Procedure ETT-ZZ-07050

### OBJECTIVES

The objectives of this procedure were to determine the reactor power level (neutron flux level) at which detectable reactivity feedback effects from fuel heating occur, and thereby establish the reactor power range in which zero power physics tests were to be performed.

### SUMMARY OF RESULTS

Detectable reactivity feedback effects from fuel heating were observed at a power level of 3.1X10E-7 amps which determined the physics testing range to be between 3.1X10E-9 and 3.1X10E-8 amps as measured by a reactivity computer picoammeter.

### DISCUSSION

With the reactor critical at a power level of approximately 1X10E-8 amps as measured on the Intermediate Range channels N35 and N36, 20 to 40 pcm of positive reactivity was added to the core by withdrawal of Control Bank D rods. The power level increased until reactivity feedback effects were observed as shown by an increase in $T_{ave}$ and a decrease in reactivity as monitored on strip chart recorders. The upper and lower limits for the physics testing range were determined by multiplying the power level value at which reactivity feedback effects were observed by factors of 0.1 and 0.01, respectively. No problems were encountered during this test.

4.4 BORON ENDPOINT DETERMINATION (S-070006)
Procedures ETT-ZZ-07061, ETT-ZZ-07062, ETT-ZZ-07063, ETT-ZZ-07064, ETT-ZZ-07065, ETT-ZZ-07066

OBJECTIVES

The objective of these test procedures was to determine the critical RCS boron concentration at various endpoint configurations (RCC configurations).

SUMMARY OF RESULTS

The critical RCS boron concentration was determined at various endpoint configurations, and the results shown in Table 4.4.1 meet the requirements within allowed tolerances as specified by the Nuclear Design Report, WCAP-10249, "The Nuclear Design and Core Physics Characteristics of the Callaway Unit 1 Nuclear Power Plant Cycle 1".

DISCUSSION

RCC banks were positioned to within 50 pcm of the desired endpoint configuration with the reactor critical in the zero power physics testing range. The controlling bank was then inserted/withdrawn to reach the desired endpoint configuration while neutron flux, reactivity, and RCS Tavg were monitored with strip chart recorders. When the neutron flux level had increased to greater than 30 percent of full scale and reactivity was constant, the controlling bank was re-positioned to re-establish the initial flux level. This process was then repeated twice for each RCC configuration. Boron samples which were taken at 15-20 minute intervals were within 6 ppm of the average and within 20 ppm of the pressurizer boron concentration. The critical RCS boron concentration at the endpoint configuration was then determined by the following equation:

$$C_{BE} = C_B - {}^{\delta C_B}/_{\delta \rho} \left( \bar{\bar{\Delta\rho}} + \alpha_{ISO} (Tavg - Tref) \right)$$

Where $C_{BE}$ = Critical RCS boron concentration at the endpoint configuration

$C_B$ = Measured average RCS critical boron concentration

${}^{\delta C_B}/_{\delta \rho}$ = Inverse boron worth from Nuclear Design data

$\bar{\bar{\Delta\rho}}$ = Measured average reactivity insertion caused by rod movement

$\alpha_{ISO}$ = Isothermal Temperature Coefficient from Nuclear Design data

$T_{avg}$ = Measured average RCS Tavg

$T_{ref}$ = 557° F

No problems were encountered during performance of these procedures.

TABLE 4.4.1

CRITICAL RCS BORON CONCENTRATIONS AT VARIOUS ENDPOINT CONFIGURATIONS

| ENDPOINT CONFIGURATION | BORON CONCENTRATION - ppm | |
|---|---|---|
| | ACCEPTANCE CRITERIA | RESULTS |
| All Rods Out (ETT-ZZ-07061) | 1299 (±50 ppm) | 1334 |
| Control Bank D at 0 (ETT-ZZ-07062) | 1236 (±10%) | 1276 |
| Control Banks D and C at 0 (ETT-ZZ-07063) | 1113 (±10%) | 1148 |
| Control Banks D, C and B at 0 (ETT-ZZ-07064) | 1020 (±10%) | 1042 |
| Control Banks D, C, B and A at 0 (ETT-ZZ-07065) | 955 (±10%) | 979 |
| All Rods In and Highest Worth Rod Stuck Out (ETT-ZZ-07066) | 674 (±10%) | 728 |

## 4.5 ISOTHERMAL TEMPERATURE COEFFICIENT MEASUREMENT (S-070007)
Procedures ETT-ZZ-07071, ETT-ZZ-07072 and ETT-ZZ-07073

### OBJECTIVES

These tests measured the isothermal temperature coefficient ($\alpha_{ISO}$) of reactivity and calculated the moderator temperature coefficient ($\alpha_M$) for rod configurations of All Rods Out (ARO), Control Bank D in (CBD = 0) and Control Banks C and D in (CBC and CBD = 0).

### SUMMARY OF RESULTS

Acceptance Criteria and results are contained in Table 4.5.1.

An additional design review criteria is that the measured values of the isothermal temperature coefficient obtained during heat-up agree within ±1 PCM/°F with those obtained during cooldown.

All design review criteria were met; however, the Safety Review Criteria for the ARO rod configuration was not met.

### DISCUSSION

These tests were performed by heating and cooling the Reactor Coolant System (RCS) by operating steam dump valves and measuring the change in reactivity caused by the change in temperature. The ratio of these changes is the isothermal temperature coefficient. The moderator temperature coefficient is determined by subtracting -1.95 PCM/°F (Doppler Coefficient) from the isothermal temperature coefficient.

The only problem encountered was the calculated moderator temperature coefficient was more positive than 0.0 PCM/°F. Since the moderator temperature coefficient was more positive than 0.0 PCM/°F the action statement of Tech Spec (T.S.) 3.1.1.3A was entered which requires that rod withdrawal limits be established (Figure 4.5.1). Start-Up Physics testing continued in accordance with T.S. 3.10.3 which allows operation below 5% power with a positive moderator temperature coefficient. A special report was filed with the NRC within 10 days as required by T.S. 3.1.1.3, Action A.3 via ULNRC-947 dated October 11, 1984.

TABLE 4.5.1

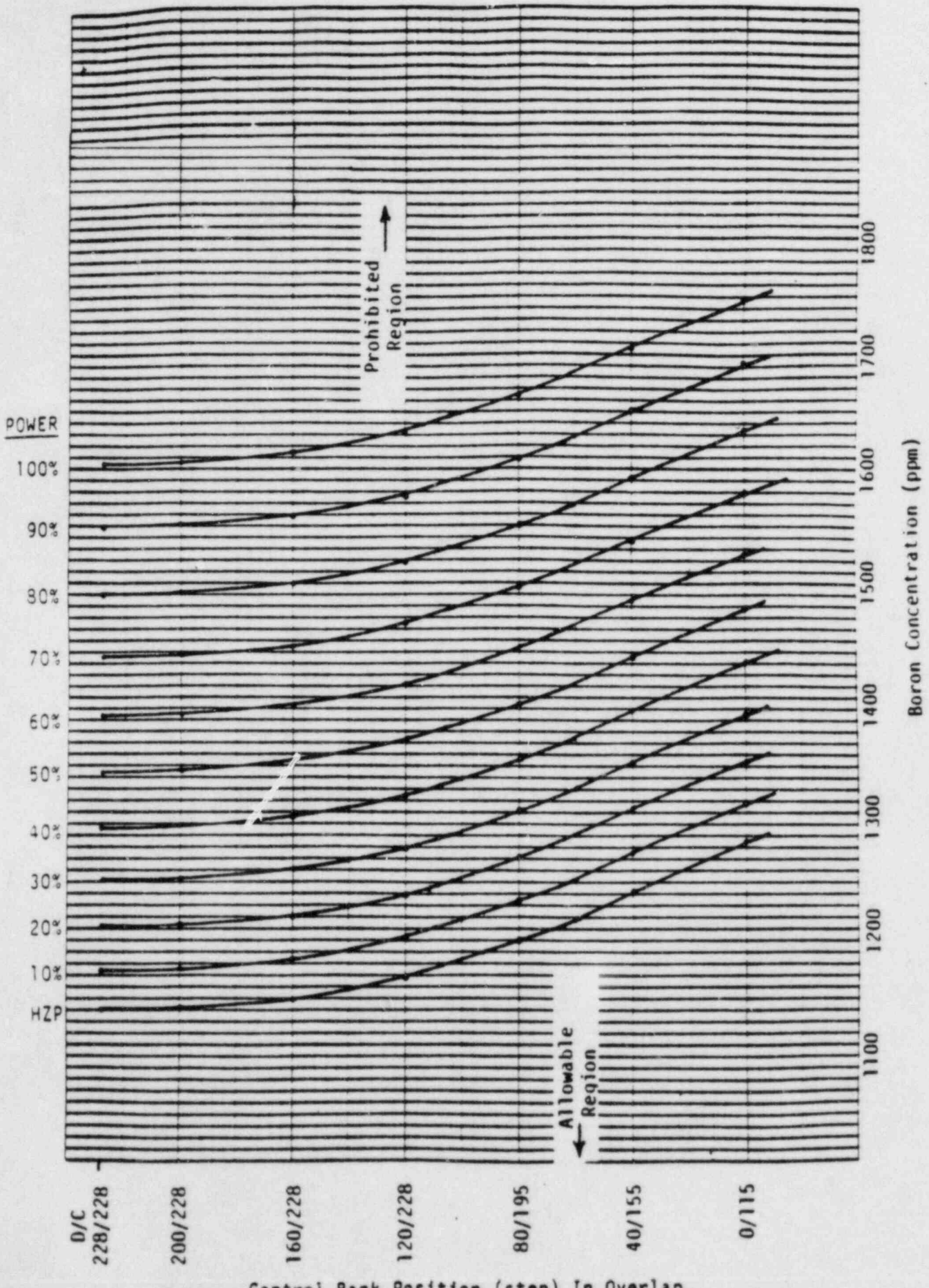
ISOTHERMAL TEMPERATURE COEFFICIENT
ACCEPTANCE CRITERIA AND RESULTS

| ROD CONFIGURATION | ACCEPTANCE CRITERIA | | | MEASURED VALUES | | |
|---|---|---|---|---|---|---|
| | SAFETY REVIEW | DESIGN REVIEW | | MEASURED | CALCULATED | MEASURED |
| | $\alpha_M$ | $\alpha_{ISO}$ | $\alpha_M$ | $\alpha_{ISO}$ | $\alpha M$ | $C_B$ (PPM) |
| ARO | $\alpha_M$ Less Positive Than 0.0 PCM/°F | -2.03±3 | 0.7±3 | -0.661 | 1.289 | 1330 |
| CBD = 0 | None | -3.36±3 | -0.95±3 | -2.20 | -0.25 | 1273 |
| CBC, CBD = 0 | None | -6.66±3 | -4.2±3 | -5.58 | -3.63 | 1148 |

ALL TEMPERATURE COEFFICIENTS ($\alpha_{ISO}$ AND $\alpha_M$) ARE IN PCM/°F.

FIGURE 4.5.1

CONTROL ROD WITHDRAWAL LIMITS TO MAINTAIN MTC < 0
(Applicable up to 6000 MWD/MTU Burnup)



4.6 RCCA OR BANK WORTH MEASUREMENT AT ZERO POWER (S-07SF08)
ETT-SF-07081 thru - 07087

OBJECTIVES

The objectives of the subject procedures were to:

1. Measure the differential and integral reactivity worths of Control Banks D, C, B, and A.

2. Measure the differential and integral reactivity worths of Shutdown Banks E, D, and C.

3. Measure the integral reactivity worth of all Control and Shutdown Banks less the predicted highest worth Rod Control Cluster Assembly (RCCA).

4. Measure the differential and integral reactivity worths of the Control Banks in normal overlap.

5. Measure the differential boron reactivity worth over the range of Control Bank travel in normal overlap.

6. Measure the differential and integral reactivity worths of the predicted worst case ejected rod at hot zero power.

7. Take a full core flux map at hot zero power.

8. Take a full core flux map at hot zero power with the predicted worst case ejected rod fully withdrawn.

SUMMARY OF RESULTS

All Acceptance Criteria for the subject procedures were met. These criteria and discussions of associated results follows:

1. The measured integral reactivity worth of Control Bank D over its entire range of travel is equal to 650 pcm (585-715).

   The measured integral reactivity worth of Control Bank D was 663 pcm.

2. The measured integral reactivity worth of Control Bank C over its entire range of travel is equal to 1240 pcm (1116-1364).

   The measured integral reactivity worth of Control Bank C was 1177 pcm.

3. The measured integral reactivity worth of Control Bank B over its entire range of travel is equal to 970 pcm (873-1067).

   The measured integral reactivity worth of Control Bank B was 1010 pcm.

4. The measured integral reactivity worth of Control Bank A over its entire range of travel is equal to 680 pcm (612-748).

   The measured integral reactivity worth of Control Bank A was 685 pcm.

5. The measured integral reactivity worth of Shutdown Bank E over its entire range of travel is equal to 870 pcm (783-957).

   The measured integral reactivity worth of Shutdown Bank E was 882 pcm.

6. The measured integral reactivity worth of Shutdown Bank D over its entire range of travel is equal to 740 pcm (666-814).

   The measured integral reactivity worth of Shutdown Bank D was 738 pcm.

7. The measured integral reactivity worth of Shutdown Bank C over its entire range of travel is equal to 960 pcm (864-1056.)

   The measured integral reactivity worth of Shutdown Bank C was 978 pcm.

8. The sum of the measured integral reactivity worths of all Control and Shutdown Banks less the predicted highest worth stuck rod is equal to 6012 pcm (6012-7348).

   The subject measurement was equal to 6416 pcm.

9. The sum of the integral reactivity worths of the Control Banks measured individually agrees with the measured integral reactivity worth of the Control Banks in normal overlap to within ± 4% of the integral reactivity worth in overlap.

   The measured integral reactivity worth of the Control Banks in overlap was 3548 pcm. The calculated difference was -0.4%.

10. The measured differential boron reactivity worth over the range of Control Bank travel in normal overlap is equal to -10.27 pcm/ppm (-11.30 to -9.24).

    The measured differential boron reactivity worth was -10.33 pcm/ppm

11. The measured integral reactivity worth of pseudo ejected rod D-12 (predicted worst case rod), increased by 10%, is less than or equal to 460 pcm.

    The measured integral reactivity worth of pseudo ejected rod D-12, increased by 10%, was 414 pcm.

DISCUSSION

Individual rod bank reactivity worth measurements were initiated with all rods out (ARO). A constant Reactor Coolant System (RCS) dilution rate was established and Control Bank D was periodically inserted approximately 10 to 20 pcm to maintain a just critical core condition. The reactivity computer was used to process the signal from one of the four power range channels and output a continuous reactivity trace, along with the associated neutron flux, on a two-pen strip chart recorder. Dilution was halted with Control Bank D fully inserted to allow for additional testing at this rod configuration. The differential and integral reactivity worths of Control Bank D were calculated using the reactivity trace and the ARO boron endpoint test result. This basic procedure was repeated for Control Bank C, Control Bank B, Control Bank A, Shutdown Bank E, Shutdown Bank D and Shutdown Bank C, in that order. Figures 4.6.1 thru 4.6.8 present the integral and differential worths of the Control Banks vs. bank position. Figures 4.6.9 thru 4.6.11 present the integral worths of the subject shutdown banks vs. bank position.

Next, the predicted most reactive stuck rod (F-10) was interchanged with Shutdown Bank A and then a portion of Shutdown Bank B, until rod F-10 was fully withdrawn. A constant RCS dilution rate was reinitiated and Shutdown Bank B periodically inserted to 25 steps while obtaining a reactivity trace. A boron endpoint was then performed by driving Shutdown Bank B to the fully inserted position. By summing the integral reactivity worth measurements of all rod banks, excluding the swap of rod F-10 with Shutdown Bank A and the upper portion of Shutdown Bank B, the measured integral reactivity worth of all Control and Shutdown Banks less the predicted highest worth stuck rod was obtained. Figure 4.6.12 presents the integral worth of Shutdown Bank B vs. bank position, excluding that portion of the bank that was swapped with rod F-10.

A rod configuration was achieved with all Shutdown Banks withdrawn and all Control Banks inserted. A constant RCS boration rate was established and the Control Banks periodically withdrawn in overlap. A continuous reactivity trace was recorded during bank withdrawal and used to calculate the integral and differential worths of the Control Banks in overlap. Figure 4.6.13 presents the integral and differential worth of the Control Banks vs. position in overlap. The average differential boron reactivity worth over Control Bank travel was obtained by calculating the least squares fit slope of a plot of critical boron concentration vs. integral reactivity worth.

Next, a rod configuration was achieved with the Shutdown Banks fully withdrawn and the Control Banks (in overlap) near the zero power insertion limit. A full core flux map was taken at this configuration. A constant RCS boration rate was established and the predicted worst case ejected rod (D-12) periodically withdrawn to 196 steps while recording the associated reactivity trace. The remaining worth of rod D-12 was measured by performing a boron endpoint, driving rod D-12 to the fully withdrawn position. RCS boron concentration was adjusted to maintain criticality with rod D-12 fully withdrawn and a full core flux map taken. Rod D-12 was inserted to its original position and the

integral and differential reactivity worths calculated using the reactivity trace and boron endpoint data. Figure 4.6.14 presents the integral worth of rod D-12 vs. position.

Flux map results are presented in Section 4.7 of the Startup Report.

