VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

R. H. LEASBURG
VICE PRESIDENT
NUCLEAR OPERATIONS

August 10, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. D. G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 478
FR/MLS:gmj
Docket Nos.: 50-280
50-281
50-338
50-339
License Nos.: DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

VEPCO NUCLEAR DESIGN
TOPICAL REPORTS

In response to the requests in the letters from Mr. Robert L. Tedesco dated May 18, 1981, May 20, 1981 and May 13, 1981 accepting the Vepco topical reports VEP-FRD-19, VEP-FRD-20, and VEP-FRD-24, respectively, for reference in licensing actions by Vepco, we have issued revised versions of these reports which include the NRC evaluation letter and its attachment between the title page and the abstract. Five (5) copies of each of these revised versions of the reports are enclosed for your use.

If you have any questions, please contact us.

Very truly yours,

R. H. Leasburg
Vice President
Nuclear Operations

Enclosure

cc: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing

Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

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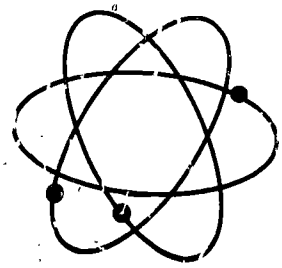
VEP-FRD-20A
JULY, 1981

Vepco

THE PDQ 07

ONE ZONE MODEL



FUEL RESOURCES DEPARTMENT
VIRGINIA ELECTRIC AND POWER COMPANY

THE PDQ07 ONE ZONE MODEL

by

J. R. RODES

NUCLEAR FUEL ENGINEERING GROUP
FUEL RESOURCES DEPARTMENT

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA

July, 1981

Recommended for Approval:

Martin L. Bowling

M. L. Bowling, Supervisor
Nuclear Fuel Engineering Group

Approved:

J. T. Rhodes

J. T. Rhodes, Director
Nuclear Fuel Engineering and
Operation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 20 1981

Mr. W. N. Thomas, Vice President
Fuel Resources
Virginia Electric Power Company
Richmond, Virginia 23261

Dear Mr. Thomas:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT VEP-FRD-20 "THE PDQ07 ONE ZONE MODEL"

The Nuclear Regulatory Commission (NRC) staff has completed its review of the Virginia Electric and Power Company (Vepco) Topical Report number VEP-FRD-20 entitled "The PDQ07 One Zone Model". The Vepco developed a coarse mesh (i.e., several mesh lines per fuel assembly), two-dimensional, two neutron energy group, diffusion-depletion calculational model, designated as the PDQ07 one zone model. This model is similar to the previously NRC accepted Vepco PDQ07 discrete (i.e., one mesh line per fuel rod) model in that it uses the NULIF, PDQ07, SHUFFLE, and HAFIT computer codes which are part of the Fuel Utilization and Performance Analysis Code (FUPAC) system obtained from the Babcock and Wilcox Company. The purpose of the one zone model is to provide a supplementary model to the PDQ07 discrete model for the more efficient (i.e., less computational time) performance of reactor physics, fuel management, and operational support analyses for the Surry and North Anna nuclear reactors. The accuracy of the one zone model is demonstrated through comparisons with both the Vepco PDQ07 discrete model predictions and measurements taken at Surry Units No. 1 and 2. Our summary of the evaluation is enclosed.

As the result of our reviews we conclude that the Vepco Licensing Topical Report VEP-FRD-20 entitled "The PDQ07 One Zone Model" dated January 1977 is acceptable for referencing in licensing actions by Vepco to the extent specified and under the limitations in the report and the enclosed evaluation.

We do not intend to repeat the review of the safety features described in the report as found acceptable herein. Our acceptance applies only to the use of features described in the topical report as discussed herein.

In accordance with established requirements, it is requested that Vepco issue a revised version of this report within three months of the receipt of this letter. This evaluation letter and its enclosure is to be included in the revised version between the title page and the abstract and the approved report will carry the identifier VEP-FRD-20.

Should NRC criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Vepco will be expected to revise and resubmit the topical report or submit justification for the continued effective applicability of the topical report without revision.

If you have any questions about the review or our conclusion, please contact us.

Sincerely,

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
Evaluation of Report
VEP-FRD-20

MAY 8 1981

Enclosure

EVALUATION OF VEPCO TOPICAL REPORT VEP-FRD-20

Report Number:	VEP-FRD-20
Report Title:	The PDQ07 One Zone Model
Report Date:	January 1977
Originating Organization:	The Virginia Electric and Power Company
Reviewed By:	Core Performance Branch/W. Brooks

The Virginia Electric and Power Company (VEPCO) has presented a number of licensing topical reports for our review in preparation for performing their own reload analysis, VEP-FRD-20 is one of the series. The Core Performance Branch has reviewed this report. Our evaluation follows.

1. Description of Report

The report includes the following:

1. A description of the Surry Units 1 and 2 cores including the fuel and burnable poison loadings for Cycles 1 and 2 of each unit.

2. A brief description of the calculational model including the homogenization procedure, the cross-section preparation techniques, and the core geometry representation for the diffusion theory code.

3. A comparison of calculated results from the one zone model to those calculated by the discrete model and those measured in the Surry reactors.

4. A summary of the results of the comparisons yielding uncertainty values to be assigned to the various calculated parameters.

Verification of the one zone model for the calculation of the following parameters is presented:

1. Assembly average radial power distributions.

2. Stuck rod power distributions.

3. Assembly and batch burnup.

4. Control bank worths.

5. Shutdown worth and stuck rod worth.

6. Critical boron concentration vs burnup and control rod configuration.

7. Differential boron worth.

8. Isothermal temperature coefficients.

In contrast to the discrete model (described in VEP-FR-19 which has been reviewed and approved) which has one mesh point for each fuel rod, guide tube or instrumentation tube, the one zone model is restricted to a few mesh points per assembly.

In practice, the assembly is divided into mesh blocks which all have the same distribution of fuel and non-fueled locations. Thus each mesh block in a particular fuel assembly has the same homogenized nuclear parameters at beginning of life. This accounts for the use of the term one zone model. Because the core can be modeled with far fewer mesh points a solution can be obtained with the one zone model much more rapidly than with the discrete model. For this reason the one zone model is often used for scoping calculations with the discrete model being used for the final analyses. In addition, the one zone model is used when the whole core must be represented as in performing stuck rod calculations.

Comparisons between measured and calculated assembly average power distributions in the form of standard deviations are presented for both the one zone and the discrete models. Representative examples of core maps showing comparisons of measurement and one zone calculations are also given. Comparisons between one zone and discrete calculations of the flux distribution for a stuck rod configuration are presented along with examples of comparisons between measured and calculated end of cycle burnup distributions.

Comparisons are presented between calculated and measured values of certain reactivity parameters including

- D and C bank works at beginning of cycle, HZP conditions;
- total shutdown worth (all rods in) and stuck rod worth;
- critical boron concentrations for the first three cycles of Surry Units 1 and 2 as a function of burnup;
- critical boron concentration as a function of control rods inserted for the first three cycles of Surry, Unit 1;
- differential boron worths for all rods out configuration for the first three cycles of Surry, Unit 1; and
- isothermal temperature coefficients as a function of rods inserted for the first three cycles of Unit 1.

For all but the last of those parameters, the results of comparisons between measurement and discrete model calculations are also given.

Finally the comparison data are summarized to obtain uncertainty values to be used for production calculations. Except as noted comparisons are to measurements. These uncertainty values are:

- assembly average power distribution - maximum standard deviation of $\sim$ 6 percent;

- stuck rod peak assembly power to within 3 percent of the discrete calculation;

- batch burnup value - maximum difference of 4.2 percent;

- stuck rod worths during initial startup of Units 1 and 2 were predicted to within 5 percent; and

- isothermal temperature coefficients are typically predicted to within 1 pcm per degree Fahrenheit with a maximum difference of 2.2 pcm. A value of 3 pcm per degree Fahrenheit is assumed for this uncertainty.

In addition to these conclusions it is also concluded that the one zone model provides adequate values for control rod bank worths in normal sequence, critical boron concentrations and differential boron worths, although the model is not intended for production calculations of these parameters.

2. Summary of Evaluation

We have reviewed the model description in the topical report and conclude that the procedures employed in the homogenization and the data inputs (cross-section, etc.) are state-of-the-art and acceptable. We have reviewed the data presented as support for the assignment of uncertainties to the various calculated parameters listed in Section 1 above. Enough data are presented to permit the conclusions relating to calculational uncertainties to be made. We concur with the values for the uncertainties as presented in the report.

3. Evaluation Procedure

The review of topical report VEP-FRD-20 has been conducted within the guidelines provided for analytical methods in the Standard Review Plan, Section 4.3. Sufficient information is provided to permit a knowledgeable person to conclude that the VEPCO model described in this report is state-of-the-art and is acceptable. Sufficient data are presented to permit the conclusion that the derived uncertainties are reasonable and are acceptable.

4. Regulatory Position

Based on our review of licensing topical report VEP-FRD-20 we conclude that it is acceptable for reference in licensing actions by VEPCO. Such reference may be made for the purpose of describing the calculational model and as support for the stated values of uncertainties in the following quantities:

- assembly average power distributions;
- assembly power distributions in the neighborhood of stuck or potentially ejected rods;
- batch and assembly burnup values;
- stuck rod worths; and
- isothermal temperature coefficients.

We further conclude that this model is an acceptable substitute for vendor calculations of the above named quantities.

We endorse the commitment made in the report by VEPCO to continue verification and model improvements in the one zone model as more data are obtained from the Surry and North Anna Reactors.