Figure 4.6.1
Control Bank D Integral
Reactivity Worth vs. Position

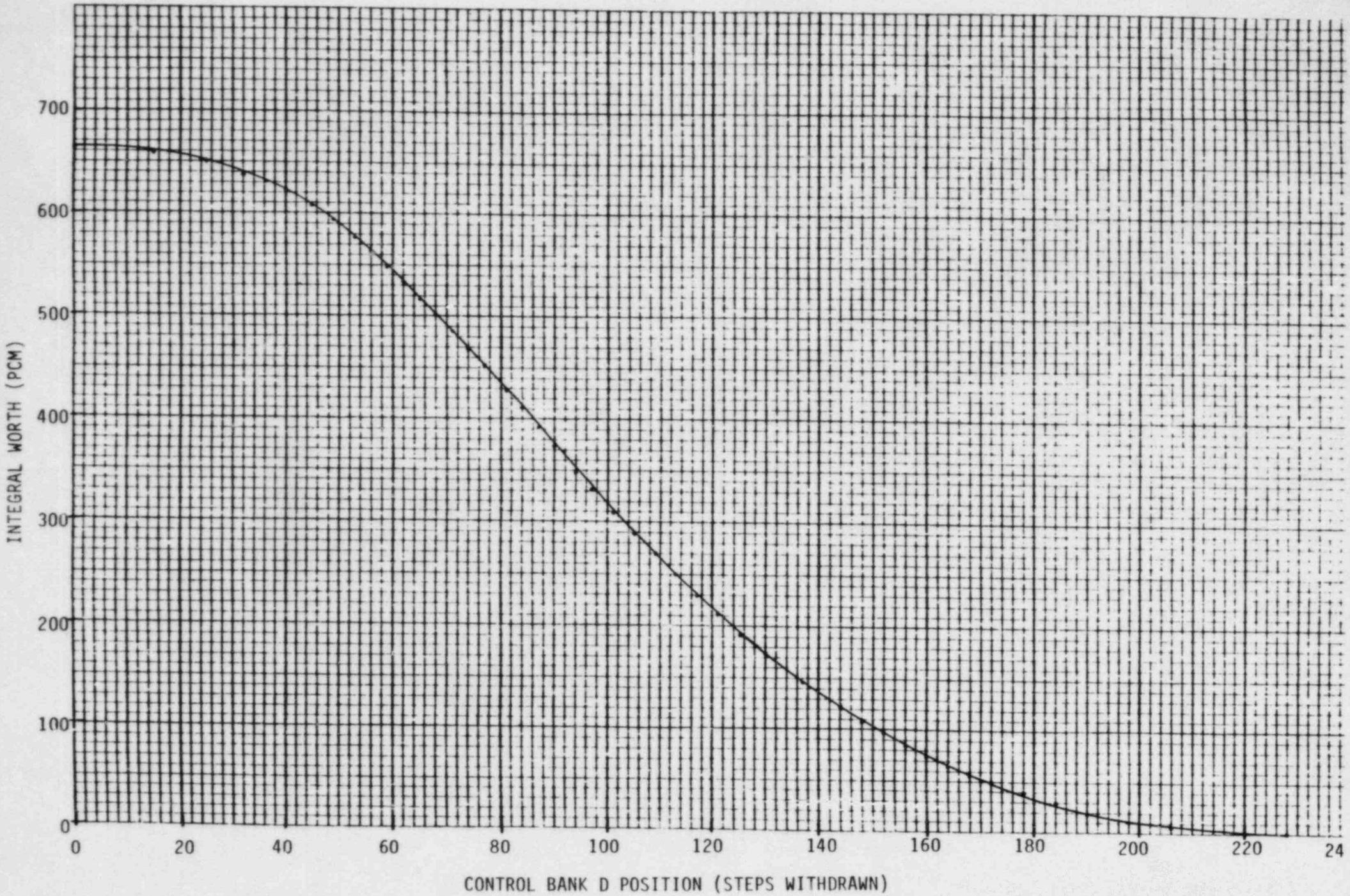


Figure 4.6.2

Control Bank D Differential
Reactivity Worth vs. Position

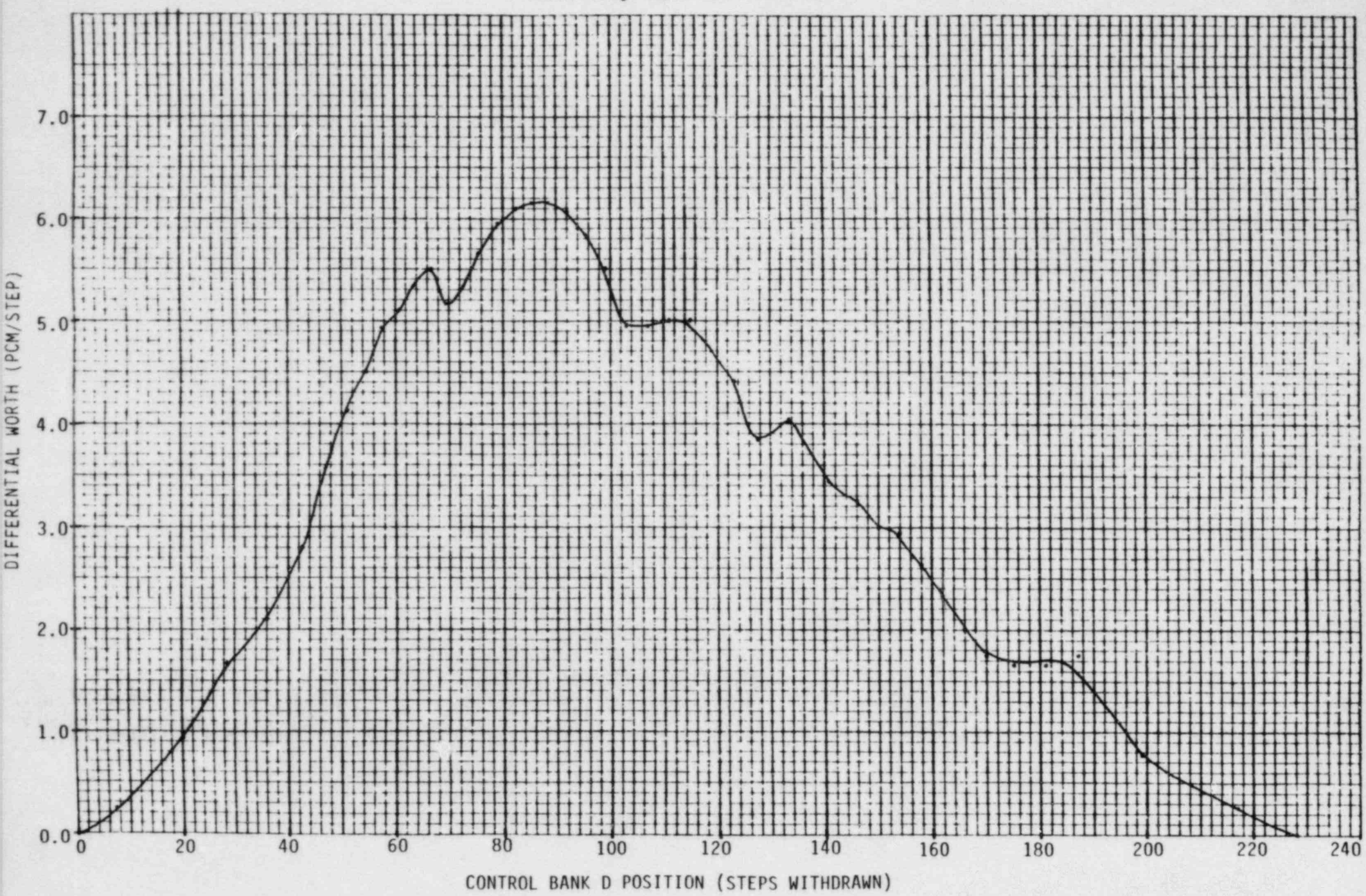


Figure 4.6.3

Control Bank C Integral
Reactivity Worth vs. Position

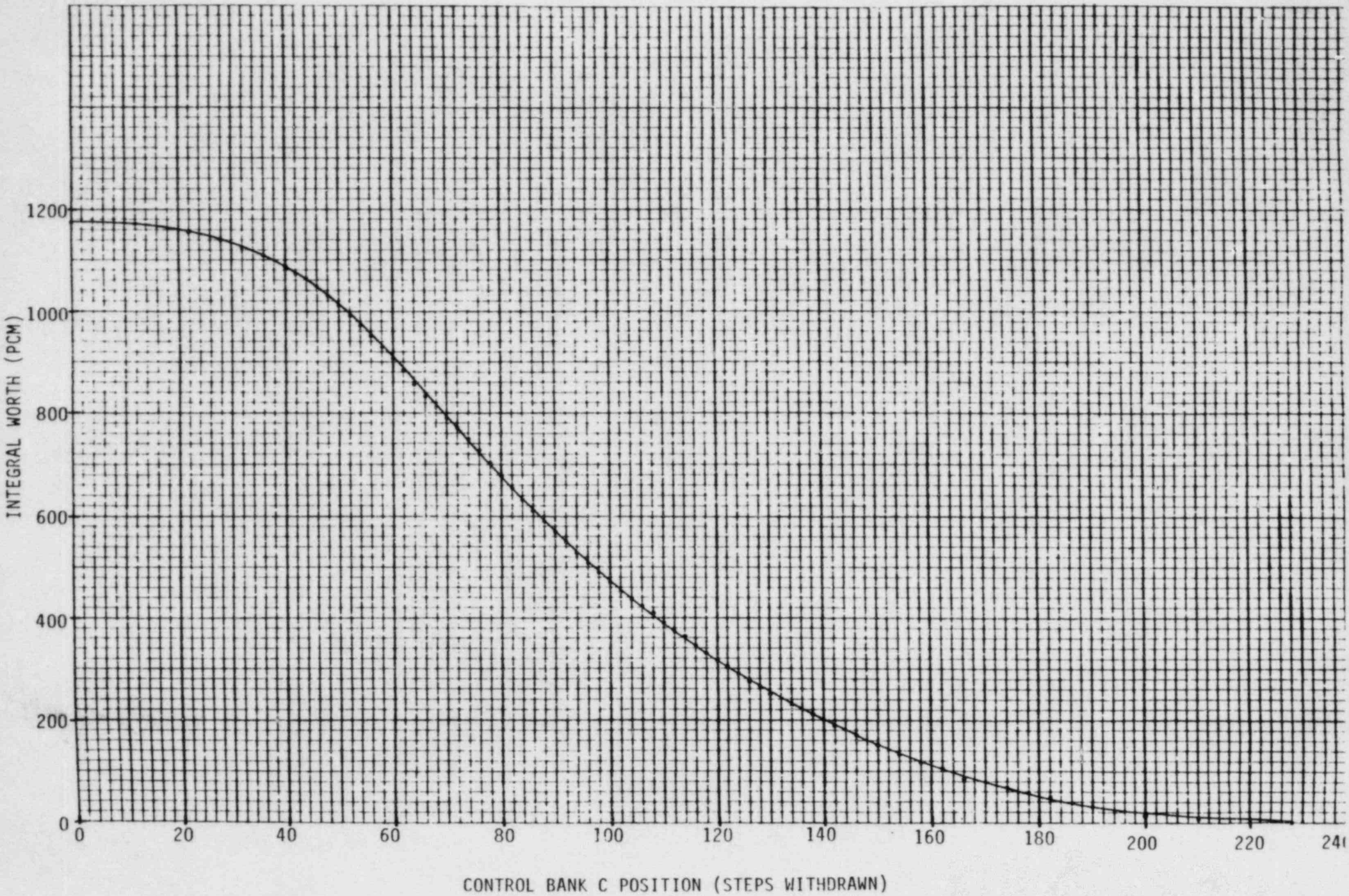


Figure 4.6.4

Control Bank C Differential
Reactivity Worth vs. Position

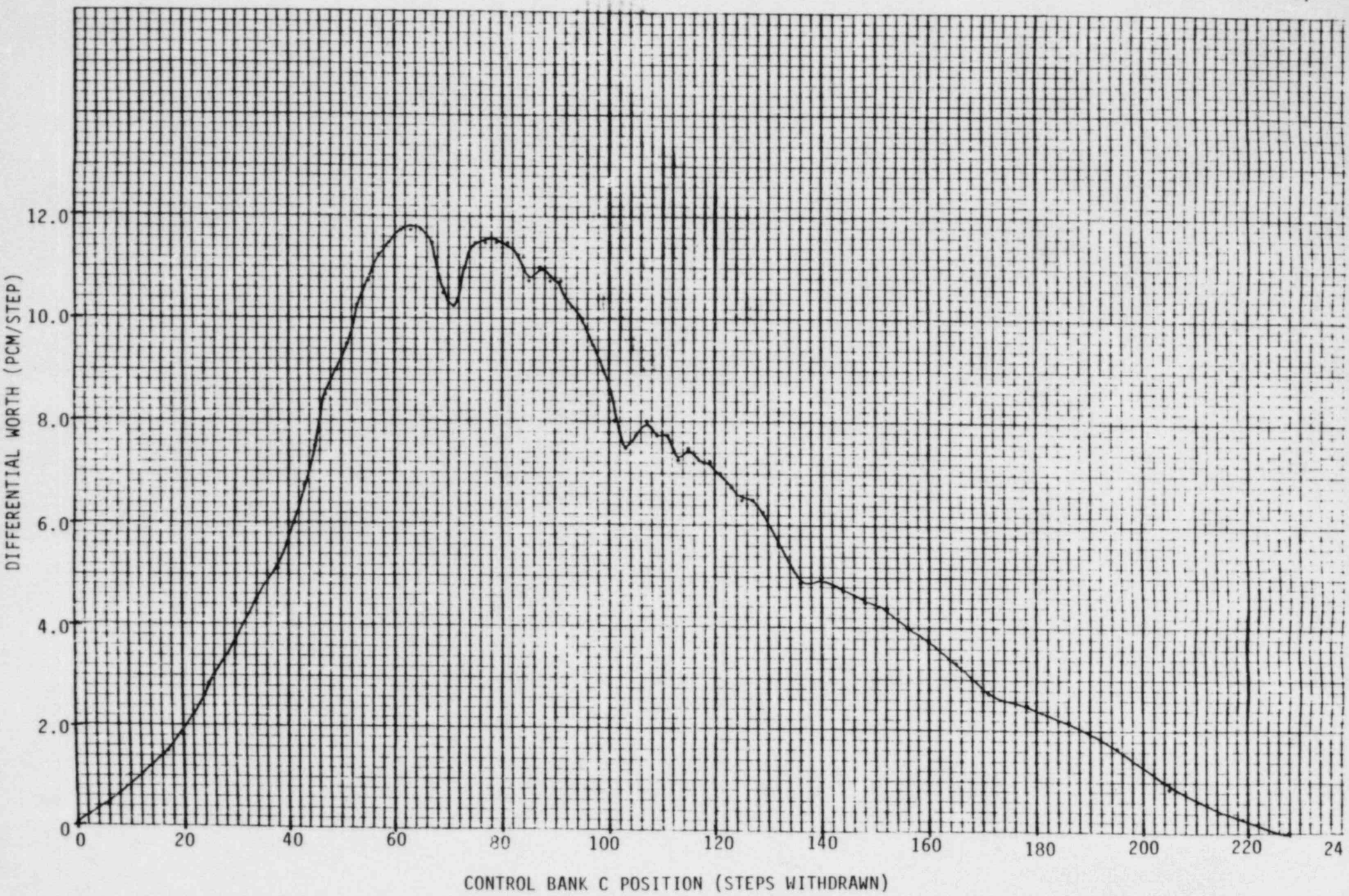


Figure 4.6.5

Control Bank B Integral
Reactivity Worth vs. Position

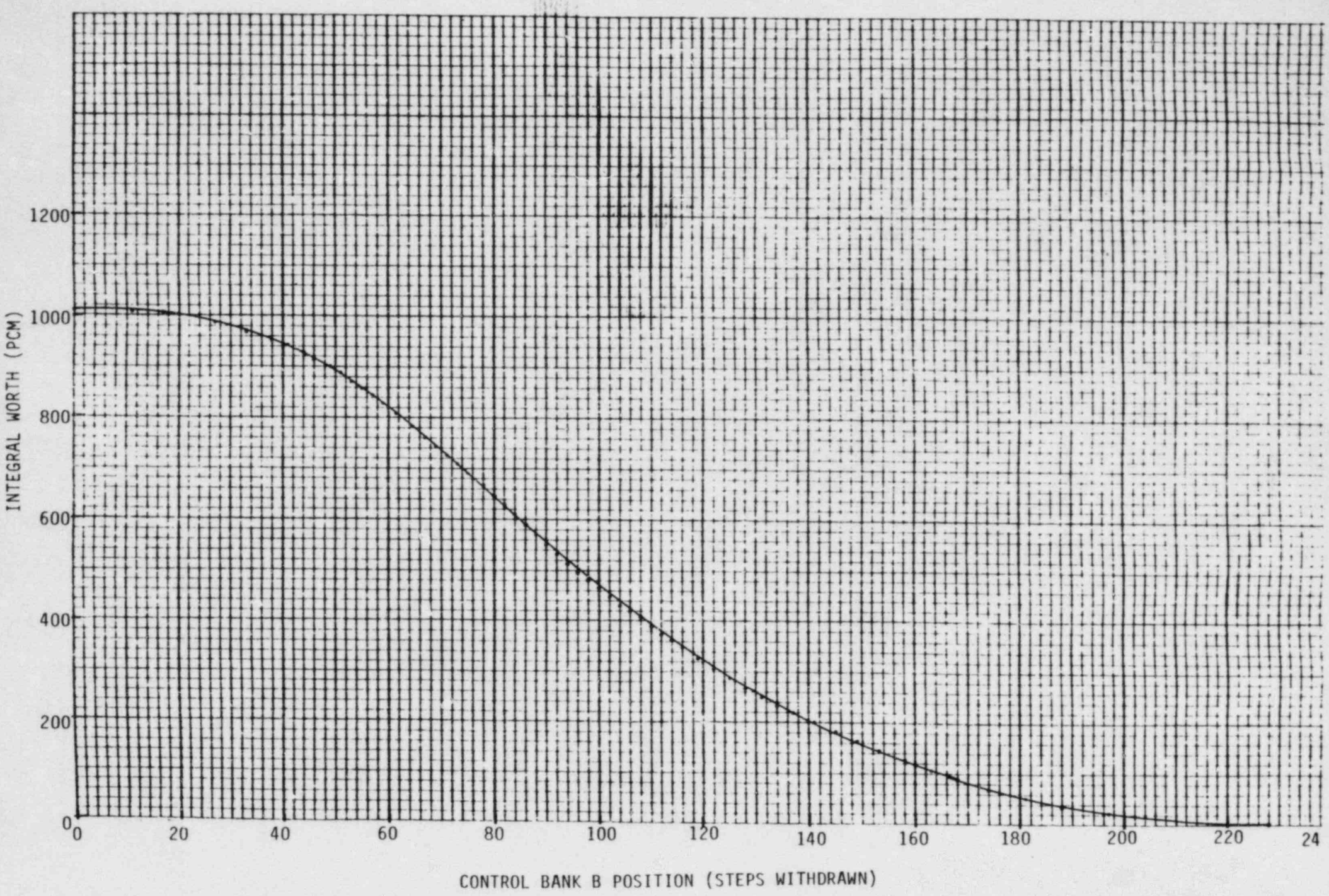


Figure 4.6.6

Control Bank B Differential
Reactivity Worth vs. Position

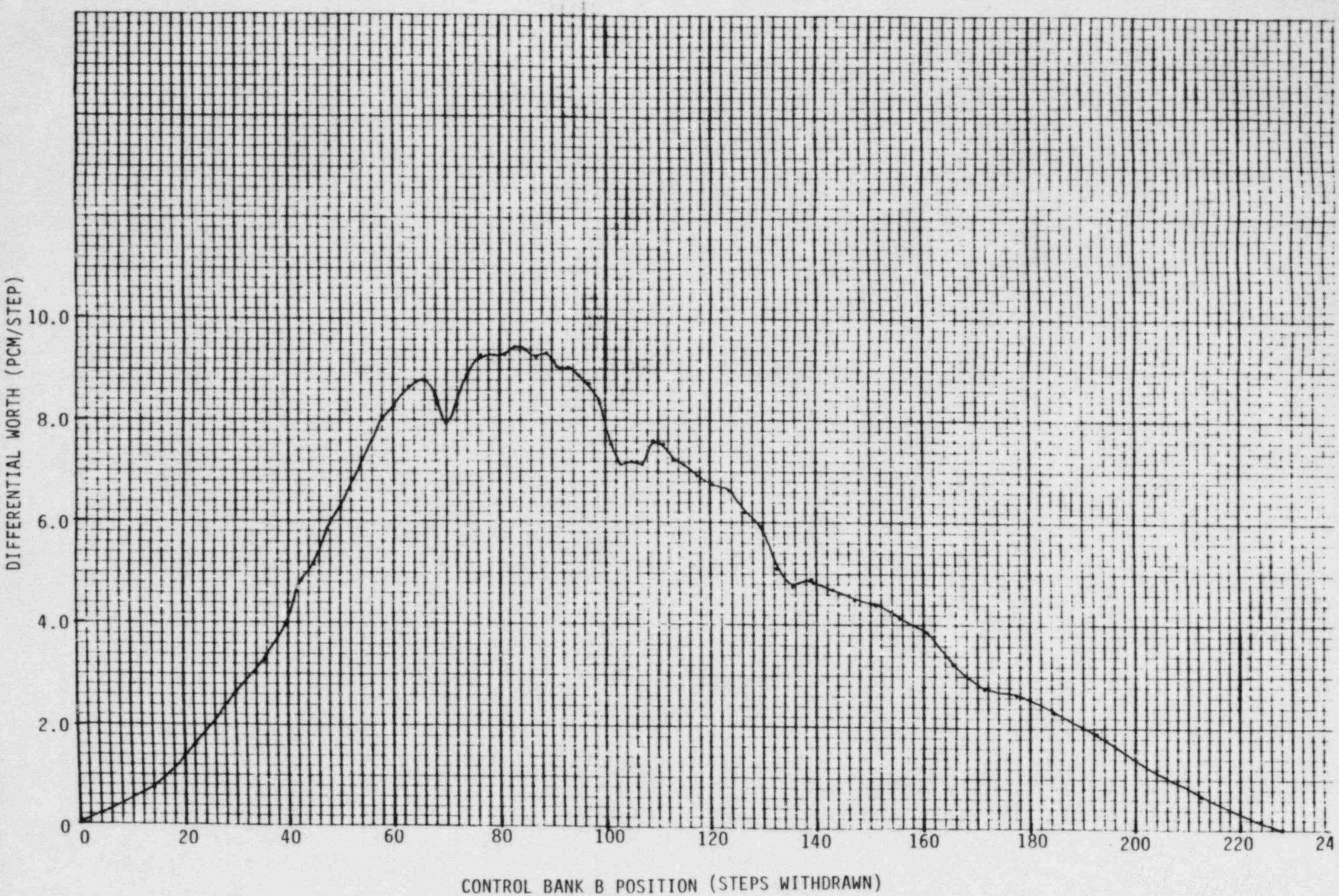


Figure 4.6.7

Control Bank A Integral
Reactivity Worth vs. Position

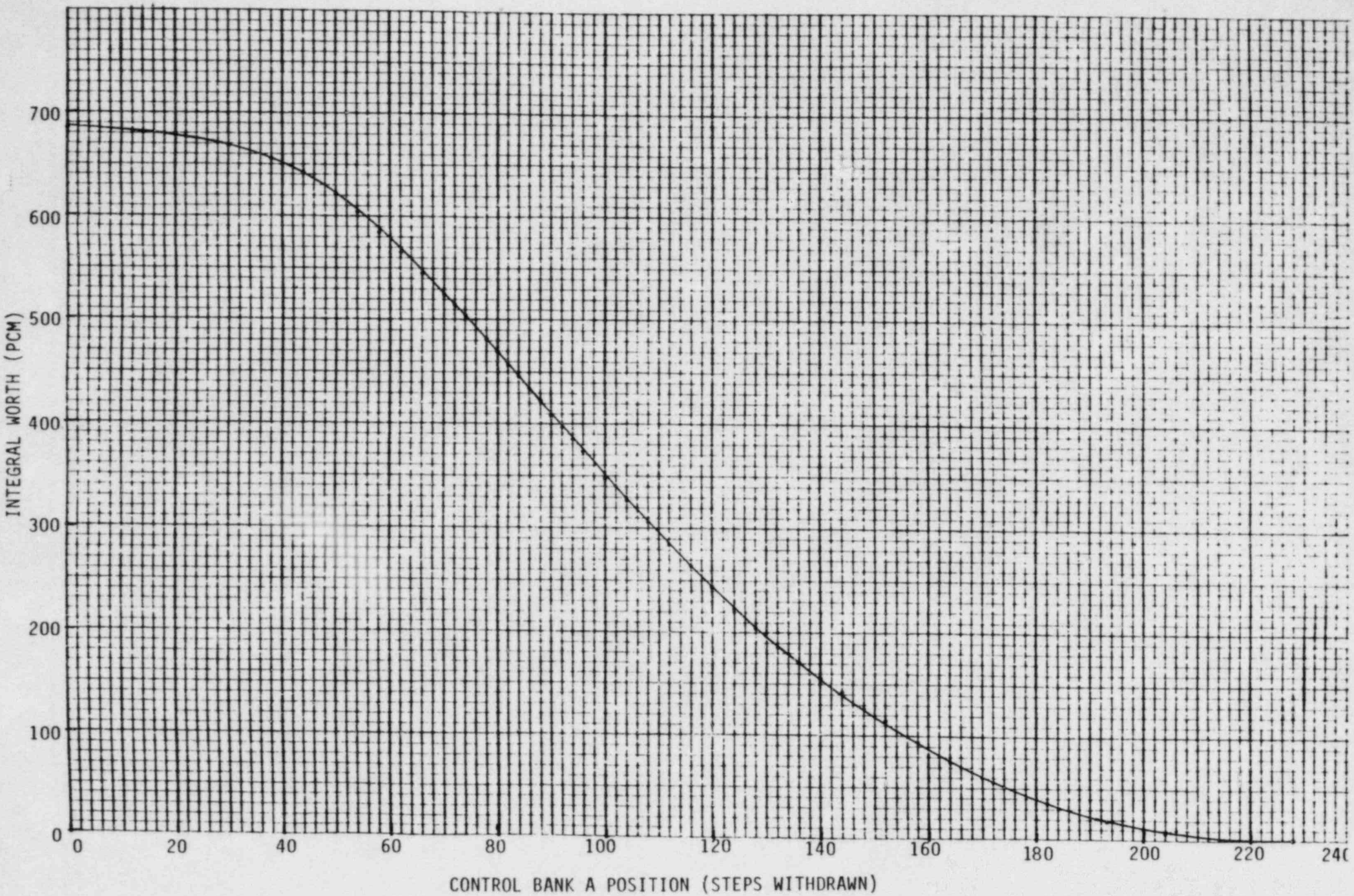


Figure 4.6.8

Control Bank A Differential
Reactivity Worth vs. Position

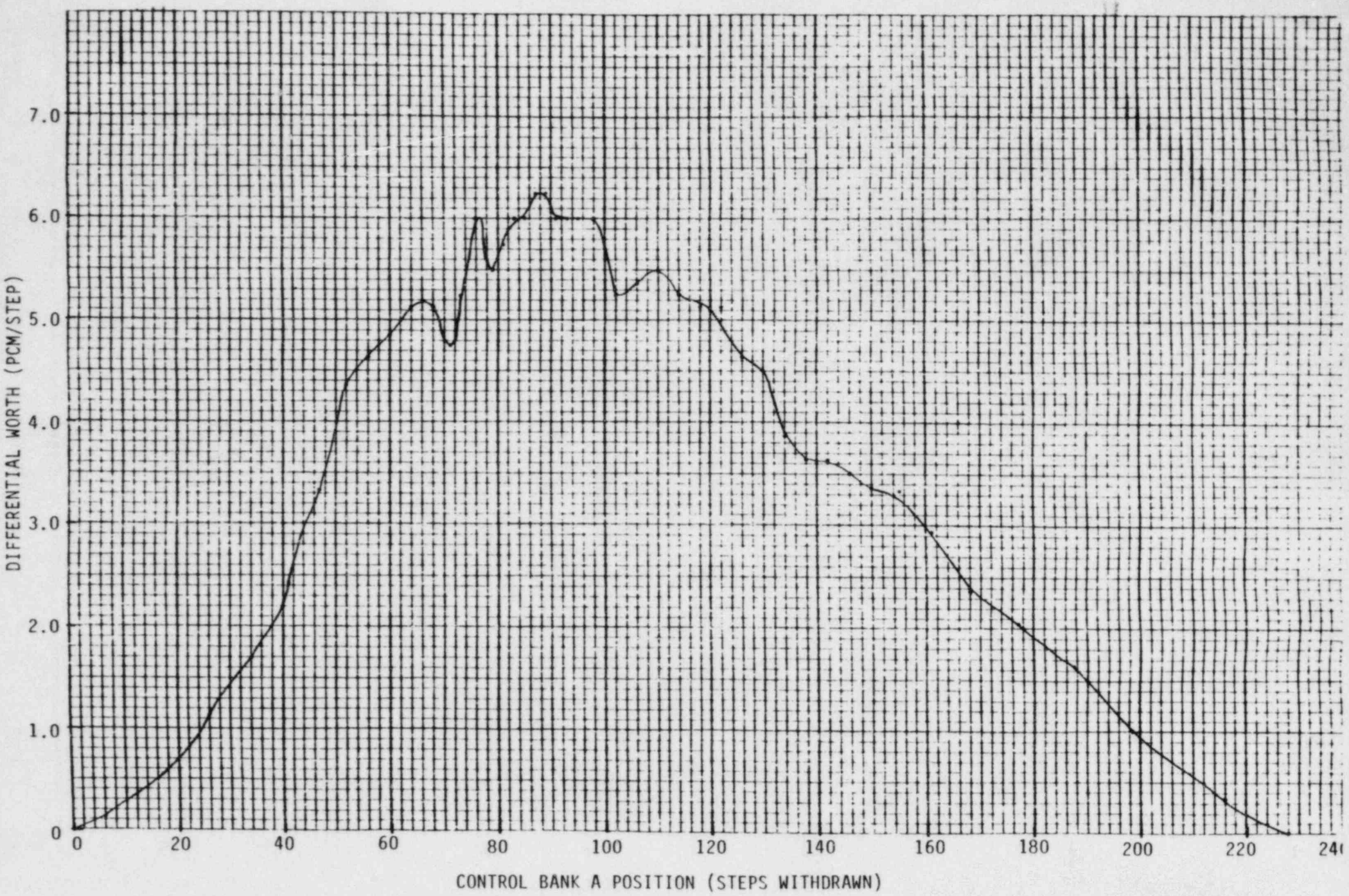


Figure 4.6.9

Shutdown Bank E Integral
Reactivity Worth vs. Position

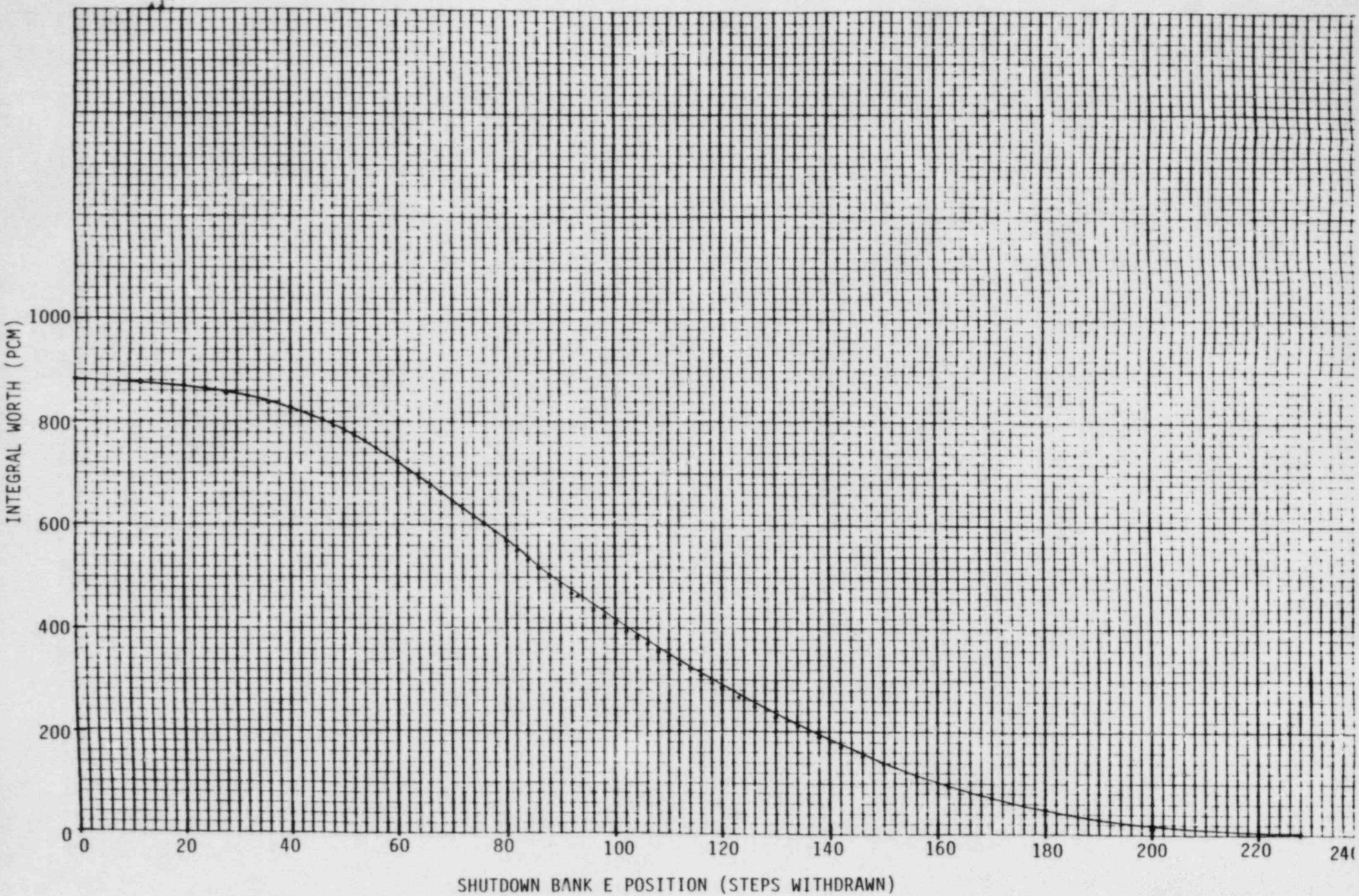


Figure 4.6.10

Shutdown Bank D Integral
Reactivity Worth vs. Position

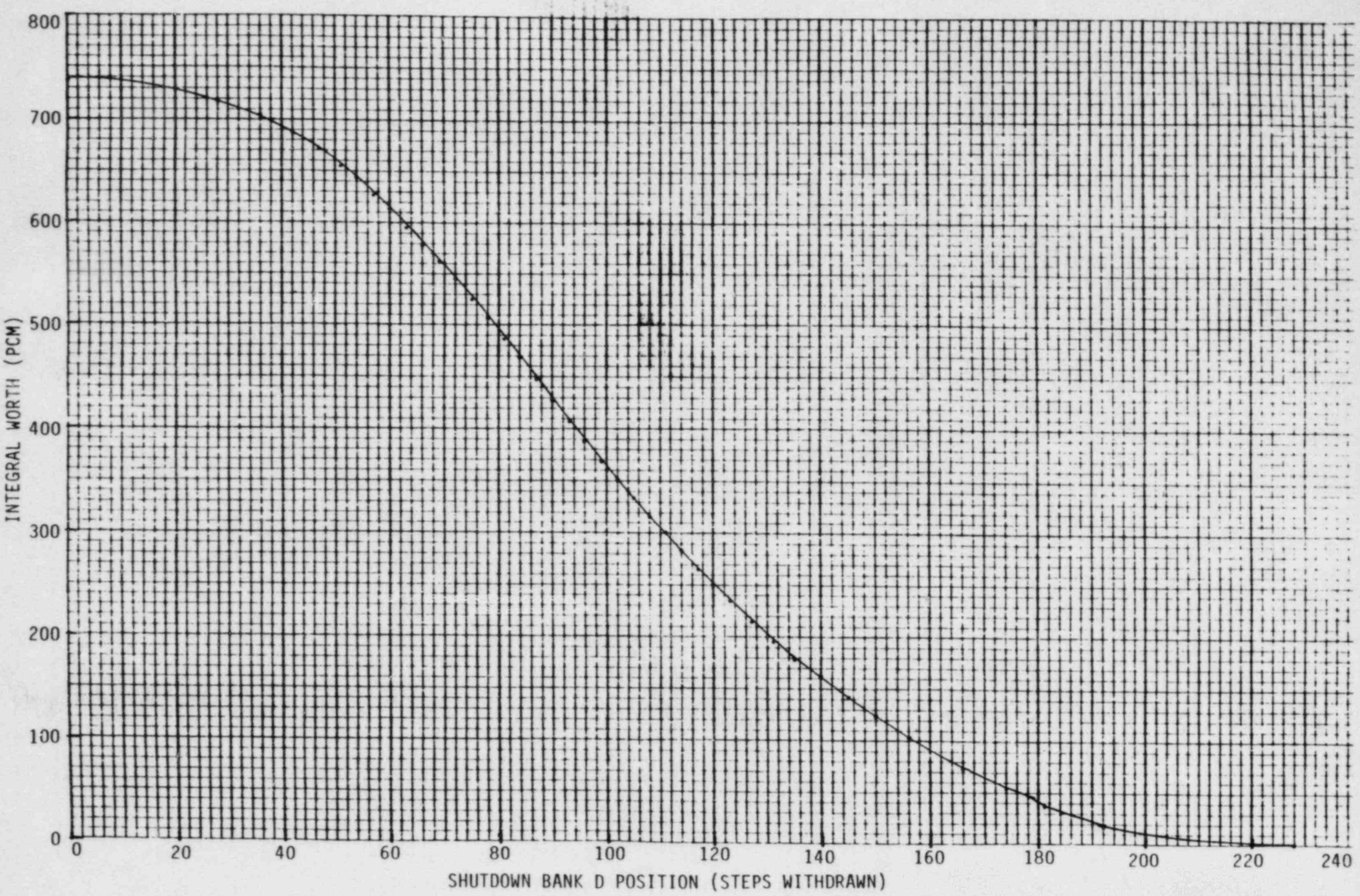


Figure 4.6.11

Shutdown Bank C Integral
Reactivity Worth vs. Position

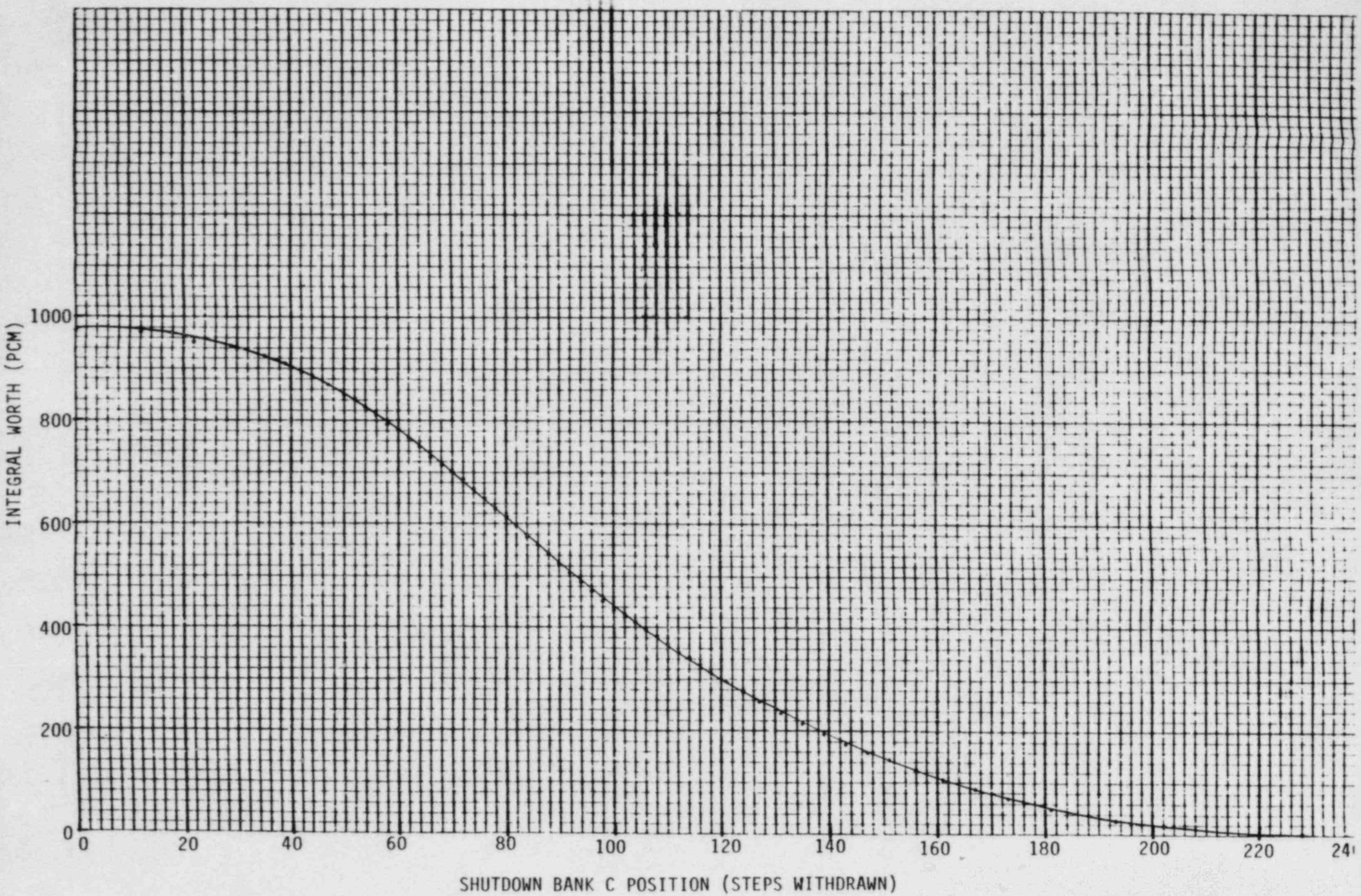


Figure 4.6.12

Shutdown Bank B Integral Reactivity Worth (Excluding Upper Portion Exchanged with Rod F-10) vs. Position

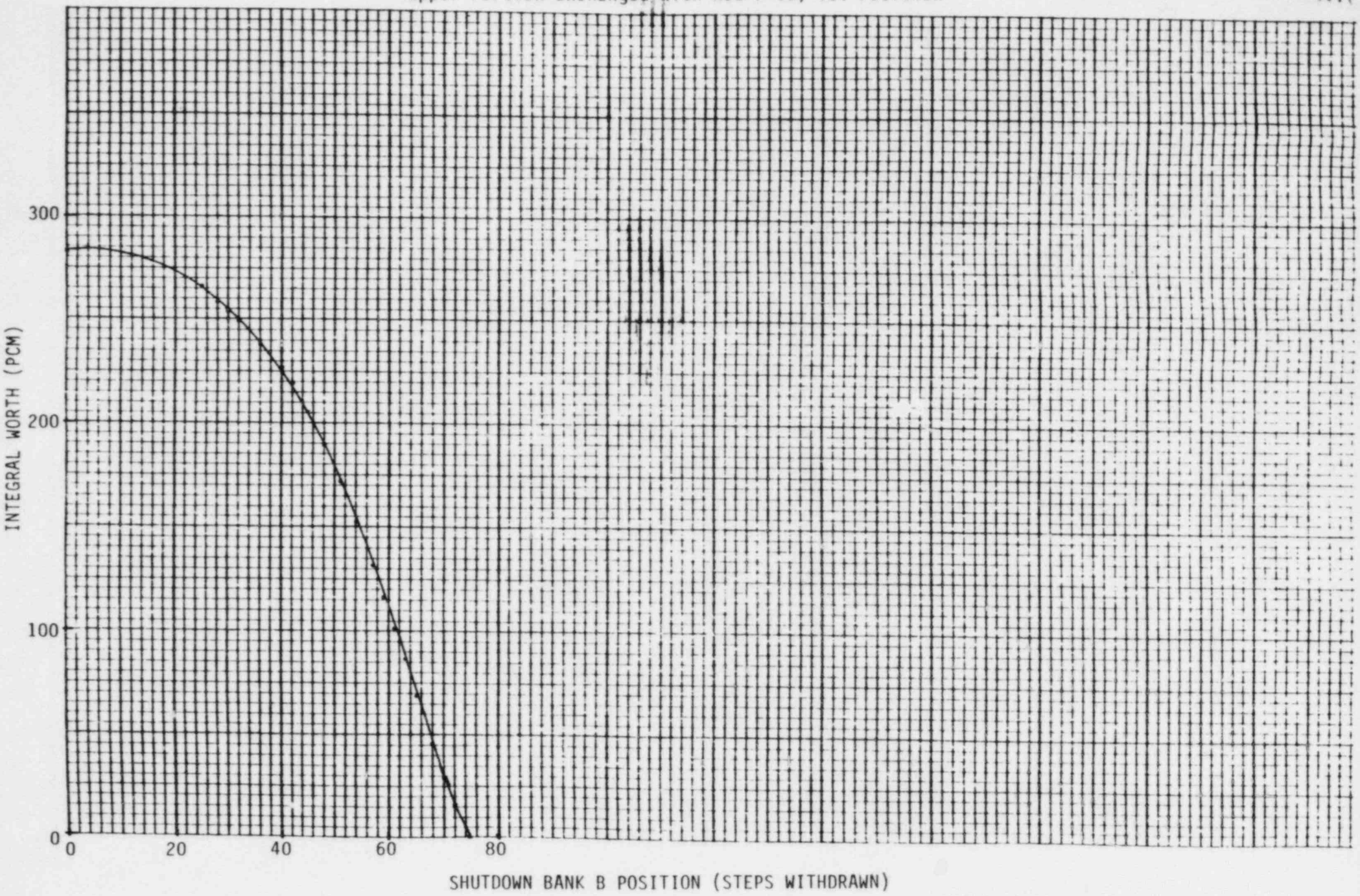


Figure 4.6.13

Integral and Differential Reactivity Worths
of Control Banks In Overlap vs. Position

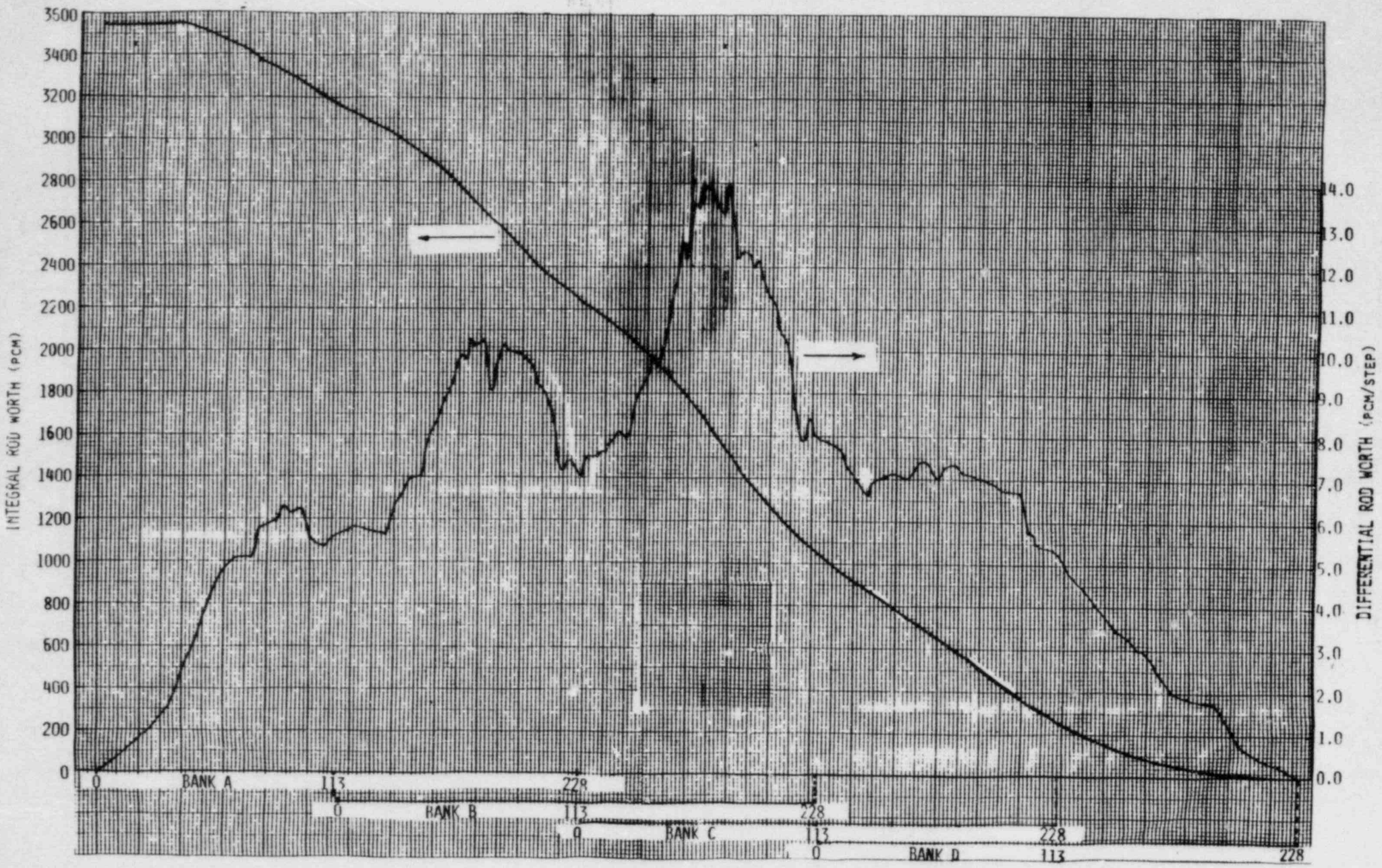
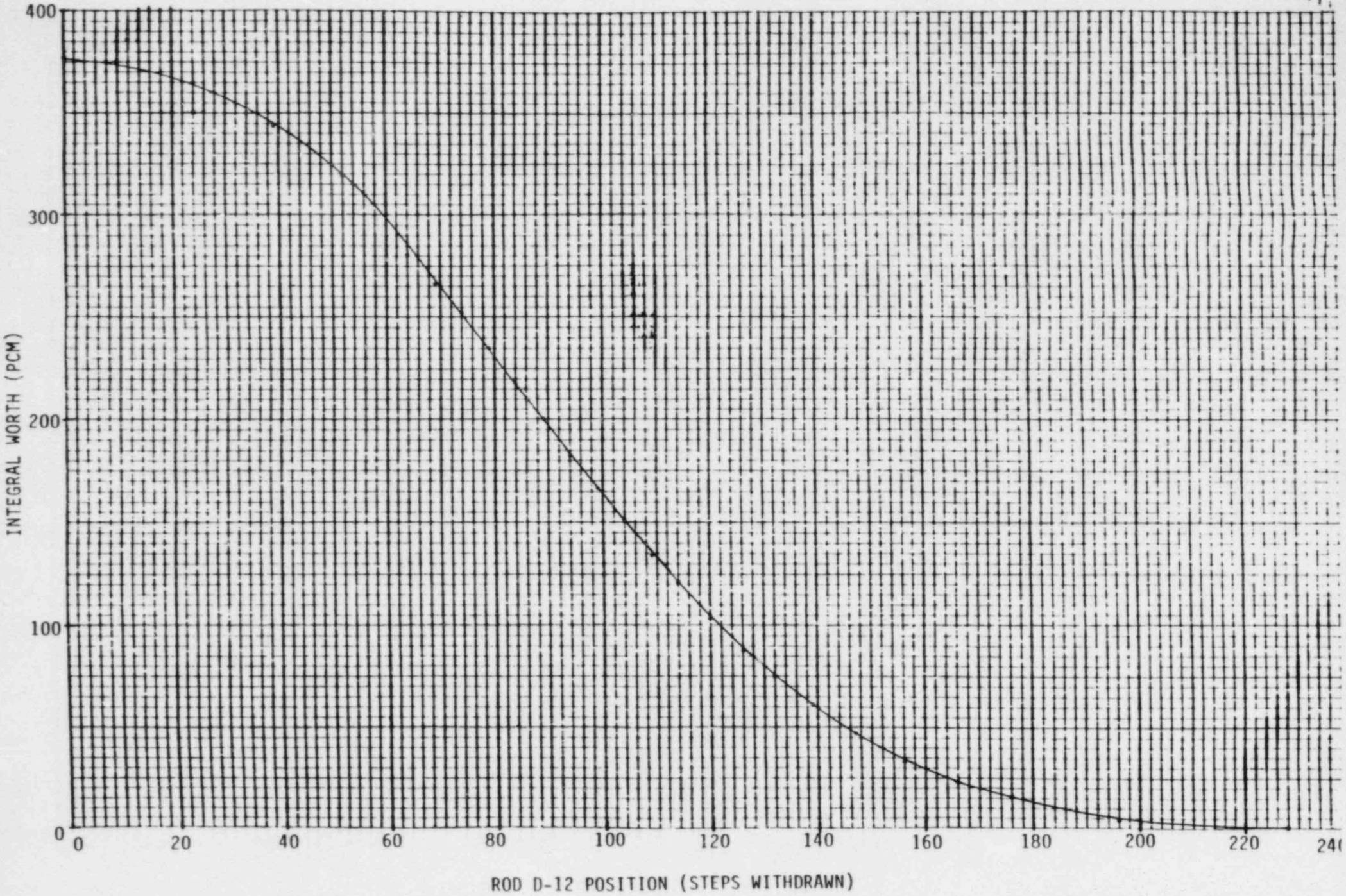


Figure 4.6.14

Integral Reactivity Worth of Pseudo
Ejected Rod D-12 vs. Position



## 4.7 INITIAL STARTUP FLUX MAPPING (S-07SR01 & S-07SR02)
Procedure ETT-SR-07010

### OBJECTIVES

The objective of this procedure was to provide guidance for all flux maps performed during initial startup testing. Flux maps were taken at Hot Zero Power (HZP), 30% power, 50% power, 75% power, 90% power, and 100% power.

### SUMMARY OF RESULTS

In general, several flux maps were performed at each power level. At HZP, flux maps were taken for the following rod configurations: All Rods Out (ARO); Control Bank D at 0 Steps; Rods at the HZP Insertion Limit; and for Rod D-12 Pseudo Ejection case. At 30% power, flux maps were taken at the Hot Full Power (HFP) Insertion Limit and for the Rod D-12 Pseudo Ejection Case. At 50% power, flux maps were taken for the ARO case and for the Rod D-8 Pseudo Rod Drop case. At 75% power, four flux maps were taken at various levels of Delta Flux to support the Axial Flux Difference (AFD) calibrations. At 90 and 100% power, flux maps were taken only for the ARO case. Data taken during the flux maps was analyzed using the Core Program (or NSSS Computer Summary Map). A summary of the safety and design review criteria and results for each power level are presented in Table 4.7.1. Figures 4.7.1, 4.7.4, 4.7.7, 4.7.10, 4.7.13, 4.7.16, 4.7.19, 4.7.22, 4.7.25, 4.7.28, 4.7.31, 4.7.34, 4.7.37, & 4.7.40 are the top 20 limiting nuclear $F_{\Delta H}$ values by region for each flux map. Figures 4.7.2, 4.7.5, 4.7.8, 4.7.11, 4.7.14, 4.7.17, 4.7.20, 4.7.23, 4.7.26, 4.7.29, 4.7.32, 4.7.35, 4.7.38 and 4.7.41 are the top 20 limiting $F_Q$ values by region for each flux map. Figures 4.7.3, 4.7.6, 4.7.9, 4.7.12, 4.7.15, 4.7.18, 4.7.21, 4.7.24, 4.7.27, 4.7.30, 4.7.33, 4.7.36, 4.7.39, and 4.7.42 are the relative errors for $F^N_{\Delta H}$ by core location for each flux map.

A number of deficiencies were identified (three at HZP, two at 30% power, and two at 50% power). Results of the Westinghouse re-analysis of the flux map data concluded that no problems existed and power escalation could continue. Further testing through the 100% power plateau was completed successfully with all safety and design review criteria being met.

### DISCUSSION

Flux maps were taken at various power levels and for various control rod configurations to verify design calculations and assure compliance with Technical Specifications.

Four flux maps were taken at HZP for different control rod configurations: ARO, CBD at 0 Steps, rods at the HZP Insertion Limit, and with rod D-12 ejected. For each map, reactor power was raised to approximately 1.5% to assure adequate detector signals. Safety review criteria were verified using the NSSS Computer Summary Map due to data link problems between the BOP computer and the Corporate computer. Design review criteria were verified later using the CORE Program, after

the data link was restored. All safety review criteria were satisfied. Three deficiencies against design review criteria were identified which were submitted to Westinghouse for resolution. For the HZP insertion limit map, the measured versus predicted distributions for $F^N_{\Delta H}$ were not within ±0.1 and the incore tilt was greater than 1.02. Westinghouse re-analysis of the flux map data determined the incore tilt to be 1.0171, which meets the acceptance criteria, and found only one assembly (N-06) which exceeded the 0.1 limit. However, this was due to high background and erratic data. For the D-12 pseudo rod ejection map, the measured versus predicted distributions for $F^N_{\Delta H}$ again were not within ±0.1. Westinghouse analysis confirmed that several assemblies exceeded this limit. However, this is not considered an uncommon occurrence for this rod configuration. In addition, this criterion was removed from the rod-out-of-bank flux maps by Westinghouse after the Callaway criteria were issued. Thus, nothing was identified that would preclude power ascension past HZP. A problem with thimble G12 was encountered during this section and was deleted from further maps. This does not affect Technical Specification operability.

Two flux maps were taken at 30% power: for rods at the HFP insertion limit and for rod D-12 ejected. All data was analyzed using the Core Program. All safety review criteria were satisfied. Two deficiencies against design review criteria were identified for the D-12 ejected rod case which were submitted to Westinghouse for resolution:

1. The measured versus predicted distributions for $F^N_{\Delta H}$ were not within ±0.1; and

2. The $F^T_Q$ (Hot Channel Factor) was not less than 2.26 (measured at 2.30).

Westinghouse analysis of the flux map data supported the $F^N_{\Delta H}$ error calculations; however, as with the pseudo ejected rod case at HZP, this is not an uncommon occurrence for a rod-out-of-bank configuration. This criterion was also removed from the rod-out-of-bank flux maps by Westinghouse. A safety evaluation was performed by Westinghouse based on the measured hot channel factor and determined that the 2.26 limit was not valid as a safety criteria and that 2.30 was within FSAR analysis limits. Thus no safety issue was represented by the test results and power escalation could continue.

Two flux maps were taken at 50% power: for all rods out (CBD ≥ 210 steps) and for the Rod D-8 pseudo dropped rod case. All safety review criteria were met but one design review criterion was not satisfied. For the pseudo dropped rod case, the measured versus predicted values of $F^N_{\Delta H}$ were not within ±0.1. Re-analysis of the flux map data by Westinghouse did indicate several $F^N_{\Delta H}$ error failures; however, as in the ejected rod cases at HZP and 30% power, this is not an uncommon occurrence for rod-out-of-bank flux maps and the criterion has been removed from this flux map. Thus, no restraints were found to prohibit further power escalation.

Four flux maps were taken at 75% to support the initial AFD calibrations. Flux maps were taken at the following levels of Delta Flux ($\Delta I$):

1. $\Delta I$ within the Target Band;

2. $\Delta I$ at approximately 50% of the negative limit;

3. $\Delta I$ at approximately 90% of the negative limit; and

4. $\Delta I$ at the maximum positive value. All safety and design review criteria were met.

Single flux maps were performed at 90% and 100% power with rods positioned to maintain $\Delta I$ within the Target Band. All acceptance criteria were met for both maps.

TABLE 4.7.1 - INITIAL STARTUP FLUX MAPPING

SHEET 1 of 4

| POWER LEVEL | CONTROL ROD CONFIGURATION | CRITERIA | MEASURED VALUE |
|---|---|---|---|
| HZP | ARO | $F^N_{\Delta H}$ Error ±0.1 | 0.076 (max.) |
| | | Incore Tilt ≤ 1.02 (D)<br>≤ 1.04 (S) | 1.012 |
| | | $F^N_{\Delta H}$ = 1.39 ±.14 | 1.4019 |
| | CBD @ 0 | Incore Tilt ≤ 1.04 (S)<br>≤ 1.02 (D) | 1.019 |
| | | $F^N_{\Delta H}$ = 1.57 ±.16 | 1.624 |
| | | | .079 (max.) |
| | | $F^N_{\Delta H}$ Error ±0.1 | |
| | Insertion Limits | Incore Tilt ≤ 1.04 (S)<br>≤ 1.02 (D) | 1.03 |
| | | $F^N_{\Delta H}$ Error ±0.1 | .1885 (max.) |
| | D-12 Ejected | $F^N_{\Delta H}$ Error ±0.1 | 1.273 (max.) |
| | | $F^T_Q \leq 7.03$ | 6.2633 |

NOTE
S = Safety Review Criteria

D = Design Review Criteria

TABLE 4.7.1 - INITIAL STARTUP FLUX MAPPING

SHEET 2 of 4

| POWER LEVEL | CONTROL ROD CONFIGURATION | CRITERIA | MEASURED VALUE |
|---|---|---|---|
| 30% Power | HFP Insertion Limit | $F^N_{\Delta H}$ Error ±0.1 | - .07181 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.01088 |
| | | $F^N_{\Delta H} \leq 1.7119$ | 1.3953 |
| | | $F_Q(z) \leq 4.64$ | 2.2196 |
| | D-12 Ejected | $F^N_{\Delta H}$ Error ±0.1 | 0.11434 (max.) |
| | | $F^T_Q \leq 3.02$ | 2.30 |
| 50% Power | CBD ≥ 210 Steps | $F^N_{\Delta H}$ Error ±0.1 | -0.06169 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.01309 |
| | | $F^N_{\Delta H} \leq 1.6864$ | 1.3733 |
| | | $F_Q(z) \leq 4.6261$ | 2.0169 |
| | D-8 Dropped | $F^N_{\Delta H}$ Error ±0.1 | 0.8531 (max.) |
| | | $F^N_{\Delta H} \leq 1.6607$ | 1.6021 |
| | | $F_Q(z) \leq 4.5655$ | 2.3647 |
| | | $F^T_{\Delta H} \leq 1.88$ | 1.6662 |

TABLE 4.7.1 - INITIAL STARTUP FLUX MAPPING

SHEET 3 of 4

| POWER LEVEL | CONTROL ROD CONFIGURATION | CRITERIA | MEASURED VALUE |
|---|---|---|---|
| 75% Power | CBD as Req'd for Δ Flux within Target Band | $F^N_{\Delta H}$ Error ±0.1 | -0.0410 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.01122 |
| | | $F^N_{\Delta H} \leq 1.5809$ | 1.3540 |
| | | $F_Q(z) \leq 3.1205$ | 1.9959 |
| | CBD as Req'd for Δ Flux at 50% of Neg. Limit | $F^N_{\Delta H}$ Error ±0.1 | -0.04621 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.00948 |
| | | $F^N_{\Delta H} \leq 1.5735$ | 1.3818 |
| | | $F_Q(z) \leq 3.0216$ | 2.1505 |
| | CBD as Req'd for Δ Flux at 90% of Neg. Limit | $F^N_{\Delta H}$ Error ±0.1 | -0.04742 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.01033 |
| | | $F^N_{\Delta H} \leq 1.5773$ | 1.3763 |
| | | $F_Q(z) \leq 3.0720$ | 2.2103 |
| | CBD as Req'd for Δ Flux at Most Pos. Value | $F^N_{\Delta H}$ Error ±0.1 | -0.07694 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.01296 |
| | | $F^N_{\Delta H} \leq 1.5761$ | 1.3605 |
| | | $F_Q(z) \leq 2.9790$ | 2.0249 |

TABLE 4.7.1 - INITIAL STARTUP FLUX MAPPING

SHEET 4 of 4

| POWER LEVEL | CONTROL ROD CONFIGURATION | CRITERIA | MEASURED VALUE |
|---|---|---|---|
| 90% Power | CBD as Req'd for ΔI within Target Band | $F^{n}_{\Delta H}$ Error ±0.1 | -0.04884 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.01273 |
| | | $F^{N}_{\Delta H}$ ≤ 1.5530 | 1.3525 |
| | | $F_{Q}(z)$ ≤ 2.5895 | 2.0003 |
| 100% Power | CBD as Req'd for ΔI within Target Band | $F^{N}_{\Delta H}$ Error ±0.1 | -0.05923 (max.) |
| | | Incore Tilt ≤ 1.02 | 1.01224 |
| | | $F^{N}_{\Delta H}$ ≤ 1.5099 | 1.3653 |
| | | $F_{Q}(z)$ ≤ 2.3686 | 2.0451 |
| | | $F^{N}_{\Delta H}$ = 1.33 ±.13 | 1.3653 |

## Figure 4.7.1 - HZP, ARO

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0024
COOLANT FLOW CORRECTION FACTOR = 1.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSLBH = 1.8037
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASY. | F.A. LOC. | FSUBHA | ROD FLW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 651 | 04-D | 1.4019 | 1.0000 | 1.8037 | 1.4019 | 0.4019 | 0.2557 | |
| 2 | 652 | 10-M | 1.3957 | 1.0000 | 1.8037 | 1.3957 | 0.4080 | 0.2924 | |
| 3 | 649 | 12-D | 1.3838 | 1.0000 | 1.8037 | 1.3838 | 0.4199 | 0.3035 | |
| 4 | 391 | 07-B | 1.3737 | 1.0000 | 1.8037 | 1.3737 | 0.4301 | 0.3131 | |
| 5 | 394 | 02-J | 1.3651 | 1.0000 | 1.8037 | 1.3651 | 0.4386 | 0.3213 | |
| 6 | 382 | 02-G | 1.3605 | 1.0000 | 1.8037 | 1.3605 | 0.4432 | 0.3257 | |
| 7 | 379 | 14-G | 1.3600 | 1.0000 | 1.8037 | 1.3600 | 0.4437 | 0.3262 | |
| 8 | 623 | 08-E | 1.3590 | 1.0000 | 1.8037 | 1.3590 | 0.4447 | 0.3272 | |
| 9 | 394 | 07-B | 1.3552 | 1.0000 | 1.8037 | 1.3552 | 0.4486 | 0.3310 | |
| 10 | 618 | 04-D | 1.3547 | 1.0000 | 1.8037 | 1.3547 | 0.4491 | 0.3315 | |
| 11 | 619 | 06-D | 1.3498 | 1.0000 | 1.8037 | 1.3498 | 0.4539 | 0.3362 | |
| 12 | 387 | 02-J | 1.3466 | 1.0000 | 1.8037 | 1.3466 | 0.4571 | 0.3394 | |
| 13 | 379 | 06-B | 1.3452 | 1.0000 | 1.8037 | 1.3453 | 0.4585 | 0.3408 | |
| 14 | 608 | 14-E | 1.3447 | 1.0000 | 1.8037 | 1.3447 | 0.4590 | 0.3413 | |
| 15 | 327 | 07-D | 1.3424 | 1.0000 | 1.8037 | 1.3424 | 0.4613 | 0.3436 | |
| 16 | 385 | 02-G | 1.3422 | 1.0000 | 1.8037 | 1.3422 | 0.4616 | 0.3439 | |
| 17 | 377 | 14-G | 1.3417 | 1.0000 | 1.8037 | 1.3417 | 0.4620 | 0.3443 | |
| 18 | 653 | 04-M | 1.3413 | 1.0000 | 1.8037 | 1.3413 | 0.4625 | 0.3448 | |
| 19 | 330 | 07-D | 1.3353 | 1.0000 | 1.8037 | 1.3353 | 0.4685 | 0.3508 | |
| 20 | 333 | 07-D | 1.3342 | 1.0000 | 1.8037 | 1.3342 | 0.4695 | 0.3519 | |

## Figure 4.7.2 - HZP, ARO

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBQE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0000
FACTOR FOR REFLECTING WATER TANK TRIP, = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FQLAR = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | FQFNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECh SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | e51 | 04-D | 26 | 2.151C | 0.9970 | 1.0000 | 4.6261 | 2.3263 | 2.2998 | 0.9886 | |
| 2 | e51 | 04-D | 27 | 2.1342 | 0.9940 | 1.0000 | 4.6122 | 2.3081 | 2.3040 | 0.9982 | |
| 3 | e52 | 12-F | 24 | 2.1400 | 0.9970 | 1.0000 | 4.6261 | 2.3144 | 2.3116 | 0.9939 | |
| 4 | e51 | 04-D | 25 | 2.1492 | 1.0000 | 1.0000 | 4.6400 | 2.3211 | 2.3189 | 0.9990 | |
| 5 | e52 | 12-F | 27 | 2.1315 | 0.9940 | 1.0000 | 4.6122 | 2.3052 | 2.3070 | 1.0009 | |
| 6 | e52 | 12-F | 25 | 2.1349 | 1.0000 | 1.0000 | 4.6400 | 2.3089 | 2.3311 | 1.0096 | |
| 7 | e48 | 12-D | 27 | 2.1141 | 0.9940 | 1.0000 | 4.6122 | 2.2864 | 2.3258 | 1.0172 | |
| 8 | e48 | 12-D | 26 | 2.1193 | 0.9970 | 1.0000 | 4.6261 | 2.2920 | 2.3341 | 1.0184 | |
| 9 | e51 | 04-D | 23 | 2.1009 | 0.9910 | 1.0000 | 4.5982 | 2.2717 | 2.3265 | 1.0241 | |
| 10 | e52 | 12-F | 23 | 2.0955 | 0.9910 | 1.0000 | 4.5982 | 2.2663 | 2.3319 | 1.0289 | |
| 11 | e48 | 12-D | 25 | 2.1141 | 1.0000 | 1.0000 | 4.6400 | 2.2864 | 2.3526 | 1.0294 | |
| 12 | e51 | 04-D | 24 | 2.1096 | 1.0000 | 1.0000 | 4.6400 | 2.2816 | 2.3594 | 1.0337 | |
| 13 | e51 | 04-D | 21 | 2.1082 | 1.0000 | 1.0000 | 4.6400 | 2.2800 | 2.3600 | 1.0351 | |
| 14 | e51 | 04-D | 22 | 2.1048 | 1.0000 | 1.0000 | 4.6400 | 2.2764 | 2.3636 | 1.0383 | |
| 15 | 391 | 07-B | 24 | 2.0874 | 0.9970 | 1.0000 | 4.6261 | 2.2575 | 2.3686 | 1.0492 | |
| 16 | e48 | 12-D | 23 | 2.0725 | 0.9910 | 1.0000 | 4.5982 | 2.2414 | 2.3568 | 1.0515 | |
| 17 | e52 | 12-F | 24 | 2.0690 | 1.0000 | 1.0000 | 4.6400 | 2.2592 | 2.3603 | 1.0533 | |
| 18 | 391 | 07-B | 27 | 2.0746 | 0.9940 | 1.0000 | 4.6122 | 2.2437 | 2.3685 | 1.0556 | |
| 19 | e51 | 04-D | 20 | 2.0945 | 1.0000 | 1.0000 | 4.6400 | 2.2656 | 2.3834 | 1.0562 | |
| 20 | e48 | 12-D | 24 | 2.0829 | 1.0000 | 1.0000 | 4.6400 | 2.2527 | 2.3873 | 1.0595 | |

Figure 4.7.1 HZP, AP.

RELATIVE ERRORS IN F SUB DELTA - CALCULATED FROM ASSISTED CRITICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.000 | 0.001 | -0.025 | -0.050 | -0.018 | C.006 | 0.005 | | | | |
| 02 | | | 0.030 | 0.0[illegible] | -0.001 | 0.000 | -0.014 | -0.030 | -0.011 | C.003 | -0.000 | -0.006 | 0.008 | | |
| 03 | | 0.023 | 0.024 | 0.026 | -0.002 | 0.000 | -0.010 | 0.012 | -0.009 | C.000 | -0.003 | -0.007 | 0.007 | C.022 | |
| 04 | | 0.018 | 0.024 | 0.019 | 0.017 | 0.016 | -0.003 | -0.017 | -0.005 | -C.009 | -0.011 | -0,025 | -0.015 | -0.005 | |
| 05 | 0.045 | 0.042 | 0.040 | 0.035 | 0.018 | 0.015 | -0.012 | -0.020 | -0.012 | -C.016 | -0.020 | -0.033 | -0.043 | -0.031 | -0.018 |
| 06 | 0.06[illegible] | 0.059 | 0.074 | 0.04[illegible] | 0.023 | 0.011 | -0.006 | -0.030 | -0.010 | -C.000 | -0.002 | -0.036 | -0.035 | -0.018 | -0.017 |
| 07 | 0.05[illegible] | 0.058 | 0.05[illegible] | 0.045 | 0.035 | 0.013 | 0.000 | -0.011 | -0.013 | C.022 | -0.016 | -0.029 | -0.044 | -0.021 | -0.021 |
| 08 | 0.04[illegible] | 0.046 | 0.047 | 0.045 | 0.042 | 0.017 | 0.004 | -0.006 | -0.015 | -C.038 | -0.021 | -0.043 | 0.017 | -0.025 | -0.005 |
| 09 | 0.04[illegible] | 0.044 | 0.045 | 0.037 | 0.032 | 0.024 | 0.015 | 0.002 | -0.013 | -C.021 | -0.012 | -0.019 | -0.017 | 0.000 | 0.015 |
| 10 | 0.02[illegible] | 0.030 | 0.043 | 0.005 | 0.021 | 0.013 | 0.027 | 0.011 | -0.003 | -C.013 | -0.016 | -0.027 | -0.027 | 0.007 | 0.007 |
| 11 | 0.01[illegible] | 0.012 | -0.003 | -0.004 | -0.016 | 0.001 | 0.011 | 0.012 | 0.002 | -C.011 | -0.014 | -0.018 | -0.019 | -0.001 | -0.000 |
| 12 | | -0.003 | -0.017 | -0.021 | -0.017 | -0.012 | -0.002 | 0.001 | -0.008 | -C.011 | -0.013 | -0.012 | -0.009 | -0.005 | |
| 13 | | -0.010 | -0.017 | -0.014 | -0.012 | -0.005 | 0.001 | -0.001 | -0.008 | -C.015 | -0.017 | -0.017 | -0.013 | -0.007 | |
| 14 | | | -0.002 | -0.008 | -0.006 | 0.001 | 0.014 | 0.001 | -0.011 | -C.019 | -0.020 | -0.021 | -0.016 | | |
| 15 | | | | | 0.002 | 0.007 | 0.002 | -0.009 | -0.013 | -C.019 | -0.020 | | | | |

## Figure 4.7.4 - HZP, CBD At 0

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0011
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.8041
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD EOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 370 | 14-G | 1.6240 | 1.0000 | 1.8041 | 1.6240 | 0.1802 | 0.1109 | |
| 2 | 391 | 07-E | 1.6076 | 1.0000 | 1.8041 | 1.6076 | 0.1965 | 0.1223 | |
| 3 | 384 | 02-J | 1.5967 | 1.0000 | 1.8041 | 1.5967 | 0.2074 | 0.1299 | |
| 4 | 382 | 02-G | 1.5875 | 1.0000 | 1.8041 | 1.5875 | 0.2166 | 0.1364 | |
| 5 | 383 | 14-J | 1.5831 | 1.0000 | 1.8041 | 1.5831 | 0.2210 | 0.1396 | |
| 6 | 379 | 09-E | 1.5661 | 1.0000 | 1.8041 | 1.5661 | 0.2380 | 0.1520 | |
| 7 | 218 | 08-D | 1.5493 | 1.0000 | 1.8041 | 1.5493 | 0.2549 | 0.1645 | |
| 8 | 330 | 07-D | 1.5427 | 1.0000 | 1.8041 | 1.5427 | 0.2615 | 0.1695 | |
| 9 | 316 | 09-D | 1.5405 | 1.0000 | 1.8041 | 1.5405 | 0.2636 | 0.1711 | |
| 10 | 219 | 08-D | 1.5383 | 1.0000 | 1.8041 | 1.5383 | 0.2658 | 0.1728 | |
| 11 | 220 | 04-H | 1.5377 | 1.0000 | 1.8041 | 1.5377 | 0.2664 | 0.1732 | |
| 12 | 323 | 04-J | 1.5341 | 1.0000 | 1.8041 | 1.5341 | 0.2700 | 0.1760 | |
| 13 | 377 | 14-G | 1.5327 | 1.0000 | 1.8041 | 1.5327 | 0.2714 | 0.1771 | |
| 14 | 333 | 07-D | 1.5299 | 1.0000 | 1.8041 | 1.5299 | 0.2742 | 0.1792 | |
| 15 | 226 | 03-H | 1.5285 | 1.0000 | 1.8041 | 1.5286 | 0.2756 | 0.1803 | |
| 16 | 317 | 09-D | 1.5278 | 1.0000 | 1.8041 | 1.5278 | 0.2763 | 0.1809 | |
| 17 | 221 | 04-H | 1.5269 | 1.0000 | 1.8041 | 1.5269 | 0.2773 | 0.1816 | |
| 18 | 213 | 05-E | 1.5249 | 1.0000 | 1.8041 | 1.5249 | 0.2792 | 0.1831 | |
| 19 | 327 | 07-D | 1.5248 | 1.0000 | 1.8041 | 1.5248 | 0.2793 | 0.1832 | |
| 20 | 315 | 09-D | 1.5227 | 1.0000 | 1.8041 | 1.5227 | 0.2814 | 0.1848 | |

Figure 4.7.5 - HZP, CBD At 0

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0011
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQA | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 376 | 14-E | 26 | 2.4645 | 0.9970 | 1.0000 | 4.6261 | 2.6654 | 1.9607 | 0.7356 | |
| 2 | 376 | 14-G | 27 | 2.4475 | 0.9940 | 1.0000 | 4.6122 | 2.6470 | 1.9652 | 0.7424 | |
| 3 | 376 | 14-G | 25 | 2.4601 | 1.0000 | 1.0000 | 4.6400 | 2.6606 | 1.9794 | 0.7440 | |
| 4 | 391 | 07-B | 26 | 2.4333 | 0.9970 | 1.0000 | 4.6261 | 2.6315 | 1.9945 | 0.7579 | |
| 5 | 384 | 02-J | 25 | 2.4290 | 0.9970 | 1.0000 | 4.6261 | 2.6259 | 2.0002 | 0.7617 | |
| 6 | 391 | 07-B | 27 | 2.4175 | 0.9940 | 1.0000 | 4.6122 | 2.6146 | 1.9976 | 0.7640 | |
| 7 | 382 | 02-G | 26 | 2.4248 | 0.9970 | 1.0000 | 4.6261 | 2.6224 | 2.0037 | 0.7641 | |
| 8 | 376 | 14-G | 28 | 2.4062 | 0.9910 | 1.0000 | 4.5982 | 2.6024 | 1.9959 | 0.7670 | |
| 9 | 391 | 07-B | 25 | 2.4254 | 1.0000 | 1.0000 | 4.6400 | 2.6231 | 2.0169 | 0.7689 | |
| 10 | 382 | 02-G | 27 | 2.4090 | 0.9940 | 1.0000 | 4.6122 | 2.6053 | 2.0069 | 0.7703 | |
| 11 | 384 | 02-J | 27 | 2.4085 | 0.9940 | 1.0000 | 4.6122 | 2.6048 | 2.0073 | 0.7706 | |
| 12 | 384 | 02-J | 25 | 2.4206 | 1.0000 | 1.0000 | 4.6400 | 2.6179 | 2.0221 | 0.7724 | |
| 13 | 382 | 02-G | 25 | 2.4151 | 1.0000 | 1.0000 | 4.6400 | 2.6120 | 2.0280 | 0.7764 | |
| 14 | 376 | 14-G | 24 | 2.4099 | 1.0000 | 1.0000 | 4.6400 | 2.6063 | 2.0337 | 0.7803 | |
| 15 | 391 | 07-B | 28 | 2.3812 | 0.9910 | 1.0000 | 4.5932 | 2.5753 | 2.0230 | 0.7855 | |
| 16 | 383 | 14-J | 26 | 2.3934 | 0.9970 | 1.0000 | 4.6261 | 2.5885 | 2.0376 | 0.7872 | |
| 17 | 383 | 14-J | 27 | 2.3842 | 0.9940 | 1.0000 | 4.6122 | 2.5785 | 2.0337 | 0.7887 | |
| 18 | 383 | 14-J | 25 | 2.3934 | 1.0000 | 1.0000 | 4.6400 | 2.5885 | 2.0515 | 0.7926 | |
| 19 | 382 | 02-G | 28 | 2.3686 | 0.9910 | 1.0000 | 4.5982 | 2.5617 | 2.0366 | 0.7950 | |
| 20 | 384 | 02-J | 22 | 2.3666 | 0.9910 | 1.0000 | 4.5982 | 2.5595 | 2.0387 | 0.7965 | |

## Figure 4.7.6 - HZP, CBD At 0

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.008 | -0.009 | -0.016 | -0.023 | -0.005 | 0.009 | 0.010 | | | | |
| 02 | | | 0.040 | 0.039 | -0.007 | -0.008 | -0.019 | -0.023 | -0.013 | 0.001 | 0.010 | 0.026 | 0.012 | | |
| 03 | | 0.055 | 0.059 | 0.065 | -0.007 | -0.009 | -0.023 | -0.027 | -0.021 | -0.006 | 0.010 | 0.028 | 0.013 | -0.004 | |
| 04 | | 0.043 | 0.039 | 0.063 | 0.006 | 0.003 | -0.031 | -0.034 | -0.019 | -0.014 | -0.006 | 0.004 | -0.003 | -0.014 | |
| 05 | 0.067 | 0.064 | 0.063 | 0.043 | 0.004 | 0.012 | -0.022 | -0.031 | -0.008 | -0.021 | -0.032 | -0.029 | -0.028 | -0.019 | -0.011 |
| 06 | 0.086 | 0.070 | 0.054 | 0.034 | 0.018 | 0.020 | -0.015 | -0.051 | -0.018 | -0.005 | -0.013 | -0.039 | -0.032 | -0.011 | -0.011 |
| 07 | 0.059 | 0.053 | 0.043 | 0.037 | 0.027 | 0.013 | -0.012 | -0.015 | -0.024 | 0.017 | -0.022 | -0.036 | -0.055 | -0.025 | -0.012 |
| 08 | 0.035 | 0.042 | 0.039 | 0.033 | 0.023 | 0.016 | 0.033 | 0.015 | -0.019 | -0.055 | -0.037 | -0.041 | -0.001 | -0.017 | 0.001 |
| 09 | 0.044 | 0.054 | 0.046 | 0.016 | 0.005 | 0.014 | 0.023 | 0.010 | -0.010 | -0.039 | -0.051 | -0.035 | -0.023 | 0.000 | 0.001 |
| 10 | 0.070 | 0.065 | 0.055 | -0.010 | -0.012 | -0.002 | 0.014 | -0.008 | -0.016 | -0.030 | -0.037 | -0.034 | -0.034 | 0.010 | 0.010 |
| 11 | 0.082 | 0.083 | 0.034 | -0.007 | -0.007 | -0.011 | -0.012 | -0.023 | -0.023 | -0.026 | -0.025 | 0.025 | 0.049 | 0.019 | 0.019 |
| 12 | | 0.056 | 0.032 | 0.013 | -0.017 | -0.031 | -0.019 | -0.010 | -0.010 | -0.026 | 0.055 | 0.162 | 0.081 | 0.005 | |
| 13 | | 0.045 | 0.032 | 0.023 | -0.018 | -0.021 | -0.011 | 0.003 | -0.017 | -0.043 | 0.028 | 0.071 | 0.038 | -0.005 | |
| 14 | | | 0.066 | 0.030 | 0.005 | -0.002 | -0.010 | -0.023 | -0.035 | -0.043 | -0.023 | -0.001 | -0.002 | | |
| 15 | | | | | 0.030 | 0.012 | -0.036 | -0.060 | -0.053 | -0.043 | -0.023 | | | | |

## Figure 4.7.7 - HZP, HZP INSERTION LIMITS

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0012
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.8041
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD BOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 216 | 08-D | 1.5256 | 1.0000 | 1.8041 | 1.5856 | 0.2184 | 0.1378 | |
| 2 | 220 | 04-H | 1.5345 | 1.0000 | 1.8041 | 1.5345 | 0.2196 | 0.1336 | |
| 3 | 219 | 08-D | 1.5750 | 1.0000 | 1.8041 | 1.5750 | 0.2291 | 0.1454 | |
| 4 | 221 | 04-H | 1.5738 | 1.0000 | 1.8041 | 1.5738 | 0.2302 | 0.1463 | |
| 5 | 391 | 07-B | 1.5655 | 1.0000 | 1.8041 | 1.5655 | 0.2386 | 0.1524 | |
| 6 | 493 | 10-C | 1.5558 | 1.0000 | 1.8041 | 1.5553 | 0.2483 | 0.1596 | |
| 7 | 316 | 09-D | 1.5453 | 1.0000 | 1.8041 | 1.5453 | 0.2588 | 0.1675 | |
| 8 | 497 | 06-C | 1.5408 | 1.0000 | 1.8041 | 1.5408 | 0.2633 | 0.1709 | |
| 9 | 317 | 09-D | 1.5367 | 1.0000 | 1.8041 | 1.5367 | 0.2674 | 0.1740 | |
| 10 | 330 | 07-D | 1.5337 | 1.0000 | 1.8041 | 1.5337 | 0.2704 | 0.1763 | |
| 11 | 315 | 09-D | 1.5303 | 1.0000 | 1.8041 | 1.5303 | 0.2738 | 0.1789 | |
| 12 | 361 | 07-C | 1.5295 | 1.0000 | 1.8041 | 1.5295 | 0.2746 | 0.1795 | |
| 13 | 364 | 07-C | 1.5287 | 1.0000 | 1.8041 | 1.5287 | 0.2753 | 0.1801 | |
| 14 | 370 | 07-C | 1.5266 | 1.0000 | 1.8041 | 1.5266 | 0.2774 | 0.1817 | |
| 15 | 333 | 07-D | 1.5251 | 1.0000 | 1.8041 | 1.5251 | 0.2790 | 0.1829 | |
| 16 | 608 | 14-E | 1.5250 | 1.0000 | 1.8041 | 1.5250 | 0.2790 | 0.1830 | |
| 17 | 373 | 07-C | 1.5248 | 1.0000 | 1.8041 | 1.5248 | 0.2793 | 0.1832 | |
| 18 | 327 | 07-D | 1.5188 | 1.0000 | 1.8041 | 1.5188 | 0.2853 | 0.1879 | |
| 19 | 214 | 05-H | 1.5152 | 1.0000 | 1.8041 | 1.5152 | 0.2889 | 0.1907 | |
| 20 | 394 | 07-B | 1.5113 | 1.0000 | 1.8041 | 1.5113 | 0.2927 | 0.1937 | |

## Figure 4.7.8 - HZP, HZP INSERTION LIMITS

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBGU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0012
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 391 | 07-B | 14 | 2.6420 | 1.0000 | 1.0000 | 4.6400 | 2.8573 | 1.7827 | 0.6239 | |
| 2 | 391 | 07-B | 13 | 2.6416 | 1.0000 | 1.0000 | 4.6400 | 2.8569 | 1.7831 | 0.6241 | |
| 3 | 391 | 07-B | 15 | 2.6161 | 1.0000 | 1.0000 | 4.6400 | 2.8293 | 1.8107 | 0.6400 | |
| 4 | 391 | 07-B | 12 | 2.6143 | 1.0000 | 1.0000 | 4.6400 | 2.8274 | 1.8126 | 0.6411 | |
| 5 | 608 | 14-E | 18 | 2.6101 | 1.0000 | 1.0000 | 4.6400 | 2.8229 | 1.8171 | 0.6437 | |
| 6 | 608 | 14-E | 19 | 2.6045 | 1.0000 | 1.0000 | 4.6400 | 2.8167 | 1.8233 | 0.6473 | |
| 7 | 391 | 07-B | 18 | 2.5959 | 1.0000 | 1.0000 | 4.6400 | 2.8075 | 1.8325 | 0.6527 | |
| 8 | 391 | 07-B | 19 | 2.5938 | 1.0000 | 1.0000 | 4.6400 | 2.8052 | 1.8348 | 0.6541 | |
| 9 | 608 | 14-E | 14 | 2.5819 | 1.0000 | 1.0000 | 4.6400 | 2.7923 | 1.8477 | 0.6617 | |
| 10 | 608 | 14-E | 15 | 2.5782 | 1.0000 | 1.0000 | 4.6400 | 2.7890 | 1.8510 | 0.6637 | |
| 11 | 608 | 14-E | 20 | 2.5785 | 1.0000 | 1.0000 | 4.6400 | 2.7887 | 1.8513 | 0.6639 | |
| 12 | 497 | 06-C | 14 | 2.5742 | 1.0000 | 1.0000 | 4.6400 | 2.7839 | 1.8560 | 0.6667 | |
| 13 | 391 | 07-B | 20 | 2.5674 | 1.0000 | 1.0000 | 4.6400 | 2.7766 | 1.8634 | 0.6711 | |
| 14 | 497 | 06-C | 15 | 2.5594 | 1.0000 | 1.0000 | 4.6400 | 2.7680 | 1.8720 | 0.6763 | |
| 15 | 608 | 14-E | 13 | 2.5590 | 1.0000 | 1.0000 | 4.6400 | 2.7676 | 1.8724 | 0.6766 | |
| 16 | 497 | 06-C | 19 | 2.5564 | 1.0000 | 1.0000 | 4.6400 | 2.7648 | 1.8752 | 0.6783 | |
| 17 | 497 | 06-C | 13 | 2.5559 | 1.0000 | 1.0000 | 4.6400 | 2.7642 | 1.8758 | 0.6786 | |
| 18 | 497 | 06-C | 18 | 2.5559 | 1.0000 | 1.0000 | 4.6400 | 2.7642 | 1.8758 | 0.6786 | |
| 19 | 391 | 07-B | 11 | 2.5555 | 1.0000 | 1.0000 | 4.6400 | 2.7638 | 1.8762 | 0.6789 | |
| 20 | 608 | 14-E | 17 | 2.5497 | 1.0000 | 1.0000 | 4.6400 | 2.7574 | 1.8826 | 0.6827 | |

Figure 4.7.9 - HZP, HZP INSERTION LIMITS

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | 0.011 | 0.001 | 0.244 | 0.704 | 0.194 | -0.002 | 0.004 | | | | |
| 02 | | | -0.147 | -0.151 | 0.013 | 0.003 | 0.200 | 0.728 | 0.145 | -0.039 | -0.074 | -0.143 | -0.165 | | |
| 03 | | -0.139 | -0.135 | -0.131 | 0.013 | 0.004 | 0.113 | 0.091 | 0.074 | -0.073 | -0.109 | -0.141 | -0.164 | -0.188 | |
| 04 | | -0.152 | -0.113 | -0.059 | 0.009 | 0.007 | -0.005 | -0.063 | -0.019 | -0.045 | -0.084 | -0.115 | -0.165 | -0.181 | |
| 05 | -0.040 | -0.068 | -0.085 | -0.031 | 0.008 | 0.265 | 0.113 | -0.033 | 0.004 | -0.023 | -0.047 | -0.112 | -0.168 | -0.100 | -0.019 |
| 06 | 0.060 | 0.034 | 0.009 | 0.003 | 0.180 | 0.739 | 0.168 | -0.035 | 0.018 | 0.037 | 0.015 | -0.099 | -0.092 | -0.027 | -0.028 |
| 07 | 0.169 | 0.112 | 0.038 | -0.007 | 0.120 | 0.181 | 0.055 | 0.025 | 0.022 | 0.123 | 0.010 | -0.034 | -0.067 | -0.026 | -0.042 |
| 08 | 0.318 | 0.204 | 0.127 | 0.039 | 0.008 | 0.044 | 0.076 | 0.063 | 0.024 | -0.045 | -0.016 | -0.064 | 0.065 | 0.035 | 0.132 |
| 09 | 0.251 | 0.191 | 0.159 | 0.054 | 0.023 | 0.062 | 0.102 | 0.073 | 0.035 | -0.010 | -0.019 | -0.041 | -0.027 | 0.107 | 0.139 |
| 10 | 0.012 | 0.060 | 0.196 | 0.016 | 0.039 | 0.065 | 0.130 | 0.048 | -0.015 | -0.005 | -0.055 | -0.102 | -0.102 | 0.005 | 0.004 |
| 11 | -0.083 | -0.082 | -0.107 | 0.009 | -0.008 | 0.003 | 0.014 | -0.013 | -0.013 | -0.040 | -0.038 | -0.063 | -0.076 | -0.102 | -0.103 |
| 12 | | -0.145 | -0.201 | -0.109 | -0.072 | -0.078 | -0.013 | 0.040 | 0.040 | -0.039 | -0.044 | -0.035 | -0.108 | -0.151 | |
| 13 | | -0.180 | -0.199 | -0.159 | -0.124 | -0.049 | 0.060 | 0.101 | 0.043 | -0.020 | -0.094 | -0.125 | -0.149 | -0.182 | |
| 14 | | | -0.149 | -0.143 | -0.107 | -0.021 | 0.138 | 0.161 | 0.078 | -0.021 | -0.115 | -0.[illegible] | -0.188 | | |

Figure 4.7.10 - HZP, ROD D-12 EJECTED

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0136
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.8004
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD BOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | TECH SPEC MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 160 | 12-D | 3.6271 | 1.0000 | 1.8004 | 3.6271 | -1.8268 | -0.5036 | * |
| 2 | 174 | 13-C | 3.1154 | 1.0000 | 1.8004 | 3.1154 | -1.3151 | -0.4221 | * |
| 3 | 173 | 13-D | 3.0478 | 1.0000 | 1.8004 | 3.0478 | -1.2474 | -0.4093 | * |
| 4 | 172 | 13-E | 2.9634 | 1.0000 | 1.8004 | 2.9634 | -1.1631 | -0.3925 | * |
| 5 | 159 | 12-E | 2.9168 | 1.0000 | 1.8004 | 2.9168 | -1.1164 | -0.3828 | * |
| 6 | 161 | 12-C | 2.8912 | 1.0000 | 1.8004 | 2.8912 | -1.0908 | -0.3773 | * |
| 7 | 147 | 11-C | 2.8706 | 1.0000 | 1.8004 | 2.8706 | -1.0703 | -0.3728 | * |
| 8 | 146 | 11-D | 2.8656 | 1.0000 | 1.8004 | 2.8656 | -1.0653 | -0.3717 | * |
| 9 | 184 | 14-E | 2.8137 | 1.0000 | 1.8004 | 2.8137 | -1.0133 | -0.3601 | * |
| 10 | 185 | 14-D | 2.7466 | 1.0000 | 1.8004 | 2.7466 | -0.9462 | -0.3445 | * |
| 11 | 132 | 10-C | 2.6786 | 1.0000 | 1.8004 | 2.6786 | -0.8782 | -0.3279 | * |
| 12 | 131 | 10-D | 2.6411 | 1.0000 | 1.8004 | 2.6411 | -0.8407 | -0.3183 | * |
| 13 | 145 | 11-E | 2.6387 | 1.0000 | 1.8004 | 2.6387 | -0.8384 | -0.3177 | * |
| 14 | 153 | 12-F | 2.6060 | 1.0000 | 1.8004 | 2.6060 | -0.8056 | -0.3091 | * |
| 15 | 171 | 13-F | 2.5604 | 1.0000 | 1.8004 | 2.5604 | -0.7601 | -0.2969 | * |
| 16 | 144 | 11-F | 2.5106 | 1.0000 | 1.8004 | 2.5106 | -0.7103 | -0.2829 | * |
| 17 | 148 | 11-B | 2.4939 | 1.0000 | 1.8004 | 2.4939 | -0.6935 | -0.2781 | * |
| 18 | 130 | 10-E | 2.3786 | 1.0000 | 1.8004 | 2.3786 | -0.5783 | -0.2431 | * |
| 19 | 152 | 12-B | 2.3380 | 1.0000 | 1.8004 | 2.3390 | -0.5376 | -0.2300 | * |
| 20 | 183 | 14-F | 2.1890 | 1.0000 | 1.8004 | 2.1890 | -0.3887 | -0.1776 | * |

Figure 4.7.11 - HZP, ROD D-12 EJECTED

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.0136
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/G(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 160 | 12-D | 20 | 5.7913 | 0.5160 | 1.0000 | 2.3944 | 6.2633 | -3.8690 | -0.6177 | * |
| 2 | 160 | 12-D | 19 | 5.7836 | 0.5194 | 1.0000 | 2.4100 | 6.2550 | -3.8450 | -0.6147 | * |
| 3 | 160 | 12-D | 21 | 5.7220 | 0.5142 | 1.0000 | 2.3859 | 6.1884 | -3.8025 | -0.6145 | * |
| 4 | 160 | 12-D | 18 | 5.7220 | 0.5231 | 1.0000 | 2.4270 | 6.1894 | -3.7614 | -0.6078 | * |
| 5 | 160 | 12-D | 22 | 5.5911 | 0.5129 | 1.0000 | 2.3799 | 6.0468 | -3.6669 | -0.6064 | * |
| 6 | 160 | 12-D | 25 | 5.3678 | 0.5048 | 1.0000 | 2.3425 | 5.8052 | -3.4627 | -0.5965 | * |
| 7 | 160 | 12-D | 26 | 5.2907 | 0.4997 | 1.0000 | 2.3188 | 5.7219 | -3.4031 | -0.5947 | * |
| 8 | 160 | 12-D | 27 | 5.1521 | 0.4953 | 1.0000 | 2.2984 | 5.5720 | -3.2737 | -0.5875 | * |
| 9 | 160 | 12-D | 28 | 4.9519 | 0.4920 | 1.0000 | 2.2831 | 5.3555 | -3.0724 | -0.5737 | * |
| 10 | 160 | 12-D | 15 | 5.4294 | 0.5440 | 1.0000 | 2.5239 | 5.8719 | -3.3479 | -0.5702 | * |
| 11 | 174 | 13-C | 20 | 5.0276 | 0.5160 | 1.0000 | 2.3944 | 5.4373 | -3.0429 | -0.5596 | * |
| 12 | 160 | 12-D | 14 | 5.3676 | 0.5515 | 1.0000 | 2.5589 | 5.8052 | -3.2463 | -0.5592 | * |
| 13 | 174 | 13-C | 21 | 4.9796 | 0.5142 | 1.0000 | 2.3859 | 5.3855 | -2.9996 | -0.5570 | * |
| 14 | 174 | 13-C | 19 | 5.0212 | 0.5194 | 1.0000 | 2.4100 | 5.4304 | -3.0204 | -0.5562 | * |
| 15 | 160 | 12-D | 29 | 4.6977 | 0.4879 | 1.0000 | 2.2637 | 5.0806 | -2.8169 | -0.5544 | * |
| 16 | 174 | 13-C | 22 | 4.9794 | 0.5129 | 1.0000 | 2.3799 | 5.2760 | -2.8960 | -0.5489 | * |
| 17 | 174 | 13-C | 18 | 4.9670 | 0.5231 | 1.0000 | 2.4270 | 5.3713 | -2.9449 | -0.5482 | * |
| 18 | 173 | 13-D | 20 | 4.8995 | 0.5160 | 1.0000 | 2.3944 | 5.2988 | -2.9045 | -0.5481 | * |
| 19 | 172 | 13-E | 20 | 4.8987 | 0.5160 | 1.0000 | 2.3944 | 5.2979 | -2.9036 | -0.5481 | * |
| 20 | 172 | 13-E | 19 | 4.9039 | 0.5194 | 1.0000 | 2.4100 | 5.3035 | -2.8935 | -0.5456 | * |

FIGURE 4.7.12 - HZP, ROD D-12 EJECTED

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | 0.455 | 0.447 | 0.798 | 1.412 | 0.641 | C.346 | 0.349 | | | | |
| 02 | | | 0.535 | 0.532 | 0.457 | 0.449 | 0.726 | 1.428 | 0.586 | C.299 | 0.193 | 0.023 | 0.039 | | |
| 03 | | 0.371 | 0.388 | 0.416 | 0.457 | 0.452 | 0.573 | 0.547 | 0.485 | 0.255 | 0.128 | 0.023 | 0.040 | 0.058 | |
| 04 | | 0.309 | 0.317 | 0.358 | 0.421 | 0.422 | 0.376 | 0.278 | 0.308 | C.197 | 0.110 | -0.002 | -0.025 | -0.015 | |
| 05 | 0.468 | 0.424 | 0.398 | 0.428 | 0.426 | 0.778 | 0.485 | 0.253 | 0.248 | C.162 | 0.086 | -0.018 | -0.103 | -0.043 | 0.026 |
| 06 | 0.608 | 0.588 | 0.569 | 0.488 | 0.692 | 1.398 | 0.501 | 0.234 | 0.243 | 0.199 | 0.143 | -0.030 | -0.047 | 0.022 | 0.021 |
| 07 | 0.759 | 0.667 | 0.540 | 0.460 | 0.586 | 0.533 | 0.303 | 0.279 | 0.128 | C.235 | 0.018 | -0.058 | -0.054 | 0.105 | 0.245 |
| 08 | 0.951 | 0.718 | 0.604 | 0.476 | 0.386 | 0.342 | 0.325 | 0.210 | 0.032 | -0.068 | -0.053 | -0.115 | 0.030 | 0.617 | 0.156 |
| 09 | 0.757 | 0.609 | 0.602 | 0.431 | 0.364 | 0.323 | 0.276 | 0.156 | 0.024 | -C.057 | -0.117 | -0.144 | -0.020 | 0.127 | -0.087 |
| 10 | 0.354 | 0.419 | 0.606 | 0.231 | 0.342 | 0.239 | 0.232 | 0.077 | 0.011 | -0.139 | -0.229 | -0.276 | -0.275 | -0.170 | -0.171 |
| 11 | 0.220 | 0.221 | 0.053 | 0.189 | 0.103 | 0.107 | C.049 | -0.009 | -0.009 | -0.269 | -0.267 | -0.300 | -0.315 | -0.242 | -0.243 |
| 12 | | 0.033 | -0.106 | -0.033 | -0.019 | -0.069 | -0.015 | 0.026 | 0.026 | -0.266 | -0.312 | -0.351 | -0.342 | -0.295 | |
| 13 | | -0.060 | -0.103 | -0.063 | -0.069 | -0.016 | 0.061 | 0.063 | -0.096 | -0.242 | -0.336 | -0.376 | -0.357 | -0.329 | |
| 14 | | | 0.010 | -0.029 | -0.016 | 0.054 | 0.168 | 0.144 | -0.069 | -0.242 | -0.328 | -0.400 | -0.371 | | |
| 15 | | | | | 0.043 | 0.092 | 0.197 | 0.225 | -0.035 | -0.244 | -0.329 | | | | |

## FIGURE 4.7.13 - 30% POWER, HFP INSERTION LIMIT

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.3075
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.7119
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHA | ROD POW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 376 | 14-G | 1.3953 | 1.0000 | 1.7119 | 1.3953 | 0.3166 | 0.2269 | |
| 2 | 391 | 07-E | 1.3946 | 1.0000 | 1.7119 | 1.3946 | 0.3173 | 0.2275 | |
| 3 | 384 | 02-J | 1.3838 | 1.0000 | 1.7119 | 1.3838 | 0.3281 | 0.2371 | |
| 4 | 382 | 02-G | 1.3820 | 1.0000 | 1.7119 | 1.3820 | 0.3299 | 0.2387 | |
| 5 | 379 | 09-B | 1.3790 | 1.0000 | 1.7119 | 1.3790 | 0.3329 | 0.2414 | |
| 6 | 214 | 05-F | 1.3675 | 1.0000 | 1.7119 | 1.3675 | 0.3444 | 0.2519 | |
| 7 | 377 | 14-E | 1.3659 | 1.0000 | 1.7119 | 1.3659 | 0.3460 | 0.2533 | |
| 8 | 394 | 07-G | 1.3654 | 1.0000 | 1.7119 | 1.3654 | 0.3465 | 0.2538 | |
| 9 | 218 | 02-D | 1.3584 | 1.0000 | 1.7119 | 1.3584 | 0.3535 | 0.2602 | |
| 10 | 392 | 09-F | 1.3576 | 1.0000 | 1.7119 | 1.3576 | 0.3543 | 0.2610 | |
| 11 | 611 | 11-E | 1.3563 | 1.0000 | 1.7119 | 1.3563 | 0.3556 | 0.2622 | |
| 12 | 387 | 02-J | 1.3546 | 1.0000 | 1.7119 | 1.3546 | 0.3573 | 0.2637 | |
| 13 | 623 | 05-E | 1.3531 | 1.0000 | 1.7119 | 1.3531 | 0.3588 | 0.2651 | |
| 14 | 385 | 02-G | 1.3528 | 1.0000 | 1.7119 | 1.3528 | 0.3591 | 0.2654 | |
| 15 | 219 | 08-D | 1.3527 | 1.0000 | 1.7119 | 1.3527 | 0.3592 | 0.2655 | |
| 16 | 213 | 02-E | 1.3518 | 1.0000 | 1.7119 | 1.3518 | 0.3600 | 0.2663 | |
| 17 | 383 | 14-J | 1.3517 | 1.0000 | 1.7119 | 1.3517 | 0.3602 | 0.2665 | |
| 18 | 220 | 04-H | 1.3513 | 1.0000 | 1.7119 | 1.3513 | 0.3605 | 0.2668 | |
| 19 | 380 | 09-E | 1.3501 | 1.0000 | 1.7119 | 1.3501 | 0.3617 | 0.2679 | |
| 20 | 221 | 04-H | 1.3457 | 1.0000 | 1.7119 | 1.3457 | 0.3662 | 0.2721 | |

## FIGURE 4.7.14 - 30% POWER, HFP INSERTION LIMITS

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUSGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSU30U = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.3078
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQA | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | E51 | 04-D | 20 | 2.0523 | 1.0000 | 1.0000 | 4.6400 | 2.2196 | 2.4204 | 1.0905 | |
| 2 | E51 | 04-D | 19 | 2.0488 | 1.0000 | 1.0000 | 4.6400 | 2.2158 | 2.4242 | 1.0940 | |
| 3 | E51 | 04-D | 21 | 2.0396 | 1.0000 | 1.0000 | 4.6400 | 2.2050 | 2.4350 | 1.1043 | |
| 4 | E52 | 12-M | 20 | 2.0386 | 1.0000 | 1.0000 | 4.6400 | 2.2047 | 2.4353 | 1.1046 | |
| 5 | E52 | 12-M | 19 | 2.0301 | 1.0000 | 1.0000 | 4.6400 | 2.1955 | 2.4445 | 1.1134 | |
| 6 | E52 | 12-M | 21 | 2.0232 | 1.0000 | 1.0000 | 4.6400 | 2.1881 | 2.4519 | 1.1206 | |
| 7 | E46 | 12-D | 19 | 2.0205 | 1.0000 | 1.0000 | 4.6400 | 2.1851 | 2.4549 | 1.1234 | |
| 8 | E51 | 04-D | 18 | 2.0195 | 1.0000 | 1.0000 | 4.6400 | 2.1840 | 2.4560 | 1.1245 | |
| 9 | E53 | 04-M | 20 | 2.0192 | 1.0000 | 1.0000 | 4.6400 | 2.1837 | 2.4563 | 1.1245 | |
| 10 | E53 | 04-M | 19 | 2.0109 | 1.0000 | 1.0000 | 4.6400 | 2.1747 | 2.4653 | 1.1337 | |
| 11 | E53 | 04-M | 21 | 2.0092 | 1.0000 | 1.0000 | 4.6400 | 2.1730 | 2.4670 | 1.1353 | |
| 12 | 391 | 07-B | 20 | 2.0076 | 1.0000 | 1.0000 | 4.6400 | 2.1712 | 2.4688 | 1.1370 | |
| 13 | E45 | 12-D | 20 | 2.0069 | 1.0000 | 1.0000 | 4.6400 | 2.1703 | 2.4697 | 1.1379 | |
| 14 | E42 | 12-D | 18 | 2.0069 | 1.0000 | 1.0000 | 4.6400 | 2.1703 | 2.4697 | 1.1379 | |
| 15 | E51 | 04-D | 22 | 2.0035 | 1.0000 | 1.0000 | 4.6400 | 2.1666 | 2.4732 | 1.1414 | |
| 16 | E52 | 12-M | 18 | 2.0029 | 1.0000 | 1.0000 | 4.6400 | 2.1661 | 2.4739 | 1.1421 | |
| 17 | E23 | 05-B | 20 | 2.0002 | 1.0000 | 1.0000 | 4.6400 | 2.1632 | 2.4768 | 1.1450 | |
| 18 | 391 | 07-B | 19 | 1.9972 | 1.0000 | 1.0000 | 4.6400 | 2.1599 | 2.4801 | 1.1482 | |
| 19 | 391 | 07-B | 21 | 1.9971 | 1.0000 | 1.0000 | 4.6400 | 2.1598 | 2.4802 | 1.1483 | |
| 20 | 376 | 14-G | 20 | 1.9944 | 1.0000 | 1.0000 | 4.6400 | 2.1569 | 2.4831 | 1.1512 | |

## FIGURE 4.7.15 - 30% POWER, HFP INSERTION LIMITS

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, ((CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | 0.011 | 0.012 | -0.015 | -0.041 | -0.014 | C.006 | 0.005 | | | | |
| 02 | | | -0.005 | -0.005 | 0.011 | C.012 | -C.002 | -0.042 | -0.001 | C.010 | 0.007 | 0.004 | -0.013 | | |
| 03 | | 0.003 | 0.000 | -0.005 | 0.010 | C.011 | 0.000 | 0.038 | 0.002 | C.015 | 0.008 | 0.005 | -0.012 | -0.029 | |
| 04 | | 0.015 | -0.005 | 0.002 | 0.013 | C.014 | 0.004 | -0.018 | 0.001 | -C.003 | -0.003 | -0.008 | -0.013 | -0.022 | |
| 05 | 0.025 | 0.020 | 0.017 | 0.018 | 0.013 | C.004 | -C.025 | -0.028 | -0.013 | -C.015 | -0.019 | -0.016 | -0.013 | -C.026 | -0.039 |
| 06 | 0.033 | C.035 | 0.044 | 0.043 | 0.019 | -0.004 | -0.026 | -0.053 | -0.017 | C.004 | 0.003 | -0.011 | -0.018 | -0.038 | -0.038 |
| 07 | 0.019 | 0.032 | 0.051 | 0.064 | 0.033 | -0.006 | -0.017 | -0.034 | -0.021 | C.034 | -0.011 | -0.003 | -0.002 | -C.010 | -0.026 |
| 08 | 0.005 | 0.022 | 0.045 | 0.047 | 0.033 | -0.004 | -C.029 | -0.039 | -0.039 | -C.058 | -0.017 | -0.024 | 0.028 | -C.014 | -0.007 |
| 09 | 0.011 | 0.017 | 0.031 | 0.041 | 0.039 | 0.015 | -C.008 | -0.033 | -0.064 | -C.043 | -0.003 | 0.000 | 0.002 | C.001 | -0.000 |
| 10 | -0.001 | 0.004 | 0.016 | 0.023 | 0.044 | C.020 | 0.014 | -0.003 | -0.027 | -C.024 | 0.000 | 0.008 | 0.008 | -0.015 | -C.015 |
| 11 | -0.014 | -0.015 | -0.004 | 0.013 | -0.006 | 0.006 | C.013 | 0.029 | 0.029 | -C.004 | -0.004 | 0.013 | 0.024 | -C.028 | -0.028 |
| 12 | | -0.004 | 0.009 | -0.002 | -0.009 | -C.013 | 0.003 | 0.022 | 0.022 | -C.004 | 0.017 | 0.025 | 0.012 | -0.023 | |
| 13 | | -0.003 | 0.009 | -0.005 | -0.007 | -0.002 | 0.010 | 0.014 | -0.004 | -C.031 | 0.005 | 0.023 | 0.005 | -0.019 | |
| 14 | | | -0.020 | -0.012 | -0.014 | C.001 | 0.022 | 0.010 | -0.010 | -C.030 | -0.011 | 0.010 | -0.002 | | |
| 15 | | | | | -0.022 | -0.004 | 0.007 | -0.006 | -0.016 | -C.030 | -0.011 | | | | |

## FIGURE 4.7.16 - 30% POWER, ROD D-12 EJECTED

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.3149
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.7098
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD FCW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 410 | 12-D | 1.5683 | 1.0000 | 1.7098 | 1.5683 | 0.1414 | 0.0902 | |
| 2 | 366 | 13-F | 1.5519 | 1.0000 | 1.7098 | 1.5519 | 0.1579 | 0.1017 | |
| 3 | 391 | 14-E | 1.5449 | 1.0000 | 1.7098 | 1.5449 | 0.1648 | 0.1067 | |
| 4 | 387 | 12-E | 1.5420 | 1.0000 | 1.7098 | 1.5420 | 0.1678 | 0.1098 | |
| 5 | 367 | 13-F | 1.5330 | 1.0000 | 1.7098 | 1.5330 | 0.1767 | 0.1153 | |
| 6 | 388 | 12-E | 1.5320 | 1.0000 | 1.7098 | 1.5320 | 0.1778 | 0.1160 | |
| 7 | 362 | 11-F | 1.5169 | 1.0000 | 1.7098 | 1.5169 | 0.1929 | 0.1271 | |
| 8 | 408 | 11-D | 1.5168 | 1.0000 | 1.7098 | 1.5168 | 0.1930 | 0.1272 | |
| 9 | 363 | 11-F | 1.5139 | 1.0000 | 1.7098 | 1.5139 | 0.1959 | 0.1294 | |
| 10 | 409 | 11-D | 1.5070 | 1.0000 | 1.7098 | 1.5070 | 0.2028 | 0.1346 | |
| 11 | 440 | 09-B | 1.4852 | 1.0000 | 1.7098 | 1.4852 | 0.2245 | 0.1512 | |
| 12 | 383 | 10-E | 1.4815 | 1.0000 | 1.7098 | 1.4815 | 0.2283 | 0.1541 | |
| 13 | 441 | 09-B | 1.4803 | 1.0000 | 1.7098 | 1.4803 | 0.2295 | 0.1550 | |
| 14 | 334 | 10-E | 1.4786 | 1.0000 | 1.7098 | 1.4786 | 0.2312 | 0.1563 | |
| 15 | 344 | 14-G | 1.4698 | 1.0000 | 1.7098 | 1.4698 | 0.2400 | 0.1633 | |
| 16 | 345 | 14-G | 1.4650 | 1.0000 | 1.7098 | 1.4650 | 0.2448 | 0.1671 | |
| 17 | 412 | 13-D | 1.4635 | 1.0000 | 1.7098 | 1.4635 | 0.2463 | 0.1683 | |
| 18 | 404 | 09-D | 1.4582 | 1.0000 | 1.7098 | 1.4582 | 0.2516 | 0.1725 | |
| 19 | 405 | 09-C | 1.4489 | 1.0000 | 1.7098 | 1.4489 | 0.2609 | 0.1801 | |
| 20 | 160 | 12-D | 1.4373 | 1.0000 | 1.7098 | 1.4373 | 0.2724 | 0.1895 | |