CLASSIFICATION/DISCLAIMER

The data and analytical techniques described in this report have been prepared for specific application by the Virginia Electric and Power Company. The Virginia Electric and Power Company makes no claim as to the accuracy of the data or technique contained in this report if used by other organizations. In addition, any use of this report or any part thereof must have the prior written approval of the Virginia Electric and Power Company.

ABSTRACT

The Virginia and Electric Power Company (Vepco) has developed a coarse mesh (i.e., several mesh lines per fuel assembly), two-dimensional, two neutron energy group, diffusion-depletion calculational model, designated as the PDQ07 one zone model. This model is similar to the previously developed Vepco PDQ07 discrete (i.e., one mesh line per fuel rod) model in that it uses the NULIF, PDQ07, SHUFFLE, and HAFIT computer codes which are part of the Fuel Utilization and Performance Analysis Code (FUPAC) system obtained from the Babcock and Wilcox Company. The purpose of the one zone model is to provide a supplementary model to the PDQ07 discrete model for the more efficient (i.e., less computational time) performance of reactor physics, fuel management, and operational support analyses for the Surry and North Anna nuclear reactors. The accuracy of the one zone model is demonstrated through comparisons with both the Vepco PDQ07 discrete model predictions and measurements taken at Surry Units No. 1 and 2.

ACKNOWLEDGEMENTS

The author would like to thank Mr. C. B. Franklin for his assistance in performing the computer calculations and data preparation required for this report and Ms. Cathy Loving for her typing of the draft and final manuscript. Special thanks is given to Mr. M. L. Smith whose technical assistance and direction greatly aided the development of the one zone model. Finally, the author wishes to acknowledge the contributions made by the many people who reviewed and provided comments on the preparation of this report.

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SECTION 1 - INTRODUCTION

The Virginia Electric and Power Company (Vepco) is currently developing the capability to perform nuclear reactor analyses for the Surry and North Anna nuclear power stations. The objective of this topical report is 1) to describe one of the computational models developed at Vepco for the purposes of reactor physics analyses and fuel management evaluation and 2) to demonstrate the accuracy of this model by comparing analytical results generated with the model to alternate calculations and to actual measurements from Surry Units No. 1 and 2.

The computational model to be described is a coarse mesh (several mesh lines per fuel assembly), two-dimensional, two neutron energy group, diffusion-depletion (with thermal-hydraulic feedback) calculational package and is designated as the PDQ07 one zone calculational model. The PDQ07 one zone model uses the NULIF[1], PDQ07[2], SHUFFLE[3], and HAFIT[4] computer codes which are part of the Fuel Utilization and Performance Analysis Code[5] (FUPAC) system obtained from the Babcock and Wilcox Company. The FUPAC system is currently used by Babcock and Wilcox to perform production reactor analysis and design. A detailed description of the input/output, functioning, and physical model of the above computer codes can be obtained from the referenced Babcock and Wilcox computer code manuals. The FUPAC system is maintained by Vepco and updated through contractual arrangements between Vepco and Babcock and Wilcox.[5]

The PDQ07 one zone model is similar to the PDQ07 discrete model[6] and was designed to provide a two-dimensional reactor physics analysis capability presently impractical or impossible with the discrete model because of excessive computer usage requirements. The one zone model was developed to require much less computer usage than the discrete model by virtue of using a fewer number of spatial mesh lines to represent the geometry of the reactor core. Reduction in running time by a factor of 5 to 12 can be realized

depending on the number of mesh lines used in the one zone model. The one zone model has also been developed to be consistent with the discrete model while:

1. making as few changes as possible to the one zone code input relative to the discrete model input, and
2. benchmarking one zone calculations to discrete calculations where practical.

Since the one zone model has many aspects which are identical to the discrete model, this topical report frequently references the PDQ07 Discrete Model Report. (See Reference 6.)

The purpose of this report is 1) to describe those aspects of the one zone model which differ significantly from the discrete model as well as to summarize the basic similarities, 2) to present the types of calculations that are intended to be performed by the one zone model, and 3) to demonstrate the model's accuracy by comparison to discrete model calculations and actual measurements performed at Surry Units No. 1 and 2.

The types of calculations that can be performed by the one zone model include:

1. Reactor Physics Analysis
   a. Two-dimensional assembly average radial power distributions
   b. Critical soluble boron concentrations as a function of burnup
   c. Nuclide concentrations as a function of burnup
   d. Integral control rod bank worths
   e. Rod worth values for abnormal positioned control rods.
   f. Moderator and doppler coefficient and defects as a function of soluble boron, burnup, average moderator temperature and control rod bank position.
2. Fuel Management Analysis
   a. Batch power and burnup sharing
   b. Fuel isotopics as a function of burnup

c. Scoping studies for the evaluation of alternative future cycle loading patterns and fuel loading requirements.

Of the above types of calculations, the ones that are of primary interest for one zone model application are the rod worth and power distribution values for abnormal positioned control rod (i.e., stuck, dropped, misaligned, and ejected rod worths), moderator and Doppler coefficients and defects, and the evaluation of alternative future cycle loading patterns and fuel loading requirements. These calculations, which are computer usage intensive, can be performed to within acceptable accuracy with the one zone model. Rod worth and power distribution values for abnormal positioned control rods generally require full core calculations which translate to very long computing requirements if the discrete model were to be used. Moderator and Doppler coefficients are needed for a wide range of reactor core conditions so that a very large number of calculations must be performed and, therefore, using the one zone model greatly reduces the overall computer usage requirement relative to the discrete model. Finally, it is planned to use the one zone model for scoping studies to generate alternate near and long term fuel cycle loading patterns and fuel loading requirements in order to identify a group of potentially acceptable loadings which can then be further analyzed by the discrete model for the purpose of selecting the most operationally and economically optimum loading.

The remainder of this report describes the Surry Units No. 1 and 2 reactor core to be modeled, the purpose and interrelationships of the various computer codes which comprise the PDQ07 one zone model, the specific modeling of the reactor core, and the comparison of calculated results with calculated results from the discrete model and/or selected reactor measurements from Cycles 1 through 3 of Surry Units No. 1 and 2, as appropriate.

SECTION 2 - CORE DESCRIPTION

2.1 INTRODUCTION

The Surry Nuclear Power Station, which currently consists of two operating units, has been selected as the operating system to be modeled for verification of the PDQ07 one zone model. The Surry Units No. 1 and 2 are identical Westinghouse designed three coolant loop pressurized water reactors with thermal ratings of 2441 Mwt. Initial criticality was achieved for Surry Unit No. 1 on July 1, 1972 and for Surry Unit No. 2 on March 7, 1973. The initial cycle for Surry Unit No. 1 was completed on October 24, 1974 and for Surry Unit No. 2 on April 26, 1975. Second cycle operation commenced on January 30, 1975 and June 14, 1975 and was completed on September 26, 1975 and April 22, 1976 for Surry Units No. 1 and 2, respectively. Third cycle operation began for Unit 1 on December 6, 1975 and was completed on October 17, 1976. For Unit 2, third cycle operation began on June 1, 1976 and is presently planned to be completed in the fall of 1977.

2.2 CORE DESIGN

The Surry cores consist of 157 fuel assemblies surrounded by a core baffle, barrel, and thermal shield and enclosed in a steel pressure vessel. The pressure inside the vessel is maintained at a nominal 2250 psia. The coolant (and moderator) is pressurized water which enters the bottom of the core at 532°F and undergoes an average rise in temperature of 65.5° F before exiting the core. The average coolant temperature is 566°F and the average linear power density of the core is 6.2 kw/ft.

Each of the 157 fuel assemblies consists of 204 fuel rods arranged in a 15 by 15 square array. The fuel used in the Surry cores consists of slightly enriched uranium dioxide fuel pellets contained within a Zircaloy-4 clad. A small gap containing pressurized helium exists between the pellets and the inner diameter of the clad. For the positions in the 15 by 15 array not occupied by

fuel rods, there are 20 guide tube locations for either solid burnable poison rods or control rods and one centrally located instrumentation tube. (See Figure 2-1.) The fuel rods in each fuel assembly are supported by seven Inconel-718 grids located along the length of the assembly. These grids are mechanically attached to the guide tubes, which are, in turn, welded to the upper and lower nozzles, and thus provide for assembly structural support.

There are 48 full-length Rod Cluster Control Assemblies (referred to as control rods) used to control core reactivity as well as five part-length rods for axial power shaping. (It should be noted that the part-length control rods are physically present but are not currently allowed to be inserted into the core.) The absorber material of the control rods is an alloy consisting of 80% silver, 15% indium, and 5% cadmium. The various control rods are arranged in and move in symmetrically located groups, or banks, as depicted in Figure 2-2. Banks D, C, B, and A are denoted as the control banks and are moved in a fixed sequential pattern to control the reactor over the power range of operation. The remaining rods, Banks SA and SB, are denoted as shutdown banks and are used to provide shutdown margin.

In addition to the control rods, a chemical (boric acid) shim is used to control excess core reactivity and to facilitate operational flexibility. Above certain concentrations of chemical shim, burnable poison rods are also used to control excess reactivity. Fresh and/or depleted burnable poison rods can also be used to shape (i.e., improve) the core power distribution. The burnable poison rods contain borosilicate in the form of Pyrex glass clad in a stainless steel tube. Burnable poison rods which may be used in any fuel assembly not under a control rod bank location, consist of clusters of either 8, 12, 16, or 20 rods which are inserted into the Zircaloy-4 control rod guide tubes.

Specific values of the principal mechanical and thermal-hydraulic parameters of the Surry core are provided in Table 2-1. A complete description of the Surry units is given in Reference 7.