## FIGURE 4.7.17 - 30% POWER, ROD D-12 EJECTED

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.3149
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 410 | 12-D | 27 | 2.1275 | 0.9940 | 1.0000 | 4.6122 | 2.3009 | 2.3113 | 1.0045 | |
| 2 | 410 | 12-D | 26 | 2.1253 | 0.9970 | 1.0000 | 4.6261 | 2.2995 | 2.3276 | 1.0126 | |
| 3 | 410 | 12-D | 28 | 2.1099 | 0.9910 | 1.0000 | 4.5982 | 2.2819 | 2.3163 | 1.0151 | |
| 4 | 366 | 13-F | 27 | 2.1095 | 0.9940 | 1.0000 | 4.6122 | 2.2814 | 2.3307 | 1.0215 | |
| 5 | 366 | 13-F | 26 | 2.1125 | 0.9970 | 1.0000 | 4.6261 | 2.2847 | 2.3414 | 1.0248 | |
| 6 | 337 | 12-E | 27 | 2.0986 | 0.9940 | 1.0000 | 4.6122 | 2.2696 | 2.3425 | 1.0321 | |
| 7 | 410 | 12-D | 25 | 2.1100 | 1.0000 | 1.0000 | 4.6400 | 2.2819 | 2.3551 | 1.0334 | |
| 8 | 366 | 13-F | 28 | 2.0867 | 0.9910 | 1.0000 | 4.5982 | 2.2589 | 2.3393 | 1.0356 | |
| 9 | 337 | 12-E | 26 | 2.0975 | 0.9970 | 1.0000 | 4.6261 | 2.2683 | 2.3576 | 1.0393 | |
| 10 | 366 | 13-F | 25 | 2.1007 | 1.0000 | 1.0000 | 4.6400 | 2.2719 | 2.3631 | 1.0424 | |
| 11 | 337 | 12-E | 29 | 2.0797 | 0.9910 | 1.0000 | 4.5982 | 2.2492 | 2.3490 | 1.0444 | |
| 12 | 391 | 14-E | 27 | 2.0659 | 0.9940 | 1.0000 | 4.6122 | 2.2559 | 2.3563 | 1.0445 | |
| 13 | 362 | 11-F | 27 | 2.0658 | 0.9940 | 1.0000 | 4.6122 | 2.2558 | 2.3564 | 1.0446 | |
| 14 | 410 | 12-D | 24 | 2.0683 | 0.9970 | 1.0000 | 4.5797 | 2.2368 | 2.3428 | 1.0474 | |
| 15 | 410 | 12-D | 33 | 2.0267 | 0.9696 | 1.0000 | 4.4942 | 2.1919 | 2.3023 | 1.0504 | |
| 16 | 362 | 11-F | 25 | 2.0659 | 0.9970 | 1.0000 | 4.6261 | 2.2559 | 2.3702 | 1.0507 | |
| 17 | 391 | 14-E | 25 | 2.0721 | 0.9910 | 1.0000 | 4.5982 | 2.2410 | 2.3573 | 1.0519 | |
| 18 | 388 | 12-E | 27 | 2.0776 | 0.9940 | 1.0000 | 4.6122 | 2.2469 | 2.3653 | 1.0527 | |
| 19 | 391 | 14-E | 25 | 2.0838 | 0.9970 | 1.0000 | 4.6261 | 2.2536 | 2.3725 | 1.0527 | |
| 20 | 410 | 12-D | 32 | 2.0432 | 0.9780 | 1.0000 | 4.5379 | 2.2097 | 2.3282 | 1.0536 | |

FIGURE 4.7.18 - 30% POWER, ROD D-12 EJECTED

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.039 | -0.039 | -0.058 | -0.076 | -0.026 | 0.013 | 0.013 | | | | |
| 02 | | | 0.049 | 0.047 | -0.038 | -0.039 | -0.029 | -0.076 | -0.000 | 0.030 | 0.019 | 0.000 | 0.013 | | |
| 03 | | 0.004 | 0.020 | 0.047 | -0.038 | -0.039 | -0.014 | 0.062 | 0.008 | 0.047 | 0.023 | 0.002 | 0.013 | 0.025 | |
| 04 | | -0.049 | 0.043 | 0.032 | 0.002 | 0.000 | 0.010 | -0.024 | 0.007 | 0.001 | -0.002 | -0.025 | -0.015 | -0.009 | |
| 05 | 0.009 | 0.015 | 0.019 | 0.028 | 0.001 | -0.002 | -0.013 | -0.019 | -0.023 | -0.029 | -0.036 | -0.044 | -0.051 | -0.061 | -0.070 |
| 06 | 0.056 | 0.055 | 0.054 | 0.053 | 0.021 | -0.005 | -0.002 | -0.009 | 0.015 | 0.025 | 0.022 | -0.026 | -0.038 | -0.069 | -0.059 |
| 07 | 0.050 | 0.057 | 0.074 | 0.092 | 0.064 | 0.040 | 0.009 | 0.028 | 0.038 | 0.114 | 0.032 | 0.004 | 0.011 | -0.025 | -0.065 |
| 08 | 0.045 | 0.052 | 0.076 | 0.086 | 0.085 | 0.075 | 0.114 | 0.045 | 0.019 | -0.008 | 0.009 | -0.030 | 0.053 | -0.060 | -0.063 |
| 09 | 0.040 | 0.035 | 0.055 | 0.062 | 0.085 | 0.096 | 0.109 | 0.056 | -0.009 | -0.006 | -0.001 | -0.002 | -0.007 | -0.034 | -0.064 |
| 10 | 0.019 | 0.024 | 0.036 | 0.034 | 0.032 | 0.058 | 0.104 | 0.059 | 0.003 | -0.017 | -0.017 | -0.010 | -0.009 | -0.037 | -0.037 |
| 11 | 0.008 | 0.008 | -0.034 | 0.015 | -0.028 | 0.014 | 0.049 | 0.069 | 0.070 | -0.040 | -0.039 | -0.042 | -0.043 | -0.009 | -0.008 |
| 12 | | -0.037 | -0.078 | -0.054 | -0.043 | -0.044 | 0.009 | 0.050 | 0.051 | -0.039 | -0.058 | -0.075 | -0.039 | 0.006 | |
| 13 | | -0.034 | -0.078 | -0.042 | -0.054 | -0.030 | 0.010 | 0.032 | -0.022 | -0.090 | -0.085 | -0.082 | -0.044 | 0.016 | |
| 14 | | | 0.035 | -0.022 | -0.026 | -0.004 | 0.030 | 0.030 | -0.023 | -0.090 | -0.089 | -0.089 | -0.048 | | |
| 15 | | | | | 0.004 | 0.015 | 0.027 | 0.026 | -0.025 | -0.089 | -0.089 | | | | |

## FIGURE 4.7.19 - 50% POWER, ARO

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.4629
COOLANT FLOW CORRECTION FACTOR = 0.0220
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0220

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.6964
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A.. LOC. | FSUBHN | POD ECH FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 651 | 04-D | 1.3733 | 1.0000 | 1.6864 | 1.3733 | 0.3131 | 0.2280 | |
| 2 | 652 | 12-M | 1.3601 | 1.0000 | 1.6864 | 1.3601 | 0.3263 | 0.2399 | |
| 3 | 584 | 02-J | 1.3574 | 1.0000 | 1.6864 | 1.3574 | 0.3290 | 0.2424 | |
| 4 | 376 | 14-G | 1.3536 | 1.0000 | 1.6864 | 1.3536 | 0.3328 | 0.2458 | |
| 5 | 214 | 05-H | 1.3528 | 1.0000 | 1.6864 | 1.3528 | 0.3336 | 0.2466 | |
| 6 | 391 | 07-E | 1.3525 | 1.0000 | 1.6864 | 1.3525 | 0.3339 | 0.2469 | |
| 7 | 379 | 04-E | 1.3453 | 1.0000 | 1.6864 | 1.3453 | 0.3411 | 0.2536 | |
| 8 | 611 | 11-E | 1.3430 | 1.0000 | 1.6864 | 1.3430 | 0.3434 | 0.2557 | |
| 9 | 218 | 08-D | 1.3427 | 1.0000 | 1.6864 | 1.3427 | 0.3437 | 0.2559 | |
| 10 | 382 | 02-G | 1.3394 | 1.0000 | 1.6864 | 1.3394 | 0.3470 | 0.2590 | |
| 11 | 387 | 02-J | 1.3393 | 1.0000 | 1.6864 | 1.3393 | 0.3471 | 0.2592 | |
| 12 | 219 | 08-J | 1.3390 | 1.0000 | 1.6864 | 1.3390 | 0.3474 | 0.2594 | |
| 13 | 213 | 08-E | 1.3374 | 1.0000 | 1.6864 | 1.3374 | 0.3490 | 0.2609 | |
| 14 | 623 | 05-B | 1.3373 | 1.0000 | 1.6864 | 1.3373 | 0.3491 | 0.2611 | |
| 15 | 377 | 14-G | 1.3355 | 1.0000 | 1.6864 | 1.3355 | 0.3509 | 0.2627 | |
| 16 | 648 | 12-D | 1.3347 | 1.0000 | 1.6864 | 1.3347 | 0.3517 | 0.2635 | |
| 17 | 394 | 07-B | 1.3344 | 1.0000 | 1.6864 | 1.3344 | 0.3520 | 0.2638 | |
| 18 | 446 | 05-F | 1.3331 | 1.0000 | 1.6864 | 1.3331 | 0.3533 | 0.2650 | |
| 19 | 220 | 04-H | 1.3329 | 1.0000 | 1.6864 | 1.3329 | 0.3535 | 0.2652 | |
| 20 | 653 | 04-M | 1.3322 | 1.0000 | 1.6864 | 1.3322 | 0.3542 | 0.2659 | |

## FIGURE 4.7.20 - 50% POWER, ARO

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.4627
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQA | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 651 | 04-D | 26 | 1.8950 | 0.9970 | 1.0000 | 4.6261 | 2.0169 | 2.6091 | 1.2936 | |
| 2 | 651 | 04-D | 27 | 1.8923 | 0.9940 | 1.0000 | 4.6122 | 2.0033 | 2.6089 | 1.3021 | |
| 3 | 651 | 04-D | 25 | 1.8924 | 1.0000 | 1.0000 | 4.6400 | 2.0142 | 2.6258 | 1.3036 | |
| 4 | 651 | 04-D | 20 | 1.8998 | 1.0000 | 1.0000 | 4.6400 | 2.0107 | 2.6293 | 1.3077 | |
| 5 | 651 | 04-D | 21 | 1.8994 | 1.0000 | 1.0000 | 4.6400 | 2.0098 | 2.6302 | 1.3082 | |
| 6 | 652 | 12-M | 26 | 1.8998 | 0.9970 | 1.0000 | 4.6261 | 2.0006 | 2.6255 | 1.3124 | |
| 7 | 652 | 12-M | 27 | 1.8940 | 0.9940 | 1.0000 | 4.6122 | 1.9900 | 2.6222 | 1.3177 | |
| 8 | 651 | 04-D | 28 | 1.8956 | 0.9910 | 1.0000 | 4.5982 | 1.9776 | 2.6206 | 1.3252 | |
| 9 | 651 | 04-D | 19 | 1.8940 | 1.0000 | 1.0000 | 4.6400 | 1.9943 | 2.6457 | 1.3260 | |
| 10 | 651 | 04-D | 22 | 1.8937 | 1.0000 | 1.0000 | 4.6400 | 1.9940 | 2.6460 | 1.3270 | |
| 11 | 652 | 12-M | 25 | 1.8926 | 1.0000 | 1.0000 | 4.6400 | 1.9931 | 2.6469 | 1.3280 | |
| 12 | 652 | 12-M | 21 | 1.8899 | 1.0000 | 1.0000 | 4.6400 | 1.9899 | 2.6501 | 1.3313 | |
| 13 | 384 | 02-J | 26 | 1.8842 | 0.9970 | 1.0000 | 4.6261 | 1.9837 | 2.6424 | 1.3321 | |
| 14 | 651 | 04-D | 24 | 1.8867 | 1.0000 | 1.0000 | 4.6400 | 1.9864 | 2.6536 | 1.3359 | |
| 15 | 652 | 12-M | 20 | 1.8865 | 1.0000 | 1.0000 | 4.6400 | 1.9862 | 2.6538 | 1.3361 | |
| 16 | 214 | 05-H | 27 | 1.8852 | 0.9940 | 1.0000 | 4.6122 | 1.9739 | 2.6383 | 1.3366 | |
| 17 | 214 | 05-H | 26 | 1.8891 | 0.9970 | 1.0000 | 4.6261 | 1.9782 | 2.6479 | 1.3385 | |
| 18 | 376 | 14-G | 26 | 1.8872 | 0.9970 | 1.0000 | 4.6261 | 1.9762 | 2.6499 | 1.3410 | |
| 19 | 384 | 02-J | 27 | 1.8815 | 0.9940 | 1.0000 | 4.6122 | 1.9699 | 2.6422 | 1.3413 | |
| 20 | 652 | 12-M | 29 | 1.8854 | 0.9910 | 1.0000 | 4.5982 | 1.9633 | 2.6349 | 1.3421 | |

FIGURE 4.7.21 - 50% POWER, ARO

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.002 | -0.001 | -0.021 | -0.041 | -0.009 | 0.015 | 0.014 | | | | |
| 02 | | | 0.008 | 0.002 | -0.002 | -0.001 | -0.009 | -0.041 | 0.004 | 0.026 | 0.027 | 0.031 | 0.009 | | |
| 03 | | 0.034 | 0.032 | 0.029 | -0.002 | -0.002 | -0.009 | 0.026 | 0.003 | 0.038 | 0.034 | 0.030 | 0.008 | -0.013 | |
| 04 | | 0.041 | 0.050 | 0.028 | 0.002 | 0.004 | -0.007 | -0.028 | -0.006 | -0.002 | 0.002 | -0.003 | -0.002 | -0.014 | |
| 05 | 0.039 | 0.043 | 0.046 | 0.033 | 0.003 | -0.012 | -0.026 | -0.038 | -0.033 | -0.029 | -0.025 | -0.021 | -0.016 | -0.015 | -0.014 |
| 06 | 0.037 | 0.041 | 0.044 | 0.036 | 0.004 | -0.023 | -0.041 | -0.053 | -0.030 | -0.014 | -0.011 | -0.014 | -0.011 | -0.014 | -0.013 |
| 07 | 0.023 | 0.028 | 0.040 | 0.053 | 0.014 | -0.026 | -0.043 | -0.045 | -0.027 | 0.009 | -0.021 | -0.010 | -0.002 | -0.006 | -0.016 |
| 08 | 0.008 | 0.026 | 0.023 | 0.027 | 0.007 | -0.021 | -0.034 | -0.045 | -0.037 | -0.047 | -0.027 | -0.036 | 0.019 | -0.018 | -0.007 |
| 09 | 0.027 | 0.047 | 0.035 | 0.022 | 0.021 | 0.002 | -0.016 | -0.034 | -0.054 | -0.042 | -0.023 | -0.015 | -0.009 | -0.001 | 0.005 |
| 10 | 0.020 | 0.027 | 0.046 | 0.013 | 0.036 | 0.008 | -0.000 | -0.013 | -0.030 | -0.031 | -0.016 | -0.010 | -0.010 | -0.007 | -0.007 |
| 11 | 0.001 | 0.000 | 0.010 | 0.002 | -0.019 | -0.006 | -0.005 | 0.005 | 0.005 | -0.015 | -0.016 | -0.000 | 0.008 | -0.019 | -0.018 |
| 12 | | 0.024 | 0.049 | 0.007 | -0.015 | -0.025 | -0.005 | 0.009 | 0.009 | -0.015 | 0.006 | 0.026 | 0.013 | -0.013 | |
| 13 | | 0.030 | 0.049 | 0.018 | 0.002 | 0.005 | 0.019 | 0.013 | -0.003 | -0.026 | 0.006 | 0.023 | 0.010 | -0.008 | |
| 14 | | | 0.003 | 0.012 | 0.002 | 0.024 | 0.057 | 0.023 | -0.007 | -0.025 | -0.004 | 0.018 | 0.007 | | |
| 15 | | | | | 0.001 | 0.024 | 0.029 | 0.001 | -0.010 | -0.025 | -0.004 | | | | |

## FIGURE 4.7.22 - 50% POWER, ROD D-8 DROPPED

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.4780
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.6607
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD BOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 260 | 05-K | 1.6021 | 1.0000 | 1.6607 | 1.6021 | 0.0585 | 0.0365 | |
| 2 | 251 | 12-L | 1.5998 | 1.0000 | 1.6607 | 1.5998 | 0.0609 | 0.0391 | |
| 3 | 256 | 12-K | 1.5966 | 1.0000 | 1.6607 | 1.5966 | 0.0641 | 0.0402 | |
| 4 | 261 | 05-K | 1.5959 | 1.0000 | 1.6607 | 1.5959 | 0.0648 | 0.0406 | |
| 5 | 207 | 09-P | 1.5838 | 1.0000 | 1.6607 | 1.5838 | 0.0768 | 0.0485 | |
| 6 | 244 | 06-L | 1.5785 | 1.0000 | 1.6607 | 1.5785 | 0.0821 | 0.0520 | |
| 7 | 242 | 04-L | 1.5761 | 1.0000 | 1.6607 | 1.5761 | 0.0846 | 0.0536 | |
| 8 | 258 | 03-K | 1.5623 | 1.0000 | 1.6607 | 1.5623 | 0.0984 | 0.0630 | |
| 9 | 220 | 10-K | 1.5479 | 1.0000 | 1.6607 | 1.5479 | 0.1128 | 0.0729 | |
| 10 | 227 | 04-P | 1.5461 | 1.0000 | 1.6607 | 1.5461 | 0.1146 | 0.0741 | |
| 11 | 228 | 05-P | 1.5454 | 1.0000 | 1.6607 | 1.5454 | 0.1153 | 0.0746 | |
| 12 | 235 | 11-P | 1.5431 | 1.0000 | 1.6607 | 1.5431 | 0.1175 | 0.0762 | |
| 13 | 263 | 11-K | 1.5384 | 1.0000 | 1.6607 | 1.5384 | 0.1223 | 0.0795 | |
| 14 | 240 | 02-L | 1.5354 | 1.0000 | 1.6607 | 1.5354 | 0.1253 | 0.0816 | |
| 15 | 259 | 11-K | 1.5321 | 1.0000 | 1.6607 | 1.5321 | 0.1286 | 0.0839 | |
| 16 | 230 | 07-M | 1.5244 | 1.0000 | 1.6607 | 1.5244 | 0.1363 | 0.0894 | |
| 17 | 276 | 02-J | 1.5232 | 1.0000 | 1.6607 | 1.5232 | 0.1375 | 0.0903 | |
| 18 | 271 | 13-K | 1.5088 | 1.0000 | 1.6607 | 1.5088 | 0.1519 | 0.1007 | |
| 19 | 250 | 11-L | 1.5000 | 1.0000 | 1.6607 | 1.5000 | 0.1607 | 0.1071 | |
| 20 | 203 | 05-P | 1.4965 | 1.0000 | 1.6607 | 1.4965 | 0.1642 | 0.1097 | |

## FIGURE 4.7.23 - 50% POWER, ROD D-8 DROPPED

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.4780
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 4.6400
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | BUFNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC MARGIN DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 236 | 12-M | 20 | 2.1865 | 0.9839 | 1.0000 | 4.5655 | 2.3647 | 2.2008 | 0.9307 | |
| 2 | 236 | 12-M | 21 | 2.1848 | 0.9833 | 1.0000 | 4.5625 | 2.3628 | 2.1996 | 0.9309 | |
| 3 | 251 | 12-L | 20 | 2.1830 | 0.9839 | 1.0000 | 4.5655 | 2.3609 | 2.2046 | 0.9338 | |
| 4 | 260 | 05-K | 26 | 2.1729 | 0.9807 | 1.0000 | 4.5503 | 2.3500 | 2.2003 | 0.9355 | |
| 5 | 251 | 12-L | 21 | 2.1772 | 0.9833 | 1.0000 | 4.5625 | 2.3546 | 2.2078 | 0.9377 | |
| 6 | 260 | 05-K | 20 | 2.1779 | 0.9839 | 1.0000 | 4.5655 | 2.3554 | 2.2101 | 0.9383 | |
| 7 | 260 | 05-K | 21 | 2.1754 | 0.9833 | 1.0000 | 4.5625 | 2.3527 | 2.2098 | 0.9382 | |
| 8 | 236 | 12-M | 19 | 2.1736 | 0.9832 | 1.0000 | 4.5620 | 2.3508 | 2.2112 | 0.9406 | |
| 9 | 251 | 12-L | 19 | 2.1713 | 0.9832 | 1.0000 | 4.5620 | 2.3482 | 2.2138 | 0.9427 | |
| 10 | 261 | 05-K | 26 | 2.1545 | 0.9807 | 1.0000 | 4.5503 | 2.3409 | 2.2094 | 0.9438 | |
| 11 | 251 | 12-L | 26 | 2.1674 | 0.9807 | 1.0000 | 4.5503 | 2.3397 | 2.2106 | 0.9448 | |
| 12 | 260 | 05-K | 25 | 2.1704 | 0.9839 | 1.0000 | 4.5652 | 2.3473 | 2.2179 | 0.9449 | |
| 13 | 207 | 09-P | 21 | 2.1691 | 0.9833 | 1.0000 | 4.5625 | 2.3459 | 2.2166 | 0.9449 | |
| 14 | 261 | 05-K | 20 | 2.1694 | 0.9839 | 1.0000 | 4.5655 | 2.3462 | 2.2193 | 0.9459 | |
| 15 | 260 | 05-K | 27 | 2.1559 | 0.9779 | 1.0000 | 4.5375 | 2.3316 | 2.2059 | 0.9461 | |
| 16 | 261 | 05-K | 21 | 2.1670 | 0.9833 | 1.0000 | 4.5625 | 2.3436 | 2.2189 | 0.9483 | |
| 17 | 207 | 09-P | 20 | 2.1667 | 0.9839 | 1.0000 | 4.5655 | 2.3433 | 2.2222 | 0.9483 | |
| 18 | 260 | 05-K | 19 | 2.1616 | 0.9832 | 1.0000 | 4.5620 | 2.3378 | 2.2243 | 0.9515 | |
| 19 | 261 | 05-K | 25 | 2.1620 | 0.9839 | 1.0000 | 4.5652 | 2.3382 | 2.2270 | 0.9524 | |
| 20 | 251 | 12-L | 25 | 2.1616 | 0.9839 | 1.0000 | 4.5652 | 2.3378 | 2.2274 | 0.9523 | |

FIGURE 4.7.24 - 50% POWER, ROD D-8 DROPPED

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, ((CALC.-MEAS.)/MEAS.)

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.164 | -0.164 | -0.152 | -0.139 | -0.065 | -0.004 | -0.004 | | | | |
| 02 | | | | -0.078 | -0.078 | -0.164 | -0.164 | -0.122 | -0.139 | -0.051 | 0.009 | 0.045 | 0.125 | 0.155 | |
| 03 | | | -0.144 | -0.132 | -0.111 | -0.164 | -0.164 | -0.102 | -0.025 | -0.053 | 0.022 | 0.071 | 0.125 | 0.155 | 0.187 |
| 04 | | | -0.188 | -0.142 | -0.146 | -0.150 | -0.150 | -0.067 | -0.093 | -0.041 | 0.018 | 0.073 | 0.133 | 0.177 | 0.197 |
| 05 | | -0.146 | -0.142 | -0.139 | -0.133 | -0.150 | -0.150 | -0.101 | -0.063 | -0.038 | 0.016 | 0.075 | 0.139 | 0.210 | 0.260 | 0.313 |
| 06 | -0.114 | -0.106 | -0.098 | -0.113 | -0.137 | -0.150 | -0.103 | -0.056 | 0.025 | 0.117 | 0.159 | 0.249 | 0.343 | 0.313 | 0.313 |
| 07 | -0.110 | -0.104 | -0.098 | -0.099 | -0.103 | -0.089 | -0.097 | -0.058 | 0.159 | 0.301 | 0.580 | 1.034 | 0.549 | 0.501 | 0.380 |
| 08 | -0.105 | -0.120 | -0.096 | -0.089 | -0.072 | -0.051 | 0.008 | -0.020 | 0.157 | 0.215 | 0.553 | 2.593 | 0.885 | 0.456 | 0.395 |
| 09 | -0.131 | -0.156 | -0.127 | -0.079 | -0.070 | -0.036 | 0.002 | 0.012 | 0.028 | 0.182 | 0.348 | 0.733 | 0.795 | 0.496 | 0.339 |
| 10 | -0.141 | -0.145 | -0.156 | -0.114 | -0.063 | -0.077 | -0.005 | 0.001 | 0.019 | 0.142 | 0.251 | 0.343 | 0.343 | 0.349 | 0.349 |
| 11 | -0.129 | -0.129 | -0.161 | -0.132 | -0.172 | -0.114 | -0.045 | 0.005 | 0.005 | 0.096 | 0.096 | 0.153 | 0.183 | 0.360 | 0.360 |
| 12 | | -0.156 | -0.181 | -0.172 | -0.162 | -0.152 | -0.074 | -0.011 | -0.011 | 0.096 | 0.076 | 0.057 | 0.110 | 0.257 | |
| 13 | | -0.142 | -0.181 | -0.150 | -0.154 | -0.134 | -0.074 | -0.027 | -0.051 | -0.094 | -0.007 | 0.037 | 0.097 | 0.194 | |
| 14 | | | -0.079 | -0.127 | -0.131 | -0.102 | -0.058 | -0.041 | -0.054 | -0.094 | -0.036 | 0.018 | 0.084 | | |
| 15 | | | | | -0.094 | -0.080 | -0.048 | -0.037 | -0.057 | -0.034 | -0.036 | | | | |

## FIGURE 4.7.25 - 75% POWER, ΔI WITHIN TARGET BAND

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7435
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.5809
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD BOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | TECH SPEC MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 214 | 08-D | 1.3540 | 1.0000 | 1.5809 | 1.3540 | 0.2269 | 0.1676 | |
| 2 | 215 | 09-D | 1.3523 | 1.0000 | 1.5809 | 1.3523 | 0.2285 | 0.1690 | |
| 3 | 216 | 04-H | 1.3483 | 1.0000 | 1.5809 | 1.3483 | 0.2325 | 0.1725 | |
| 4 | 210 | 05-H | 1.3483 | 1.0000 | 1.5809 | 1.3483 | 0.2326 | 0.1725 | |
| 5 | 217 | 04-K | 1.3467 | 1.0000 | 1.5809 | 1.3467 | 0.2342 | 0.1739 | |
| 6 | 277 | 07-D | 1.3397 | 1.0000 | 1.5809 | 1.3397 | 0.2412 | 0.1800 | |
| 7 | 452 | 12-D | 1.3384 | 1.0000 | 1.5809 | 1.3384 | 0.2424 | 0.1811 | |
| 8 | 454 | 12-B | 1.3368 | 1.0000 | 1.5809 | 1.3368 | 0.2441 | 0.1826 | |
| 9 | 455 | 04-D | 1.3361 | 1.0000 | 1.5809 | 1.3361 | 0.2448 | 0.1832 | |
| 10 | 293 | 07-E | 1.3350 | 1.0000 | 1.5809 | 1.3350 | 0.2459 | 0.1842 | |
| 11 | 461 | 04-D | 1.3344 | 1.0000 | 1.5809 | 1.3344 | 0.2464 | 0.1847 | |
| 12 | 209 | 03-E | 1.3342 | 1.0000 | 1.5809 | 1.3342 | 0.2466 | 0.1848 | |
| 13 | 273 | 09-D | 1.3320 | 1.0000 | 1.5809 | 1.3320 | 0.2489 | 0.1868 | |
| 14 | 456 | 12-M | 1.3279 | 1.0000 | 1.5809 | 1.3279 | 0.2529 | 0.1905 | |
| 15 | 270 | 04-J | 1.3278 | 1.0000 | 1.5809 | 1.3278 | 0.2531 | 0.1906 | |
| 16 | 462 | 12-P | 1.3263 | 1.0000 | 1.5809 | 1.3263 | 0.2546 | 0.1920 | |
| 17 | 262 | 05-J | 1.3246 | 1.0000 | 1.5809 | 1.3246 | 0.2562 | 0.1934 | |
| 18 | 274 | 04-G | 1.3245 | 1.0000 | 1.5809 | 1.3245 | 0.2564 | 0.1936 | |
| 19 | 318 | 05-F | 1.3244 | 1.0000 | 1.5809 | 1.3244 | 0.2564 | 0.1936 | |
| 20 | 265 | 05-J | 1.3241 | 1.0000 | 1.5809 | 1.3241 | 0.2568 | 0.1939 | |

FIGURE 4.7.26 - 75% POWER, ΔI WITHIN TARGET BAND

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBGU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7435
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 3.1205
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 214 | 08-D | 20 | 1.8455 | 1.0000 | 1.0000 | 3.1205 | 1.9959 | 1.1246 | 0.5634 | |
| 2 | 210 | 05-H | 20 | 1.8439 | 1.0000 | 1.0000 | 3.1205 | 1.9942 | 1.1263 | 0.5648 | |
| 3 | 215 | 08-D | 20 | 1.8433 | 1.0000 | 1.0000 | 3.1205 | 1.9935 | 1.1269 | 0.5653 | |
| 4 | 452 | 12-D | 20 | 1.8431 | 1.0000 | 1.0000 | 3.1205 | 1.9934 | 1.1271 | 0.5654 | |
| 5 | 216 | 04-H | 20 | 1.8431 | 1.0000 | 1.0000 | 3.1205 | 1.9933 | 1.1271 | 0.5655 | |
| 6 | 455 | 04-D | 20 | 1.8416 | 1.0000 | 1.0000 | 3.1205 | 1.9917 | 1.1287 | 0.5667 | |
| 7 | 217 | 04-H | 20 | 1.8409 | 1.0000 | 1.0000 | 3.1205 | 1.9910 | 1.1295 | 0.5673 | |
| 8 | 454 | 12-D | 20 | 1.8405 | 1.0000 | 1.0000 | 3.1205 | 1.9905 | 1.1299 | 0.5676 | |
| 9 | 210 | 05-H | 21 | 1.8395 | 1.0000 | 1.0000 | 3.1205 | 1.9894 | 1.1311 | 0.5685 | |
| 10 | 216 | 04-H | 21 | 1.8395 | 1.0000 | 1.0000 | 3.1205 | 1.9894 | 1.1311 | 0.5686 | |
| 11 | 461 | 04-D | 20 | 1.8390 | 1.0000 | 1.0000 | 3.1205 | 1.9889 | 1.1315 | 0.5689 | |
| 12 | 214 | 08-D | 21 | 1.8382 | 1.0000 | 1.0000 | 3.1205 | 1.9881 | 1.1324 | 0.5696 | |
| 13 | 217 | 04-H | 21 | 1.8373 | 1.0000 | 1.0000 | 3.1205 | 1.9870 | 1.1335 | 0.5704 | |
| 14 | 215 | 08-D | 21 | 1.8361 | 1.0000 | 1.0000 | 3.1205 | 1.9857 | 1.1348 | 0.5715 | |
| 15 | 214 | 08-D | 19 | 1.8334 | 1.0000 | 1.0000 | 3.1205 | 1.9828 | 1.1377 | 0.5738 | |
| 16 | 455 | 04-D | 21 | 1.8332 | 1.0000 | 1.0000 | 3.1205 | 1.9826 | 1.1378 | 0.5739 | |
| 17 | 210 | 05-H | 19 | 1.8322 | 1.0000 | 1.0000 | 3.1205 | 1.9816 | 1.1389 | 0.5747 | |
| 18 | 216 | 04-H | 19 | 1.8321 | 1.0000 | 1.0000 | 3.1205 | 1.9814 | 1.1390 | 0.5749 | |
| 19 | 455 | 04-D | 19 | 1.8315 | 1.0000 | 1.0000 | 3.1205 | 1.9807 | 1.1397 | 0.5754 | |
| 20 | 215 | 08-D | 19 | 1.8312 | 1.0000 | 1.0000 | 3.1205 | 1.9805 | 1.1400 | 0.5756 | |

FIGURE 4.7.27 - 75% POWER, ΔI WITHIN TARGET BAND

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | 0.003 | 0.003 | -0.015 | -0.032 | 0.004 | 0.031 | 0.031 | | | | |
| 02 | | | 0.009 | 0.010 | 0.002 | 0.003 | -0.006 | -0.032 | 0.003 | 0.020 | 0.013 | -0.001 | 0.001 | | |
| 03 | | 0.006 | 0.007 | 0.008 | 0.001 | 0.003 | -0.007 | 0.020 | -0.004 | 0.010 | 0.004 | -0.002 | 0.000 | 0.002 | |
| 04 | | 0.003 | 0.008 | 0.005 | 0.008 | 0.010 | -0.009 | -0.026 | -0.005 | -0.005 | -0.006 | -0.019 | -0.014 | -0.011 | |
| 05 | 0.032 | 0.020 | 0.013 | 0.013 | 0.009 | 0.005 | -0.019 | -0.023 | -0.012 | -0.015 | -0.018 | -0.025 | -0.031 | -0.025 | -0.019 |
| 06 | 0.053 | 0.039 | 0.025 | 0.024 | 0.012 | 0.000 | -0.015 | -0.031 | -0.008 | 0.005 | 0.003 | -0.023 | -0.024 | -0.018 | -0.018 |
| 07 | 0.035 | 0.030 | 0.028 | 0.034 | 0.019 | -0.001 | -0.016 | -0.018 | -0.007 | 0.034 | -0.006 | -0.017 | -0.022 | -0.014 | -0.019 |
| 08 | 0.018 | 0.022 | 0.024 | 0.025 | 0.017 | 0.002 | -0.001 | -0.015 | -0.013 | -0.034 | -0.013 | -0.030 | 0.013 | -0.020 | -0.006 |
| 09 | 0.022 | 0.025 | 0.024 | 0.021 | 0.019 | 0.015 | 0.011 | -0.003 | -0.020 | -0.020 | -0.005 | -0.011 | -0.011 | -0.001 | 0.009 |
| 10 | 0.023 | 0.023 | 0.024 | 0.008 | 0.021 | 0.014 | 0.024 | 0.010 | -0.004 | -0.012 | -0.010 | -0.014 | -0.014 | 0.009 | 0.010 |
| 11 | 0.021 | 0.020 | 0.002 | 0.002 | -0.009 | 0.001 | 0.011 | 0.020 | 0.020 | -0.012 | -0.012 | -0.015 | -0.016 | 0.010 | 0.010 |
| 12 | | 0.008 | -0.005 | -0.013 | -0.015 | -0.020 | -0.001 | 0.012 | 0.012 | -0.012 | -0.015 | -0.020 | -0.011 | 0.004 | |
| 13 | | 0.005 | -0.005 | -0.003 | -0.013 | -0.005 | 0.006 | 0.005 | -0.006 | -0.020 | -0.013 | -0.010 | -0.006 | 0.000 | |
| 14 | | | 0.017 | 0.005 | 0.002 | 0.012 | 0.026 | 0.014 | -0.004 | -0.020 | -0.010 | -0.001 | -0.001 | | |

## FIGURE 4.7.28 - 75% POWER, ΔI AT 50% NEGATIVE VALUE

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0600

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7678
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.5735
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD EOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | TECH SPEC MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 214 | 08-D | 1.3818 | 1.0000 | 1.5735 | 1.3818 | 0.1917 | 0.1387 | |
| 2 | 215 | 08-D | 1.3796 | 1.0000 | 1.5735 | 1.3796 | 0.1939 | 0.1406 | |
| 3 | 216 | 04-H | 1.3729 | 1.0000 | 1.5735 | 1.3729 | 0.2006 | 0.1461 | |
| 4 | 210 | 05-H | 1.3715 | 1.0000 | 1.5735 | 1.3715 | 0.2020 | 0.1473 | |
| 5 | 217 | 04-H | 1.3707 | 1.0000 | 1.5735 | 1.3707 | 0.2028 | 0.1480 | |
| 6 | 209 | 08-E | 1.3633 | 1.0000 | 1.5735 | 1.3633 | 0.2102 | 0.1542 | |
| 7 | 277 | 07-D | 1.3586 | 1.0000 | 1.5735 | 1.3586 | 0.2149 | 0.1582 | |
| 8 | 273 | 09-D | 1.3558 | 1.0000 | 1.5735 | 1.3558 | 0.2178 | 0.1606 | |
| 9 | 293 | 07-E | 1.3540 | 1.0000 | 1.5735 | 1.3540 | 0.2195 | 0.1621 | |
| 10 | 276 | 04-J | 1.3526 | 1.0000 | 1.5735 | 1.3526 | 0.2209 | 0.1633 | |
| 11 | 292 | 02-J | 1.3513 | 1.0000 | 1.5735 | 1.3513 | 0.2223 | 0.1645 | |
| 12 | 275 | 12-J | 1.3499 | 1.0000 | 1.5735 | 1.3499 | 0.2236 | 0.1656 | |
| 13 | 274 | 04-G | 1.3491 | 1.0000 | 1.5735 | 1.3491 | 0.2244 | 0.1663 | |
| 14 | 265 | 05-J | 1.3443 | 1.0000 | 1.5735 | 1.3443 | 0.2292 | 0.1705 | |
| 15 | 262 | 05-J | 1.3437 | 1.0000 | 1.5735 | 1.3437 | 0.2299 | 0.1711 | |
| 16 | 318 | 05-F | 1.3437 | 1.0000 | 1.5735 | 1.3437 | 0.2299 | 0.1711 | |
| 17 | 287 | 09-E | 1.3425 | 1.0000 | 1.5735 | 1.3425 | 0.2310 | 0.1721 | |
| 18 | 263 | 05-G | 1.3418 | 1.0000 | 1.5735 | 1.3418 | 0.2318 | 0.1727 | |
| 19 | 260 | 05-G | 1.3411 | 1.0000 | 1.5735 | 1.3411 | 0.2324 | 0.1733 | |
| 20 | 288 | 14-G | 1.3405 | 1.0000 | 1.5735 | 1.3405 | 0.2330 | 0.1738 | |

FIGURE 4.7.29 - 75% POWER, ΔI AT 50% NEGATIVE VALUE

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFACR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7678
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 2.0216
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | FQRNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 452 | 12-D | 19 | 1.9884 | 1.0000 | 1.0000 | 2.0216 | 2.1505 | 0.8711 | 0.4051 | |
| 2 | 452 | 12-D | 19 | 1.9884 | 1.0000 | 1.0000 | 2.0216 | 2.1505 | 0.8711 | 0.4051 | |
| 3 | 210 | 05-H | 20 | 1.9873 | 1.0000 | 1.0000 | 2.0216 | 2.1492 | 0.8723 | 0.4059 | |
| 4 | 210 | 05-H | 19 | 1.9869 | 1.0000 | 1.0000 | 2.0216 | 2.1488 | 0.8728 | 0.4062 | |
| 5 | 214 | 08-D | 20 | 1.9860 | 1.0000 | 1.0000 | 2.0216 | 2.1479 | 0.8737 | 0.4068 | |
| 6 | 454 | 12-D | 19 | 1.9857 | 1.0000 | 1.0000 | 2.0216 | 2.1475 | 0.8741 | 0.4070 | |
| 7 | 454 | 12-D | 19 | 1.9857 | 1.0000 | 1.0000 | 2.0216 | 2.1475 | 0.8741 | 0.4070 | |
| 8 | 215 | 08-D | 20 | 1.9857 | 1.0000 | 1.0000 | 2.0216 | 2.1474 | 0.8742 | 0.4084 | |
| 9 | 214 | 08-D | 19 | 1.9857 | 1.0000 | 1.0000 | 2.0216 | 2.1473 | 0.8742 | 0.4084 | |
| 10 | 215 | 08-D | 19 | 1.9814 | 1.0000 | 1.0000 | 2.0216 | 2.1429 | 0.8787 | 0.4101 | |
| 11 | 209 | 05-E | 19 | 1.9809 | 1.0000 | 1.0000 | 2.0216 | 2.1423 | 0.8792 | 0.4104 | |
| 12 | 216 | 04-H | 20 | 1.9805 | 1.0000 | 1.0000 | 2.0216 | 2.1419 | 0.8797 | 0.4107 | |
| 13 | 217 | 04-H | 20 | 1.9782 | 1.0000 | 1.0000 | 2.0216 | 2.1394 | 0.8822 | 0.4123 | |
| 14 | 209 | 08-E | 20 | 1.9781 | 1.0000 | 1.0000 | 2.0216 | 2.1393 | 0.8823 | 0.4124 | |
| 15 | 216 | 04-H | 19 | 1.9769 | 1.0000 | 1.0000 | 2.0216 | 2.1380 | 0.8835 | 0.4133 | |
| 16 | 455 | 04-D | 19 | 1.9755 | 1.0000 | 1.0000 | 2.0216 | 2.1366 | 0.8850 | 0.4142 | |
| 17 | 217 | 04-H | 19 | 1.9746 | 1.0000 | 1.0000 | 2.0216 | 2.1356 | 0.8860 | 0.4149 | |
| 18 | 461 | 04-D | 19 | 1.9726 | 1.0000 | 1.0000 | 2.0216 | 2.1336 | 0.8880 | 0.4162 | |
| 19 | 210 | 05-H | 18 | 1.9712 | 1.0000 | 1.0000 | 2.0216 | 2.1319 | 0.8897 | 0.4173 | |
| 20 | 455 | 04-D | 20 | 1.9697 | 1.0000 | 1.0000 | 2.0216 | 2.1302 | 0.8914 | 0.4185 | |

## FIGURE 4.7.30 - 75% POWER, ΔI AT 50% NEGATIVE VALUE

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | 0.004 | 0.004 | -0.015 | -0.034 | 0.004 | 0.033 | 0.032 | | | | |
| 02 | | | 0.016 | 0.016 | 0.003 | 0.004 | -0.009 | -0.034 | 0.001 | 0.021 | 0.017 | 0.010 | 0.008 | | |
| 03 | | 0.016 | 0.017 | 0.019 | 0.005 | 0.004 | -0.010 | 0.011 | -0.007 | 0.010 | 0.009 | 0.009 | 0.008 | 0.007 | |
| 04 | | 0.011 | 0.023 | 0.011 | 0.001 | 0.002 | -0.013 | -0.029 | -0.010 | -0.003 | -0.004 | -0.012 | -0.004 | -0.004 | |
| 05 | 0.036 | 0.027 | 0.020 | 0.013 | 0.001 | -0.003 | -0.022 | -0.026 | -0.020 | -0.021 | -0.021 | -0.021 | -0.020 | -0.016 | -0.011 |
| 06 | 0.055 | 0.038 | 0.022 | 0.017 | 0.004 | -0.007 | -0.016 | -0.030 | -0.013 | -0.003 | -0.003 | -0.020 | -0.017 | -0.011 | -0.011 |
| 07 | 0.034 | 0.028 | 0.023 | 0.024 | 0.010 | -0.005 | -0.012 | -0.019 | -0.012 | 0.024 | -0.012 | -0.017 | -0.020 | -0.010 | -0.014 |
| 08 | 0.015 | 0.022 | 0.022 | 0.018 | 0.007 | -0.002 | -0.004 | -0.015 | -0.017 | -0.038 | -0.020 | -0.035 | 0.014 | -0.018 | -0.007 |
| 09 | 0.023 | 0.031 | 0.026 | 0.017 | 0.014 | 0.011 | 0.007 | -0.005 | -0.020 | -0.026 | -0.018 | -0.015 | -0.010 | -0.002 | 0.003 |
| 10 | 0.025 | 0.027 | 0.030 | 0.007 | 0.022 | 0.011 | 0.018 | 0.002 | -0.011 | -0.017 | -0.014 | -0.012 | -0.012 | 0.010 | 0.010 |
| 11 | 0.022 | 0.021 | 0.004 | -0.001 | -0.015 | -0.006 | -0.001 | 0.002 | 0.002 | -0.012 | -0.013 | -0.006 | -0.001 | 0.017 | 0.018 |
| 12 | | 0.014 | 0.007 | -0.009 | -0.020 | -0.032 | -0.011 | 0.004 | 0.004 | -0.012 | -0.003 | -0.007 | 0.002 | 0.006 | |
| 13 | | 0.018 | 0.007 | 0.006 | -0.010 | -0.002 | 0.005 | 0.006 | -0.001 | -0.012 | -0.001 | 0.002 | 0.001 | -0.001 | |
| 14 | | | 0.033 | 0.014 | 0.006 | 0.017 | 0.032 | 0.016 | -0.001 | -0.012 | -0.005 | 0.001 | 0.000 | | |
| 15 | | | | | 0.022 | 0.026 | 0.020 | 0.009 | 0.000 | -0.012 | -0.005 | | | | |

## FIGURE 4.7.31 - 75% POWER, ΔI AT 90% NEGATIVE VALUE

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7552
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.5773
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | COD ECW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | TECH SPEC MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 214 | 08-D | 1.3763 | 1.0000 | 1.5773 | 1.3763 | 0.2010 | 0.1461 | |
| 2 | 215 | 08-D | 1.3742 | 1.0000 | 1.5773 | 1.3742 | 0.2031 | 0.1478 | |
| 3 | 210 | 05-H | 1.3715 | 1.0000 | 1.5773 | 1.3715 | 0.2059 | 0.1501 | |
| 4 | 216 | 04-H | 1.3707 | 1.0000 | 1.5773 | 1.3707 | 0.2066 | 0.1508 | |
| 5 | 217 | 04-H | 1.3686 | 1.0000 | 1.5773 | 1.3686 | 0.2087 | 0.1525 | |
| 6 | 277 | 07-D | 1.3586 | 1.0000 | 1.5773 | 1.3586 | 0.2187 | 0.1610 | |
| 7 | 209 | 03-E | 1.3573 | 1.0000 | 1.5773 | 1.3573 | 0.2200 | 0.1621 | |
| 8 | 293 | 07-B | 1.3557 | 1.0000 | 1.5773 | 1.3557 | 0.2217 | 0.1635 | |
| 9 | 273 | 09-D | 1.3538 | 1.0000 | 1.5773 | 1.3538 | 0.2235 | 0.1651 | |
| 10 | 292 | 02-J | 1.3510 | 1.0000 | 1.5773 | 1.3510 | 0.2263 | 0.1675 | |
| 11 | 276 | 04-J | 1.3502 | 1.0000 | 1.5773 | 1.3502 | 0.2271 | 0.1682 | |
| 12 | 274 | 04-G | 1.3486 | 1.0000 | 1.5773 | 1.3486 | 0.2288 | 0.1696 | |
| 13 | 263 | 05-G | 1.3448 | 1.0000 | 1.5773 | 1.3448 | 0.2325 | 0.1729 | |
| 14 | 275 | 12-J | 1.3445 | 1.0000 | 1.5773 | 1.3445 | 0.2328 | 0.1732 | |
| 15 | 260 | 05-G | 1.3445 | 1.0000 | 1.5773 | 1.3445 | 0.2329 | 0.1732 | |
| 16 | 318 | 05-F | 1.3423 | 1.0000 | 1.5773 | 1.3423 | 0.2350 | 0.1751 | |
| 17 | 265 | 05-J | 1.3422 | 1.0000 | 1.5773 | 1.3422 | 0.2351 | 0.1752 | |
| 18 | 289 | 09-B | 1.3422 | 1.0000 | 1.5773 | 1.3422 | 0.2352 | 0.1752 | |
| 19 | 262 | 05-J | 1.3418 | 1.0000 | 1.5773 | 1.3413 | 0.2355 | 0.1755 | |
| 20 | 288 | 14-G | 1.3381 | 1.0000 | 1.5773 | 1.3381 | 0.2392 | 0.1788 | |

FIGURE 4.7.32 - 75% POWER, ΔI AT 90% NEGATIVE VALUE

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBGU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7552
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 3.0720
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | FUFNUF FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 210 | 05-F | 19 | 2.0437 | 1.0000 | 1.0000 | 3.0720 | 2.2103 | 0.8617 | 0.3899 | |
| 2 | 210 | 05-F | 20 | 2.0382 | 1.0000 | 1.0000 | 3.0720 | 2.2043 | 0.8677 | 0.3937 | |
| 3 | 214 | 08-B | 19 | 2.0381 | 1.0000 | 1.0000 | 3.0720 | 2.2042 | 0.8678 | 0.3937 | |
| 4 | 215 | 08-D | 19 | 2.0357 | 1.0000 | 1.0000 | 3.0720 | 2.2017 | 0.8704 | 0.3953 | |
| 5 | 455 | 04-D | 19 | 2.0352 | 1.0000 | 1.0000 | 3.0720 | 2.2011 | 0.8710 | 0.3957 | |
| 6 | 210 | 05-F | 18 | 2.0334 | 1.0000 | 1.0000 | 3.0720 | 2.1991 | 0.8729 | 0.3969 | |
| 7 | 216 | 04-H | 19 | 2.0329 | 1.0000 | 1.0000 | 3.0720 | 2.1986 | 0.8734 | 0.3973 | |
| 8 | 452 | 12-D | 19 | 2.0327 | 1.0000 | 1.0000 | 3.0720 | 2.1984 | 0.8737 | 0.3974 | |
| 9 | 461 | 04-D | 19 | 2.0324 | 1.0000 | 1.0000 | 3.0720 | 2.1980 | 0.8740 | 0.3976 | |
| 10 | 217 | 04-H | 19 | 2.0306 | 1.0000 | 1.0000 | 3.0720 | 2.1961 | 0.8759 | 0.3989 | |
| 11 | 209 | 08-E | 19 | 2.0305 | 1.0000 | 1.0000 | 3.0720 | 2.1960 | 0.8760 | 0.3989 | |
| 12 | 454 | 12-D | 19 | 2.0299 | 1.0000 | 1.0000 | 3.0720 | 2.1953 | 0.8767 | 0.3993 | |
| 13 | 214 | 08-D | 20 | 2.0296 | 1.0000 | 1.0000 | 3.0720 | 2.1939 | 0.8781 | 0.4002 | |
| 14 | 215 | 08-C | 20 | 2.0263 | 1.0000 | 1.0000 | 3.0720 | 2.1914 | 0.8806 | 0.4018 | |
| 15 | 455 | 04-D | 18 | 2.0262 | 1.0000 | 1.0000 | 3.0720 | 2.1914 | 0.8806 | 0.4019 | |
| 16 | 214 | 08-C | 18 | 2.0262 | 1.0000 | 1.0000 | 3.0720 | 2.1914 | 0.8806 | 0.4019 | |
| 17 | 216 | 04-H | 18 | 2.0257 | 1.0000 | 1.0000 | 3.0720 | 2.1908 | 0.8812 | 0.4022 | |
| 18 | 216 | 04-H | 20 | 2.0257 | 1.0000 | 1.0000 | 3.0720 | 2.1908 | 0.8812 | 0.4022 | |
| 19 | 215 | 08-D | 18 | 2.0239 | 1.0000 | 1.0000 | 3.0720 | 2.1889 | 0.8832 | 0.4035 | |
| 20 | 461 | 04-D | 18 | 2.0235 | 1.0000 | 1.0000 | 3.0720 | 2.1884 | 0.8837 | 0.4033 | |

## FIGURE 4.7.33 - 75% POWER, ΔI AT 90% NEGATIVE VALUE

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.003 | -0.002 | -0.020 | -0.037 | 0.002 | -0.031 | 0.030 | | | | |
| 02 | | | 0.020 | 0.021 | -0.003 | -0.002 | -0.012 | -0.037 | -0.001 | 0.019 | 0.020 | 0.021 | 0.014 | | |
| 03 | | 0.021 | 0.022 | 0.025 | -0.004 | -0.003 | -0.013 | 0.011 | -0.009 | 0.008 | 0.014 | 0.020 | 0.013 | 0.007 | |
| 04 | | 0.013 | 0.031 | 0.011 | -0.006 | -0.004 | -0.014 | -0.030 | -0.012 | -0.010 | -0.000 | -0.009 | -0.003 | -0.005 | |
| 05 | 0.031 | 0.028 | 0.025 | 0.015 | -0.005 | -0.005 | -0.023 | -0.029 | -0.025 | -0.021 | -0.018 | -0.021 | -0.022 | -0.019 | -0.016 |
| 06 | 0.044 | 0.036 | 0.029 | 0.017 | 0.002 | -0.004 | -0.014 | -0.032 | -0.015 | -0.002 | 0.000 | -0.021 | -0.021 | -0.014 | -0.015 |
| 07 | 0.027 | 0.023 | 0.020 | 0.022 | 0.008 | -0.001 | -0.003 | -0.013 | -0.011 | 0.027 | -0.011 | -0.020 | -0.026 | -0.015 | -0.019 |
| 08 | 0.010 | 0.014 | 0.009 | 0.012 | 0.005 | 0.003 | 0.002 | -0.009 | -0.015 | -0.039 | -0.019 | -0.034 | 0.011 | -0.023 | -0.011 |
| 09 | 0.017 | 0.024 | 0.016 | 0.009 | 0.009 | 0.010 | 0.010 | -0.001 | -0.015 | -0.023 | -0.014 | -0.016 | -0.014 | -0.005 | 0.002 |
| 10 | 0.025 | 0.024 | 0.023 | 0.005 | 0.013 | 0.010 | 0.019 | 0.002 | -0.009 | -0.011 | -0.013 | -0.019 | -0.019 | 0.004 | 0.005 |
| 11 | 0.025 | 0.025 | 0.012 | 0.000 | -0.007 | -0.002 | 0.000 | -0.002 | -0.002 | -0.006 | -0.007 | -0.001 | 0.004 | 0.007 | 0.007 |
| 12 | | 0.021 | 0.017 | -0.002 | -0.011 | -0.022 | -0.010 | -0.000 | -0.001 | -0.006 | 0.010 | 0.018 | 0.015 | 0.003 | |
| 13 | | 0.024 | 0.017 | 0.013 | -0.001 | -0.003 | 0.002 | 0.001 | -0.005 | -0.014 | 0.005 | 0.013 | 0.008 | 0.001 | |
| 14 | | | 0.033 | 0.021 | 0.015 | 0.017 | 0.021 | 0.010 | -0.003 | -0.013 | -0.006 | 0.002 | 0.001 | | |
| 15 | | | | | 0.030 | 0.026 | 0.015 | 0.010 | 0.000 | -0.013 | -0.005 | | | | |

## FIGURE 4.7.34 - 75% POWER, ΔI AT MOST POSITIVE VALUE

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7593
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.5761
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD BOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 214 | 08-D | 1.3605 | 1.0000 | 1.5761 | 1.3605 | 0.2156 | 0.1585 | |
| 2 | 215 | 08-D | 1.3586 | 1.0000 | 1.5761 | 1.3586 | 0.2175 | 0.1601 | |
| 3 | 209 | 08-E | 1.3449 | 1.0000 | 1.5761 | 1.3449 | 0.2312 | 0.1719 | |
| 4 | 273 | 09-D | 1.3409 | 1.0000 | 1.5761 | 1.3409 | 0.2352 | 0.1754 | |
| 5 | 210 | 05-H | 1.3398 | 1.0000 | 1.5761 | 1.3398 | 0.2363 | 0.1764 | |
| 6 | 277 | 07-D | 1.3390 | 1.0000 | 1.5761 | 1.3390 | 0.2371 | 0.1770 | |
| 7 | 318 | 05-F | 1.3350 | 1.0000 | 1.5761 | 1.3350 | 0.2411 | 0.1806 | |
| 8 | 293 | 07-B | 1.3344 | 1.0000 | 1.5761 | 1.3344 | 0.2417 | 0.1811 | |
| 9 | 263 | 05-G | 1.3316 | 1.0000 | 1.5761 | 1.3316 | 0.2444 | 0.1836 | |
| 10 | 260 | 03-G | 1.3315 | 1.0000 | 1.5761 | 1.3315 | 0.2446 | 0.1837 | |
| 11 | 292 | 02-J | 1.3314 | 1.0000 | 1.5761 | 1.3314 | 0.2447 | 0.1838 | |
| 12 | 317 | 10-E | 1.3277 | 1.0000 | 1.5761 | 1.3277 | 0.2484 | 0.1871 | |
| 13 | 320 | 05-K | 1.3270 | 1.0000 | 1.5761 | 1.3270 | 0.2490 | 0.1877 | |
| 14 | 454 | 12-C | 1.3265 | 1.0000 | 1.5761 | 1.3265 | 0.2496 | 0.1881 | |
| 15 | 452 | 12-D | 1.3245 | 1.0000 | 1.5761 | 1.3245 | 0.2516 | 0.1899 | |
| 16 | 396 | 12-E | 1.3237 | 1.0000 | 1.5761 | 1.3237 | 0.2524 | 0.1907 | |
| 17 | 316 | 11-F | 1.3221 | 1.0000 | 1.5761 | 1.3221 | 0.2540 | 0.1921 | |
| 18 | 259 | 09-E | 1.3210 | 1.0000 | 1.5761 | 1.3210 | 0.2551 | 0.1931 | |
| 19 | 275 | 12-J | 1.3209 | 1.0000 | 1.5761 | 1.3209 | 0.2552 | 0.1932 | |
| 20 | 258 | 09-E | 1.3208 | 1.0000 | 1.5761 | 1.3208 | 0.2553 | 0.1933 | |

## FIGURE 4.7.35 - 75% POWER, ΔI AT MOST POSITIVE VALUE

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBGE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBGU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.7593
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 2.6554
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | PENALTY FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC MARGIN DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 214 | 08-D | 33 | 1.8723 | 0.9750 | 1.0000 | 2.8790 | 2.0249 | 0.8541 | 0.4712 | |
| 2 | 214 | 08-D | 32 | 1.8770 | 0.9750 | 1.0000 | 2.8782 | 2.0300 | 0.8482 | 0.4720 | |
| 3 | 215 | 08-D | 27 | 1.8703 | 0.9750 | 1.0000 | 2.8790 | 2.0227 | 0.8563 | 0.4728 | |
| 4 | 215 | 08-D | 32 | 1.8749 | 0.9750 | 1.0000 | 2.8782 | 2.0278 | 0.8504 | 0.4736 | |
| 5 | 209 | 08-E | 33 | 1.8631 | 0.9750 | 1.0000 | 2.8790 | 2.0150 | 0.8640 | 0.4734 | |
| 6 | 209 | 08-E | 32 | 1.8672 | 0.9750 | 1.0000 | 2.8782 | 2.0194 | 0.8588 | 0.4797 | |
| 7 | 214 | 08-D | 28 | 1.8864 | 0.9810 | 1.0000 | 2.8279 | 2.0401 | 0.7878 | 0.4842 | |
| 8 | 215 | 08-D | 28 | 1.8843 | 0.9810 | 1.0000 | 2.8279 | 2.0378 | 0.7901 | 0.4858 | |
| 9 | 214 | 08-D | 34 | 1.8443 | 0.9720 | 1.0000 | 2.8699 | 1.9946 | 0.8752 | 0.4889 | |
| 10 | 215 | 08-D | 34 | 1.8423 | 0.9720 | 1.0000 | 2.8699 | 1.9924 | 0.8774 | 0.4905 | |
| 11 | 214 | 08-D | 29 | 1.8700 | 0.9870 | 1.0000 | 2.8157 | 2.0224 | 0.7933 | 0.4911 | |
| 12 | 452 | 12-D | 32 | 1.8527 | 0.9740 | 1.0000 | 2.8782 | 2.0037 | 0.8145 | 0.4913 | |
| 13 | 273 | 09-D | 33 | 1.8468 | 0.9750 | 1.0000 | 2.8790 | 1.9973 | 0.8817 | 0.4915 | |
| 14 | 273 | 09-D | 32 | 1.8515 | 0.9780 | 1.0000 | 2.8782 | 2.0024 | 0.8758 | 0.4923 | |
| 15 | 214 | 08-D | 27 | 1.8817 | 0.9940 | 1.0000 | 2.8371 | 2.0350 | 0.8020 | 0.4924 | |
| 16 | 215 | 08-D | 29 | 1.8690 | 0.9870 | 1.0000 | 2.8157 | 2.0202 | 0.7955 | 0.4923 | |
| 17 | 454 | 12-D | 32 | 1.8502 | 0.9720 | 1.0000 | 2.8782 | 2.0010 | 0.8772 | 0.4933 | |
| 18 | 209 | 08-E | 34 | 1.8381 | 0.9720 | 1.0000 | 2.8699 | 1.9879 | 0.8819 | 0.4939 | |
| 19 | 215 | 08-D | 27 | 1.8796 | 0.9940 | 1.0000 | 2.8371 | 2.0328 | 1.0043 | 0.4940 | |
| 20 | 452 | 12-D | 33 | 1.8428 | 0.9750 | 1.0000 | 2.8790 | 1.9930 | 0.8861 | 0.4948 | |

## FIGURE 4.7.36 - 75% POWER, ΔI AT MOST POSITIVE VALUE

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | 0.005 | 0.005 | -0.009 | -0.023 | 0.013 | 0.040 | 0.040 | | | | |
| 02 | | | -0.016 | -0.017 | 0.004 | 0.005 | -0.002 | -0.023 | 0.008 | 0.026 | 0.011 | -0.016 | -0.023 | | |
| 03 | | -0.006 | -0.007 | -0.008 | 0.004 | 0.005 | 0.006 | 0.020 | 0.008 | 0.012 | -0.003 | -0.015 | -0.023 | -0.030 | |
| 04 | | -0.002 | -0.001 | -0.004 | -0.013 | -0.013 | 0.019 | 0.020 | 0.007 | -0.005 | -0.009 | -0.020 | -0.027 | -0.032 | |
| 05 | 0.033 | 0.019 | 0.011 | 0.003 | -0.013 | -0.011 | -0.006 | -0.009 | -0.017 | -0.017 | -0.017 | -0.026 | -0.035 | -0.024 | -0.012 |
| 06 | 0.061 | 0.048 | 0.035 | 0.022 | 0.000 | -0.009 | -0.017 | -0.027 | -0.008 | 0.003 | 0.003 | -0.022 | -0.021 | -0.012 | -0.012 |
| 07 | 0.044 | 0.039 | 0.034 | 0.036 | 0.016 | -0.006 | -0.015 | -0.018 | -0.005 | 0.033 | -0.004 | -0.009 | -0.014 | -0.004 | -0.010 |
| 08 | 0.027 | 0.035 | 0.032 | 0.029 | 0.014 | -0.002 | -0.007 | -0.015 | -0.014 | -0.032 | -0.012 | -0.026 | 0.028 | -0.009 | 0.013 |
| 09 | 0.036 | 0.046 | 0.039 | 0.023 | 0.016 | 0.013 | 0.008 | -0.006 | -0.025 | -0.022 | -0.010 | -0.010 | -0.003 | 0.017 | 0.035 |
| 10 | 0.032 | 0.035 | 0.045 | 0.003 | 0.021 | 0.011 | 0.025 | 0.008 | -0.003 | -0.014 | -0.012 | -0.018 | -0.018 | 0.025 | 0.025 |
| 11 | 0.022 | 0.022 | -0.006 | -0.005 | -0.023 | -0.000 | 0.013 | 0.017 | 0.017 | -0.008 | -0.007 | -0.036 | -0.050 | 0.016 | 0.016 |
| 12 | | 0.003 | -0.013 | -0.018 | -0.018 | -0.010 | 0.006 | 0.017 | 0.017 | -0.003 | -0.043 | -0.062 | -0.060 | -0.019 | |
| 13 | | -0.003 | -0.014 | -0.009 | -0.021 | -0.002 | 0.018 | 0.017 | 0.011 | 0.002 | -0.031 | -0.043 | -0.042 | -0.041 | |
| 14 | | | 0.014 | 0.000 | -0.001 | 0.015 | 0.040 | 0.029 | 0.015 | 0.002 | -0.004 | -0.011 | -0.024 | | |
| 15 | | | | | 0.020 | 0.028 | 0.035 | 0.030 | 0.018 | 0.003 | -0.004 | | | | |

FIGURE 4.7.37 - 95% POWER, ARO

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.9959
COOLANT FLOW CORRECTION FACTOR = 0.0211
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0211

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.5530
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD BOW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 210 | 05-H | 1.3525 | 1.0000 | 1.5530 | 1.3525 | 0.2005 | 0.1483 | |
| 2 | 209 | 08-E | 1.3491 | 1.0000 | 1.5530 | 1.3491 | 0.2039 | 0.1512 | |
| 3 | 214 | 08-D | 1.3472 | 1.0000 | 1.5530 | 1.3472 | 0.2059 | 0.1529 | |
| 4 | 215 | 08-D | 1.3462 | 1.0000 | 1.5530 | 1.3462 | 0.2068 | 0.1536 | |
| 5 | 273 | 09-D | 1.3434 | 1.0000 | 1.5530 | 1.3434 | 0.2097 | 0.1561 | |
| 6 | 216 | 04-H | 1.3418 | 1.0000 | 1.5530 | 1.3418 | 0.2112 | 0.1574 | |
| 7 | 217 | 04-H | 1.3409 | 1.0000 | 1.5530 | 1.3409 | 0.2122 | 0.1582 | |
| 8 | 258 | 09-E | 1.3381 | 1.0000 | 1.5530 | 1.3381 | 0.2149 | 0.1606 | |
| 9 | 259 | 09-E | 1.3378 | 1.0000 | 1.5530 | 1.3378 | 0.2153 | 0.1609 | |
| 10 | 277 | 07-D | 1.3377 | 1.0000 | 1.5530 | 1.3377 | 0.2153 | 0.1609 | |
| 11 | 317 | 10-E | 1.3345 | 1.0000 | 1.5530 | 1.3345 | 0.2185 | 0.1639 | |
| 12 | 318 | 05-F | 1.3342 | 1.0000 | 1.5530 | 1.3342 | 0.2188 | 0.1640 | |
| 13 | 452 | 12-D | 1.3325 | 1.0000 | 1.5530 | 1.3325 | 0.2205 | 0.1655 | |
| 14 | 262 | 05-J | 1.3319 | 1.0000 | 1.5530 | 1.3319 | 0.2212 | 0.1660 | |
| 15 | 265 | 05-J | 1.3316 | 1.0000 | 1.5530 | 1.3316 | 0.2215 | 0.1663 | |
| 16 | 454 | 12-D | 1.3315 | 1.0000 | 1.5530 | 1.3315 | 0.2215 | 0.1664 | |
| 17 | 276 | 04-J | 1.3285 | 1.0000 | 1.5530 | 1.3285 | 0.2245 | 0.1690 | |
| 18 | 455 | 04-D | 1.3276 | 1.0000 | 1.5530 | 1.3276 | 0.2255 | 0.1699 | |
| 19 | 274 | 04-G | 1.3267 | 1.0000 | 1.5530 | 1.3267 | 0.2264 | 0.1706 | |
| 20 | 461 | 04-D | 1.3265 | 1.0000 | 1.5530 | 1.3265 | 0.2265 | 0.1708 | |

FIGURE 4.7.38 - 90% POWER, ARO

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBQE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.8959
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 2.5895
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 210 | 05-H | 20 | 1.8495 | 1.0000 | 1.0000 | 2.5895 | 2.0003 | 0.5892 | 0.2946 | |
| 2 | 209 | 08-E | 20 | 1.8460 | 1.0000 | 1.0000 | 2.5895 | 1.9965 | 0.5930 | 0.2970 | |
| 3 | 210 | 05-H | 21 | 1.8432 | 1.0000 | 1.0000 | 2.5895 | 1.9935 | 0.5960 | 0.2990 | |
| 4 | 210 | 05-H | 19 | 1.8393 | 1.0000 | 1.0000 | 2.5895 | 1.9892 | 0.6003 | 0.3013 | |
| 5 | 209 | 08-E | 21 | 1.8385 | 1.0000 | 1.0000 | 2.5895 | 1.9884 | 0.6012 | 0.3023 | |
| 6 | 214 | 08-D | 20 | 1.8382 | 1.0000 | 1.0000 | 2.5895 | 1.9881 | 0.6014 | 0.3025 | |
| 7 | 452 | 12-D | 20 | 1.8382 | 1.0000 | 1.0000 | 2.5895 | 1.9880 | 0.6015 | 0.3026 | |
| 8 | 215 | 08-D | 20 | 1.8370 | 1.0000 | 1.0000 | 2.5895 | 1.9867 | 0.6028 | 0.3034 | |
| 9 | 454 | 12-D | 20 | 1.8365 | 1.0000 | 1.0000 | 2.5895 | 1.9861 | 0.6034 | 0.3038 | |
| 10 | 209 | 08-E | 19 | 1.8359 | 1.0000 | 1.0000 | 2.5895 | 1.9855 | 0.6040 | 0.3042 | |
| 11 | 215 | 04-H | 20 | 1.8347 | 1.0000 | 1.0000 | 2.5895 | 1.9842 | 0.6053 | 0.3051 | |
| 12 | 273 | 09-D | 20 | 1.8341 | 1.0000 | 1.0000 | 2.5895 | 1.9835 | 0.6060 | 0.3055 | |
| 13 | 217 | 04-H | 20 | 1.8335 | 1.0000 | 1.0000 | 2.5895 | 1.9829 | 0.6066 | 0.3059 | |
| 14 | 258 | 09-E | 20 | 1.8332 | 1.0000 | 1.0000 | 2.5895 | 1.9826 | 0.6069 | 0.3061 | |
| 15 | 455 | 04-D | 20 | 1.8327 | 1.0000 | 1.0000 | 2.5895 | 1.9821 | 0.6074 | 0.3065 | |
| 16 | 259 | 09-E | 20 | 1.8327 | 1.0000 | 1.0000 | 2.5895 | 1.9820 | 0.6075 | 0.3065 | |
| 17 | 452 | 12-D | 19 | 1.8326 | 1.0000 | 1.0000 | 2.5895 | 1.9820 | 0.6075 | 0.3065 | |
| 18 | 461 | 04-D | 20 | 1.8310 | 1.0000 | 1.0000 | 2.5895 | 1.9802 | 0.6093 | 0.3077 | |
| 19 | 454 | 12-D | 19 | 1.8309 | 1.0000 | 1.0000 | 2.5895 | 1.9801 | 0.6094 | 0.3077 | |
| 20 | 258 | 09-E | 19 | 1.8306 | 1.0000 | 1.0000 | 2.5895 | 1.9801 | 0.6094 | 0.3078 | |

FIGURE 4.7.39 - 90% POWER, ARO

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | E | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.003 | -0.002 | -0.015 | -0.027 | 0.008 | 0.035 | 0.034 | | | | |
| 02 | | | 0.011 | 0.012 | -0.003 | -0.002 | -0.007 | -0.028 | 0.006 | 0.024 | 0.019 | 0.010 | 0.003 | | |
| 03 | | 0.013 | 0.014 | 0.015 | -0.004 | -0.003 | -0.006 | 0.018 | -0.001 | 0.014 | 0.011 | 0.009 | 0.003 | -0.003 | |
| 04 | | 0.011 | 0.019 | 0.006 | -0.006 | -0.005 | -0.005 | -0.020 | -0.004 | -0.006 | -0.005 | -0.014 | -0.009 | -0.011 | |
| 05 | 0.036 | 0.027 | 0.022 | 0.012 | -0.005 | -0.007 | -0.021 | -0.023 | -0.014 | -0.019 | -0.025 | -0.025 | -0.023 | -0.017 | -0.011 |
| 06 | 0.054 | 0.042 | 0.031 | 0.020 | 0.002 | -0.009 | -0.022 | -0.034 | -0.012 | -0.002 | -0.005 | -0.022 | -0.017 | -0.011 | -0.010 |
| 07 | 0.044 | 0.034 | 0.025 | 0.023 | 0.011 | -0.009 | -0.022 | -0.024 | -0.014 | 0.023 | -0.010 | -0.012 | -0.019 | -0.009 | -0.013 |
| 08 | 0.035 | 0.027 | 0.016 | 0.018 | 0.009 | -0.005 | -0.009 | -0.019 | -0.013 | -0.040 | -0.020 | -0.024 | 0.017 | -0.017 | -0.004 |
| 09 | 0.033 | 0.030 | 0.023 | 0.015 | 0.014 | 0.009 | 0.005 | -0.007 | -0.021 | -0.029 | -0.026 | -0.016 | -0.009 | 0.001 | 0.009 |
| 10 | 0.027 | 0.028 | 0.030 | 0.009 | 0.019 | 0.012 | 0.017 | 0.004 | -0.009 | -0.019 | -0.019 | -0.021 | -0.021 | 0.005 | 0.006 |
| 11 | 0.024 | 0.024 | 0.010 | 0.003 | -0.006 | -0.001 | 0.004 | 0.009 | 0.009 | -0.011 | -0.012 | -0.017 | -0.018 | 0.001 | 0.002 |
| 12 | | 0.017 | 0.011 | -0.004 | -0.013 | -0.022 | -0.003 | 0.010 | 0.010 | -0.011 | -0.013 | -0.018 | -0.011 | -0.002 | |
| 13 | | 0.016 | 0.011 | 0.007 | -0.007 | -0.002 | 0.011 | 0.011 | 0.003 | -0.009 | -0.007 | -0.007 | -0.008 | -0.005 | |
| 14 | | | 0.024 | 0.012 | 0.005 | 0.017 | 0.034 | 0.021 | 0.004 | -0.009 | -0.003 | 0.003 | -0.000 | | |
| 15 | | | | | 0.017 | 0.025 | 0.028 | 0.017 | 0.006 | -0.009 | -0.002 | | | | |

## FIGURE 4.7.40 - 100% POWER, ARO

CONSTANT FACTOR INCLUDED IN THE CALCULATION OF ACTUAL ENTHALPY RISE, FSUBH
MEASUREMENT UNCERTAINTY FACTOR, FSUBHU = 1.0000

RATIO OF ACTUAL POWER TO RATED POWER, P = 0.9795
COOLANT FLOW CORRECTION FACTOR = 0.0092
COOLANT TEMP. CORRECTION FACTOR = 0.0000
COMBINED FLOW AND TEMPERATURE FACTOR = 1.0092

CONSTANT MULTIPLIER FOR TECH SPEC FSUBH = 1.5099
* INDICATES VIOLATION OF TECH SPECS

20 LOWEST FRACTIONAL MARGINS FOR ENTHALPY RISE FACTORS

| ORDER | FUEL ASM. | F.A. LOC. | FSUBHN | ROD POW FACTOR | TECH SP. FSUBH | ACTUAL FSUBH | TECH SPEC MARGIN DIFF. | TECH SPEC MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 210 | 05-H | 1.3653 | 1.0000 | 1.5099 | 1.3653 | 0.1446 | 0.1059 | |
| 2 | 214 | 08-D | 1.3627 | 1.0000 | 1.5099 | 1.3627 | 0.1472 | 0.1080 | |
| 3 | 215 | 08-D | 1.3622 | 1.0000 | 1.5099 | 1.3622 | 0.1477 | 0.1084 | |
| 4 | 209 | 06-E | 1.3551 | 1.0000 | 1.5099 | 1.3551 | 0.1547 | 0.1142 | |
| 5 | 215 | 04-H | 1.3489 | 1.0000 | 1.5099 | 1.3489 | 0.1609 | 0.1193 | |
| 6 | 277 | 07-D | 1.3488 | 1.0000 | 1.5099 | 1.3488 | 0.1611 | 0.1195 | |
| 7 | 217 | 04-H | 1.3485 | 1.0000 | 1.5099 | 1.3485 | 0.1614 | 0.1197 | |
| 8 | 273 | 09-D | 1.3477 | 1.0000 | 1.5099 | 1.3477 | 0.1622 | 0.1203 | |
| 9 | 262 | 05-J | 1.3436 | 1.0000 | 1.5099 | 1.3436 | 0.1662 | 0.1237 | |
| 10 | 265 | 05-J | 1.3434 | 1.0000 | 1.5099 | 1.3434 | 0.1665 | 0.1239 | |
| 11 | 276 | 04-J | 1.3386 | 1.0000 | 1.5099 | 1.3386 | 0.1713 | 0.1280 | |
| 12 | 318 | 05-F | 1.3374 | 1.0000 | 1.5099 | 1.3374 | 0.1725 | 0.1290 | |
| 13 | 260 | 05-G | 1.3371 | 1.0000 | 1.5099 | 1.3371 | 0.1728 | 0.1293 | |
| 14 | 263 | 05-G | 1.3368 | 1.0000 | 1.5099 | 1.3368 | 0.1731 | 0.1295 | |
| 15 | 274 | 04-G | 1.3358 | 1.0000 | 1.5099 | 1.3358 | 0.1740 | 0.1303 | |
| 16 | 317 | 10-E | 1.3315 | 1.0000 | 1.5099 | 1.3315 | 0.1784 | 0.1340 | |
| 17 | 242 | 09-F | 1.3313 | 1.0000 | 1.5099 | 1.3313 | 0.1785 | 0.1341 | |
| 18 | 258 | 09-E | 1.3299 | 1.0000 | 1.5099 | 1.3299 | 0.1800 | 0.1354 | |
| 19 | 259 | 09-E | 1.3296 | 1.0000 | 1.5099 | 1.3296 | 0.1803 | 0.1356 | |
| 20 | 275 | 12-J | 1.3293 | 1.0000 | 1.5099 | 1.3293 | 0.1806 | 0.1358 | |

FIGURE 4.7.41 - 100% POWER, ARO

CONSTANT FACTORS INCLUDED IN THE CALCULATION OF ACTUAL HEAT FLUX, FQFNCR
ENGINEERING FACTOR, FSUBQE = 1.0300
MEASUREMENT UNCERTAINTY FACTOR, FSUBQU = 1.0500

PARAMETERS FOR THE FINAL ACCEPTANCE CRITERIA
RATIO OF ACTUAL POWER TO RATED POWER, P = 0.9795
FACTOR FOR REFUELING WATER TANK TEMP. = 1.0000
CONSTANT MULTIPLIER FOR TECH SPEC FSUBQ = 2.3686
* INDICATES VIOLATION OF TECH SPECS FOR F.A.C.

20 LOWEST FRACTIONAL MARGINS FOR HEAT FLUX PEAKING FACTORS CORRESPONDING TO THE FINAL ACCEPTANCE CRITERIA

| ORDER | FUEL ASM. | F.A. LOC. | AXL. LOC. | FSUBQN | K(Z)/C(Z) FACTOR | BURNUP FACTOR | TECH SP. FSUBQ | ACTUAL FQFNCR | TECH SPEC DIFF. | MARGIN FRACT. | VIOL. |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 210 | 05-H | 20 | 1.8009 | 1.0000 | 1.0000 | 2.3686 | 2.0451 | 0.3235 | 0.1582 | |
| 2 | 210 | 05-H | 19 | 1.8076 | 1.0000 | 1.0000 | 2.3686 | 2.0415 | 0.3271 | 0.1603 | |
| 3 | 214 | 08-D | 20 | 1.8070 | 1.0000 | 1.0000 | 2.3686 | 2.0408 | 0.3278 | 0.1606 | |
| 4 | 215 | 08-D | 20 | 1.8064 | 1.0000 | 1.0000 | 2.3686 | 2.0401 | 0.3285 | 0.1610 | |
| 5 | 214 | 08-D | 19 | 1.8016 | 1.0000 | 1.0000 | 2.3686 | 2.0350 | 0.3336 | 0.1640 | |
| 6 | 209 | 08-E | 20 | 1.8014 | 1.0000 | 1.0000 | 2.3686 | 2.0347 | 0.3339 | 0.1641 | |
| 7 | 215 | 06-D | 19 | 1.8009 | 1.0000 | 1.0000 | 2.3686 | 2.0342 | 0.3344 | 0.1644 | |
| 8 | 209 | 06-E | 19 | 1.8000 | 1.0000 | 1.0000 | 2.3686 | 2.0332 | 0.3354 | 0.1649 | |
| 9 | 210 | 05-H | 21 | 1.6707 | 1.0000 | 1.0000 | 2.3686 | 2.0319 | 0.3367 | 0.1657 | |
| 10 | 214 | 06-D | 21 | 1.8707 | 1.0000 | 1.0000 | 2.3686 | 2.0232 | 0.3454 | 0.1707 | |
| 11 | 273 | 09-D | 20 | 1.8703 | 1.0000 | 1.0000 | 2.3686 | 2.0229 | 0.3459 | 0.1710 | |
| 12 | 215 | 06-D | 21 | 1.8701 | 1.0000 | 1.0000 | 2.3686 | 2.0225 | 0.3461 | 0.1711 | |
| 13 | 210 | 05-H | 13 | 1.8695 | 1.0000 | 1.0000 | 2.3686 | 2.0218 | 0.3468 | 0.1715 | |
| 14 | 277 | 07-D | 20 | 1.8685 | 1.0000 | 1.0000 | 2.3686 | 2.0208 | 0.3478 | 0.1721 | |
| 15 | 273 | 09-D | 19 | 1.8680 | 1.0000 | 1.0000 | 2.3686 | 2.0203 | 0.3483 | 0.1724 | |
| 16 | 209 | 06-E | 21 | 1.8662 | 1.0000 | 1.0000 | 2.3686 | 2.0183 | 0.3503 | 0.1736 | |
| 17 | 277 | 07-D | 19 | 1.8655 | 1.0000 | 1.0000 | 2.3686 | 2.0176 | 0.3510 | 0.1740 | |
| 18 | 214 | 08-D | 18 | 1.8599 | 1.0000 | 1.0000 | 2.3686 | 2.0115 | 0.3571 | 0.1776 | |
| 19 | 215 | 06-D | 18 | 1.8592 | 1.0000 | 1.0000 | 2.3686 | 2.0107 | 0.3579 | 0.1780 | |
| 20 | 209 | 08-E | 13 | 1.8570 | 1.0000 | 1.0000 | 2.3686 | 2.0084 | 0.3603 | 0.1794 | |

FIGURE 4.7.42 - 100% POWER, ARO

RELATIVE ERRORS IN F SUB DELTA H CALCULATED FROM WEIGHTED THEORETICAL FACTORS, (CALC.-MEAS.)/MEAS.

| | R | P | N | M | L | K | J | H | G | F | E | D | C | B | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 01 | | | | | -0.001 | -0.001 | -0.017 | -0.032 | 0.007 | 0.037 | 0.037 | | | | |
| 02 | | | 0.020 | 0.020 | -0.001 | -0.001 | -0.010 | -0.032 | 0.003 | 0.026 | 0.024 | 0.020 | 0.015 | | |
| 03 | | 0.021 | 0.022 | 0.024 | -0.001 | -0.001 | -0.011 | 0.009 | -0.005 | 0.016 | 0.018 | 0.020 | 0.015 | 0.010 | |
| 04 | | 0.016 | 0.028 | 0.016 | -0.001 | -0.001 | -0.012 | -0.026 | -0.010 | -0.006 | 0.003 | -0.003 | -0.001 | -0.004 | |
| 05 | 0.032 | 0.028 | 0.025 | 0.016 | -0.001 | -0.009 | -0.020 | -0.031 | -0.024 | -0.020 | -0.017 | -0.019 | -0.022 | -0.018 | -0.014 |
| 06 | 0.044 | 0.037 | 0.029 | 0.019 | -0.001 | -0.017 | -0.025 | -0.043 | -0.020 | -0.003 | -0.001 | -0.021 | -0.020 | -0.014 | -0.014 |
| 07 | 0.031 | 0.026 | 0.022 | 0.023 | 0.007 | -0.010 | -0.014 | -0.024 | -0.018 | 0.022 | -0.016 | -0.020 | -0.025 | -0.013 | -0.016 |
| 08 | 0.019 | 0.022 | 0.014 | 0.015 | 0.009 | -0.004 | -0.010 | -0.018 | -0.024 | -0.043 | -0.024 | -0.036 | 0.013 | -0.018 | -0.004 |
| 09 | 0.026 | 0.034 | 0.024 | 0.013 | 0.012 | 0.007 | 0.003 | -0.012 | -0.032 | -0.033 | -0.019 | -0.019 | -0.014 | 0.001 | 0.011 |
| 10 | 0.029 | 0.030 | 0.034 | 0.003 | 0.015 | 0.007 | 0.015 | -0.002 | -0.019 | -0.019 | -0.016 | -0.023 | -0.023 | 0.010 | 0.010 |
| 11 | 0.025 | 0.025 | 0.010 | -0.003 | -0.015 | -0.004 | 0.001 | 0.002 | 0.002 | -0.006 | -0.006 | -0.011 | -0.013 | 0.010 | 0.010 |
| 12 | | 0.022 | 0.019 | 0.001 | -0.012 | -0.020 | -0.005 | 0.006 | 0.006 | -0.006 | -0.005 | -0.003 | 0.001 | 0.008 | |
| 13 | | 0.025 | 0.019 | 0.015 | -0.001 | 0.001 | 0.011 | 0.010 | 0.002 | -0.009 | -0.001 | 0.004 | 0.005 | 0.007 | |
| 14 | | | 0.032 | 0.019 | 0.011 | 0.021 | 0.035 | 0.019 | 0.003 | -0.009 | 0.001 | 0.011 | 0.009 | | |
| 15 | | | | | 0.023 | 0.028 | 0.024 | 0.014 | 0.004 | -0.009 | 0.001 | | | | |

4.8 NATURAL CIRCULATION TEST (S-090024)
Procedure ETT-ZZ-09240

OBJECTIVES

The objectives of the subject procedure were:

1. To demonstrate the length of time required to stabilize natural circulation.

2. To demonstrate core flow distribution during natural circulation using the incore detection system.

3. To collect data for insertion into the training simulator program.

SUMMARY OF RESULTS

The acceptance criteria for the test and discussions of associated results follows.

1. Natural circulation has been demonstrated.

   This criterion was met as verified by the following parameters.

   a. Steam pressure - Stable

   b. RCS hot leg temperature - Stable and approximately equal to core exit thermocouple temperature.

   c. Core exit thermocouple temperature - Stable

   d. RCS cold leg temperature - Near saturation temperature for steam pressure.

2. The measured core $\Delta T$ as a function of core power under natural circulation conditions is less than 29.5°F at 3.0% Rated Thermal Power (RTP).

   The measured core $\Delta T$ was 42.5°F at 3.0 RTP. This criterion was not met.

DISCUSSION

The natural circulation demonstration test was first attempted on October 12, 1984. When the minimum power level required by procedure (3% RTP as measured by core $\Delta T$) was reached, the Power Range Nuclear Instrumentation indicated greater than 10% power. This inactivated the P-7 permissive signal and a reactor trip would have resulted had the reactor coolant pumps been tripped. Testing was suspended.