2.3 FUEL LOADING

The initial and reload quarter core fuel loadings (i.e., initial enrichments and density, previous cycle location if appropriate, beginning of cycle burnup, and number of fresh or depleted burnable poison rods present) for both Surry units are provided in Figures 2-3 through 2-12. It should be noted that the fuel loadings for Cycle 1 of both Surry units are identical. The fuel management strategy employed in the initial cycle of operation of each unit was the checkerboard loading of the two lower enriched fuel batches in the center of the core and the highest enriched fuel batch around the periphery of the core. After the first cycle, the fuel management became more complicated as the result of the need to minimize the impact of fuel densification (which was most severe in the lower density, lower prepressurization Batches 1, 2, and 3). Generally a modified out-in strategy was followed wherein higher enrichment fresh fuel was loaded on the core periphery with lower enrichment fresh fuel (one-burned fuel and twice-burned fuel) checkerboard loaded in the inner region of the core. An exception to this was in the third cycle of Unit No. 1 where no fresh fuel was loaded on the periphery. The only fresh fuel was 16 lower enrichment assemblies loaded in the inner region of the core.

Table 2-1

SURRY CORE DESCRIPTION

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total core heat output, Mwt	2441
Heat generated in fuel, %	97.4
System operating pressure, psi	2250
Total coolant flow rate, lb/hr (gpm)	100.7×10^6 (265,500)
Coolant Temperatures, °F (@100% power)	
Nominal inlet	532
Average rise in the core	65.5
Average in the core	566
Nominal outlet of hot channel	642
Average linear power density, Kw/ft	6.2

MECHANICAL DESIGN PARAMETERS

Fuel Assemblies		
Design	Canless 15 x 15	
Number	157	
Rod pitch, inches	0.563	
Overall dimensions, inches	8.426 x 8.426	
Number of grids per assemly (material)	7 (Inconel-718)	
Number of instrumentation tubes	1	
Fuel Rods		
Number	32,028	
Number of rods/assembly	204	
	Batch 1,2,4,5	Batch 3
Outside diameter, inches	0.422	0.422
Diametrical gap, inches	0.0075	0.0085
Clad thickness, inches	0.0243	0.0243
Clad material	Zircaloy-4	
Fuel Pellets		
Material	Sintered UO_2	
Density (% of theoretical) and Enrichment (w/o U235)	See Figures 2-4 through 2-12	
	Batch 1,2,4,5	Batch 3
Outer diameter	0.3659	0.3649
Control Rod Assemblies		
Neutron absorber	5% Cd-15% In-80% Ag	
Cladding Material	Type 304 SS-Cold worked	
Clad thickness, inches	0.019	
Number (full length)	48	
Number of rods per assembly	20	

Table 2-1
(Continued)

Burnable Poison Rods	
Material	Pyrex glass
Content B_2O_3 (w/o)	12.5
Core Structure	
Core barrel I.D./O.D., inches	133.875/137.875
Thermal shield I.D./O.D., inches	142.625/148.000
Core diameter, inches (approximate)	119.5
Reflector thickness (approximate) and composition	
Top - Water plus steel, in.	10
Bottom - Water plus steel, in.	10
Side - Water plus steel, in.	15

FIGURE 2-1

CROSS SECTIONAL VIEW OF SURRY FUEL ASSEMBLIES

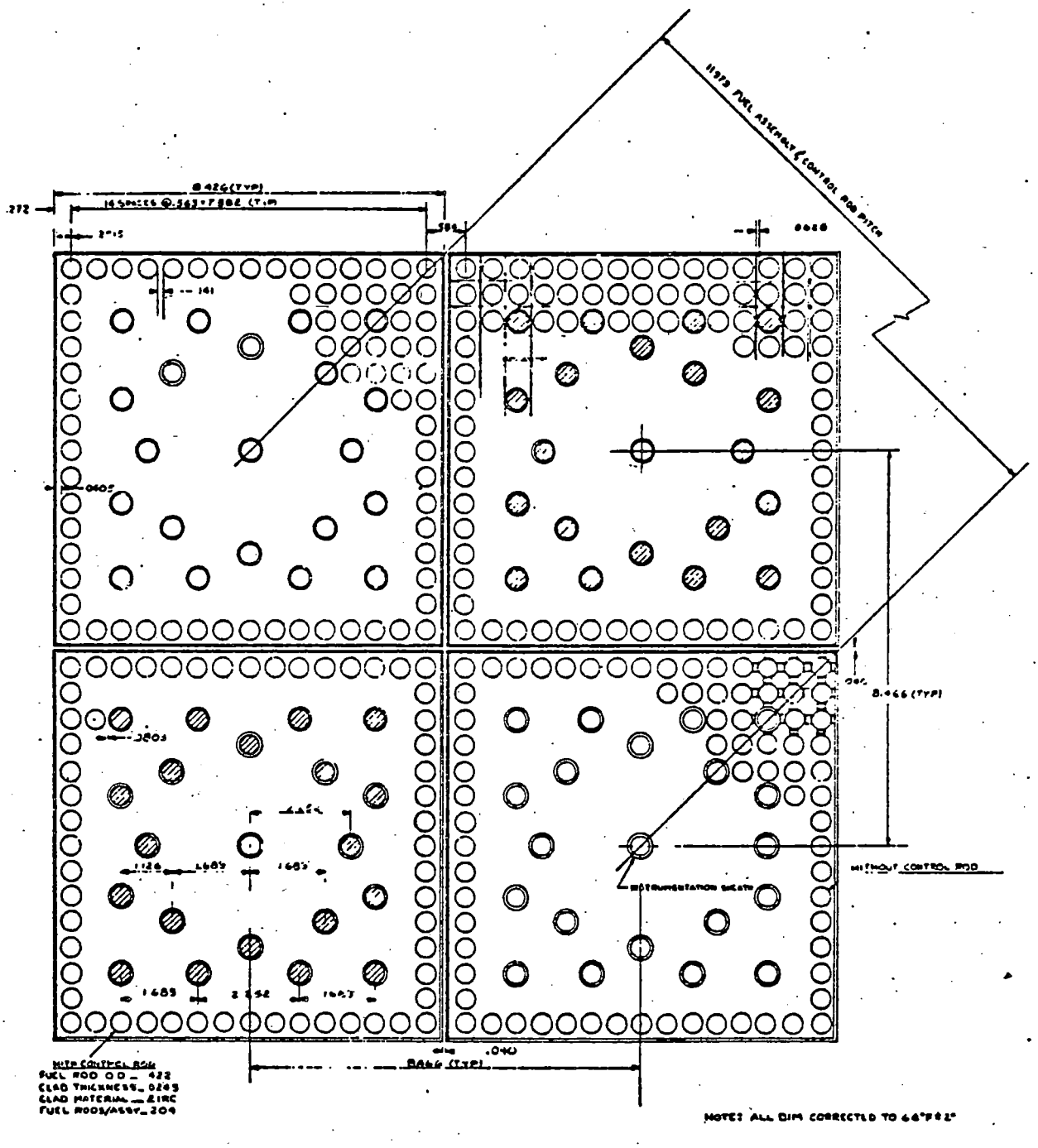
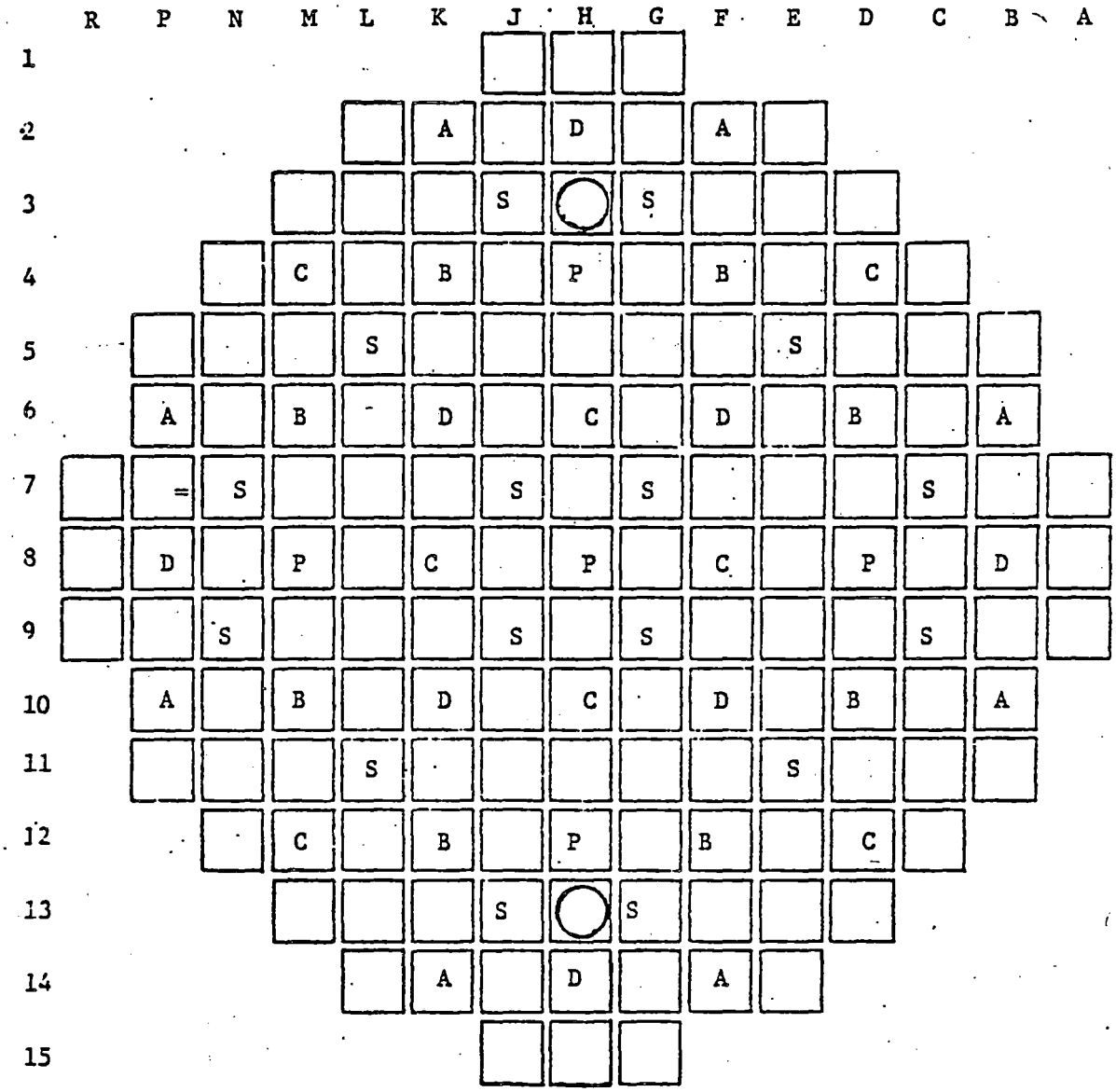