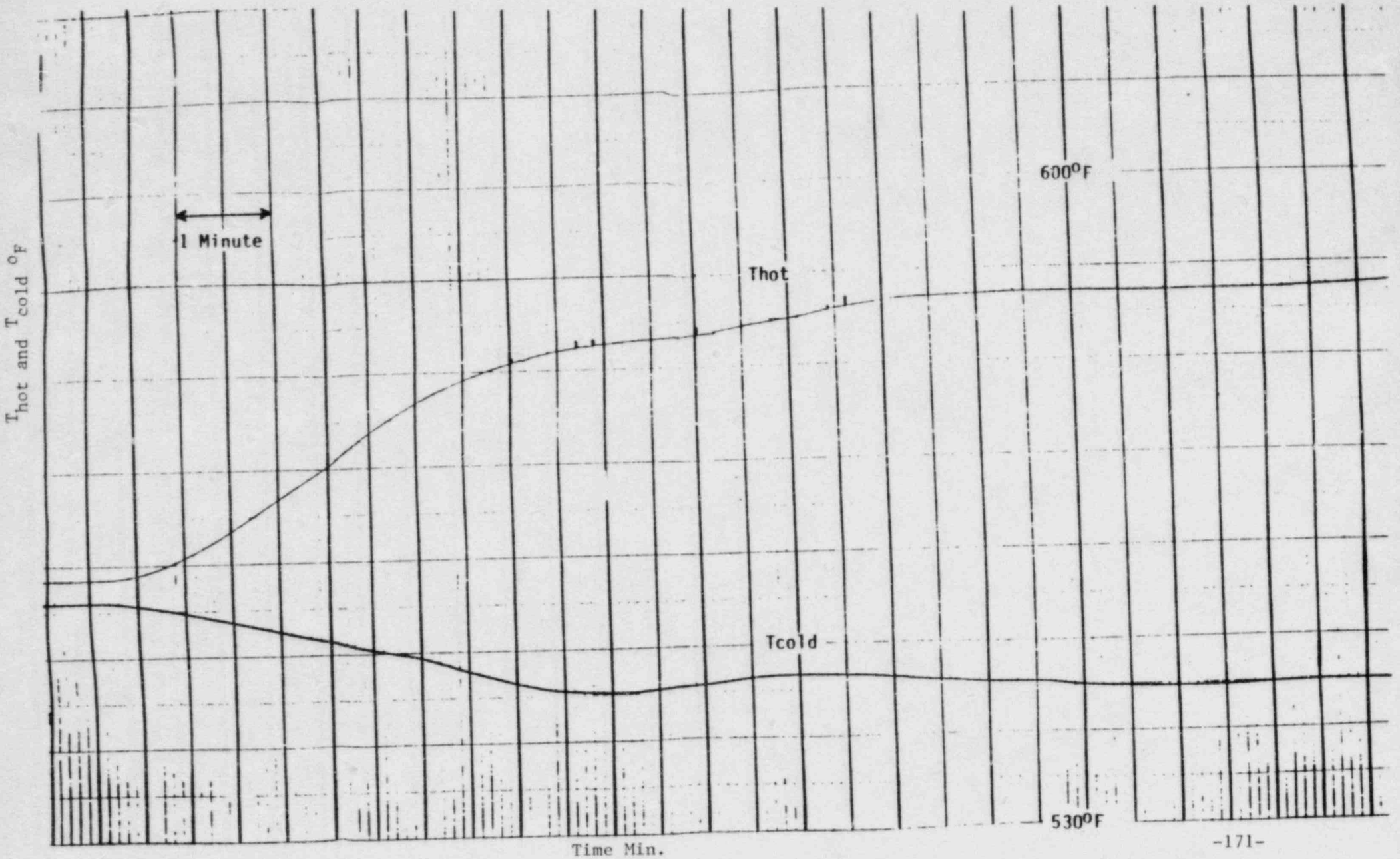
On October 14, 1984 reactor power was increased to just over 3.0% RTP, as indicated by core $\Delta T$, and stabilized. The Power Range Nuclear Instrumentation was calibrated to indicate twice the measured thermal power which resolved the P-7 problem. The reactor coolant pumps were then simultaneously tripped. The plant responded as expected with two exceptions. First, pressurizer level did not initially decrease, as observed on the training simulator, but instead remained constant for approximately 30 seconds and then, as anticipated, increased until being checked by a pressurizer PORV. Second, core $\Delta T$ stabilized at a higher value than expected, as delineated in the "Summary of Results". The data was transmitted to Westinghouse, who found the results acceptable and could see no safety issue represented by the test results. Figure 4.8.1 presents a copy of the strip chart recording for $T_{hot}$ and $T_{cold}$, Loop 1.

Approximately eleven minutes elapsed from time of pump trips to the establishment of stable natural circulation. Natural circulation was maintained for approximately one hour, during which time a thermocouple map was performed to demonstrate core flow distribution. Natural circulation was terminated by the insertion of the Control Banks.

Throughout this test, data was obtained on strip chart recorders and NSSS Computer trend blocks for benchmarking of the training simulator.

Figure 4.8.1

RCS Thot and Tcold (Loop 1)
During Natural Circulation Transient



## 5.1 DYNAMIC AUTOMATIC STEAM DUMP CONTROL TEST (S-07AB02)
Procedure ETT-AB-07020

### OBJECTIVES

The objectives of this procedure were to verify the proper closed loop operation of both reactor coolant system Tavg control modes (plant trip and load rejection) of the steam dump control system, to demonstrate the adequacy of the controller setpoints and to obtain final settings for steam pressure control of the steam dump valves.

### SUMMARY OF RESULTS

This test satisfactorily demonstrated the ability of the steam dump control system, operating in the Tavg mode (plant trip and load rejection controller) and manual pressure control modes, to maintain a stable steam pressure appropriate to plant conditions, or as selected by the operator.

Plant trip controller response was tested by simulating a P-4 reactor trip to the steam dump controller and raising power to approximately 6%. Tavg was verified to be stable at a programmed Tavg of 561°F.

Load rejection controller response was tested by raising power to 6% while the steam dumps were in the steam pressure mode. A large load rejection was then simulated and the steam dumps placed in the Tavg mode. Tavg was then monitored and verified to be stable at the programmed Tavg of 561±2°F.

The steam pressure controller was tested by inputting a setpoint corresponding to 1092 psig, placing the steam dump controller in auto in the steam pressure mode, and verifying steam header pressure was controlled at 1092±25 psig at approximately 6% power.

### DISCUSSION

The steam dump control system can be operated either in the steam pressure control or average temperature control mode. In the former, steam pressure control is based upon the comparison between system pressure and a manual setpoint. In the latter (average temperature) mode, control of the steam dump valves originates either from a load rejection controller or a plant trip controller, depending on plant conditions. If the plant should trip, the P-4 control signal will originate in the plant trip controller; if a load rejection is experienced then the control signal comes from the load rejection controller. Both of these modes base control action upon the measured reactor coolant system average temperature which is compared to the no load average temperature in the plant trip controller, and to the reference temperature signal derived from turbine first stage impulse chamber pressure. The control action of the load rejection controller is characterized by a 2°F controller deadband and another 2°F to provide the proper valve position, whereas the plant trip controller has no programmed deadband.

The steam dump system test consisted of three parts: the plant trip controller test, the load rejection controller test and verification of the pressure mode of control. In each the general approach was first to verify the ability of the controller to maintain stable steady state conditions and then to test the capability to cope with transient situations.

With reactor power at 1%, a P-4 reactor trip was simulated. The reactor coolant average temperature was raised 3°F above the no load value (557°F) using control rods, then the steam dump system was placed in the average temperature (Tavg) mode to see if it would restore Tavg to approximately 558±2°F. Next, reactor power was raised to 6% by pulling rods to determine if the steam dump system would stabilize Tavg at approximately 562±2°F. In all cases the plant trip controller responded properly in reducing the Tavg to its required acceptance criteria.

To verify the ability of the steam pressure controller to cope with transients, the system was placed in the steam pressure control mode at 1% reactor power with a steam header pressure controller setpoint of 1092 Psig. Then reactor power was raised to 5% using rods. The steam dump valves modulated to maintain a stable 1092 Psig throughout the transient.

A problem was experienced at 6% power prior to testing the load rejection controller initially due to the C-9 "Condenser Available" interlock lost due to low condenser vacuum. This problem was corrected by placing an additional vacuum pump and circulating water pump in service.

In order to test the load rejection controller, it was necessary to activate the sudden loss of load interlock to obtain selection of this mode of control. Then, with the reactor at approximately 6% power, the system was placed in the Tavg mode to verify Tavg was maintained at 561±2°F. No problems were encountered with this portion of the test.

## 5.2 OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION (S-07SE0?) Procedure ETT-SE-07020

### OBJECTIVES

The objectives of this test were to determine, prior to core loading, the pulse amplifier attenuator and discriminator voltage settings for the Source Range channels and to determine the high voltage power supply operating voltage settings for the Source Range channels. Prior to criticality, the objectives of this test were to establish the initial trip setpoints for the intermediate range and power range channels, to determine and record the precritical settings for the SOURCE RANGE, INTERMEDIATE RANGE and POWER RANGE channels, and to determine the gain settings for the Flux Deviation Averaging amplifiers. During power escalation, the objectives of this test were to determine the amount of overlap between the Source and Intermediate Range channels and the Intermediate and Power Range channels, and to determine the Power Range Detector currents versus core power. At full power, the objectives of this test were to check the Intermediate and Power Range operating detector voltages, to record the final operational settings for the Source, Intermediate, and Power Range channels, and to set the Intermediate Range detectors compensation voltages following a shutdown.

### SUMMARY OF RESULTS

All adjustments and settings were performed in accordance with the appropriate I&C controlling procedure. The Power Range channel gain setpoints were made based on the result of calimetrics. All data was obtained and analyzed without incident. The shutdown after Fuel-Burnup portion of the test was performed because the 502 EFPH burnup requirement was satisfied.

### DISCUSSION

Nuclear Instrumentation setpoints for pulse amp high voltage and discriminator settings were set in accordance with the appropriate I&C controlling procedure. This was completed before core loading. Prior to criticality, the remaining setpoints for Nuclear Instrumentation System were set. These settings and adjustments are shown on Table 5.2.1.

When the Intermediate Range channels indicated $1X10^{-10}$ amps, the overlap region between the Source and Intermediate Range channels was determined. Prior to criticality, the Intermediate Range channel indicated a value less than $10^{-11}$ amps, therefore the acceptance criteria of a minimum of 1½ decades of overlap between the Source and Intermediate Range channels was met.

During the power ascension phase, data was taken at the 10, 30, 50, 75 and 90 percent power plateaus to determine the amount of overlap between the Intermediate and Power Range channels and to determine the Power Range Detector Current versus core power. At the 10 percent plateau, the Intermediate Range channel indicated an average of $7X10^{-5}$ amps, compared to $10^{-10}$ amps at zero power. Therefore, the acceptance

criteria of a minimum of 1½ decades of overlap between the Intermediate and Power Range channels was met.

At each plateau, calorimetric power calculations were performed and the NI channels were adjusted. The Power Range channel current versus core power level is plotted for each plateau on Figures 5.2.1 through 5.2.4.

At 100 percent power, the operating voltage for each Intermediate and Power Range detector was determined. The final operational settings for the Nuclear Instrumentation are shown on Table 5.2.3. After the reactor was shutdown, the compensation voltage for each Intermediate Range Detector was adjusted to -21.27 vdc and -21.77 vdc for detectors N35 and N36, respectively.

TABLE 5.2.1

INITIAL
NUCLEAR INSTRUMENTATION SETTINGS AND ADJUSTMENTS

Source Range

Attenuator Setting

N31 -6 db

N32 -9 db

Discriminator Voltage

N31 -0.45 volts d.c.

N32 -3.70 volts d.c.

High Flux Alarm

N31 200 cps

N32 200 cps

High Flux Trip

N31 $1X10^{5}$ cps

N32 $1X10^{5}$ cps

Detector Voltage

N31 1875 volts d.c.

N32 1880 volts d.c.

Background Count Rate

N31 <1 cps

N32 <1 cps

Detector Voltage

Bistable Trip

N31 1760 volts d.c.

N32 1760 volts d.c.

Intermediate Range

P-6 Bistable Trip

N35 $1X10^{-10}$ amps

N36 $1X10^{-10}$ amps

Compensating Voltage

N35 -40.00 volts d.c.

N36 39.81 volts d.c.

High Voltage Setting

N35 .800 volts d.c.

N36 .800 volts d.c.

Rod Stop Bistable Trip

N35 $2.08X10^{-5}$ amps

N36 $2.03X10^{-5}$ amps

Level Trip Bistable

N35 $2.5X10^{-5}$ amps

N36 $2.5X10^{-5}$ amps

High Voltage Bistable Trip

N35 8.000 volts d.c.

N36 8.004 volts d.c.

Compensation Voltage

Bistable Trip

N35 -0.001 volts d.c.

N36 -0.002 volts d.c.

TABLE 5.2.1 (cont'd)

INITIAL

NUCLEAR INSTRUMENTATION SETTINGS AND ADJUSTMENTS

Power Range

Hi Power Level Bistable Trip

N41 109.0 %

N42 25.0 %

N43 24.0 %

N44 24.0 %

Overpower Rod Stop Bistable Trip

N41 103.3 %

N42 103.0 %

N43 102.5 %

N44 103.0 %

Low Range Bistable Trip

N41 25.0 %

N42 25.0 %

N43 25.0 %

N44 25.0 %

High Flux Rate Trip (negative)

N41 4.0 %

N42 4.0 %

N43 4.5 %

N44 4.0 %

High Flux Rate Trip (positive)

N41 4.0 %

N42 4.5 %

N43 4.5 %

N44 4.0 %

Time Constant (Flux Rate)

N41 2.5 sec.

N42 2.5 sec.

N43 2.5 sec.

N44 2.55 sec.

P-10 Bistable Trip

N41 10.0 %

N42 10.0 %

N43 10.5 %

N44 10.0 %

P-8 Bistable Trip

N41 48.3 %

N42 48.2 %

N43 47.9 %

N44 48.3 %

High Voltage Setting

N41 799.99 volts d.c.

N42 800.00 volts d.c.

N43 800.01 volts d.c.

N44 800.00 volts d.c.

P-9 Bistable Trip

N41 50.3 %

N42 50.2 %

N43 50.0 %

N44 50.0 %

Comparator Trip Setting

N41 2.0 %

N42 2.03 %

N43 2.5 %

N44 2.02 %

TABLE 5.2.3

FINAL

NUCLEAR INSTRUMENTATION SETTINGS AND ADJUSTMENTS

Power Range

Hi Power Level Bistable Trip

N41 109.0 %
N42 109.0 %
N43 109.0 %
N44 108.5 %

Overpower Rod Stop Bistable Trip

N41 103.3 %
N42 103.0 %
N43 102.5 %
N44 103.0 %

Low Range Bistable Trip

N41 25.0 %
N42 25.0 %
N43 25.2 %
N44 25.25 %

High Flux Rate Trip (negative)

N41 2.5 %
N42 2.5 %
N43 2.5 %
N44 2.5 %

High Flux Rate Trip (positive)

N41 2.5 %
N42 2.5 %
N43 2.5 %
N44 2.5 %

Time Constant (Flux Rate)

N41 2.5 sec.
N42 2.5 sec.
N43 2.5 sec.
N44 2.5 sec.

P-10 Bistable Trip

N41 10.1 %
N42 10.0 %
N43 10.0 %
N44 10.25 %

P-8 Bistable Trip

N41 48.2 %
N42 48.0 %
N43 48.25 %
N44 48.25 %

High Voltage Setting

N41 799.8 volts d.c.
N42 800.0 volts d.c.
N43 800.9 volts d.c.
N44 800.3 volts d.c.

P-9 Bistable Trip

N41 50.2 %
N42 50.0 %
N43 50.5 %
N44 50.0 %

Comparator Trip Setting

N41 2.0 %
N42 2.3 %
N43 2.5 %
N44 2.02 %

Figure 5.2.1
Total Detector Current Versus Reactor Power
Power Range Channel 41

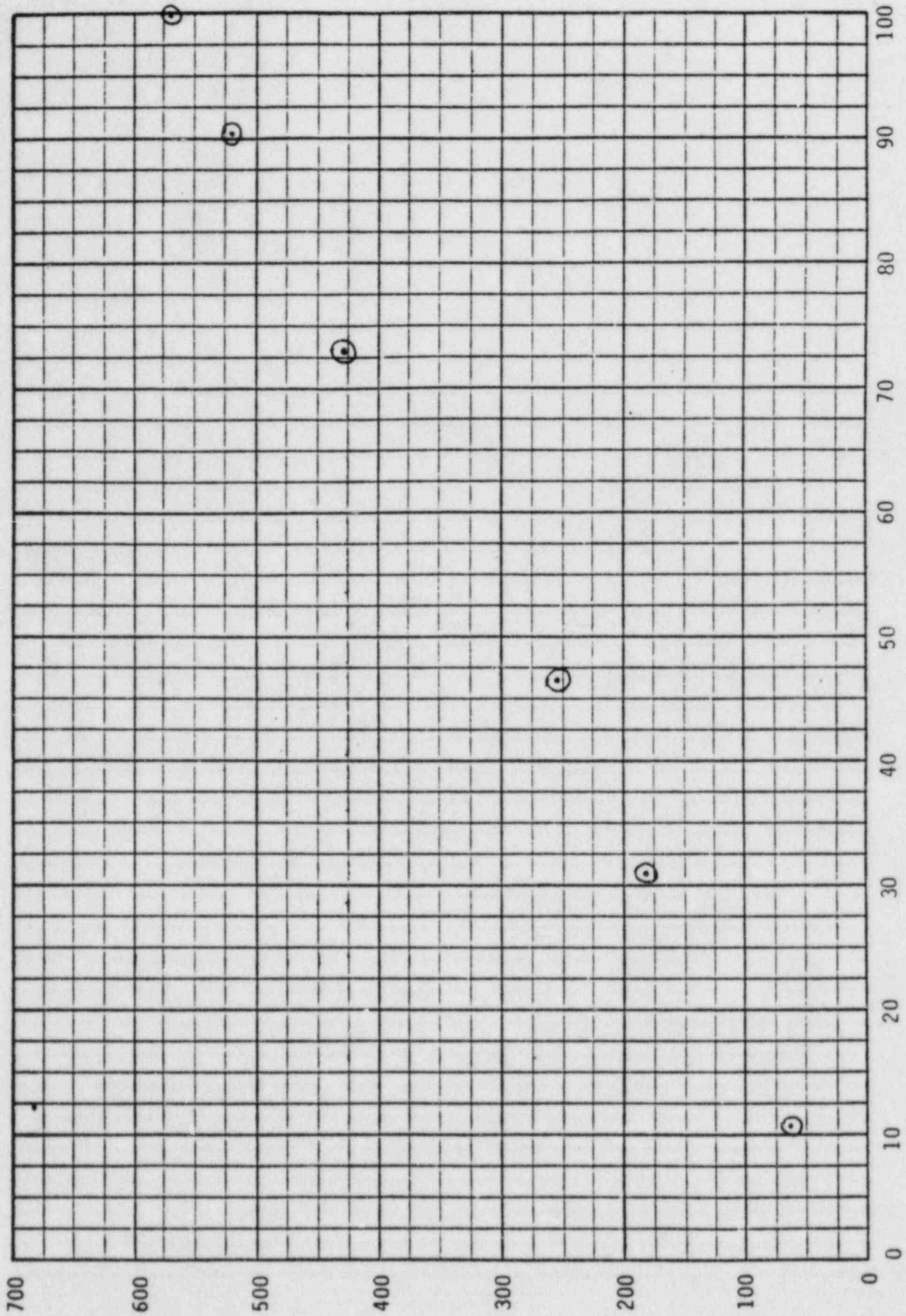


Figure 3.2.2
Total Detector Current Versus Reactor Power
Power Range Channel 42

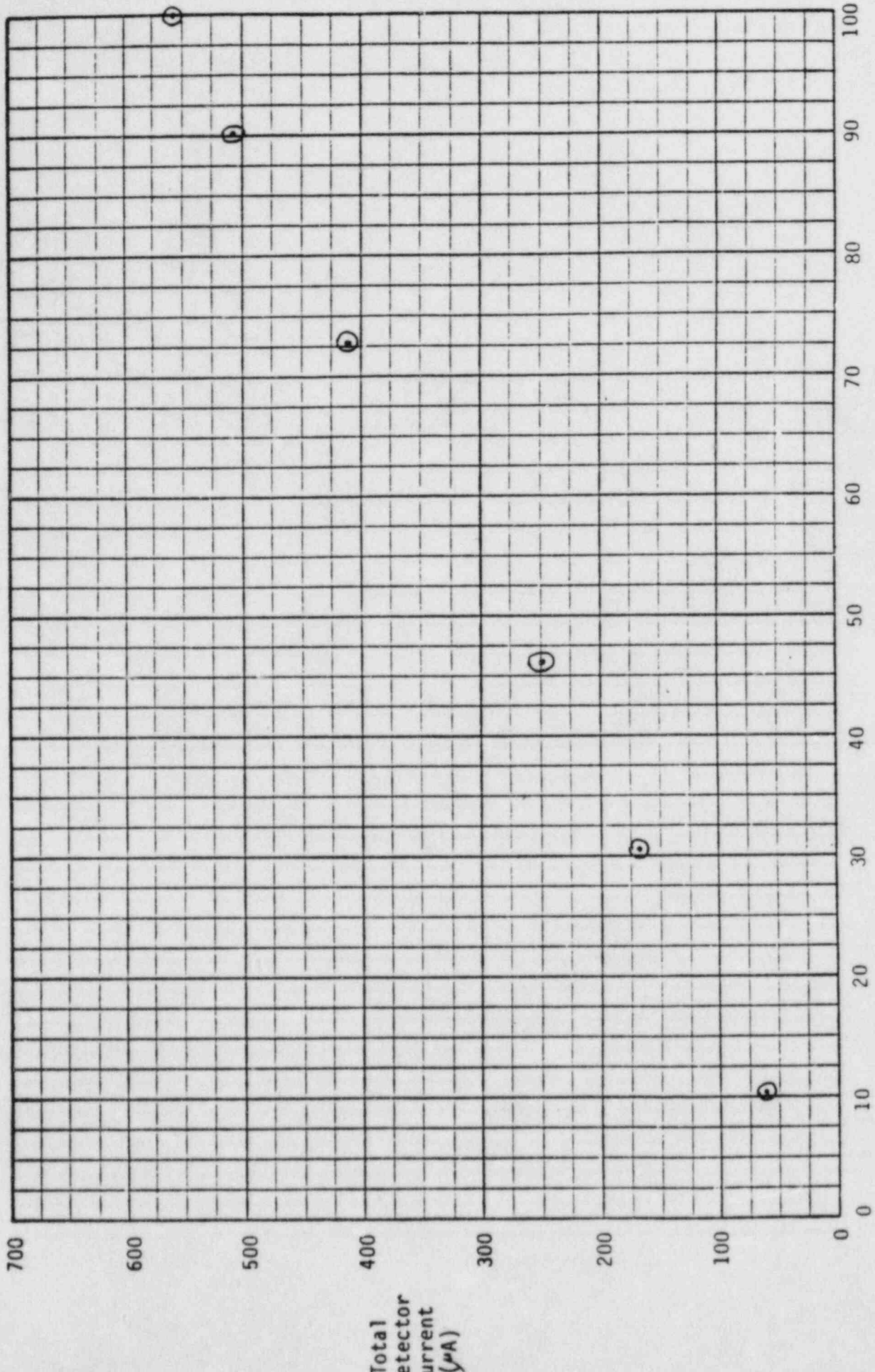


Figure 3.2.3
Total Detector Current Versus Reactor Power
Power Range Channel 43

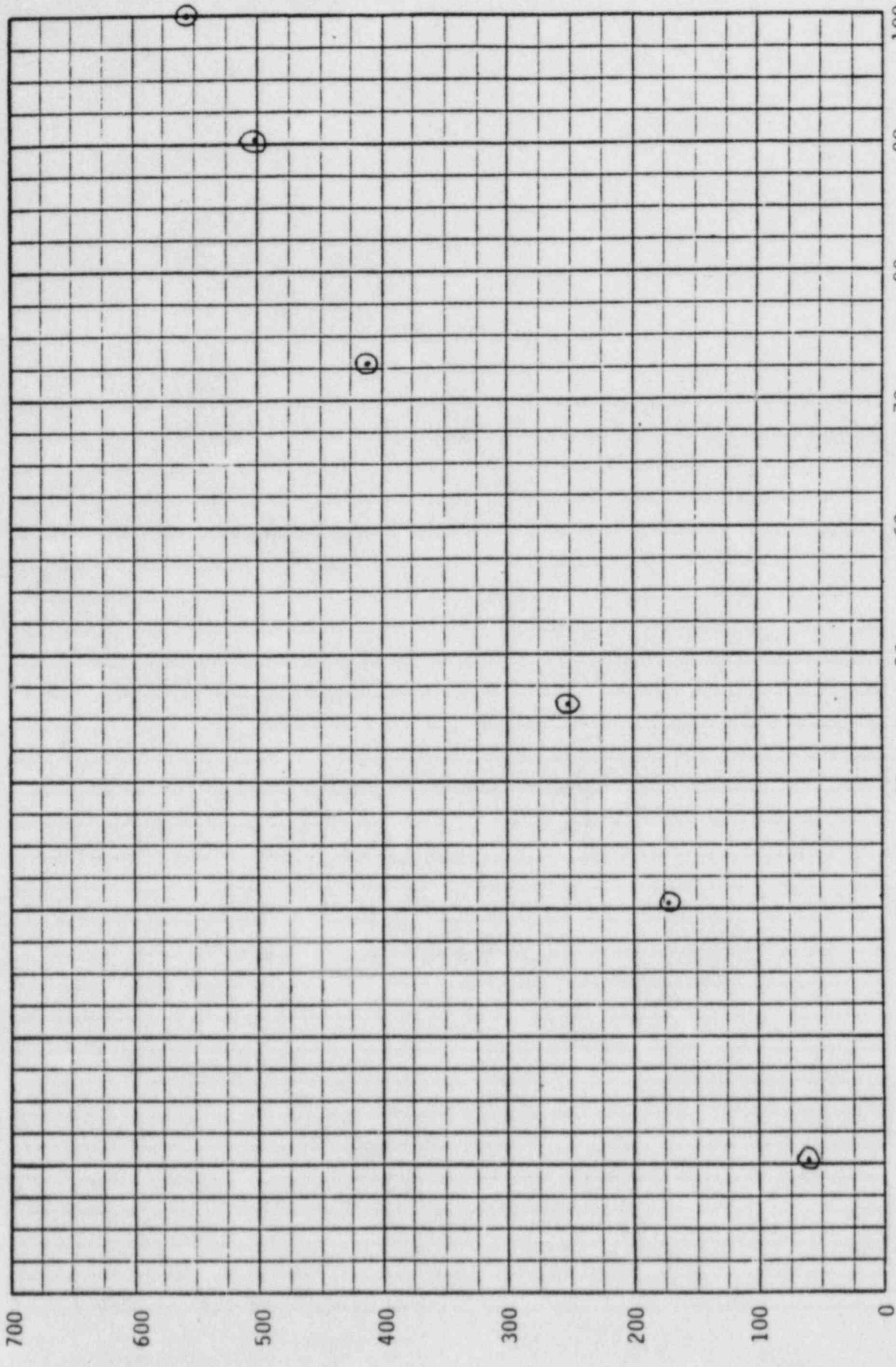
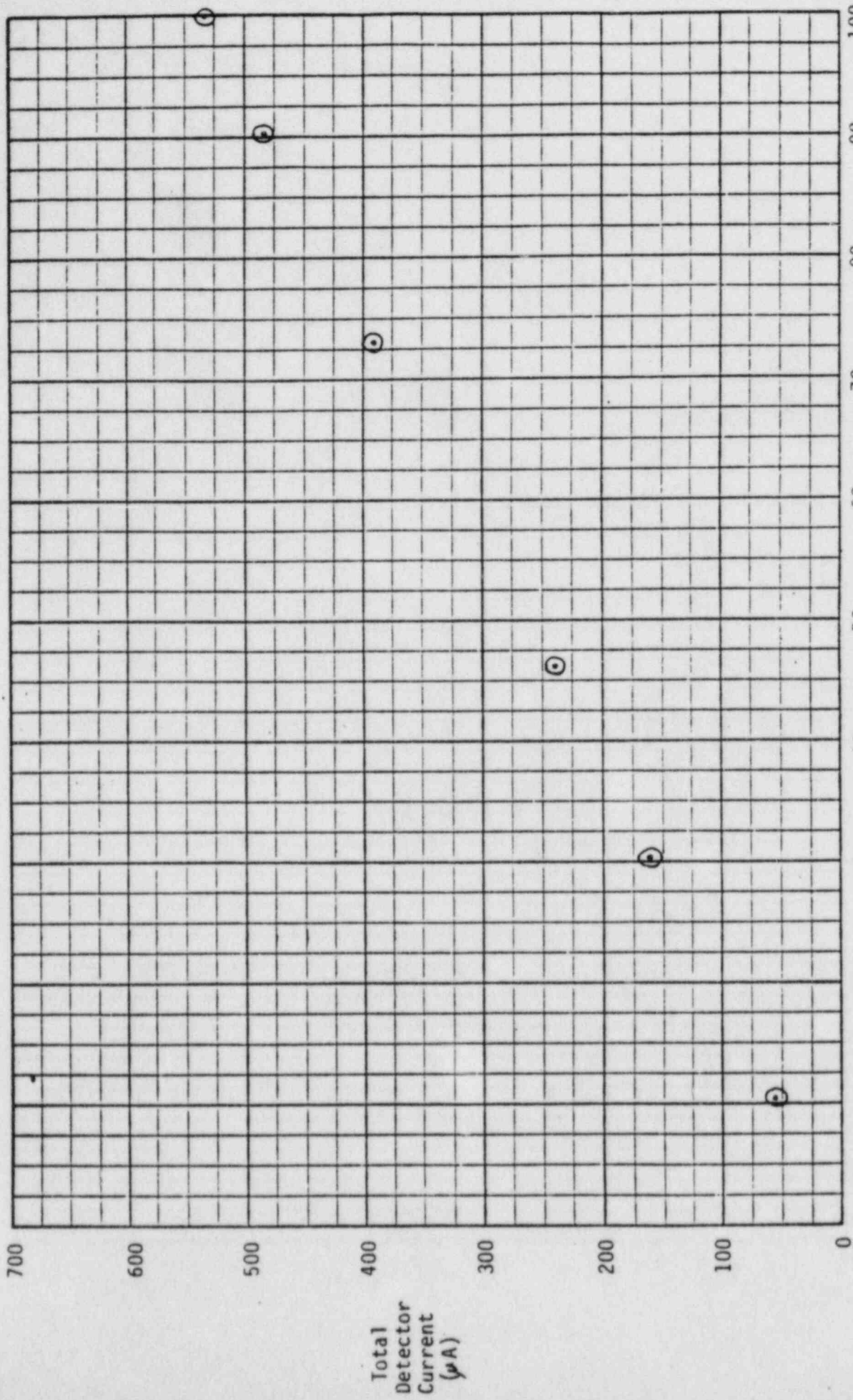


Figure 5.2.4
Total Detector Current Versus Reactor Power
Power Range Channel 44



## 5.3 SHUTDOWN AND MAINTENANCE OF HOT STANDBY EXTERNAL TO THE CONTROL ROOM (S-070014)

Procedure ETT-ZZ-07100

### OBJECTIVE

The objective of this test was to demonstrate the capability to shut down the plant from ≥10% generator power and maintain the reactor in a safe Hot Standby condition from outside the control room.

### SUMMARY OF RESULTS

The acceptance criteria for the shutdown from outside the main control room test were:

1. The Hot Standby conditions shall be maintained for at least 30 minutes after all controls have been assumed at the Auxiliary Shutdown panels following a shutdown from ≥10% generator power.

2. Hot Standby is maintained by the minimum shift crew from outside the control room assuming no fire is present in the main control room.

3. Controls and instrumentation external to the control room are utilized in shutting down to Hot Standby.

4. System, equipment, and instrumentation response is as expected for a normal plant trip from the initial conditions of this test.

All acceptance criteria for this test were satisfactorily demonstrated with no significant problems.

### DISCUSSION

The shutdown was initiated by the Shift Supervisor from outside the control room by manually tripping the reactor and turbine. The initial turbine generator power was 125.4MW. No operator actions were required by control room personnel. Only those controls and instrumentation external to the Control Room were utilized in maintaining Hot Standby conditions. The Technical Specifications minimum shift crew maintained the plant in Hot Standby for 41 minutes and then control was transferred back to the control room. System, equipment and instrument response was as expected for a normal plant trip. Since the acceptance criteria were satisfied, the test successfully demonstrated the capability to safely shut down the plant from outside the main control room.

## 5.4 AUTOMATIC STEAM GENERATOR LEVEL CONTROL (S-07AB01)

Procedure ETT-AB-07010

### OBJECTIVES

The objectives of this test were as follows:

1. To demonstrate the level control stability of the steam generator feedwater bypass valves in automatic control below 20% thermal power.

2. To demonstrate the stability of the steam generator level control when transferring control from the feedwater valves.

3. To demonstrate proper response of the automatic steam generator level control system during plant transients at power levels of 30%, 75%, and 100% with adjustments being made as required to optimize system performance.

4. To verify Main Feedwater Pump (MFP) performance and to demonstrate proper operation of the turbine driven feedwater pump speed control during power escalation.

5. To verify proper automatic programming of the steam generator level during power escalation.

### SUMMARY OF RESULTS

The acceptance criteria were as follows:

1. Steam generator level overshoot/undershoot shall be less than ± 4% following a 5% step level decrease.

2. Steam generator level overshoot/undershoot shall be less than ± 4% following a 5% step level increase.

3. Steam Generator Level shall return to and remain within ± 2% of the level setpoint within 3 x level controller reset time period (T36) following a step 5% level or level setpoint change.

4. Steam generator level shall return to and remain within ± 2% of the level setpoint within 3 x level controller reset time period (T31) following a step 5% level or level setpoint change.

5. Steam generator level shall return to and remain within ± 2% of the reference level, within 3 x level controller reset time period (T31) following a transfer of level controls.

6. Level control variables shall not incur sustained or diverging oscillation.

7. Main Feedwater Pump discharge pressure oscillations are less than ± 3%.

8. The Main Feedwater control valve stem position (% valve open) shall stabilize within the acceptable ranges.

9. The Main Feedwater Pump discharge pressure oscillations are less than ± 3%, 2 minutes after a steam flow change.

10. The Main feedwater Pump Total Developed Head (TDH) will be 2387 ± 238 feet for a flow capacity ≥ 17,620 gpm.

With the exception of minor deficiencies encountered during conduct of the tests, all acceptance criteria were satisfied. Several deficiencies were reviewed by Westinghouse and found acceptable "as is". All objectives identified above were satisfied with the exception of number 3 at the 75% power level. This was performed at the 100% power level.

Table 5.4.1 summarizes the final settings of the main feedwater pump speed and the steam generator level control systems.

DISCUSSION

The level control capability of the feedwater bypass valves was verified by manually inducing 5% steam generator level decreases and increases into each steam generator. All controller responses were satisfactory at 0% reactor power.

At approximately 18% reactor power, the level control system was transferred from the bypass valves to the main feedwater valves. Steam generator levels were then monitored to verify that they remained within the narrow range span of 50 ± 2%. After the initial transfer, none of the controllers met the acceptance criteria; however, during a subsequent retest the acceptance criteria were satisfactorily met. Additionally, the gain and reset time constants were adjusted for all steam generators as shown on Table 5.4.1.

Main feedwater pump speed controller setpoints were checked at the 30%, 75%, and 100% power levels to verify proper operation. During all power level tests, the main feed valves positioned themselves more open than predicted for a given feedwater flow. Subsequent Westinghouse evaluation showed that the unexpected valve positions were acceptable and continued power operation could proceed at all levels.

Characteristic main feedwater pump operating data was obtained to verify if the main feedwater pumps met the vendor operating curves at the 30%, 50%, 75%, and 100% power levels. With one exception, it was determined that feedwater pump curves were satisfied at all power levels. During the 50% power level test, main feedwater pump "A" suction flow computer point indication was indicating low. Subsequent repair of the computer point and a retest verified that suction flow was acceptable for determining pump flow.

During the large load reduction of 50% from 75% full reactor power per procedure ETT-ZZ-07101, the ability to monitor steam generator level and pump speed response on all steam generators was interrupted due to unexpected steam dump closure. It was determined that the aforementioned responses would be verified during the large load reduction of 50% from 100% full reactor power per ETT-ZZ-07102. Steam generator level and pump speed control responded satisfactorily during the 100% full power load reduction test.

During the large load reduction of 50% power from 100% full reactor power, acceptance criteria (No. 9 above) was not satisfied when it was observed that main feedwater pump discharge pressure oscillated up to as much as ± 6.5% beyond two (2) minutes of the initiation of the transient. Subsequent evaluation of this anomaly was evaluated by the architect engineer and found to be acceptable.

TABLE 5.4.1
CONTROLLER SETPOINTS DETERMINED BY TESTS

1. Feedwater Bypass Valve Setpoints (18% Reactor Power)

| | S.G.No. 1 AE-LK-550 | S.G.No. 2 AE-LK-560 | S.G.No. 3 AE-LK-570 | S.G.No. 4 AE-LK-580 |
|---|---|---|---|---|
| T30 (Seconds) (Lag Time Constant) | 5 | 5 | 5 | 5 |
| T36 (Seconds) (Reset Time Constant) | 690 | 690 | 690 | 690 |
| Gain K34 ($\frac{\%\ \text{Lift}}{\%\ \text{Power}}$) (Bypass Valve Controller) | 1.25 | 1.50 | 1.50 | 1.25 |
| Gain K35 ($\frac{\%\ \text{Lift}}{\%\ \text{Power}}$) (Nuclear Feed Forward) | 2.0 | 2.0 | 2.0 | 2.0 |

2. Feedwater Control Valve Setpoints (100% Reactor Power)

| | S.G.No. 1 AE-FK-510 | S.G.No.2 AE-FK-520 | S.G.No. 3 AE-FK-530 | S.G.No. 4 AE-FK-540 |
|---|---|---|---|---|
| Gain K30 ($\frac{\%\ \text{Power}}{\%\ \text{Level}}$) | 2.6 | 2.6 | 2.6 | 2.6 |
| T31 (Seconds) (Level Compensation Reset Time Constant) | 200 | 200 | 200 | 200 |
| Gain K31 ($\frac{\%\ \text{Power}}{\%\ \text{Flow}}$) (Flow Compensation) | 1 | 1 | 1 | 1 |
| T32 (Seconds) (Flow Compensation Reset Time Constant) | 200 | 200 | 200 | 200 |

3. Feedwater Pump Speed Control Setpoints (100% Reactor Power)

| | MFWP A<br>FC-SK-509 A(B) | MFWP B<br>FC-SK-509 A(C) |
|---|---|---|
| Gain K32 ( $\frac{\text{psi}}{\text{lbm/hr}}$)<br>(Delta P Setpoint Controller) | 10E-5 | 10E-5 |
| Delta P Setpoint (psi) | 45 | 45 |
| T34 (Seconds) | 120 | 120 |
| Gain K33 (rpm/psi)<br>(Delta P Controller) | 3.6 | 3.6 |
| T35 (Seconds)<br>(Reset Time Constant) | 100 | 100 |

## 5.5 CALIBRATION OF STEAM AND FEEDWATER FLOW INSTRUMENTATION AT POWER (S-070018)

Procedure ETT-ZZ-07180

### OBJECTIVES

The objectives of this test were to calibrate the steam flow transmitters using feedwater flow as a reference and to perform a cross-check verification of all signals indicating feedwater and steam flow. This was to be done at the 30, 50, 75 and 100 percent power plateaus.

### SUMMARY OF RESULTS

Through the 50 percent power plateau, the acceptance criteria was that the steam flow/feedwater flow mismatch alarm did not activate. This criterion was met.

For the 75 and 100 percent plateaus, the acceptance criteria were that the steam flow/feedwater flow mismatch alarm did not activate, that the main control board indications of steam flow were within ±4.0% of the main control board indications of feedwater flow, and that the test feedwater flow instrument and the plant feedwater flow and steam flow instruments should agree to within ±2.5% and 3.0%, respectively, of the ideal values. At the 75 percent plateau, the steam flow/feedwater flow mismatch alarm was observed and it was determined that two steam flow and one feedwater flow transmitter did not agree with the tolerance band of the ideal indication. Four main control board steam flow transmitters did not agree to within 4.0% of the main control board feedwater flow indication. The appropriate actions were taken to re-calibrate the flow transmitters. At 100 percent power, the steam flow/feedwater flow mismatch alarm did not actuate. The main control board indications of steam flow and feedwater flow were within ±4.0%. The steam flow and feedwater flows agreed to within the respective tolerance band with the exception of two steam flow transmitters. The two steam flow transmitters were retested and the results indicated that the acceptance criteria was met.

### DISCUSSION

Four test instruments were installed on the main feedwater flow loops at the No. 2 tap to establish a reference test flow with which the respective feedflow and steam flow loop transmitters were compared and to establish the necessity for re-calibration.

All data was obtained without incident and re-calibration was necessary only at the 75% power plateau.

The final portion of the procedure required a "zero point" retest. This retest does not impact the test results, as the transmitters are now properly calibrated at 100% power, and will be performed during the first scheduled outage of sufficient length.

5.6 BIOLOGICAL SHIELD TEST (S-070016)
Procedure ETT-ZZ-07160

OBJECTIVES

The objectives of this test were as follows:

A. To measure and record the neutron and gamma ray radiation levels in accessible areas of the plant where radiation levels above background were anticipated.

B To determine locations, if any, where shielding was deficient.

C. To ensure that plant personnel would not be subjected to overexposure from radiation as a result of inadequate shielding.

SUMMARY OF RESULTS

No shielding deficiencies were identified as a result of this test.

The Reactor Building will be posted as follows:

A. Very High Radiation Area (>1000mR/hr) - the entire RB at >50% power and inside the Biological Shield at Mode 1.

B. High Radiation Area (100 - 1000mR/hr) - outside the Biological Shield at Mode 1 up to 50% power and inside the Biological Shield at Mode 2.

C. Radiation Area (<100mR/hr) - outside the Biological Shield at Modes 2-6 and inside the Biological Shield at Modes 3-6.

Several areas in the Auxiliary Building are posted as radiation Areas due to radiation emanating from piping within these areas and not as a result of radiation passing through plant shielding.

DISCUSSION

Neutron and gamma radiation surveys were performed prior to fuel load, during low power physics testing, at 30% power, at 50% power, at 75% power, and at 100% power. Survey points were designated on biological shield walls inside Containment, general access areas inside Containment, along the Containment outer wall, and throughout the Auxiliary Building. Shielding was surveyed for voids, none of which were detected. Test results were reviewed at each power level to determine if additional shielding was required prior to proceeding to the next power level. Several of the reviews identified the need to add new survey points to the survey at higher power levels.

Survey readings at 100% power showed radiation levels existing at the following locations within the Reactor Building:

2068 level - General area radiation levels of <100 mR/hr were found in areas back from the edge overlooking the Refueling Cavity. The neutron component made up the majority of the total dose in these areas.

Radiation levels at the handrail overlooking the Refueling Cavity were 200 - 500 mR/hr with the gamma component being the majority of the total exposure.

2047 level - General area radiation levels of <100 mR/hr were found in areas outside of the Refueling Cavity handrail with the neutron component being the majority of the total dose.

Radiation levels at the edge of the Refueling Cavity were 1200 - 3000 mR/hr with the highest levels being at the Plant north end of the cavity. The gamma component was the majority of the toal exposure. These high radiation levels necessitated the posting of the entire Reactor Building as a Very High Radiation Areas at >50% power.

2033 level - General area radiation levels in the Flux Mapping Drive area were ~20 mR/hr with the neutron component being the majority of the total dose.

2026 level - General area radiation levels of <5 mR/hr were found in areas outside of the Biological Shield with the gamma/neutron components essentially equal.

Radiation levels outside the Biological Shield on the 2026 level above floor grating over 2000 level openings into Biological Shield at the RCS loops A, B, and C were 200-300mR/hr. These high levels are due to radiation coming from inside the Biological Shield from below the 2026 level. The gamma component is the majority of the total exposure.

Radiation levels inside the Pressurizer Cubicle were 5 - 160mR/hr with the gamma component being the majority of the total exposure.

2000 level - General area radiation levels of <10 mR/hr were found

in areas outside of the Biological Shield with the gamma component being the majority of the total exposure. Localized areas outside of the Shield (i.e. access openings in the Shield and penetrations through the Shield) had radiation levels of 250 - 250mR/hr with the gamma component being the majority of the total exposure.

Radiation levels of ~10,000mR/hr were found just inside the Biological Shield access openings with

the gamma component being the majority of the total exposure.

Radiation levels in the Incore Instrumentation Tunnel were not assessed. The entrance to this area is posted as an Exclusion Area and personnel access is not allowed.

Survey readings at 100% power showed radiation levels existing at the following locations within the Auxiliary Building:

2047 level - Approximately 1 mR/hr of neutron radiation was found on the Plant east side of the Containment Access where the Access Tube penetrates the Containment wall.

Area 5 - Approximately 0.2 - 0.4 mR/hr of neutron radiation was found on the Containment wall where steam and main feed lines penetrated.

## 5.7 THERMAL POWER MEASUREMENT AND STATEPOINT DATA COLLECTION (S-07SC03)

Procedures ETT-SC-07031, ETT-SC-07032, ETT-SC-07033, ETT-SC-07034, and ETT-SC-07035

### OBJECTIVE

The objective of these procedures was to determine Reactor Thermal Power and obtain the control and protection instrumentation calibration data at steady-state power levels (statepoints) at 30%, 50%, 75%, 90% and 100% testing plateaus.

### SUMMARY OF RESULTS

Statepoint data was taken at 30%, 50%, 75%, 90% and 100% testing plateaus and is summarized in Table 5.7.1.

### DISCUSSION

Statepoint data collection, consisting of three sets of readings taken under stable plant conditions at 15-minute intervals, was performed at nominal power levels of 30%, 50%, 75%, 90% and 100% power. Included in the data were accurate measurements of the parameters needed to perform a calorimetric calculation of reactor thermal power.

TABLE 5.7.1

THERMAL POWER MEASUREMENT SUMMARY

| Nominal Power | | 30% | 50% | 75% | 90% | 100% |
|---|---|---|---|---|---|---|
| Thermal Power(%) | | 27.04% | 44.53 | 69.8 | 88.27 | 95.51 |
| Feedwater | Loop 1 | $7.98 \times 10^5$ | $1.50 \times 10^6$ | $2.65 \times 10^6$ | $3.34 \times 10^6$ | $3.75 \times 10^6$ |
| Flow | Loop 2 | $7.61 \times 10^5$ | $1.42 \times 10^6$ | $2.54 \times 10^6$ | $3.28 \times 10^6$ | $3.63 \times 10^6$ |
| (lbm/hr) | Loop 3 | $8.39 \times 10^5$ | $1.54 \times 10^6$ | $2.75 \times 10^6$ | $3.37 \times 10^6$ | $3.76 \times 10^6$ |
| | Loop 4 | $7.91 \times 10^5$ | $1.48 \times 10^6$ | $2.59 \times 10^6$ | $3.29 \times 10^6$ | $3.67 \times 10^6$ |
| Feedwater | Loop 1 | 223 | 337 | 422 | 426 | 448 |
| Temp. | Loop 2 | 226 | 339 | 423 | 426 | 450 |
| (°F) | Loop 3 | 225 | 339 | 424 | 426 | 449 |
| | Loop 4 | 226 | 336 | 423 | 423 | 449 |
| Steam | Loop 1 | 986 | 1000 | 992 | 985 | 975 |
| Generator | Loop 2 | 990 | 1005 | 998 | 1001.7 | 975 |
| Pressure | Loop 3 | 981 | 1000 | 992 | 995 | 975 |
| (psig) | Loop 4 | 984 | 995 | 1042 | 1000 | 975 |

5.8 POWER COEFFICIENT DETERMINATION (S-070008)
Procedure ETT-ZZ-07081, ETT-ZZ-07082, ETT-ZZ-07083 and ETT-ZZ-07084

OBJECTIVES

These procedures verify nuclear design prediction of the Doppler only power coefficient through correlative measurements of reactor coolant system temperature and core thermal power at power levels of 30, 50, 75 and 90% power.

SUMMARY OF RESULTS

This procedure was to verify that the absolute values of the calculated and predicted values of the Doppler coefficient verification factor, $C_m$ and $C_p$ respectively, are within 0.5°F/% of each other at 30, 50, 75 and 90% rated thermal power. The following values were obtained:

| POWER LEVEL (%) | $\lvert\lvert C_m \rvert - \lvert C_p \rvert\rvert$ (°F/%) |
|---|---|
| 28.8 | 0.61 |
| 48.5 | 0.418 |
| 71.9 | 0.165 |
| 88.9 | 0.073 |

The test results were satisfactory for all power levels excluding the 30% plateau. The value of 0.61°F/% at 30% exceeded the 0.5°F/% limit, however no further retesting at 30% power was performed pending performance at the 50% plateau. Acceptable results at the 50% plateau cleared the deficiency for the 30% plateau.

DISCUSSION

This procedure was performed at nominal power levels of 30, 50, 75 and 90 percent of rated thermal power. The Doppler power coefficient verification is performed by varying plant load approximately 47 MWe utilizing the turbine controls on the main turbine EHC panel. Three successive load decreases/increases were performed, allowing $T_{avg}$ and $\Delta T$ to stabilize prior to each load swing. No changes in rod position or RCS boron concentration were made during the load changes.

After the load swings were performed, values for $T_{avg}$ and $\Delta T$ were taken from the strip chart recorders (or computer points, if required). Using this data, and the power readings obtained prior to and after the load swings, the average measured Doppler coefficient verification factor ($C_m$) was calculated, where $C_m$ is effectively the measured ratio of change in $T_{avg}$ vs. change in reactor power. This value of $C_m$ is then compared with the predicted Doppler coefficient verification factor ($C_p$). $C_p$ is calculated using the Doppler only power coefficient ($\alpha_{DOP}$), the isothermal temperature coefficient ($\alpha_{ISO}$), and the difference ($\delta$) in the measured isothermal temperature coefficient ($\alpha_{ISO\ M}$ - as determined in procedure ETT-ZZ-07071) versus the predicted isothermal temperature coefficient at ARO,HZP. The values of $C_m$ and $C_p$ are compared to determine if the acceptance criteria were met.

For the 30% plateau, the difference between $C_m$ and $C_p$ was 0.61°F/% which exceeded the acceptance criteria. Since this was not a safety review criteria and the procedure was to be repeated at 50% power, a retest was not performed.

Testing at the 50% plateau determined the difference in $C_m$ and $C_p$ to be 0.41°F/% which satisfied the acceptance criteria and cleared the deficiency for the 30% plateau. Further testing at the 75% power level determined the difference in $C_m$ and $C_p$ to be 0.165°F/% (as determined by Westinghouse analysis). Testing at 90% power determined the difference in $C_m$ and $C_p$ to be 0.07°F/%. Thus as power was increased, the agreement between predicted and measured values improved.

## 5.9 PSEUDO ROD EJECTION AND PSEUDO ROD DROP AT POWER (S-07SF09)
Procedure ETT-SF-07091, ETT-SF-07092

### OBJECTIVES

The objectives of the subject procedures were:

1. To obtain a full core flux map at approximately 30% Rated Thermal Power (RTP) with Control Bank D (CBD) near the Hot Full Power (HFP) insertion limit.

2. To determine the integral reactivity worth of the predicted worst case ejected rod at approximately 30% RTP.

3. To obtain a full core flux map at approximately 30% RTP with the predicted worst case ejected rod fully withdrawn and CBD positioned near the HFP insertion limit.

4. To obtain a full core flux map at approximately 50% RTP with CBD at greater than or equal to 210 steps.

5. To obtain a full core flux map at approximately 50% RTP with the predicated worst case dropped rod fully inserted and CBD at greater than or equal to 210 steps.

6. To observe the response of incore instrumentation to a power anomaly caused by a control rod below normal bank alignment.

### SUMMARY OF RESULTS

The Acceptance Criterion for the subject procedures was that the measured integral reactivity worth of pseudo ejected rod D-12 with CBD near the HFP insertion limit, increased by a factor of 10%, is less than or equal to 23 pcm. The test resulted in a value of 14 pcm, thus meeting the acceptance criterion.

### DISCUSSION

At approximately 30% RTP, with CBD near the HFP insertion limit, a full core flux map was performed.

The worth of the pseudo ejected rod was measured as follows. The control rod lift coil disconnect switches for all rods in CBD except D-12 were placed in the disconnect position. Control Bank D was withdrawn, moving only rod D-12, to the fully withdrawn position. This caused a corresponding increase in Tavg, balancing the positive reactivity addition by rod withdrawal. After Tavg stabilized, rod D-12 was inserted to its original position. The measured change in Tavg before and after rod withdrawal was multiplied by a best estimate isothermal temperature coefficient, yielding the worth of pseudo ejected rod D-12.

Rod D-12 was then borated to the fully withdrawn position and a full core flux map was performed.

At approximately 50% RTP, with CBD at greater than 210 steps, a full core flux map was performed.

The control rod lift coil disconnect switches for all rods in Shutdown Bank E except D-8 were placed in the disconnect position. Reactor Coolant System dilution was initiated and Shutdown Bank E was inserted, moving only rod D-8. At intervals of approximately 50 steps of rod insertion, a one-pass partial flux map was obtained to observe the response of incore instrumentation. With rod D-8 fully inserted, a full core flux map was performed.

Flux map results are presented in section 4.7 of the Startup Report.

5.10 AUTOMATIC REACTOR CONTROL (S-07SF05)
Procedure FTT-SF-07050

OBJECTIVES

The objective of this procedure was to verify the performance of the automatic reactor control system in maintaining reactor coolant average temperature within acceptable steady state limits.

SUMMARY OF RESULTS

The acceptance criteria for the automatic reactor control test were:

1. No manual intervention is required to bring plant conditions to equilibrium values following initiation of a 6°F temperature transient.

2. Tavg returns to within ±1.5°F of Tref following initiation of a 6°F temperature transient.

3. Rod motion is inhibited by application of the appropriate inhibit inputs.

As demonstrated by the data given in Tables 5.10.1 and 5.10.2, taken from the analysis of strip chart recordings, the performance of the reactor rod control met all acceptance criteria and no instability or inadequacy in the response of the system was noted. As a result, no adjustments to the rod control system were required.

DISCUSSION

Two multi-channel strip chart recorders were used to record such parameters as Tavg, rod speed demand voltage, steam pressure, and pressurizer pressure during automatic operation of the rod control system. Initially, Tavg was matched to the reference temperature (which acts as the setpoint). The control system was then placed in automatic mode to test its ability to maintain a stable temperature within ±1.5°F of Tref. When satisfactory steady state operation was verified, the rod control system was subjected to temperature transients both to determine if recovery to within ±1.5°F of Tref would occur and to evaluate the stability of the system during the recovery. The first transient consisted of increasing Tavg to 6°F above the reference temperature, then putting the system in automatic mode. A recovery to the reference temperature was made in approximately 120 seconds with no instability noted. The second transient was similar except Tavg was decreased to 6°F below Tref. Recovery occurred within approximately 90 seconds in a controlled manner with no abnormal indications. Because no problems or abnormalities of behavior were identified, no adjustments to the system were necessary.

TABLE 5.10.1
AUTOMATIC REACTOR CONTROL

TRANSIENT-1: Tavg INCREASE

| | | |
|---|---|---|
| Tavg INCREASE | 6°F | (peak to peak) |

| Pressurizer Pressure Response | |
|---|---|
| Initial Pressure (psig) | 2240 |
| Minimum Pressure (psig) | 2238 |
| Maximum Pressure (psig) | 2245 |
| Final Pressure (psig) | 2238 |

| Pressurizer Level Response | Level | Setpoint |
|---|---|---|
| Initial Level | 28 | 27.5 |
| Maximum Level | 36 | 37.5 |
| Minimum Level | 28 | 27.5 |
| Final Level | 36 | 37.5 |

| | |
|---|---|
| Steam Generator Level behavior: | No anomalous changes |
| Oscillation Amplitude (°F) | 0 |
| Period (Sec.) | approximately infinite |
| Time to reach Tref ±1.5°F | approximately 120 sec. |

TABLE 5.10.2
AUTOMATIC REACTOR CONTROL
TRANSIENT-2: Tavg DECREASE

| | | |
|---|---|---|
| Tavg DECREASE | | 6°F (peak to peak) |
| Pressurizer Pressure Response | | |
| Initial Pressure (psig) | 2230 | |
| Minimum Pressure (psig) | 2193 | |
| Maximum Pressure (psig) | 2245 | |
| Final Pressure (psig) | 2245 | |
| Pressure Level Response | Level | Setpoint |
| Initial level (%) | 45 | 43.5 |
| Maximum level (%) | 45.3 | 44 |
| Minimum level (%) | 36 | 34.5 |
| Final level (%) | 33.5 | 33 |
| Steam Generator level behavior: | No anomalous changes | |
| Oscillation Amplitude (°F) | 2 | |
| Period (Sec.) | approximately 300 sec. | |
| Time to reach Tref ±1.5°F | approximately 90 sec. | |

## 5.11 OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE INSTRUMENTATION (S-07SF06)

Procedure ETT-SF-07060

### OBJECTIVES

The objectives of this test were to align the $\Delta T$ and $T_{avg}$ process instrumentation under isothermal conditions prior to initial criticality, to align this instrumentation at power levels up to approximately 75% power, to extrapolate values to 100% power and to check the alignment of the $\Delta T$ and $T_{avg}$ instrumentation at 100% power.

### SUMMARY OF RESULTS

The results of measurements taken for this test are shown on Table 5.11.1.

### DISCUSSION

This test was performed by taking R/E converter data for $T_{hot}$, $T_{cold}$, $\Delta T$ and $T_{avg}$ and by taking spare $T_{hot}$ and $T_{cold}$ RTD readings. These readings were taken prior to initial criticality, at 30%, 50%, 75%, 90%, and 100% power. R/E converter data for $T_{hot}$ and $T_{cold}$ were compared to spare $T_{hot}$ and $T_{cold}$ RTD readings. Also, a $T_{avg}$ was calculated from $T_{hot}$ and $T_{cold}$ and compared to the R/E converter reading of $T_{avg}$.

Prior to 90% power, the data for $\Delta T$ and $T_{avg}$ was extrapolated to 100%. Based on the $\Delta T$ extrapolation, the $\Delta T$ summators were rescaled. The $T_{avg}$ for loop 3 of 588.7°F exceeded the maximum 100% $T_{avg}$ of 588.5°F, so this was rechecked using the 90% data. Using this data, the 100% extrapolated $\Delta T$'s and $T_{avg}$'s were recalculated. The 100% $\Delta T$'s did not exceed the 59.4°F limit; however, the loop 3 $T_{avg}$ of 588.522°F still exceeded the 588.5°F limit. It was determined that this was due to a problem with the $T_{hot}$ readings for loop 3. When the data from the $T_{hot}$ R/E converter was compared to the spare $T_{hot}$ RTD it was discovered that errors had been introduced by normalization factors applied to this RTD and its associated spare. Westinghouse reviewed this apparent discrepancy and stated that the value of 588.5°F applies to the average $T_{avg}$, and individual loop $T_{avg}$'s are expected to vary above and below the average $T_{avg}$ by as much as 1°F. Also, the excess of 0.022°F is insignificant considering the number of significant digits (especially since this is a projected value). In addition, Westinghouse stated, that although the active and spare hot leg RTD's for loop 3 varied by more than 1.2°F no recalibration need be performed since the $T_{avg}$ is acceptable and since Westinghouse and the RdF Co. (the RTD manufacturer) are still reviewing the previously mentioned normalization values.

The only other problem encountered was with the loop 4 $T_{cold}$ RTD. It was not functioning at 30%, 50% and 75% power. It was repaired and no problems were encountered at 90% and 100% power.

TABLE 5.11.1
OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE INSTRUMENTATION

| Parameters (Units) | STATEPOINT (% POWER) | | | | | |
|---|---|---|---|---|---|---|
| | Prior to Criticality | 30% | 50% | 75% | 90% | 100% |
| Thermal Power (%) | N/A | 27 | 41.9 | 69.8 | 88.3 | 100 |
| $T_{hot}$ (°F) | | | | | | |
| Loop 1 | 556.1 | 572.3 | 581.1 | 600.4 | 609.4 | 615.3 |
| Loop 2 | 555.9 | 571.6 | 579.6 | 599.3 | 607.9 | 614.3 |
| Loop 3 | 556.1 | 571.6 | 580.4 | 599.8 | 608.5 | 616.3 |
| Loop 4 | 555.5 | 571.8 | 580.3 | 599.9 | 608.4 | 614.5 |
| $T_{cold}$ (°F) | | | | | | |
| Loop 1 | 555.9 | 554.8 | 556.3 | 559.1 | 559.2 | 560.2 |
| Loop 2 | 556.5 | 555.0 | 555.8 | 559.0 | 558.8 | 559.6 |
| Loop 3 | 555.8 | 555.0 | 556.5 | 559.3 | 559.5 | 560.0 |
| Loop 4 | 556.2 | 32.2* | * | 38.5* | 559.3 | 559.7 |
| $T_{avg}$ (°F) | | | | | | |
| Loop 1 | 555.9 | 560.1 | 568.3 | 579.8 | 584.3 | 587.9 |
| Loop 2 | 556.2 | 559.7 | 567 8 | 579.2 | 583.6 | 587.0 |
| Loop 3 | 556.0 | 560.1 | 569.1 | 580.4 | 584.8 | 588.3 |
| Loop 4 | 555.9 | 559.7 | 568.3 | 579.5 | 585.9 | 587.2 |

* TE-440 B, Loop 4 $T_{cold}$, was not functioning and was repaired.

## 5.12 STARTUP ADJUSTMENTS OF REACTOR CONTROL SYSTEM (S-07SF07)
Procedure ETT-SF-07070

### OBJECTIVES

The objective of this test was to determine the Tavg program which would result in design steam pressure at full load and therefore optimum plant efficiency without exceeding pressure limitations for the turbine or temperature limits for the maximum allowable Tavg. This was to be accomplished by making adjustments to the reference Tavg (Tref) programming module as required.

### SUMMARY OF RESULTS

Data for this test was obtained at Hot Standby conditions (no load, Tavg at 557+0, -5°F), 30% full reactor power, 50%, 75% and 100% full reactor power.

Acceptance criteria for this test stipulated that full load Tavg could not exceed 592.5°F. This criterion was satisfactorily met.

It was also stipulated that full load steam generator pressure must have been determined to be within ±10 psi of the design pressure of 1000 psia (985 psig). This criterion was also satisfactorily met.

No adjustments to the Tref programming module were necessary at the 50%, 75%, and 100% power levels. Table 5.12.1 summarizes the data obtained in meeting the above acceptance criteria.

### DISCUSSION

This test was performed in conjunction with the Thermal Power Measurement and Steady-state Data Collection test procedures (ETT-SC-07030 series) at the 30%, 50% and 75% power levels. ETT-SF-07060 was utilized to obtain 100% reactor power data.

At the conclusion of obtaining data at the 50% (46.7% actual) power level, it was verified that actual turbine impulse pressure was within ±50 psi of the design impulse pressure at 50% (46.7% actual) power. Therefore, no gain adjustment was required relative to the Tref programming module. The extrapolated Tavg at 100% power was determined to be 588.5°F, resulting in no adjustments to the Tavg module.

Utilizing average steam generator pressures obtained at the 0%, 30%, and 50% plateaus, steam generator pressure was extrapolated up to 100% reactor power. Upon completion of the thermal power measurement test at 75% (69.8% actual) power (ETT-SC-07033), it was determined that the actual average steam generator pressure was within ±10 psi of the extrapolated pressure at the same power level, 1020.3 versus 1022 psia, respectively.

Upon completion of ETT-SF-07060, 100% (96.77% actual) power data was analyzed. Impulse pressure was not within 10 psi of the design impulse pressure at 100% (96.77% actual) reactor power. However,

because Tref was already at its maximum (588.5°F extrapolated), and because the vendor did not require the actual impulse pressure to be within 10 psi of the design pressure, the scaling factors were not adjusted. Adjustment would have resulted in lowering Tavg and steam generator pressure and subsequently overall plant efficiency.

With reactor power at 99.91% (based on calorimetric), it was determined that average steam generator pressure was within the design pressure, 1000.5 psia versus 1000 ±10 psia respectively. The acceptance criterion was therefore satisfied.

Utilizing the methodology of ETT-SF-07060, Tavg at 100% reactor power was extrapolated based on data obtained at 0.0%, 41.89%, 69.8%, 81.14%, 88.27% and 96.77% reactor power. The extrapolated value, considering all loops, was 588.5°F. Considering the 1.5°F deadband associated with the rod control system, it was then projected that the maximum Tavg would not exceed 590.0°F which is below the safety review criteria upper limit of 592.5°F.

TABLE 5.12.1

SUMMARY OF STARTUP ADJUSTMENTS
TO REACTOR CONTROL SYSTEM

1. AC-JY505, AC-JY505A, AC-TY505

   No adjustments required

2. Reactor Power Level versus Average Steam Generator Pressure and Impulse Pressure

| Power (%) | Avg. Pressure (psia) | Impulse Pressure (psia) |
|---|---|---|
| 0 | 1071 | N/A |
| 27.0 | 998 | 187 |
| 46.7 | 1029 | 326.2 |
| 69.8 | 1020 | 485 |
| 96.8 | 1003.8 | 694.5 |
| 99.9 | *1000.5 | 712.7 |

*Acceptance Criteria 1000±10 psia

3. Extrapolated Tavg at 100% Power

   **Tavg = 588.5 ± 1.5°F deadband = 590.0°F

   **Acceptance Criteria = 592.5°F Maximum

## 5.13 INCORE/EXCORE CALIBRATION (S-07SE03)
Procedure ESP-ZZ-00006

### OBJECTIVES

The objectives of this procedure were to determine the relationship between the excore power range detector currents and incore axial offset, and to provide calibration data for the Nuclear Instrumentation System (NIS) top and bottom detector isolation amplifiers.

### SUMMARY OF RESULTS

Acceptance criteria for this procedure are:

1. Calibration data has been obtained for NIS top and bottom detector isolation amplifiers.

2. The NIS top and bottom isolation amplifiers have been calibrated by I&C.

These acceptance criteria were met satisfactorily.

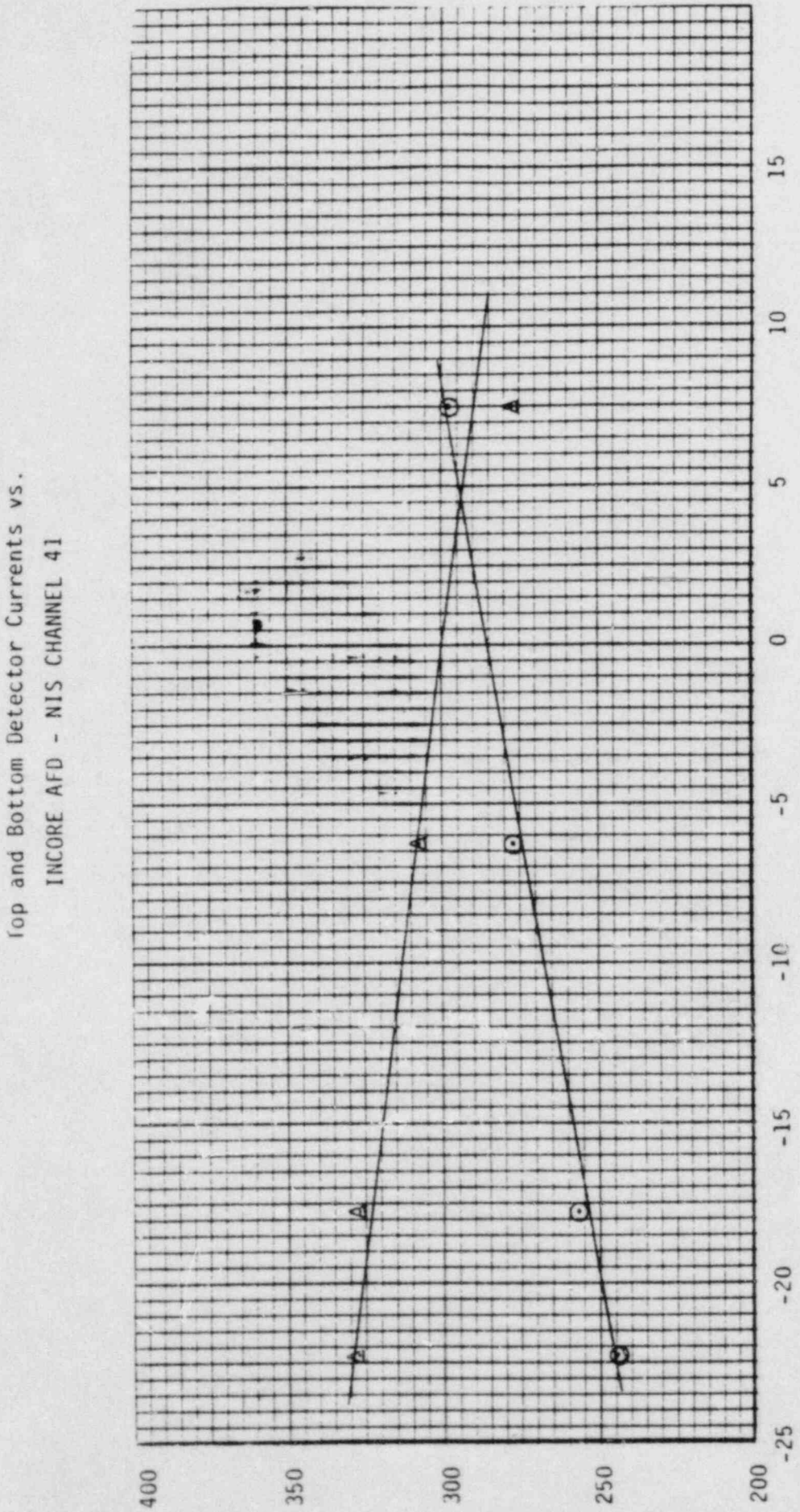
### DISCUSSION

The power range nuclear instrumentation system produces an output proportional to the currents in the top and bottom detectors of each channel. These signals are used for delta flux indication, upper and lower quadrant power indication, NSSS computer monitoring of delta flux, and generation of the delta flux penalty to the OT$\Delta$T trip setpoint. Calibration of the isolation amplifiers associated with each detector is required so that excore indication reflects actual core conditions as measured by incore flux maps.

With power at approximately 75%, axial offset was varied by changing the position of the control banks (as compensation for boration or dilution of the RCS). Flux maps were taken at different axial offsets. Concurrent with the flux maps, calorimetrics were run and detector currents were recorded. Axial offsets for the 4 flux maps were -6.35%, -17.60%, -22.37%, and +7.51%.

From the data taken during the flux maps, normalized detector currents were calculated. These were plotted with incore axial flux differences on the horizontal axis, and a least squares fit was used to calculate the relationship between them (See Figures 5.13.1 - 5.13.4). From this relationship, expected detector currents at 100% power with 0% axial flux difference were obtained. This was used by the I&C Department to calibrate the upper and lower isolation amplifiers for the power range detectors.

Figure 5.13.1


Top and Bottom Detector Currents vs.
INCORE AFD - NIS CHANNEL 41
Detector Current (μA)
400
350
300
250
200
-25
-20
-15
-10
-5
0
5
10
15
INCORE AFD (%)

Top and Bottom Detector Currents vs.

INCORE AFD - NIS CHANNEL 42

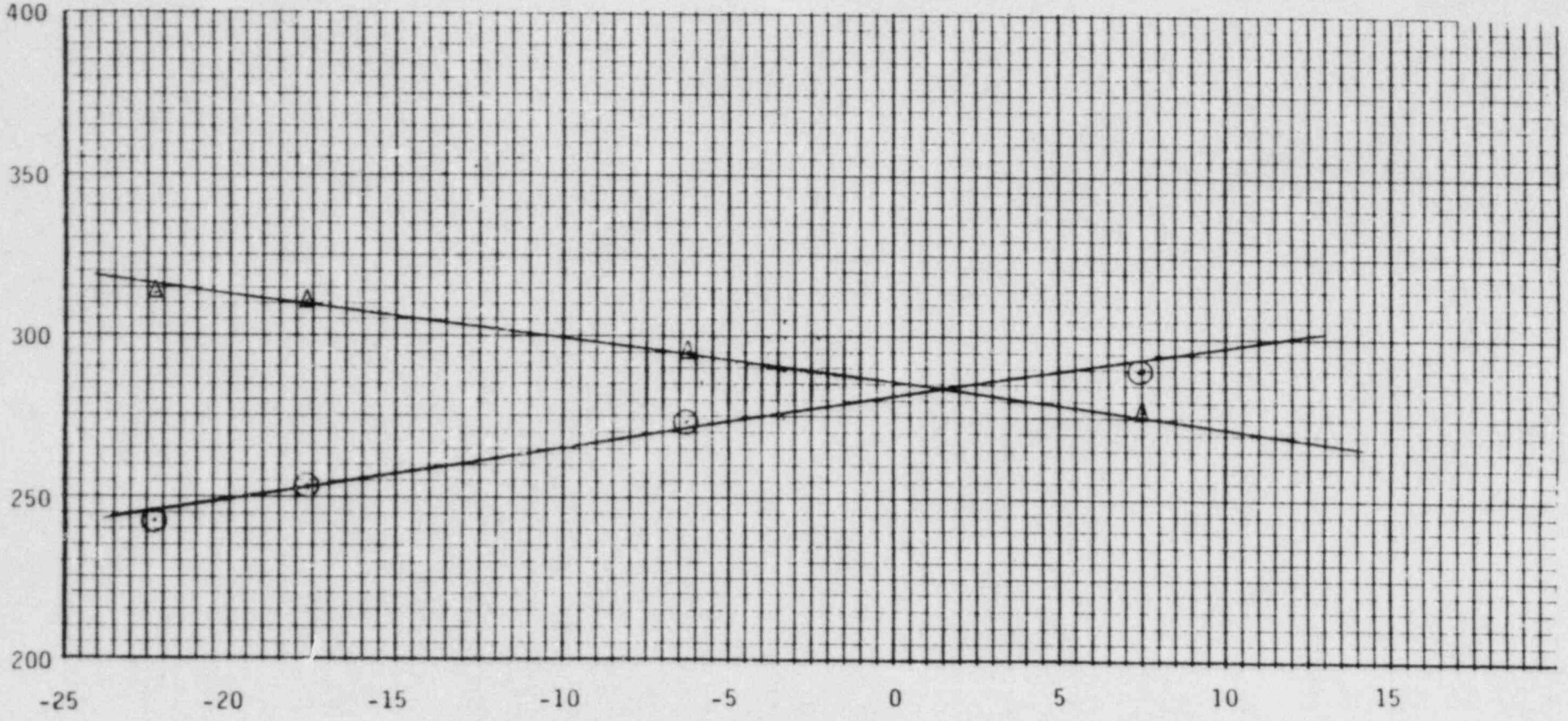
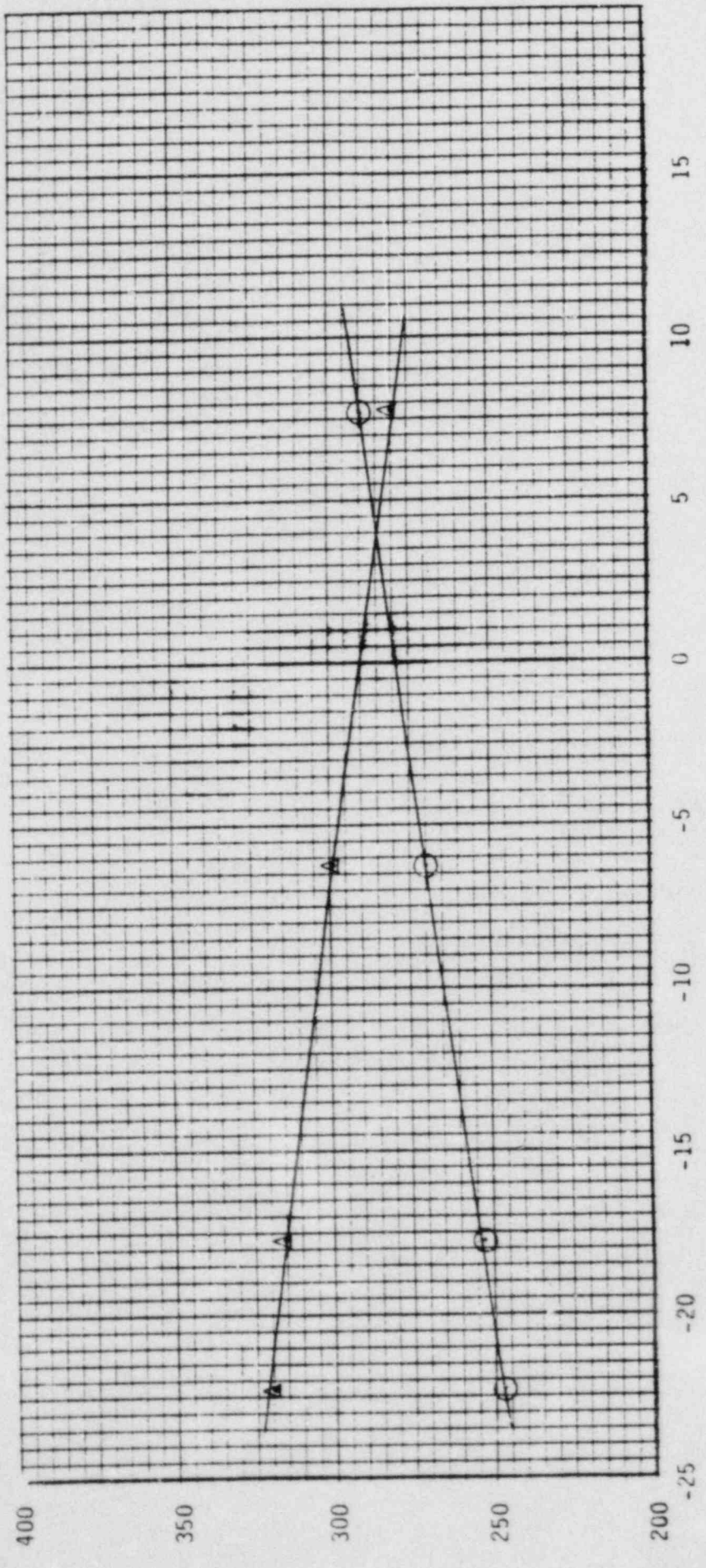



Figure 5.13.2

Figure 5.13.3


Top and Bottom Detector Currents vs.
INCORE AFD - NIS CHANNEL 43
Detector Current (μA)
400
350
300
250
200
15
10
5
0
-5
-10
-15
-20
-25
INCORE AFD (%)

Top and Bottom Detector Currents vs.
INCORE AFD - NIS CHANNEL 44

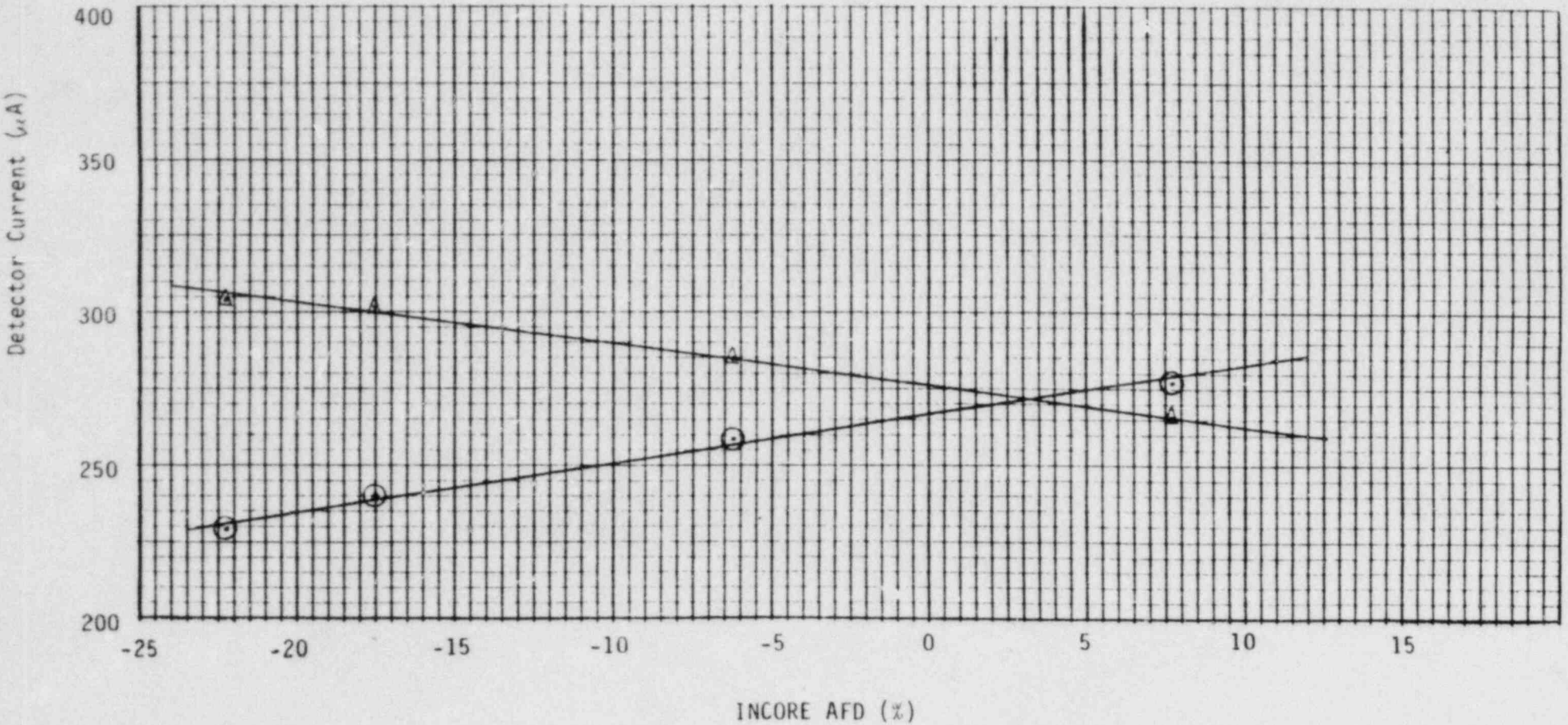



Figure 5.13.4

5.14 LOAD SWING TESTS (S-070009)
Procedures ETT-ZZ-07091, ETT-ZZ-07092, ETT-ZZ-07093

OBJECTIVES

The objective of these tests was to verify proper nuclear plant transient response, including automatic control performance at the 30%, 75%, and 100% nominal power levels when 10% step load changes were introduced at the turbine generator.

SUMMARY OF RESULTS

The following criteria were specified for these tests:

1. Reactor and turbine are not tripped.
2. Safety injection is not initiated.
3. Neither steam generator power-operated nor safety valves lift.
4. No manual intervention is required to bring plant conditions to a steady state.
5. Plant variables do not incur sustained or diverging oscillations.
6. Nuclear power undershoot/overshoot is less than 3% for the load decrease/increase.

The initial and final conditions of recorded plant variables in response to a 10% load decrease followed by a 10% load increase beginning at the 30%, 75%, 90% (Retest) and 100% power levels is presented in Table 5.14.1.

During the 30% power load swing test, all acceptance criteria were met with the exception of 4 and 6 above, during the load increase.

At the onset of the 10% decrease from 75% nominal reactor power, acceptance criterion 4 above could not be met which resulted in a load decrease retest at 90% nominal reactor power. Criterion 4 was then satisfied. The 10% load increase acceptance criteria back to the 75% power level were satisfied.

During the 100% power load swing test, all acceptance criteria were satisfied.

DISCUSSION

The load swing tests were conducted per procedure by basically making rapid 10% turbine generator load decreases and increases via the turbine Stand-by Load Set. During the course of the load decreases/increases, strip chart recordings of essential plant parameters were taken so that their response could be analyzed and compared to the stabilized recorded values previous to and after the load swings.

In addition to the acceptance criteria specified above (1-6), the following parameters were observed to determine whether or not vendor review was required if the indicated margin was exceeded:

-Primary pressure swings less than ±50 psi.
-Steam generator level swings less than ±10% of the initial level.
-Steam generator pressure overshoots/undershoots the final value by less than ±25 psi.
-Minimum delta T margin to trip occurs in Steady State before load decrease/after load increase.
-Tavg doesn't undershoot/overshoot its final value.

For the 30% power load swing, acceptance criteria were satisfied during the load decrease. During the load increase back to 30% power from 20% power, manual intervention was required due to Xe build-up after the load decrease. With Bank D starting at 164 steps, the load increase was induced. With Bank D at 228 steps (fully withdrawn) and reactor power at approximately 27.5%, the reactor operator had to begin dilution to override the Xe build-up. It was therefore not possible to verify if nuclear power overshoot was less than 3%. Subsequent satisfactory performance of ETT-ZZ-07092 during the load increase up to 75% reactor power closed this deficiency.

Several parameters exceeded their predicted values during the 30% power load swing which required vendor review. Steam generator level deviated to -11% (Loop 4) during the load decrease. Westinghouse considered this deviation to be no restraint to power escalation above 30%. When the 10% load increase (7.5% actual) parameters were analyzed, pressurizer pressure was observed to decrease by -108 psig; additionally, all steam generator levels deviated between +11% and -12%. These two load increase deviations were reviewed by the NSSS Vendor and found attributable to a Tavg decreasing transient which occurred primarily due to discontinued rod movement at the C-11 control interlock auto rod stop. The vendor considered these two deviations to be no restraint to continued power escalation.

During the 75% power load swing test, a rapid decrease of 10% was not observed due to a 7°F error artificially existing between Tref and auctioneered Tave. This error automatically armed the steam dumps and produced an excessive load on the reactor. Manual intervention was necessary to reset the C-7 control interlock, and subsequent load reduction (10%) was completed. As the load was increased from 65% power up to 75% power, it was observed that all acceptance criteria were met.

Based on satisfactory response of the rod control system and plant response, it was determined to retest load reduction at the 90% power level after the Tref summing amplifier was recalibrated. Results of the 90% load decrease (actually started at 88% and went to 76.5%), met all design review criteria. The additional 1.5% power drop was caused by a 1.5°F error between Tave and Tref.

During the load decrease (75% to 65% power), several parameters exceeded their predicted values. Due to the artificial error (Tave/Tref) of 7°F, and subsequent steam dump opening, ΔT began increasing and Tave dropped. Due to the decrease in Tave, the ΔT margin to trip increased above minimum and Tave undershot its predicted value. During the subsequent load increase (65% to 75% power), the response of secondary pressure was sluggish due to the existing Tave/Tref error.

Westinghouse representatives saw no problem due to the parameter abnormalities which would preclude further power escalation above 75% reactor power. During the 88% to 76.5% load reduction, these same parameters were within their predicted range.

The final load swing test (100% to 90% to 100%) was performed. All acceptance criteria were satisfactorily met. Additionally, the observed parameters all fell within the predicted range requiring no notification to and evaluation by the vendor.

TABLE 5.14.1
10% LOAD SWING DECREASE/INCREASE SUMMARIES

A. 10% LOAD DECREASE FROM 30% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 284 | 170 |
| Power - Nuclear | (%) | 29.5 | 18 |
| $T_{avg}$ (auctioneered) | | 564.5 | 561 |
| $T_{ref}$ | (F°) | 565 | 562 |
| ΔT (% FULL POWER) | | 34 | 22 |
| ΔT Overpower Setpoint (% FULL POWER) | | 100 | 109 |
| ΔT Overtemp. Setpoint (% FULL POWER) | | 144 | 148 |
| Pressurizer Pressure | (psig) | 2235 | 2235 |
| Pressurizer Level | (%) | 34 | 32 |
| Steam Header Pressure | (psig) | 1000 | 1015 |
| Steam Flow - Loop 1 | (MPPH) | 1.1 | .8 |
| Steam Flow - Loop 2 | (MPPH) | .95 | .62 |
| Steam Flow - Loop 3 | (MPPH) | 1.1 | .8 |
| Steam Flow - Loop 4 | (MPPH) | .95 | .8 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 50 | 51 |
| Steam Generator Level-Loop 4 | (%) | 50 | 51 |
| Feedwater Temperature-Loop 1 | (°F) RE | 303.2 | 2612 |
| Feedwater Temperature-Loop 2 | (°F) RE | 302 | 267 |
| Feedwater Temperature-Loop 3 | (°F) RE | 304 | 269 |
| Feedwater Temperature-Loop 4 | (°F) RE | 125 | 268 |
| Feedwater Flow - Loop 1 | (MPPH) | .75 | .3 |
| Feedwater Flow - Loop 2 | (MPPH) | .6 | .25 |
| Feedwater Flow - Loop 3 | (MPPH) | .9 | .42 |
| Feedwater Flow - Loop 4 | (MPPH) | .8 | .4 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1000 | 1160 |
| Feed Pump A Speed | (RPM) | 3650 | 3600 |
| Feed Pump B Speed | (RPM) | 0 | 0 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 205 | 152 |
| Boron Concentration | (PPM) | 1103 | 1163 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANGE: 3 MINUTES

TABLE 5.14.1 (cont.)