FIGURE 2-2

CONTROL ROD BANK LOCATIONS



CONTROL ROD ASSEMBLY BANKS

Function	Number of Assemblies
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown (S)	16
Part Length (P)	5
	53

◯ = SOURCE ASSEMBLY LOCATIONS

Figure 2-3

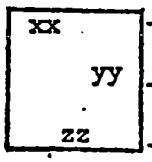
SURRY UNITS 1 AND 2 -- CYCLE 1

FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh
J	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh	3 0 Fresh
K	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	
L	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	3 0 Fresh	
M	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh		
N	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	3 0 Fresh			
P	1 0 Fresh	3 0 Fresh	3 0 Fresh	3 0 Fresh				
R	3 0 Fresh	3 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (53)	1.85	94
2 (52)	2.55	93
3 (52)	3.10	92

LEGEND



---Batch No.

---Initial Burnup (MWD/MTU)

---Previous Location (If applicable)

Figure 2-4

SURRY UNIT 1 - CYCLE 2
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 15231 H08	4B 0 Fresh	1 12373 H14	4B 0 Fresh	2 15398 H13	4B 0 Fresh	1 14546 H12	4C 0 Fresh
J	4B 0 Fresh	2 16557 L08	4A 0 Fresh	2 14339 K13	2 16191 K11	2 14438 L12	4C 0 Fresh	4C 0 Fresh
K	1 12373 P08	4A 0 Fresh	1 11120 M12	4A 0 Fresh	2 16031 J12	1 14053 K12	4C 0 Fresh	
L	4B 0 Fresh	2 14339 N10	4A 0 Fresh	2 16883 J08	2 16723 J10	4C 0 Fresh	4C 0 Fresh	
M	2 15398 N08	2 16191 L10	2 16031 M09	2 16723 K09	4A 0 Fresh	4C 0 Fresh		
N	4B 0 Fresh	2 14438 M11	1 14053 M10	4C 0 Fresh	4C 0 Fresh			
P	1 14546 M08	4C 0 Fresh	4C 0 Fresh	4C 0 Fresh				
R	4C 0 Fresh	4C 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (21)	1.85	94
2 (52)	2.55	93
4A (20)	1.85	95
4B (12)	2.60	95
4C (52)	3.35	95

LEGEND

xx
yy
zz

xx ---Batch No.
yy ---Initial Burnup (MWD/MTU)
zz ---Previous Location (If applicable)

Figure 2-5

SURRY UNIT 2 - CYCLE 2
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 16535 H08	3 15490	3 11309	2 18101	4A 0 Fresh	2 18399	2 16912	4B 0 Fresh
J	3 15490 P09	4A 0 Fresh	2 17745 K11	4A 0 Fresh	2 15805 K13	3 14745 N11	4B 0 Fresh	4B 0 Fresh
K	3 11309 R08	2 17745 L10	3 15490 J14	2 18257 J10	4A 0 Fresh	2 17582 J12	4B 0 Fresh	
L	2 18101 L08	4A 0 Fresh	2 18257 K09	4A 0 Fresh	2 15929 L12	4B 0 Fresh	4B 0 Fresh	
M	4A 0 Fresh	2 15805 N10	4A 0 Fresh	2 15929 M11	4A 0 Fresh	4B 0 Fresh		
N	2 18399 J08	3 14745 L13	2 17582 M09	4B 0 Fresh	4B 0 Fresh			
P	2 16912 N08	4B 0 Fresh	4B 0 Fresh	4B 0 Fresh				
R	4B 0 Fresh	4B 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (1)	1.85	94
2 (52)	2.55	93
3 (20)	3.10	92
4A (32)	2.60	94
4B (52)	3.10	95

LEGEND

xx
yy
zz

---Batch No.
---Initial Burnup (MWD/MTU)
---Previous Location (If applicable)

Figure 2-6

SURRY UNIT 1 - CYCLE 3
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 15107 Cy1 K08	3 10312 Cy1 H01	1 14044 Cy1 E11	3 14088 Cy1 G14	3 12099 Cy1 J14	3 12099 Cy1 K14	3 8266 Cy1 L14	4C 6273 Cy2 H01
J	3 10312 Cy1 A08	4B 8647 Cy2 L08	3 8880 Cy1 C12	4A 7897 Cy2 J10	3 13341 Cy1 E03	1 12709 Cy1 J13	4C 7123 Cy2 F14	4C 4901 Cy2 G01
K	1 14044 Cy1 L11	3 8880 Cy1 M03	4A 6358 Cy2 M12	5 0	5 0	3 8062 Cy1 J15	4C 7253 Cy2 G02	
L	3 14088 Cy1 P09	4A 7897 Cy2 K09	5 0	3 12099 Cy1 P10	4A 7403 Cy2 K11	4C 7646 Cy2 E03	4C 5619 Cy2 D03	
M	3 14088 Cy1 P07	3 13341 Cy1 C05	5 0	4A 7403 Cy2 L10	3 8266 Cy1 B05	4C 5193 Cy2 E02		
N	3 12099 Cy1 P06	1 13709 Cy1 N09	3 8062 Cy1 R09	4C 7646 Cy2 E05	4C 5193 Cy2 B05			
P	3 8266 Cy1 P05	4C 7123 Cy2 P06	4C 7253 Cy2 B07	4C 5619 Cy2 C04				
R	4C 6273 Cy2 A08	4C 4901 Cy2 A07						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (13)	1.85	94
3 (52)	3.10	92
4A (20)	1.85	95
4B (4)	2.60	95
4C (52)	3.35	95
5 (16)	2.10	95

LEGEND

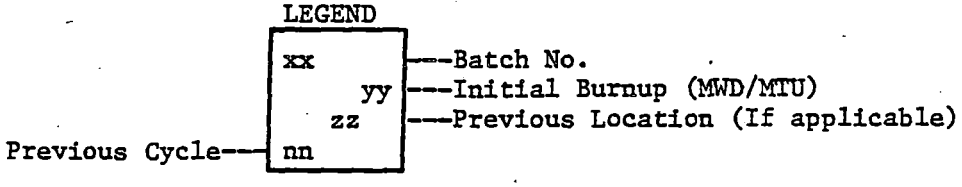


Figure 2-7

SURRY UNIT 2 - CYCLE 3
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 16480 Cy1 K08	4B 10349 Cy2 J14	4A 10870 Cy2 G09	3 13293 Cy1 K14	1 15061 Cy1 J13	4B 10349 Cy2 G14	1 13597 Cy1 H14	4B 6969 Cy2 H01
J	4B 10349 Cy2 P07	1 15061 Cy1 M09	4B 10303 Cy2 L13	4A 11092 Cy2 K12	4B 8540 Cy2 K14	4A 11040 Cy2 J11	4B 5730 Cy2 J15	5 0 Fresh
K	4A 10870 Cy2 J09	4B 10303 Cy2 N11	4A 10874 Cy2 M08	3 9118 Cy1 L14	1 15441 Cy1 K12	3 8867 Cy1 J15	4B 6031 Cy2 L14	
L	3 13293 Cy1 P06	4B 11092 Cy2 M10	3 9118 Cy1 P11	1 16227 Cy1 L09	3 9805 Cy1 M13	4B 6889 Cy2 M13	5 0 Fresh	
M	1 15061 Cy1 N07	4B 8540 Cy2 P10	1 15441 Cy1 M10	3 9805 Cy1 N12	3 13293 Cy1 P10	5 0 Fresh		
N	4B 10349 Cy2 P09	4A 11040 Cy2 L09	3 8867 Cy1 R09	4B 6889 Cy2 N12	5 0 Fresh			
P	1 13597 Cy1 P08	4B 5730 Cy2 R09	4B 6031 Cy2 P11	5 10 Fresh				
R	4B 6969 Cy2 A08	5 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (25)	1.85	94
3 (32)	3.10	92
4A (24)	2.60	94
4B (52)	3.10	95
5 (24)	3.10	95

LEGEND

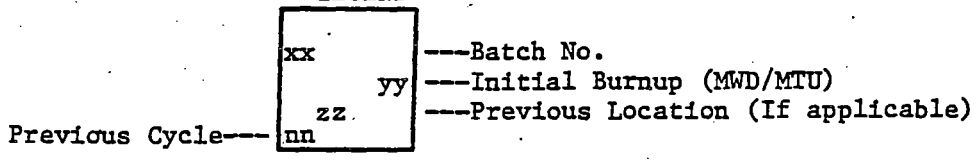


Figure 2-8

SURRY UNITS 1 AND 2 — CYCLE 1
Burnable Poison Rod Loading

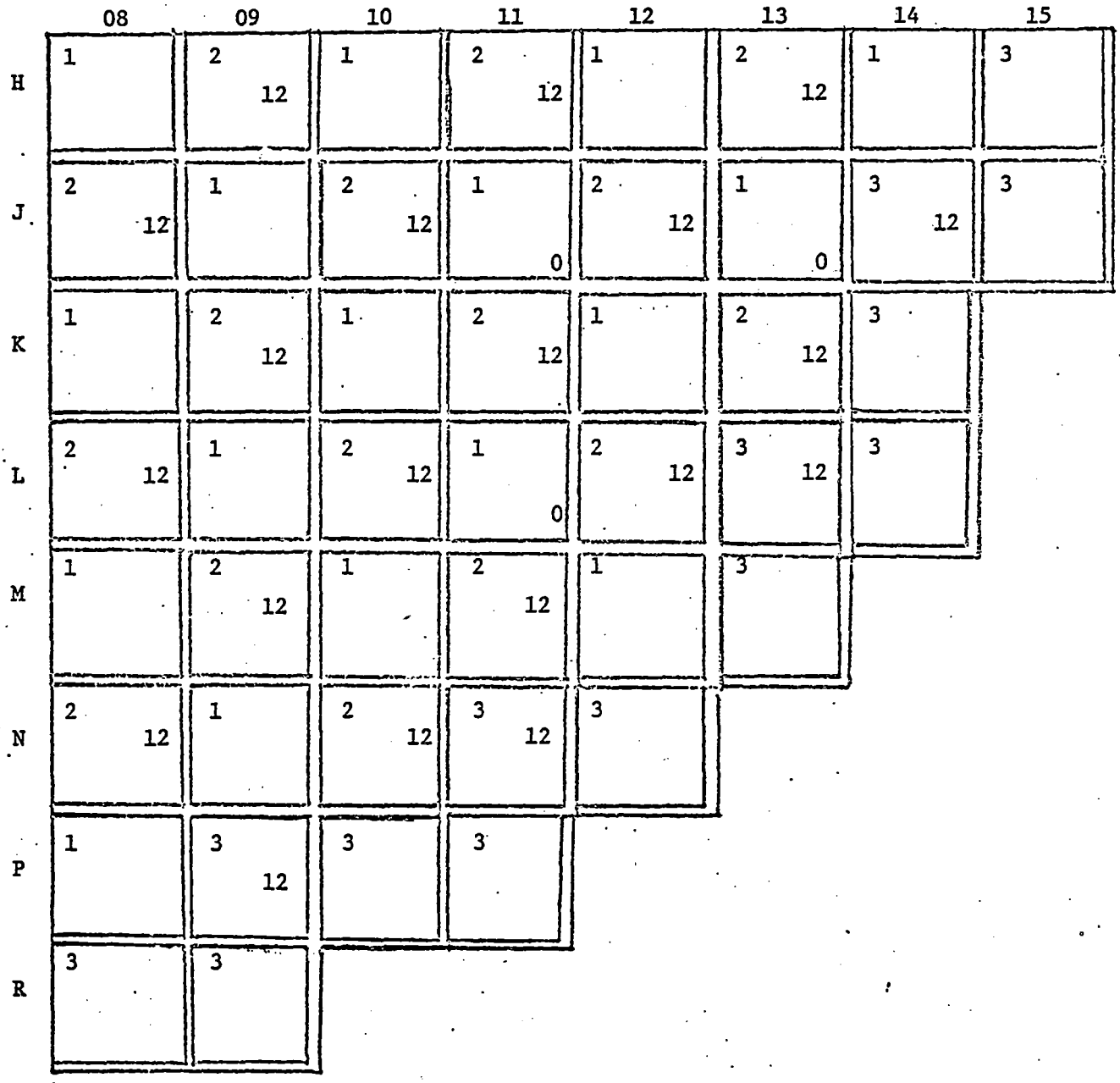


Figure 2-9

SURRY UNIT 1 -- CYCLE 2
BURNABLE POISON ROD LOADING

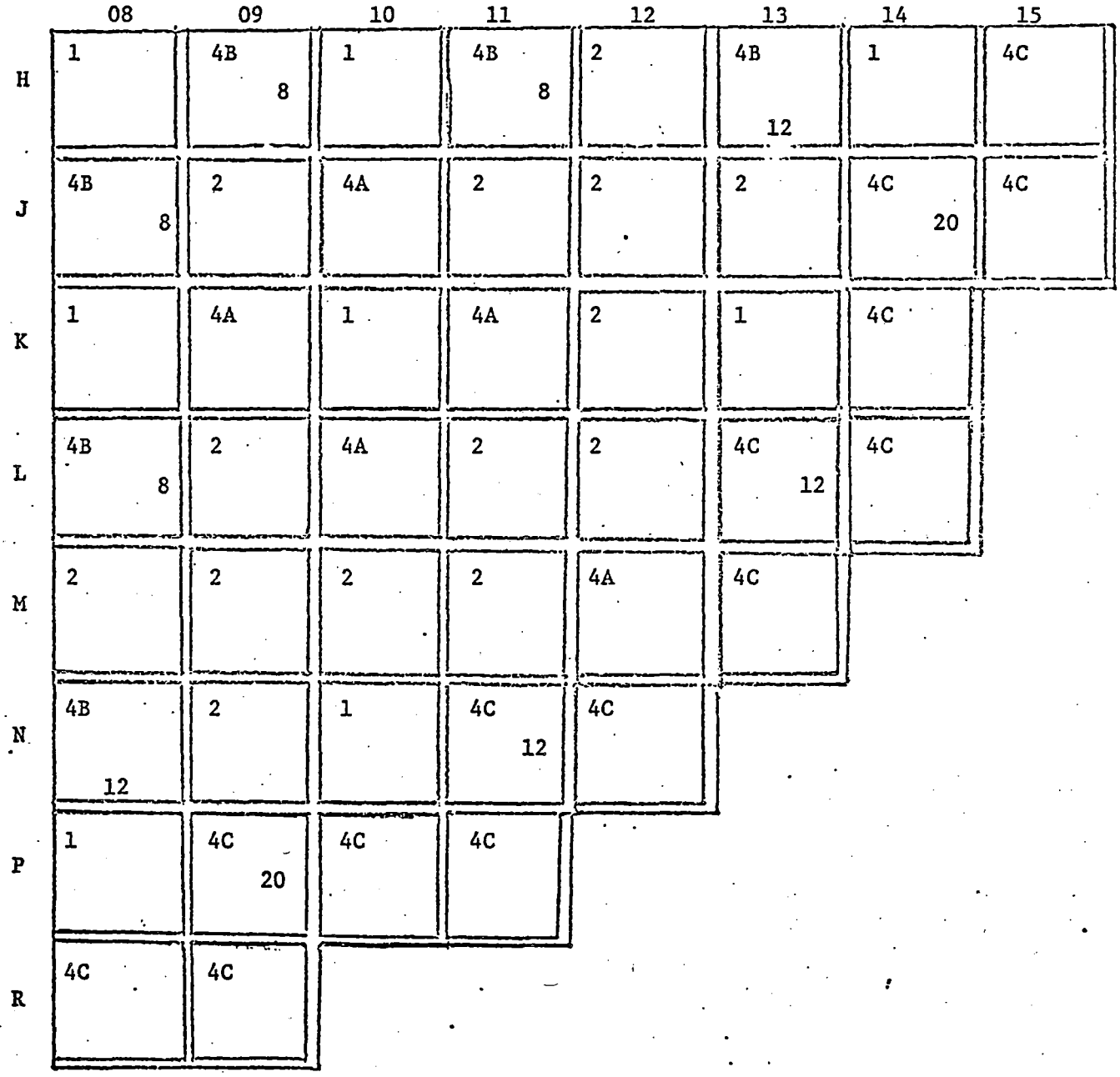