10% LOAD SWING DECREASE/INCREASE SUMMARIES

B. 10% LOAD INCREASE FROM 20% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 172 | 280 |
| Power - Nuclear | (%) | 19.4 | 29.5 |
| $T_{avg}$ (auctioneered) | | 563 | 565 |
| $T_{ref}$ | (F°) | 562 | 564 |
| $\Delta T$ (% FULL POWER) | | 22 | 32 |
| $\Delta T$ Overpower Setpoint (% FULL POWER) | | 109 | 110 |
| $\Delta T$ Overtemp. Setpoint (% FULL POWER) | | 147 | 144 |
| Pressurizer Pressure | (psig) | 2240 | 2240 |
| Pressurizer Level | (%) | 33 | 36 |
| Steam Header Pressure | (psig) | 1040 | 1020 |
| Steam Flow - Loop 1 | (MPPH) | .8 | 1.06 |
| Steam Flow - Loop 2 | (MPPH) | .6 | 1.01 |
| Steam Flow - Loop 3 | (MPPH) | .7 | 1.10 |
| Steam Flow - Loop 4 | (MPPH) | .65 | 0.98 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 50 | 50 |
| Steam Generator Level-Loop 4 | (%) | 50 | 50 |
| Feedwater Temperature-Loop 1 | (°F) RE | 267.7 | 293.6 |
| Feedwater Temperature-Loop 2 | (°F) RE | 266 | 292 |
| Feedwater Temperature-Loop 3 | (°F) RE | 268 | 294 |
| Feedwater Temperature-Loop 4 | (°F) RE | 185? | 125 |
| Feedwater Flow - Loop 1 | (MPPH) | .31 | .72 |
| Feedwater Flow - Loop 2 | (MPPH) | .21 | .60 |
| Feedwater Flow - Loop 3 | (MPPH) | .45 | .80 |
| Feedwater Flow - Loop 4 | (MPPH) | .35 | .72 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1150 | 1110 |
| Feed Pump A Speed | (RPM) | 3600 | 3700 |
| Feed Pump B Speed | (RPM) | 0 | 0 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 163 | 154 |
| Boron Concentration | (PPM) | 1113 | 1113 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANGE: 5 MINUTES

TABLE 5.14.1 (cont.)
10% LOAD SWING DECREASE/INCREASE SUMMARIES

C. 10% LOAD DECREASE FROM 75% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 850 | 755 |
| Power - Nuclear | (%) | 75 | 65 |
| $T_{avg}$ (auctioneered) | | 582 | 579 |
| $T_{ref}$ | (F°) | 581 | 571 |
| ΔT (% FULL POWER) | | 81% | 75% |
| ΔT Overpower Setpoint (% FULL POWER) | | 110 | 110 |
| ΔT Overtemp. Setpoint (% FULL POWER) | | 121 | 128 |
| Pressurizer Pressure | (psig) | 2235 | 2240 |
| Pressurizer Level | (%) | 58 | 48 |
| Steam Header Pressure | (psig) | 1000 | 1000 |
| Steam Flow - Loop 1 | (MPPH) | 2.9 | 2.6 |
| Steam Flow - Loop 2 | (MPPH) | 2.6 | 2.3 |
| Steam Flow - Loop 3 | (MPPH) | 2.9 | 2.6 |
| Steam Flow - Loop 4 | (MPPH) | 2.8 | 2.5 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 50 | 50 |
| Steam Generator Level-Loop 4 | (%) | 50 | 50 |
| Feedwater Temperature-Loop 1 | (°F) RE | 416 | 407 |
| Feedwater Temperature-Loop 2 | (°F) RE | 414 | 405 |
| Feedwater Temperature-Loop 3 | (°F) RE | 416 | 407 |
| Feedwater Temperature-Loop 4 | (°F) RE | 415 | 405 |
| Feedwater Flow - Loop 1 | (MPPH) | 2.8 | 2.45 |
| Feedwater Flow - Loop 2 | (MPPH) | 2.6 | 2.4 |
| Feedwater Flow - Loop 3 | (MPPH) | 2.9 | 2.55 |
| Feedwater Flow - Loop 4 | (MPPH) | 2.7 | 2.4 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1160 | 1140 |
| Feed Pump A Speed | (RPM) | 4080 | 3980 |
| Feed Pump B Speed | (RPM) | 4050 | 3950 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 181 | 150 |
| Boron Concentration | (PPM) | 921 | 921 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANGE: 1 MINUTES

TABLE 5.14.1 (cont.)

10% LOAD SWING DECREASE/INCREASE SUMMARIES

D. 10% LOAD INCREASE FROM 65% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 755 | 855 |
| Power - Nuclear | (%) | 65 | 75 |
| $T_{avg}$ (auctioneered) | | 579 | 580 |
| $T_{ref}$ | (F°) | 571 | 581 |
| $\Delta T$ (% FULL POWER) | | 75% | 84 |
| $\Delta T$ Overpower Setpoint (% FULL POWER) | | 110 | 110 |
| $\Delta T$ Overtemp. Setpoint (% FULL POWER) | | 128 | 124 |
| Pressurizer Pressure | (psig) | 2240 | 2235 |
| Pressurizer Level | (%) | 48 | 57 |
| Steam Header Pressure | (psig) | 1000 | 1000 |
| Steam Flow - Loop 1 | (MPPH) | 2.6 | 2.9 |
| Steam Flow - Loop 2 | (MPPH) | 2.3 | 2.65 |
| Steam Flow - Loop 3 | (MPPH) | 2.6 | 2.9 |
| Steam Flow - Loop 4 | (MPPH) | 2.5 | 2.8 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 50 | 50 |
| Steam Generator Level-Loop 4 | (%) | 50 | 50 |
| Feedwater Temperature-Loop 1 | (°F) RE | 407 | 417 |
| Feedwater Temperature-Loop 2 | (°F) RE | 405 | 415 |
| Feedwater Temperature-Loop 3 | (°F) RE | 407 | 417 |
| Feedwater Temperature-Loop 4 | (°F) RE | 405 | 415 |
| Feedwater Flow - Loop 1 | (MPPH) | 2.45 | 2.8 |
| Feedwater Flow - Loop 2 | (MPPH) | 2.4 | 2.6 |
| Feedwater Flow - Loop 3 | (MPPH) | 2.55 | 2.9 |
| Feedwater Flow - Loop 4 | (MPPH) | 2.4 | 2.7 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1140 | 1150 |
| Feed Pump A Speed | (RPM) | 3980 | 4100 |
| Feed Pump B Speed | (RPM) | 3950 | 4100 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 150 | 194.5 |
| Boron Concentration | (PPM) | 921 | 921 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANGE: 3 MINUTES

TABLE 5.14.1 (cont.)
10% LOAD SWING DECREASE/INCREASE SUMMARIES

E. 10% LOAD DECREASE FROM 90% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 994 | 865 |
| Power - Nuclear | (%) | 88 | 76.5 |
| $T_{avg}$ (auctioneered) | | 586 | 583 |
| $T_{ref}$ | (F°) | 586 | 583 |
| ΔT (% FULL POWER) | | 88 | 77 |
| ΔT Overpower Setpoint (% FULL POWER) | | 109 | 99 |
| ΔT Overtemp. Setpoint (% FULL POWER) | | 118 | 121 |
| Pressurizer Pressure | (psig) | 2235 | 2235 |
| Pressurizer Level | (%) | 61 | 56 |
| Steam Header Pressure | (psig) | 995 | 995 |
| Steam Flow - Loop 1 | (MPPH) | 3.4 | 3.1 |
| Steam Flow - Loop 2 | (MPPH) | 3.1 | 2.7 |
| Steam Flow - Loop 3 | (MPPH) | 3.3 | 2.9 |
| Steam Flow - Loop 4 | (MPPH) | 3.3 | 2.9 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 50 | 50 |
| Steam Generator Level-Loop 4 | (%) | 50 | 49 |
| Feedwater Temperature-Loop 1 | (°F) RE | 428.1 | 398.8 |
| Feedwater Temperature-Loop 2 | (°F) RE | 426 | 397 |
| Feedwater Temperature-Loop 3 | (°F) RE | 428 | 403 |
| Feedwater Temperature-Loop 4 | (°F) RE | 426 | 401 |
| Feedwater Flow - Loop 1 | (MPPH) | 3.25 | 2.85 |
| Feedwater Flow - Loop 2 | (MPPH) | 3.1 | 2.65 |
| Feedwater Flow - Loop 3 | (MPPH) | 3.3 | 2.9 |
| Feedwater Flow - Loop 4 | (MPPH) | 3.25 | 2.8 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1160 | 1140 |
| Feed Pump A Speed | (RPM) | 4400 | 4200 |
| Feed Pump B Speed | (RPM) | 4300 | 4100 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 216/215 | 190/181 |
| Boron Concentration | (PPM) | 912 | 912 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANCE: 4.75 MINUTES

TABLE 5.14.1 (cont.)

10% LOAD SWING DECREASE/INCREASE SUMMARIES

F. 10% LOAD DECREASE FROM 100% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 1175 | 1062 |
| Power - Nuclear | (%) | 100 | 90 |
| $T_{avg}$ (auctioneered) | | 587.3 | 586 |
| $T_{ref}$ | (F°) | 588.5 | 585 |
| $\Delta T$ (% FULL POWER) | | 99 | 90 |
| $\Delta T$ Overpower Setpoint (% FULL POWER) | | 108 | 108 |
| $\Delta T$ Overtemp. Setpoint (% FULL POWER) | | 110 | 114 |
| Pressurizer Pressure | (psig) | 2235 | 2235 |
| Pressurizer Level | (%) | 61.5 | 60 |
| Steam Header Pressure | (psig) | 975 | 990 |
| Steam Flow - Loop 1 | (MPPH) | 3.8 | 3.55 |
| Steam Flow - Loop 2 | (MPPH) | 3.7 | 3.3 |
| Steam Flow - Loop 3 | (MPPH) | 3.8 | 3.4 |
| Steam Flow - Loop 4 | (MPPH) | 3.75 | 3.35 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 51 | 51 |
| Steam Generator Level-Loop 4 | (%) | 50 | 50 |
| Feedwater Temperature-Loop 1 | (°F) RE | 440.3 | 431.8 |
| Feedwater Temperature-Loop 2 | (°F) RE | 438 | 430 |
| Feedwater Temperature-Loop 3 | (°F) RE | 440 | 432 |
| Feedwater Temperature-Loop 4 | (°F) RE | * | 430 |
| Feedwater Flow - Loop 1 | (MPPH) | 3.77 | 3.38 |
| Feedwater Flow - Loop 2 | (MPPH) | 3.66 | 3.28 |
| Feedwater Flow - Loop 3 | (MPPH) | 3.83 | 3.47 |
| Feedwater Flow - Loop 4 | (MPPH) | 3.75 | 3.35 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1160 | 995 |
| Feed Pump A Speed | (RPM) | 4750 | 4500 |
| Feed Pump B Speed | (RPM) | 4620 | 4210 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 198 | 159 |
| Boron Concentration | (PPM) | 888 | 888 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANGE: 13 MINUTES

* Computer Point Not Available

TABLE 5.14.1 (cont.)
10% LOAD SWING DECREASE/INCREASE SUMMARIES

G. 10% LOAD INCREASE FROM 90% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 1062 | 1175 |
| Power - Nuclear | (%) | 90 | 100 |
| $T_{avg}$ (auctioneered) | | 586 | 587.5 |
| $T_{ref}$ | (F°) | 585 | 588.5 |
| ΔT (% FULL POWER) | | 90 | 100 |
| ΔT Overpower Setpoint (% FULL POWER) | | 108 | 108 |
| ΔT Overtemp. Setpoint (% FULL POWER) | | 114 | 110 |
| Pressurizer Pressure | (psig) | 2235 | 2235 |
| Pressurizer Level | (%) | 60 | 61.5 |
| Steam Header Pressure | (psig) | 990 | 970 |
| Steam Flow - Loop 1 | (MPPH) | 3.55 | 3.8 |
| Steam Flow - Loop 2 | (MPPH) | 3.3 | 3.7 |
| Steam Flow - Loop 3 | (MPPH) | 3.4 | 3.8 |
| Steam Flow - Loop 4 | (MPPH) | 3.35 | 3.75 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 51 | 51 |
| Steam Generator Level-Loop 4 | (%) | 50 | 51 |
| Feedwater Temperature-Loop 1 | (°F) RE | 431.8 | 440.7 |
| Feedwater Temperature-Loop 2 | (°F) RE | 430 | 439 |
| Feedwater Temperature-Loop 3 | (°F) RE | 432 | 440 |
| Feedwater Temperature-Loop 4 | (°F) RE | 430 | 439 |
| Feedwater Flow - Loop 1 | (MPPH) | 3.38 | 3.87 |
| Feedwater Flow - Loop 2 | (MPPH) | 3.28 | 3.65 |
| Feedwater Flow - Loop 3 | (MPPH) | 3.47 | 3.88 |
| Feedwater Flow - Loop 4 | (MPPH) | 3.35 | 3.79 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 995 | 1060 |
| Feed Pump A Speed | (RPM) | 4500 | 4675 |
| Feed Pump B Speed | (RPM) | 4210 | 4600 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 159 | 213 |
| Boron Concentration | (PPM) | 888 | 888 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANGE: 13 MINUTES

5.15 LARGE LOAD REDUCTION TESTS (S-070010)
Procedure ETT-ZZ-07101, ETT-ZZ-07102

OBJECTIVES

The objectives of these tests were:

1. To verify the ability of the primary plant, secondary plant, and the automatic reactor control systems to sustain a 50% step level reduction from 75% and 100% full power, and

2. To obtain test data for the evaluation of interaction between the control system setpoint changes as required to improve transient response based on active plant operation.

SUMMARY OF RESULTS

The following acceptance criteria were specified:

1. Reactor and turbine do not trip

2. Safety injection is not initiated

3. Pressurizer safety valves do not lift

4. Steam generator safety valves do not lift

5. No manual intervention is required to bring plant conditions to equilibrium values following the transient

In addition to the above acceptance criteria, the following plant parameters were observed to identify the need to contact the NSSS Vendor should one of the values fall outside their predicted range:

- Tavg peaks less than 5°F above the steady state initial value.

- Tavg undershoot is less than 3°F below the steady state final value. No Tave oscillations during steam dump. Oscillations less than 5°F peak-to-peak and decreasing after termination of the steam dump.

- Primary pressure swings less than +80 psi and -100 psi from the initial pressure.

- Steam generator level swings less than 25% from the initial level.

- Maximum rod speed in occurs for about 30 seconds.

- Steam dump actuates and may modulate, but does not cycle on and off.

- Steam dump shuts off within 8 minutes.

During the 50% load reduction from 75% reactor power, all acceptance criteria above were satisfactorily met with the exception of No. 5. Several observed parameters did not fall within their predicted range of value.

During the 50% load reduction from 100% reactor power, all acceptance criteria were satisfactorily met. One of the observed parameters identified above did not meet its predicted time span.

Table 5.15.1 provides a summary of the initial and final plant conditions during conduct of these tests.

DISCUSSION

Large load reduction tests were conducted per procedure by rapidly reducing turbine load 50% from the 75% and 100% reactor power levels respectively. Load reduction was achieved utilizing the Stand-by Load Set. Strip chart recorders and computer trend blocks were employed to obtain preselected data.

The large load reduction of 50% from 75% reactor power was actually initiated from the 72% reactor power level. The procedure required reactor power to be 75 (+0, -5%). When the condenser unavailable control interlock C-9 came on, steam dump actuation was prevented. This resulted in Main Feedwater Control Valve and Steam Generator oscillations requiring the operator to take manual control. Additionally, turbine load was increased from 260 to 300 MWe in order to reduce the oscillations. Acceptance criterion No. 5 above therefore was not met; however, all other acceptance criteria were satisfied.

C-9 came on due to condenser vacuum exceeding the 4.9 inches of Hg absolute setpoint associated with condenser vacuum bistable. Based on engineering evaluation, it was decided to reverify the need to relax the setpoint based on load reduction of 50% from the 100% power level (ETT-ZZ-07102). During conduct of ETT-ZZ-07102, condenser vacuum only reached 4.3 inches of Hg absolute. It was then concluded that due to acceptance criterion No. 5 being satisfied at the 100% load reduction, the same criteria was satisfied for the 75% load reduction.

Several parameters fell outside their predicted ranges during the reduction from 75% power. The first was that the primary pressure swing exceeded its lower limit of -100 psig by 8 psig. The second was that the steam dump did not shut off within 8 minutes. Finally, the steam dump cycled on and off. These were reviewed and were determined to result from the C-9 interlock problem. These parameters were not associated with acceptance criteria and were provided for design review purposes only. A determination was made by the Onsite Review Committee that retest would be performed during the large load reduction from 100% power.

The initial attempt to conduct the large load reduction from 100% reactor power was not completed due to manually opening the PORV block valves during the transient. (PORV block valves are normally closed). This action resulted in reactor trip due to low pressurizer pressure.

ETT-ZZ-07102 was then revised to require the PORV block valves to be opened prior to initiation of the load reduction. The second attempt resulted in all acceptance criteria and the plant parameter predicted values and responses being satisfactorily met.

As mentioned previously, the results of the 100% power load reduction provided satisfactory results to demonstrate that manual intervention, as previously encountered during the performance of ETT-ZZ-07101, was not necessary therefore satisfying acceptance criteria No. 5 for the 75% reactor power large load reduction.

TABLE 5.15.1

LARGE LOAD REDUCTION SUMMARY

A. 50% LOAD REDUCTION FROM 75% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 856 | 300 |
| Power - Nuclear | (%) | 72 | 27 |
| $T_{avg}$ (auctioneered) | | 578 | 568 |
| $T_{ref}$ | (F°) | 577 | 566 |
| ΔT (% FULL POWER) | | 84.3 | 38.7 |
| ΔT Overpower Setpoint (% FULL POWER) | | 110 | 108 |
| ΔT Overtemp. Setpoint (% FULL POWER) | | 117 | 135 |
| Pressurizer Pressure | (psig) | 2248 | 2180 |
| Pressurizer Level | (%) | 57 | 43 |
| Steam Header Pressure | (psig) | 1013 | 1092 |
| Steam Flow - Loop 1 | (MPPH) | 27.8 | 12.2 |
| Steam Flow - Loop 2 | (MPPH) | 26.4 | 9.1 |
| Steam Flow - Loop 3 | (MPPH) | 29.0 | 12.5 |
| Steam Flow - Loop 4 | (MPPH) | 28.3 | 12.5 |
| Steam Generator Level-Loop 1 | (%) | 50 | 52 |
| Steam Generator Level-Loop 2 | (%) | 50 | 56 |
| Steam Generator Level-Loop 3 | (%) | 51 | 54 |
| Steam Generator Level-Loop 4 | (%) | 51 | 56 |
| Feedwater Temperature-Loop 1 | (°F) RE | 416.6 | 342.5 |
| Feedwater Temperature-Loop 2 | (°F) RE | 414.7 | 343.5 |
| Feedwater Temperature-Loop 3 | (°F) RE | 416.7 | 341.9 |
| Feedwater Temperature-Loop 4 | (°F) RE | 415.1 | 344.1 |
| Feedwater Flow - Loop 1 | (MPPH) | 27.8 | 9.6 |
| Feedwater Flow - Loop 2 | (MPPH) | 27.4 | 9.6 |
| Feedwater Flow - Loop 3 | (MPPH) | 29.3 | 11.4 |
| Feedwater Flow - Loop 4 | (MPPH) | * | * |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1084 | 1136 |
| Feed Pump A Speed | (RPM) | 3600 | 4020 |
| Feed Pump B Speed | (RPM) | 0 | 0 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 175 |
| Control Rod Pos. - Bank D | (Steps) | 209 | 78 |
| Boron Concentration | (PPM) | ** | ** |

* NOT RECORDED

** Computer Point Not Functioning

TABLE 5.15.1 (cont.)

LARGE LOAD REDUCTION SUMMARY

B. 50% LOAD REDUCTION FROM 100% REACTOR POWER

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Load (MW) | | 1230(1) | 537 |
| Power - Nuclear | (%) | 100 | 58 |
| $T_{avg}$ (auctioneered) | | 592 | 579.3 |
| $T_{ref}$ | (F°) | 591 | 577 |
| ΔT (% FULL POWER) | | 93 | 63 |
| ΔT Overpower Setpoint (% FULL POWER) | | 108 | 108 |
| ΔT Overtemp. Setpoint (% FULL POWER) | | 108 | 120 |
| Pressurizer Pressure | (psig) | 2197 | 2128 |
| Pressurizer Level | (%) | 61 | 50 |
| Steam Header Pressure | (psig) | 982.5 | 1000 |
| Steam Flow - Loop 1 | (MPPH) | 3.67 | 2.3 |
| Steam Flow - Loop 2 | (MPPH) | 3.55 | 2.14 |
| Steam Flow - Loop 3 | (MPPH) | 3.74 | 2.38 |
| Steam Flow - Loop 4 | (MPPH) | 3.84 | 2.50 |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 50 | 53 |
| Steam Generator Level-Loop 4 | (%) | 50 | 52 |
| Feedwater Temperature-Loop 1 | (°F) RE | 438.6 | 384.7 |
| Feedwater Temperature-Loop 2 | (°F) RE | 436.9 | 383.5 |
| Feedwater Temperature-Loop 3 | (°F) RE | 438.1 | 385.8 |
| Feedwater Temperature-Loop 4 | (°F) RE | 436.8 | 383.7 |
| Feedwater Flow - Loop 1 | (MPPH) | 3.70 | 2.21 |
| Feedwater Flow - Loop 2 | (MPPH) | 3.55 | 2.02 |
| Feedwater Flow - Loop 3 | (MPPH) | 3.65 | 2.21 |
| Feedwater Flow - Loop 4 | (MPPH) | 3.82 | 2.30 |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1172.5 | 1185 |
| Feed Pump A Speed | (RPM) | 5520 | 4640 |
| Feed Pump B Speed | (RPM) | 0 | 0 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 192 |
| Control Rod Pos. - Bank D | (Steps) | 169 | 77 |
| Boron Concentration | (PPM) | 1046 | 1000 (2) |

(1) Computer Trend Block Not Functioning. Valve Recorded on Strip Chart

(2) Computer Point Indication

## 5.16 PLANT TRIP FROM 100% POWER (S-070011)
Procedure ETT-ZZ-07110

### OBJECTIVES

The objectives of this procedure were to verify the ability of the primary and secondary plant to sustain a trip from 100% power and to bring the plant to stable conditions following the transient and to determine the overall response time of the reactor coolant hot leg RTDs. In addition, the data taken during this test was evaluated to determine possible changes in control system settings in order to improve transient response based on actual plant operation.

### SUMMARY OF RESULTS

The acceptance criteria for the 100% power trip test were:

1. All rods must be fully inserted following the trip.
2. The overall hot leg RTD response time determined from recorder traces must be no greater than 6.9 seconds.
3. The pressurizer and steam generator safety valves must not lift.
4. Safety injection is not initiated.
5. Feedwater flow isolated by receipt of a feedwater isolation signal prior to no load $T_{avg}$ being reached.
6. $T_{avg}$ steadies out at or above no load $T_{avg}$ without manual intervention on feed flow.
7. The steam dump valves modulate closed.
8. The minimum pressurizer level stays greater than or equal to 20% of level.
9. The minimum pressurizer pressure is greater than or equal to 2000 psig.
10. Nuclear flux drops to less than or equal to 15% within 2 seconds after initiation of the trip.

During and immediately following the trip, plant and control system response was found to be acceptable with the exception of criteria 7, 8, and 9. Failure was attributed to a defective card in the steam dump circuitry of the Plant Trip Controller. This card was replaced and a retest of the steam dump Plant Trip Controller was performed satisfactorily with the steam dumps operating properly with a signal from the Plant Trip Controller. Acceptance Criteria 8 and 9 had been satisfied prior to initiation of steam dumps by manual transfer to the steam pressure mode.

Calculation of the hot leg RTD response time yielded a response time of approximately 7.5 seconds. Vendor analysis concluded that the 7.5 seconds

(based on conservative interpretation of the test data) obtained during the test is an acceptable DNBR (i.e. DNBR > 1.30) for the OTΔT response to a Rod Withdrawal at Power and the OPΔT response to a Small Steam Break. The 6.9 second acceptance criterion was based on a faster responding RTD. The vendor performed analyses showed acceptability (i.e. DNBR > 1.30) for both transients with longer RTD response times. The new analysis resulted in a new acceptance criterion of 8.4 seconds. No other unresolved questions were identified and the procedure results were accepted satisfactorily.

Table 5.16.1 gives parameter values prior to the trip and final values following stabilization at no load conditions.

DISCUSSION

High speed, multi-channel strip chart recorders were used to collect data on various plant parameters which included $T_{hot}$, $T_{cold}$, turbine trip event, reactor power, Auctioneered $T_{avg}$, $T_{ref}$, steam and feed flows, pressurizer pressure and pressurizer level. The trip was caused by opening the main generator breakers which resulted in a immediate turbine trip followed by a reactor trip.

During the performance of the 100% Plant Trip Test, a steam dump arming signal was not generated by the Plant Trip Controller during the transient. Cooldown of the secondary proceeded via the steam generator atmospheric relief valves until the steam dump valves could be placed in the "Steam Pressure" mode of operation. This action resulted in a larger drop in $T_{avg}$ than would normally be observed if controlled by the Plant Trip Controller programmed cooldown based on $T_{avg}-T_{no\ load}$ mismatch. Since steam pressure was still high when the steam dump control switch was placed in the steam pressure mode, the steam dumps opened causing $T_{avg}$ to drop to approximately 550°F, pressurizer level to drop to approximately 18% and pressurizer pressure to drop to approximately 1975 psig. Acceptance criteria 8 and 9 had been satisfied prior to initiation of steam dumps by manual transfer to the steam pressure mode. Steam dumps were reset to prevent further decrease. The Plant Trip Controller failure was traced to a defective card in the steam dump circuitry. This card was replaced and a retest of the steam dump Plant Trip Controller was performed satisfactorily, with the steam dumps operating properly with a signal for the Plant Trip Controller.

Determination of RTD response time was done utilizing the hot leg temperature and nuclear flux data from the strip chart recorders. The response time is defined as the interval of time between the point where neutron flux decreased by 50% of its initial value to the point where the hot leg temperature has decreased by 33 1/3% of the initial loop ΔT value. The response time was determined to be approximately 6.6 seconds. Due to the poor quality of the $T_{hot}$ trace, the data was analyzed using the most conservative values for $T_{hot}$ and Nuclear Flux and the response time was determined to be 7.5 seconds. Because this value exceeds the acceptance criterion, the vendor was asked to evaluate the data, determine if the test results are acceptable, and determine if the 6.9 second acceptance criterion can be increased to reflect measurement errors.

The Vendor Response is as follows:

"The original Acceptance Criterion (6.9 seconds) was based on a plant trip LOFTRAN analysis using standard RTD lag times. Because Callaway uses RdF Corporation RTDs, longer lag times should have been used. Analysis of the limiting cases for OPΔT and OTΔT protection (OPΔT is small steam break: OTΔT is Rod Withdrawal at Power), justify the use of longer lag times. The plant trip was reanalyzed with longer lag times, resulting in a new acceptance criterion of 8.4 seconds. A measured time response of less than this indicates that the RTD is faster than analyzed; thus, the safety analysis is still conservative."

No other unresolved questions were identified and the procedure results were accepted satisfactorily.

TABLE 5.16.1
PLANT PARAMETERS DURING PLANT TRIP TEST

| PARAMETER (UNITS) | INITIAL | DURING TRANSIENT MIN | DURING TRANSIENT MAX | FINAL |
|---|---|---|---|---|
| Nuclear Power (%) | 99.5 | 0 | 99.5 | 0 |
| Tavg (Auctioneered High °F) | 588.5 | 550 | 588.5 | 557 |
| Tref (°F) | 588.5 | — | — | 557 |
| ΔT (%) | 98 | 2 | 98 | 2 |
| OPΔT (%) | 109 | — | — | 109 |
| OTΔT (%) | 109 | — | — | 150 |
| Pressurizer Pressure (psig) | 2230 | 1975 | 2230 | 2240 |
| Pressurizer Level (%) | 62 | 18 | 63 | 25 |
| Steam Header Pressure (psig) | 980 | — | — | 985 |
| Steam Generator Level (%) | | | | |
| Loop 1 | 50 | NR<0 | 50 | 50 |
| Loop 2 | 50 | NR<0 | 50 | 50 |
| Loop 3 | 50 | NR<0 | 50 | 50 |
| Loop 4 | 50 | NR<0 | 50 | 50 |
| Steam Flow ($x10^6$ Rb/hr) | | | | |
| Loop 1 | 3.8 | — | — | 0.6 |
| Loop 2 | 3.6 | — | — | 0.0 |
| Loop 3 | 3.85 | — | — | 0.5 |
| Loop 4 | 3.8 | — | — | 0.25 |
| Feedwater Flow ($x10^6$ Rb/hr) | | | | |
| Loop 1 | 3.9 | 0 | 3.9 | 0 |
| Loop 2 | 3.6 | 0 | 3.6 | 0 |
| Loop 3 | 3.85 | 0 | 3.85 | 0 |
| Loop 4 | 3.75 | 0 | 3.75 | 0 |
| Feedwater Temperature (°F) | | | | |
| Loop 1 | 439.2 | — | — | 436.9 |
| Loop 2 | 437.3 | — | — | 432.6 |
| Loop 3 | 439.1 | — | — | 437.4 |
| Loop 4 | 437.4 | — | — | 435.7 |

## 5.17 RODS DROP AND PLANT TRIP (S-070012)

Procedure ETT-ZZ-07120

### OBJECTIVES

This procedure demonstrated operation of the negative rate trip circuitry by dropping 2 control rods from a common rod bank. In addition, it provided an opportunity to review plant response and control systems behavior to a plant trip from an intermediate power level prior to the plant trip from 100% power.

### SUMMARY OF RESULTS

The acceptance criteria for this test were:

1. The reactor shall trip as a result of the negative rate trip function.
2. All rods shall drop.
3. The pressurizer safety valves shall not lift.
4. The steam generator safety valves shall not lift.
5. Safety injection is not initiated.

All acceptance criteria were met.

### DISCUSSION

The negative rate trip circuitry was tested by simultaneously dropping rods D-4 and M-12 of Control Bank D. These rods were dropped by pulling the movable gripper coil fuses, and then simultaneously pulling the stationary gripper coil fuses. The rod drop and subsequent plant trip were monitored by using strip chart recorders and trend blocks from the plant computer. Plant response and control systems behavior were observed by the test director and the on-shift operations crew. No problems were encountered in this test.

## 5.18 LOSS OF HEATER DRAIN PUMPS TEST (S-070017)
Procedure ETT-ZZ-07170

### OBJECTIVES

The objective of this test was to verify proper nuclear plant response to a loss of heater drain pumps incident.

### SUMMARY OF RESULTS

The acceptance criteria were as follows:

1. The reactor and turbine must not trip
2. Safety injection is not initiated
3. Steam pressure remains less than 1150 psig and neither safety valves nor relief valves lift
4. Reactor coolant system pressure remains less than 2450 psig and neither pressurizer relief valves nor safety valves lift
5. No manual intervention should be required to bring the plant condition to steady state.

All acceptance criteria were satisfactorily met. Table 5.18.1 identifies those plant parameters which were monitored during the test including initial and final conditions.

### DISCUSSION

With the plant operating at 980 MWe, Heater Drain Pump A and B were tripped simultaneously. Brush recorders were utilized to monitor and chart the response of pre-selected plant parameters.

At approximately fourteen (14) minutes into the event, the reactor operator was forced to take manual control of the rod control system due to the C-11 control interlock not blocking Bank D at 223 steps in auto. The operator's action prevented misalignment of the rod step counting system. Approximately eighteen (18) minutes after the heater drain pumps trip, generator electrical output began oscillating due to continued operation without the heater drain pumps and due to heater extraction isolation caused by high level in the high pressure heaters. The reactor operator reduced load approximately 10 MWe to dampen the oscillation.

A review of recorded parameters showed that plant stability was attained approximately four minutes after tripping the heater drain pumps. Manual intervention was only required, as addressed above, fourteen and eighteen minutes into the event, due to abnormal configuration of the plant.

All acceptance criteria were satisfied.

TABLE 5.18.1
LOSS OF HEATER DRAIN PUMPS TEST SUMMARY

| PLANT PARAMETER | | INITIAL CONDITIONS | FINAL CONDITION |
|---|---|---|---|
| Plant Operating Level | (MWe-Gross) | 980 | 942 |
| Power - Nuclear | (%) | 86 | 88 |
| $T_{avg}$ (auctioneered) | | 583 | 584 |
| $T_{ref}$ | (F°) | 585 | 585 |
| ΔT (% FULL POWER) | | 85 | 85 |
| ΔT Overpower Setpoint | (% FULL POWER) | 108 | 108 |
| ΔT Overtemp. Setpoint | (% FULL POWER) | 116 | 88 |
| Pressurizer Pressure | (psig) | 2235 | 2235 |
| Pressurizer Level | (%) | 59 | 58 |
| Steam Header Pressure | (psig) | 990 | 965 |
| Steam Flow - Loop 1 | (LB/HR) | $3.3 \times 10^6$ | $3.45 \times 10^6$ |
| Steam Flow - Loop 2 | (LB/HR) | $3.05 \times 10^6$ | $3.15 \times 10^6$ |
| Steam Flow - Loop 3 | (LB/HR) | $3.2 \times 10^6$ | $3.35 \times 10^6$ |
| Steam Flow - Loop 4 | (LB/HR) | $3.15 \times 10^6$ | $3.25 \times 10^6$ |
| Steam Generator Level-Loop 1 | (%) | 50 | 50 |
| Steam Generator Level-Loop 2 | (%) | 50 | 50 |
| Steam Generator Level-Loop 3 | (%) | 50 | 50 |
| Steam Generator Level-Loop 4 | (%) | 50 | 50 |
| Feedwater Temperature-Loop 1 | (°F) | 426.4 | 42[illegible].4 |
| Feedwater Temperature-Loop 2 | (°F) | 424.7 | 421.5 |
| Feedwater Temperature-Loop 3 | (°F) | 426.4 | 422.9 |
| Feedwater Temperature-Loop 4 | (°F) | 424.8 | 421.5 |
| Feedwater Flow - Loop 1 | (LB/HR) | $3.15 \times 10^6$ | $3.25 \times 10^6$ |
| Feedwater Flow - Loop 2 | (LB/HR) | $3.0 \times 10^6$ | $3.1 \times 10^6$ |
| Feedwater Flow - Loop 3 | (LB/HR) | $3.2 \times 10^6$ | $3.35 \times 10^6$ |
| Feedwater Flow - Loop 4 | (LB/HR) | $3.[illegible] \times 10^6$ | $3.3 \times 10^6$ |
| Feed Pump Disch. Hdr. Pres. | (psig) | 1160 | 1140 |
| Feed Pump A Speed | (RPM) | 4200 | 4400 |
| Feed Pump B Speed | (RPM) | 4200 | 4400 |
| Control Rod Pos. - Bank C | (Steps) | 228 | 228 |
| Control Rod Pos. - Bank D | (Steps) | 193½ | 226 |
| Boron Concentration | (PPM) | 894 | 894 |

TIME TO REACH EQUILIBRIUM FOLLOWING LOAD CHANGE: 4 MINUTES

5.19 POWER ASCENSION THERMAL EXPANSION AND DYNAMIC TEST (S-070015)
Procedure ETT-ZZ-07150

OBJECTIVES

The thermal expansion portion of this test was conducted to verify that during plant heatup, the piping was free to move thermally as designed and that it returned to its original position upon cooldown of the system. The dynamic portion of this test was conducted to monitor the dynamic response of the affected systems to the following transients: Pressurizer Power-Operated Relief Valves opening, Main Steam Atmospheric Dump Valves opening, and a plant trip from 100% power.

SUMMARY OF RESULTS

As systems heated up, the major restraint to pipe thermal growth was insulation, which was removed or trimmed as necessary to accommodate the design thermal movements. No major modifications were required to piping supports to meet the acceptance criteria for thermal movement and stress. Slightly larger than expected movements were observed on several systems during the plant trip from 50% power; however, all stresses remained within acceptable limits.

DISCUSSION

Pipe movements during heat-ups, cooldowns, and transients were monitored during the test using lanyard-type motion transducers at selected positions. Visual observations and readings were also taken during thermal plateaus at spring-can supports and snubbers. During the Pressurizer PORV Dynamic Test, pipes were monitored for vibration and loading using accelerometers and clevis pin load cells in conjunction with lanyard-type motion transducers.

Piping position data was recorded at ambient temperature and also when the Reactor Coolant System (RCS) reached 350°F and 557°F plateaus. In conjunction with this, data was taken on the Main Steam and Main Feedwater System at every 50°F increment of the system temperature and walked down every 100°F to verify that the piping was not restricted from moving thermally.

Potential or actual interferences were either evaluated to be acceptable or were removed prior to continuing to the next plateau. Only minor non design-change modifications were required to permit piping to expand thermally as designed i.e., insulation trimming, etc.

Transient data was collected during the opening of the Pressurizer PORV's, Main Steam Atmospheric Dump Valves and Main Steam Condenser Dump Valves prior to Initial Criticality. These data and the thermal movement data taken on primary systems were taken as Hot Functional Testing Retests and each received an engineering evaluation prior to Initial Criticality. During the heatup of the secondary plant, Main Feedwater

and Main Steam Systems were monitored for thermal movements and the Main Steam System was monitored for its response to a plant trip from 50% power. Data received an engineering evaluation by the architect-engineer, Bechtel Stress Engineering, prior to increasing the system temperature to the next plateau.

During the plant trip from 50% power, lines leading to the Condenser Dump Valves were observed to move more than expected for that transient. Bechtel evaluated these data and extrapolated the results to a 100% Power Trip. This evaluation indicated that the stress level at a header-to-sweepolet weld could be in excess of the allowable stress level on the Condenser Dump Valves header depending on the actual weld profile. Bechtel provided three (3) options to reconcile this condition in letter BLUE 1801. UENE chose the option that Bechtel recommended in the above letter which was the addition of a pipe hanger which is being performed under CMP 85-0069.

## 5.20 LOOSE PARTS MONITORING SYSTEM TEST (S-09SQ01)
Procedure ETT-SQ-09010

### OBJECTIVES

The objective of this test was to obtain a base noise "signature" for the Reactor Coolant System for future comparison with real or suspected loose parts signals by recording the channel audio outputs.

### SUMMARY OF RESULTS

Two recordings were obtained at both 0% and 100% power.

### DISCUSSION

Recordings were obtained using installed components of the Vibration and Loose Parts Monitoring System.

The loose parts monitoring system (LPMS) monitors the reactor coolant system (RCS) for the presence of metallic loose parts. It consists of 12 active instrumentation channels, each comprising a piezoelectric accelerometer (sensor), signal conditioning, and diagnostic equipment.

Two redundant sensors are fastened mechanically to the RCS at each of the following potential loose parts collection regions:

Reactor pressure vessel - upper head region

Reactor pressure vessel - lower head region

Each steam generator - reactor coolant inlet region

The output signal from each accelerometer is amplified b a preamplifier and amplifier. The amplified signal is processed through a discriminator to eliminate noises and signals not indicative of loose parts, and the processed signal is compared to a preset alarm setpoint. Loose parts detection is accomplished at a frequency of approximately 25 kHz, where background signals from the RCS are low. Spurious alarming due to control rod stepping is prevented by a contact that activates an alarm disable circuit.

If a measured signal exceeds the preset alarm level, audible and visible alarms at the LPMS console in the control room are activated. The LPMS automatically initiates recording, on a four-channel tape recorder, the signals from the alarmed channel and three other channels in proximity to the alarmed channel. A microprocessor records the arrival times and energy amplitudes of the first and subsequent impact signals that reach various sensors. This enables the approximate location and size of the loose part to be determined. The LPMS also has provision for audio monitoring of any channel. Audio monitoring, including comparison of the audio signal with a previously recorded

"signature" audio signal, if there is an ambiguity, will be performed to confirm the presence of a loose part.

The on-line sensitivity of the LPMS is such that the system will detect a loose part that weights from 0.25 to 30 pounds and impacts with a kinetic energy of 0.5 feet pounds on the inside surface of the RCS pressure boundary within 3 feet of a sensor.

5.21 PLANT PERFORMANCE TEST POWER ASCENSION (S-090007)
Procedure ETT-ZZ-09070

OBJECTIVES

The objectives of this test were:

1. To monitor balance-of-plant and electrical systems under loaded conditions during power ascension testing.

2. To demonstrate the ability of the ventilation systems to maintain ambient temperatures within design limits during power ascension testing.

3. To monitor concrete temperatures surrounding hot penetrations.

SUMMARY OF RESULTS

The following acceptance criteria were specified for this test:

The Containment air coolers maintain Containment air temperatures equal to or less than 120°F during performance of this test at 50% and 100% full reactor power.

The above acceptance criteria were satisfactory met during conduct of the test. Table 5.21.1 lists the Containment cooler air inlet temperatures at the 50% and 100% reactor power levels.

Additional plant operating data was taken at the 0%, 30%, 50%, 75%, and 100% reactor power plateaus. The purpose of obtaining the additional data was to verify whether or not various plant parameters were operating within expected design ranges. No discrepancies were identified which would impact the safe operation of the plant.

DISCUSSION

Data required by this test was obtained at the power levels described above, and documented. Plant operating data taken consisted of the following systems operating parameters:

Main Steam
Feedwater
Feedwater Heater Extraction, Drain and Vent
Cavity Cooling
Containment Fan Cooling
Station Service and Emergency Service Water Transformer
Electrical Load Centers (13.8KV, 480V)
AC Inverters
125V AC and DC
≥ 250 HP Motor Current
Containment Cooling

Auxiliary Building Ventilation
Control Building Ventilation
Steam Tunnel Ventilation
Auxiliary Feedpump Room Ventilation
Auxiliary Boiler Room Ventilation
Central Chilled Water
ESW Pump House and Cooling Tower Electrical Room Ventilation
Radwaste Building Ventilation
Turbine Building Ventilation
Condensate

Additionally, concrete temperatures were measured in those areas which contained hot penetrations.

Several parameters were found to be outside their predicted design ranges. During hot zero power conditions, it was noted that reactor cavity cooling exit temperatures reached 22.7°F above the design maximum value of 160°F. Upon further investigation, it was determined that several hatchways in the cavity area were left open, thereby reducing air flow through the cavity. Once the hatchways were secured, the cavity cooling exit temperature requirements were satisfied.

The remainder of those system parameters which did not meet their design ranges were investigated and found to be, in some cases, associated with defective instrumentation and computer problems. In all cases where parameters exceeded or fell outside their design range, corrective action was initiated.

TABLE 5.21.1

CONTAINMENT COOLING DATA SUMMARY

| (1) MEASUREMENT NO. | * INLET TEMPERATURES (°F) FAN "A" | FAN "B" | FAN "C" | FAN "D" |
|---|---|---|---|---|
| 1. (50% Power) | 84.9 | 82.9 | 84.0 | 90.6 |
| 2. (50% Power) | 85.8 | 83.6 | 84.9 | 91.1 |
| 3. (100% Power) | 80.0 | 70.0 | 98.0 | 78.0 |
| 4. (100% Power) | 80.0 | 70.0 | 75.0 | 90.0 |
| 5. (100% Power) | 80.0 | 70.0 | 80.0 | 90.0 |

* ACCEPTANCE CRITERIA = ≤ 120°F

(1) Lapse Time Between Measurement No. 1 and No. 2 = 9 hr 32 min

Lapse Time Between Measurement No. 3 and No. 4 = 9 hr 33 min

Lapse Time Between Measurement No. 4 and No. 5 = 4 hr 5 min

DMB

UNION ELECTRIC COMPANY
1901 Gratiot Street, St. Louis

Donald F. Schnell
Vice President

March 19, 1985

PRIORITY ROUTING
First | Second
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Regional Administrator
Office of Inspection & Enforcement
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Region III
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Glen Ellyn, IL 60137

ULNRC- 1062

Dear Mr. Keppler:

DOCKET NUMBER 50-483
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FACILITY OPERATING LICENSE NPF-30
STARTUP REPORT

The enclosed Startup Report is submitted pursuant to sections 6.9.1.1, 6.9.1.2, and 6.9.1.3 of the Callaway Unit 1 Technical Specifications.

Very truly yours,

Donald F. Schnell

Donald F. Schnell

drs
Enclosure

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CALLAWAY PLANT, UNIT 1

STARTUP REPORT

MARCH 19, 1985

DOCKET NUMBER 50-483

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