Figure 2-10

SURRY UNIT 2 - CYCLE 2
BURNABLE POISON ROD LOADING

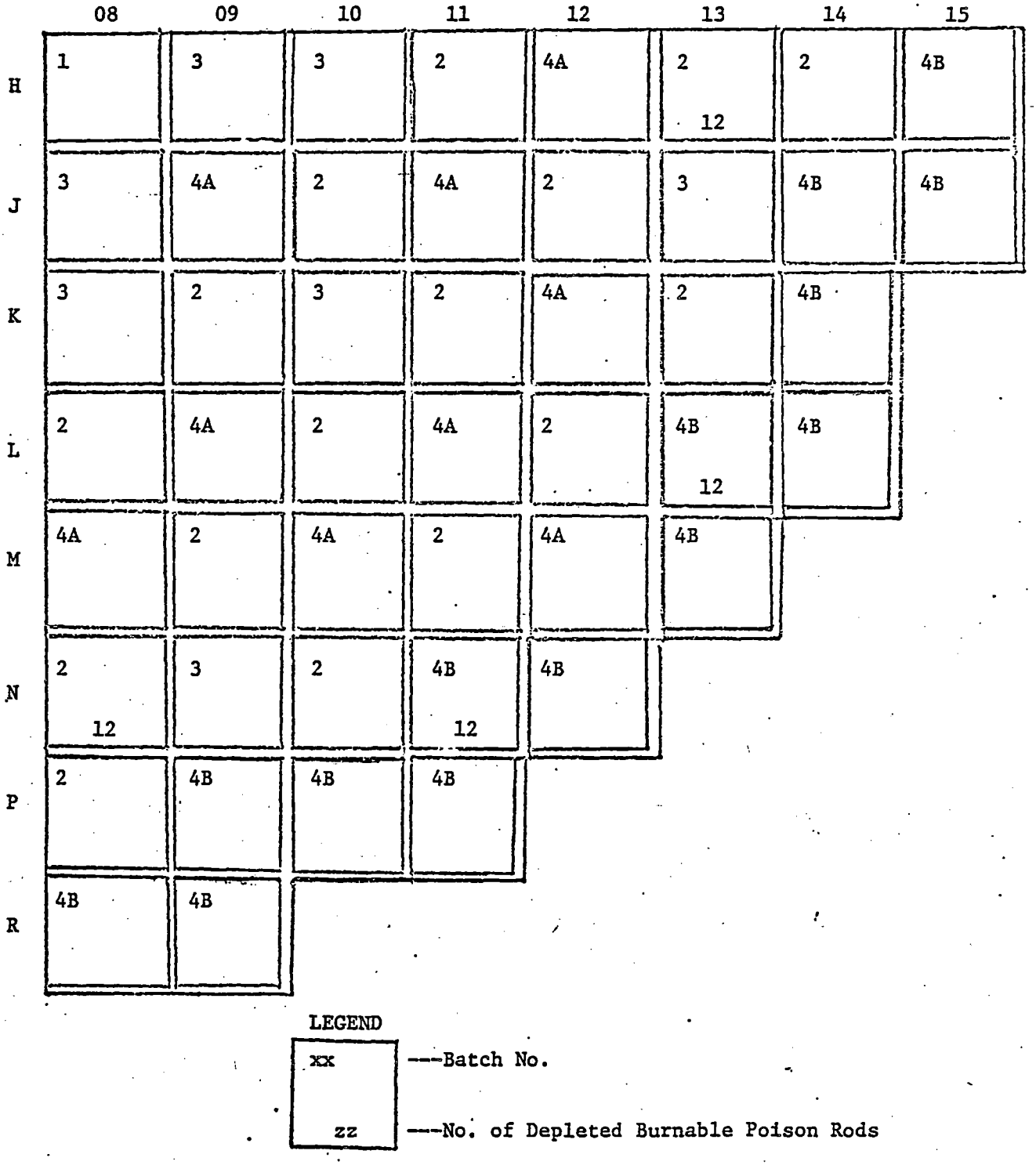
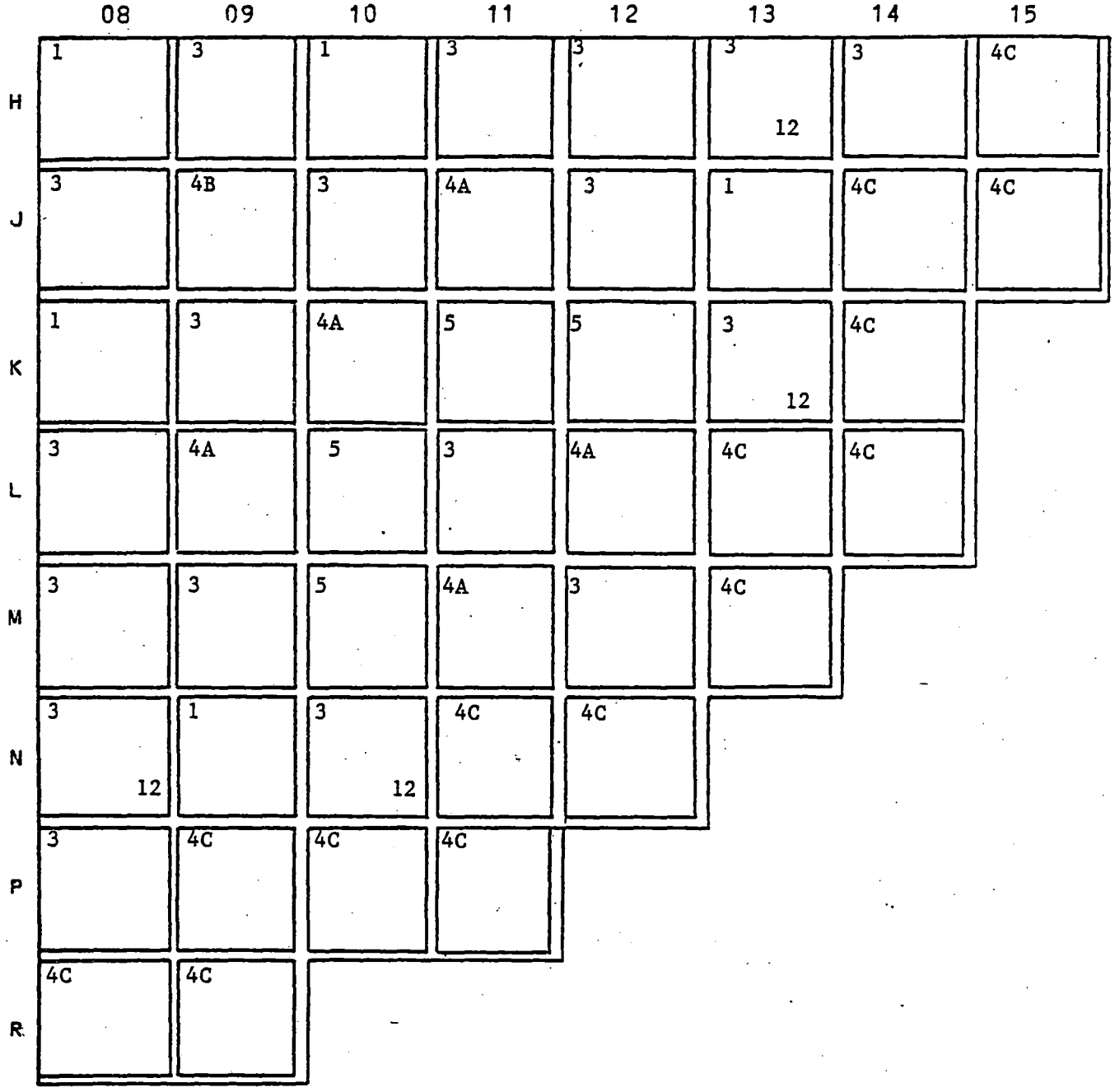


Figure 2-11

SURRY UNIT 1 - CYCLE 3
BURNABLE POISON ROD LOADING



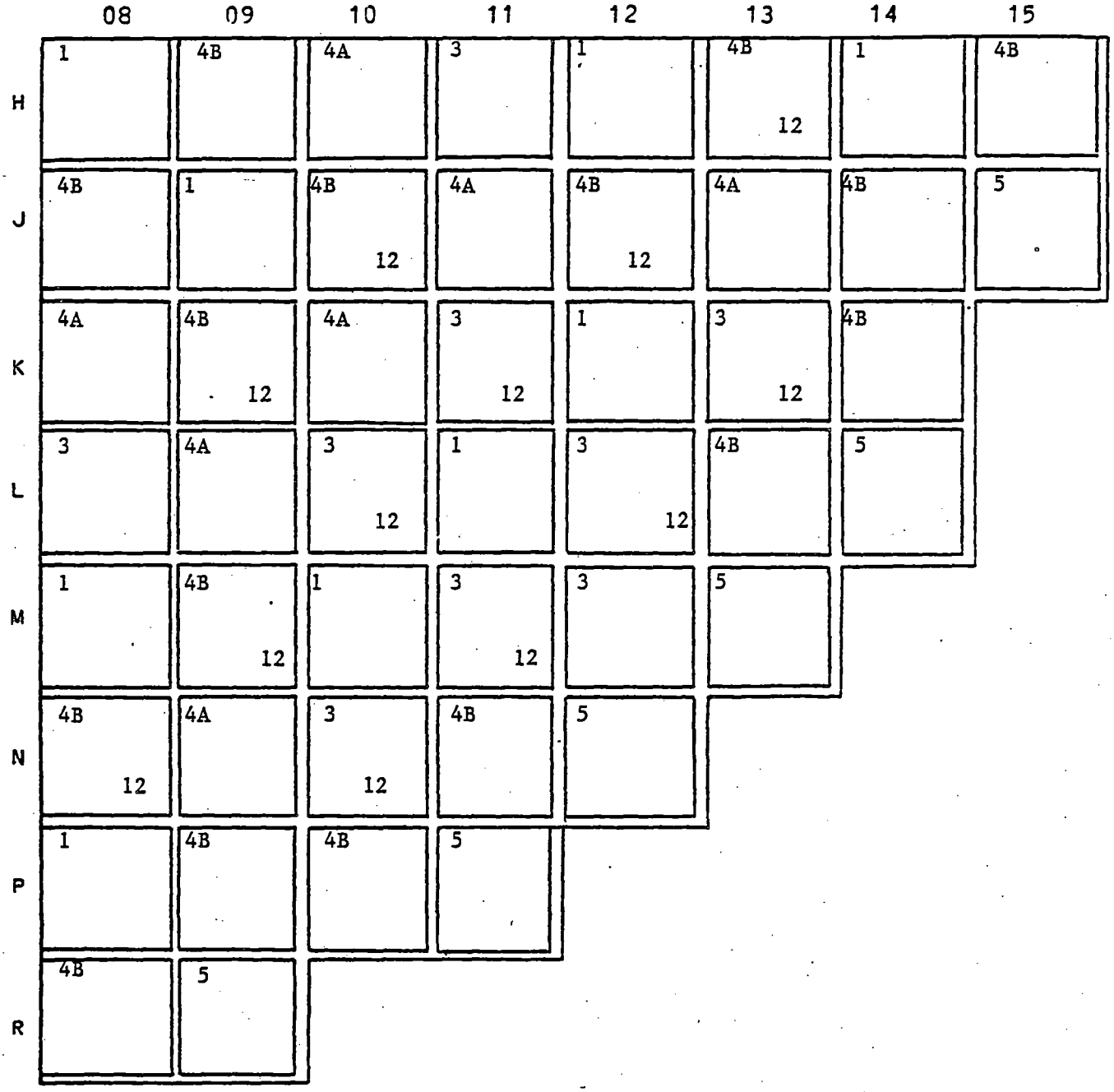
LEGEND

xx —Batch No.

zz —No. of Depleted Burnable Poison Rods

Figure 2-12

SURRY UNIT 2 - CYCLE 3
BURNABLE POISON ROD LOADING



SECTION 3 - MODEL DESCRIPTION

3.1 INTRODUCTION

The PDQ07 one zone model incorporates a few-group, diffusion-depletion theory model, with thermal-hydraulic feedback, to perform spatial neutron flux and material distribution calculations in two-dimensions (x-y) throughout the reactor core as a function of burnup. This calculation is performed at each mesh point represented in the geometry of the reactor core. Furthermore, each fuel assembly is represented by a specified array of mesh blocks formed by the intersection of the mesh lines in the x and y directions (i.e., mesh points). Each mesh block is used to represent a material composition whereas the neutron flux is calculated at each mesh point.

The material compositions of each assembly are homogenized so that each mesh block within an assembly represents an equal number of fuel rods, guide tubes, etc. Therefore, the concentrations of the various nuclides would be initially the same in all mesh blocks within a fresh fuel assembly. Hence, when PDQ07 is using this type of representation, it is called a one zone model. The initial concentrations of nuclides in the homogenized compositions are independent of the number of mesh blocks used to represent an assembly because changing the mesh block size (and hence, the number of mesh points per assembly) only changes the number of points where the neutron flux is calculated. As the assembly undergoes depletion with power operation, however, the material compositions change in each mesh block according to the neutron flux associated with that mesh block. Thus, the calculated flux is dependent to some extent on the mesh block size.

The one zone model performs calculations in several steps. First, a fine-group neutron flux spectrum and the appropriate cross sections as a function of neutron energy are calculated for each material composition by a cross section generating code, such as NULIF. Then the fine-group flux

spectrum is used to spectrum weight and collapse the fine-group cross sections into two neutron energy groups (denoted as the fast and thermal groups or simply two group). The spectrum weighted two-group cross sections associated with each material composition as well as for the baffle and reflector are then used to perform an iterative diffusion theory calculation of the neutron flux as a function of spatial position. Solution of the diffusion theory equations consists of estimating an initial source distribution and eigenvalue, computing the flux in each group at each mesh point, and then recomputing the source and eigenvalue. This process is repeated until the change in flux and/or eigenvalue between successive iterations meets a predetermined convergence criterion. From the converged neutron flux and cross sections, the core power distribution is determined, and subsequently the fuel and moderator temperature distributions are calculated. Thermal feedback effects are included in the diffusion theory calculation by recalculating the neutron cross sections, power distribution, and fuel and moderator temperature distribution iteratively until both the required nuclear and thermal convergence are achieved.

The neutron flux in the core is not only a function of energy and position but is also a function of changes in the nuclide concentrations and cross sections which vary with burnup. The initial nuclide depletion calculation is performed with the intial two-group fluxes and microscopic absorption and fission cross sections for the nuclides in each mesh block that vary with burnup. The neutron flux is then recalculated based upon these new values of nuclide concentrations and cross sections. This process is repeated over an interval of depletion steps until the desired burnup is achieved.

Theoretically, a simultaneous calculation of the neutron flux (at each depletion step) as a function of both space and energy should be performed since the leakage into or out of a given mesh block affects the neutron energy spectrum and, consequently, the spectrum-weighted two-group cross

sections. However, the leakage effect can be approximated by applying a fundamental mode axial buckling to the calculations used to generate the spectrum-weighted two-group cross sections. This approximation allows for the calculational separability of spatial and energy effects and is appropriate since the leakage effect generally has a much smaller effect on the neutron flux calculation than does the properties of the material composition (particularly in large PWR cores where the flux spectrum does not change substantially between adjacent fuel assemblies). Therefore, the separability treatment of space and energy is a valid and accepted assumption for large PWR cores.

Several interrelated computer codes are used to perform the calculations outlined above. The computer codes comprising the PDQ07 one zone model and their interrelationships are presented in the flow chart of Figure 3-1. The PDQ07 computer code itself is the principal reactor analysis calculational tool in the PDQ07 one zone model and is used to perform the two-group, two-dimensional diffusion theory calculations. The other codes provide either input data, data manipulation, or use the PDQ07 code output. As indicated in Figure 3-1, the NULIF computer code is used to calculate the required two-group spectrum-weighted cross sections. The HAFIT computer code formats these cross sections for use in the PDQ07 code (as HARMONY tablesets). The SHUFFLE computer code is a data manipulation code that takes appropriate end-of-cycle nuclide concentrations from the PDQ07 computer code and shuffles this data in the reactor core according to a specified scheme which duplicates calculationally the actual replacement and movement of fuel assmeblies in the reactor core as the result of a refueling.

The remainder of this section describes in greater detail the functioning of each of the computer codes used in the PDQ07 one zone model.

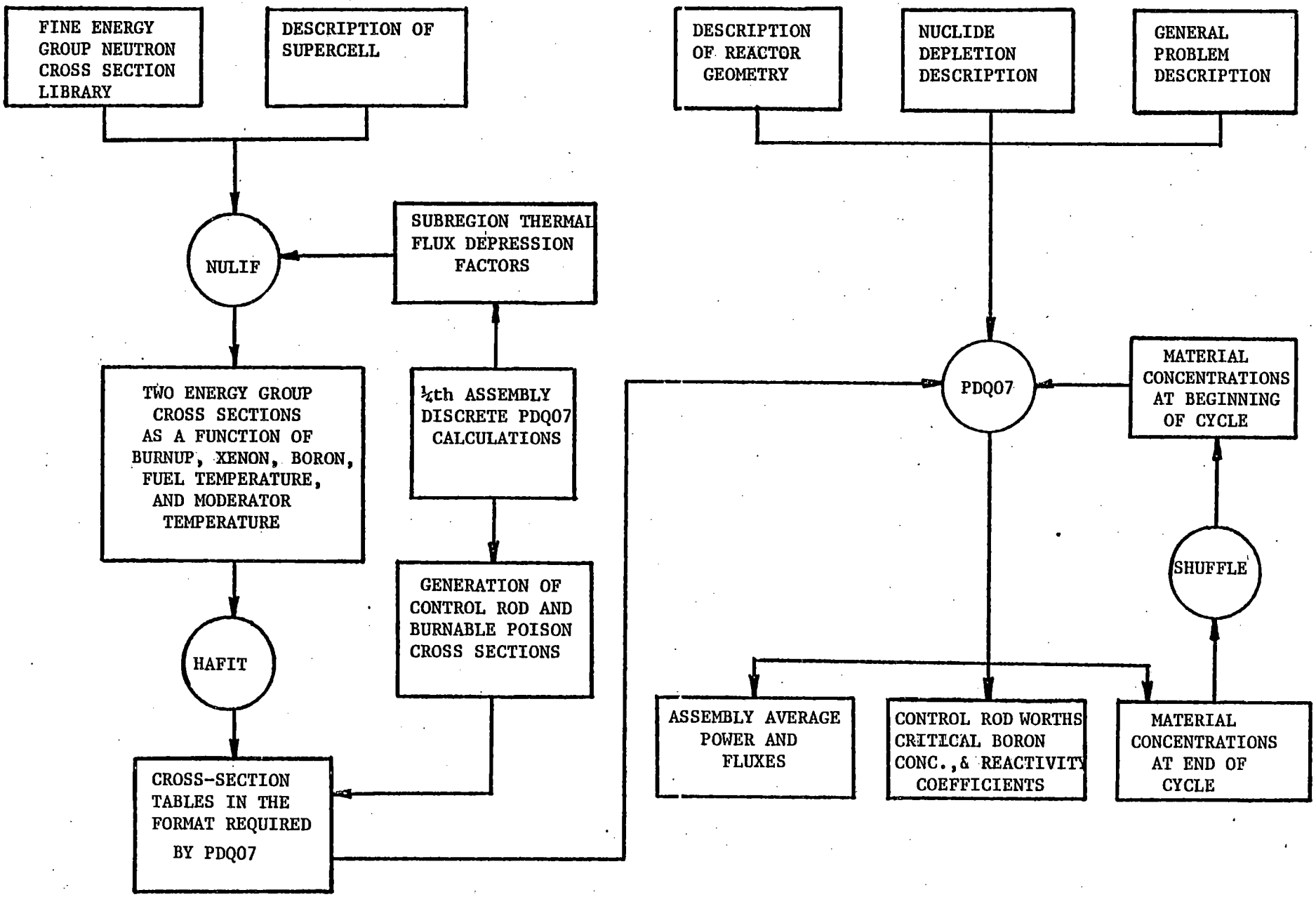
3.2 CROSS SECTION PREPARATION

3.2.1 FINE ENERGY GROUP CROSS SECTION DATA:

The source of basic nuclear cross section data for the NULIF computer

Figure 3-1

FLOWCHART FOR THE ONE ZONE MODEL



code calculations is the standard fine-group cross section library used by Babcock and Wilcox (see Reference 1). This cross section library was supplied by Babcock and Wilcox as part of the FUPAC system.

The library contains cross sections for 31 fast and 80 thermal energy groups with a thermal energy cutoff of 1.85 eV. The fast library contains smooth cross sections, resonance parameters, and an (n, 2n) inelastic scattering matrix for each nuclide. The thermal library contains temperature-dependent cross sections for each thermal energy group and temperature-dependent thermal scattering kernels (both isotropic and anistropic kernels for the bound atom model). The contents of the files in the cross section library are listed in Table 3-1.

The standard fine-group cross section library contains cross section data for all structural materials, fissionable isotopes, fission products, and the moderator-coolant (water) used in the reactor core. The constituents of the library are listed in Table 3-2.

The NULIF code is used to calculate composition-dependent energy spectra and then collapse the fine-energy group cross sections to produce two-group cross sections for each unit cell.

3.2.2 FEW-GROUP CROSS SECTION PREPARATION:

The NULIF computer code calculates two-group spectrum-weighted cross sections for each type of unit cell or groups of unit cells that are present in the reactor core. A unit cell can be either a fuel rod, a control guide tube, a control rod, or a burnable poison rod, and the moderator associated with each rod. A supercell is defined as a representative group of unit cells comprising, for instance, a fuel assembly. For the supercell group, the fuel rod unit cell is designated as the central cell (or cell) and any other unit cell types present in a particular fuel assembly are designated as subregion-X cells (or subcells). The supercell option is used to represent the fuel assembly in the one zone

Table 3-1

CONTENTS OF FINE-ENERGY GROUP CROSS SECTION LIBRARY

FILE 1 GENERAL LIBRARY DATA

TAPE LABEL
MATERIAL CONTENTS
EPITHERMAL GROUP STRUCTURE
THERMAL GROUP STRUCTURE
DELAYED NEUTRON PRECURSOR DATA
FISSION SOURCE DISTRIBUTION DATA
GENERAL MATERIAL PARAMETERS
TEMPERATURE LIST
FISSION PRODUCT YIELDS
RESONANCE ISOTOPE DATA
FISSION SPECTRUM DATA
DELAYED NEUTRON DATA

FILE 2 FAST CROSS SECTION DATA

GROUP DATA
GENERAL MATERIAL PARAMETERS
GENERAL UNRESOLVED RESONANCE DATA
UNRESOLVED RESONANCE PARAMETERS
RESOLVED RESONANCE PARAMETERS
SMOOTH CROSS SECTION DATA

FILE 3 THERMAL CROSS SECTION DATA

GENERAL MATERIAL PARAMETERS
SLOWING-DOWN SOURCE DATA
SMOOTH CROSS SECTION DATA
ISOTROPIC SCATTERING KERNEL
ANISOTROPIC SCATTERING KERNEL

Table 3-2

FINE ENERGY GROUP CROSS SECTION LIBRARY CONSTITUENTS

HYDROGEN-1	PROMETHIUM-149
BORON-10	SAMARIUM-149
BORON-11	URANIUM-234
CARBON-12	URANIUM-235
NITROGEN-14	URANIUM-236
OXYGEN-16	URANIUM-238
SODIUM-23	NEPTUNIUM-237
NATURAL MAGNESIUM	NEPTUNIUM-239
ALUMINUM-27	PLUTONIUM-239
NATURAL SILICON	PLUTONIUM-240
NATURAL CHLORINE	PLUTONIUM-241
NATURAL POTASSIUM	PLUTONIUM-242
NATURAL CALCIUM	AMERICIUM-241
NATURAL CHROMIUM	AMERICIUM-243
MANGANESE-55	BURNABLE POISON (B10)
NATURAL IRON	NON-SAT U233 FISSION PRODUCTS
NATURAL NICKEL	RAP-SAT U233 FISSION PRODUCTS
NATURAL ZIRCONIUM	SLOW-SAT U233 FISSION PRODUCTS
NATURAL MOLYBDENUM	NON-SAT U235 FISSION PRODUCTS
SILVER-107	RAP-SAT U235 FISSION PRODUCTS
SILVER-109	SLOW-SAT U235 FISSION PRODUCTS
CADMIUM-113	NON-SAT PU239 FISSION PRODUCTS
IODINE-135	RAP-SAT PU239 FISSION PRODUCTS
XENON-135	SLOW-SAT PU239 FISSION PRODUCTS

model because only homogenized groups of unit cells can be represented due the larger-than-discrete mesh description used in the one zone geometry representation of the reactor core. The homogenization is performed in a manner that results in all the mesh blocks within a given assembly initially having the same material composition. All the cross sections used in the one zone model are generated with either the supercell option (fuel assemblies) or the heterogeneous cell option (baffle and reflector) of the NULIF computer code with the exception of burnable poison and control rod cross sections. The technique used in the calculation of these cross sections will be discussed in Sections 3.2.5 and 3.2.6 below.

The calculation of the neutron energy spectrum and the spectrum-weighted two-group cross sections for each supercell is described in detail in Reference 1. Those aspects which involve calculations using the supercell option are described below.

With the supercell option of the NULIF code, the material compositions of the central cell and the various subregion-X cells must be homogenized together before the fine-group neutron flux is calculated, since the NULIF code does not perform a spatial calculation for the various subregions. Because of this, a method must be employed to represent the heterogeneous nature of the supercell. This is done by inputting appropriate thermal flux depression factors for each subregion-X cell relative to the supercell. These flux depression factors are generated by a detailed spatial calculation (i.e., a quarter assembly discrete PDQ07 calculation where each fuel rod, thimble cell, and water channel associated with the fuel assembly is explicitly represented). From this detailed spatial calculation, the ratios of the thermal flux in the average fuel cell, thimble cell, and water gap relative to the thermal flux in the entire assembly are determined. The above flux depression factors are then combined in NULIF along with those normally calculated by NULIF for the central cell (i.e., the flux distribution in the fuel pellet, clad, and moderator regions of the central

cell) to give the overall flux depression factors to be applied over each of the 80 thermal fine-groups for each nuclide in the supercell.

NULIF calculates the neutron flux in the supercell for each of 31 fast and 80 thermal energy fine groups. Macroscopic and microscopic cross sections are then determined for one fast and one thermal energy group by collapsing these 111 fine groups based on the neutron flux and cross sections calculated for each fine group. Cross sections are collapsed into two groups for use in PDQ07 calculations. (It has been determined that the use of two groups is adequate for large thermal reactors such as the Surry and North Anna reactors so that the use of more energy groups in PDQ07 would result in substantially longer computer execution times without a corresponding benefit in accuracy.)

The neutron energy spectrum calculated by NULIF for a supercell depends on the material concentrations (i.e., the nuclide concentration or number density) in the unit cell. The material concentrations change during the operation of the reactor as a result of:

1) Depletion of the material
2) Changes in the soluble boron (chemical shim) and xenon concentrations
3) Changes in material temperature

The neutron spectrum is also dependent on the temperature of the fuel due to Doppler broadening of the resonance absorption peaks. The NULIF code is used to calculate the effect of both changes in material concentrations and in the fuel and moderator temperatures on the neutron spectrum and spectrum-weighted two-group cross sections.

NULIF calculates the depletion of supercells based on the spectrum-weighted neutron cross sections and the neutron flux. As the material is depleted, the material concentrations change. This change in concentrations affects both the neutron flux and the neutron spectrum and therefore, requires the frequent recalculation of the spectrum-weighted cross sections.

3.2.3 GENERATION OF FUEL ASSEMBLY FEW-GROUP CROSS SECTIONS:

Fuel assembly cross sections are generated using the supercell option

in NULIF for an assembly with no burnable poison or control rods present.

The NULIF input consists of:

1) Fuel cell dimensions (pellet diameter, clad inside diameter, clad outside diameter, and fuel rod pitch) control rod guide tube and instrument channel dimensions, and the assembly water gap area.

2) Material concentrations for the fuel pellet, clad, gap and moderator as well as the various subregions.

3) Average temperature for the fuel, clad, and moderator.

4) Average power density

5) Description for depletion calculations

6) Description for other calculations to obtain cross sections as a function of moderator and fuel temperature, soluble boron, and xenon

NULIF calculations are then made for the supercell to determine the dependence of the two-group cross sections for each fuel enrichment on:

1) Burnup

2) Soluble boron concentration

3) Xenon concentration

4) Moderator temperature

5) Average fuel temperature

Sets of HARMONY cross section tables based on these NULIF calculations are prepared by the HAFIT code. These tables represent:

1) Microscopic fast and thermal energy group absorption and fission cross sections as a function of burnup, soluble boron concentration, and xenon concentration.

2) Macroscopic fast transport and removal, and thermal transport cross sections as a function of burnup, soluble boron concentration, and xenon concentration

3) The effect of fuel and moderator temperature changes on the macroscopic cross sections

3.2.4 GENERATION OF BURNABLE POISON (BP) FEW-GROUP CROSS SECTIONS

The burnable poison (BP) cross sections are not calculated directly by NULIF with the supercell option because of calculational inefficiencies.

Normally, the BP cross sections would be calculated using the two-group cross sections for fuel assemblies containing BP by performing a NULIF supercell calculation (as described in Section 3.2.3 above) where one of the subregions contains the appropriate amount of BP. In other words, the cross sections would be generated in the same straightforward manner as for assemblies without BP. However, the fact that the heterogeneous effect of the various subregions are accounted for by applying thermal flux depression factors determined from quarter assembly discrete PDQ calculations must be considered. In general, for assemblies containing no BP, these flux depression factors do not change significantly as a function of burnup so that the flux depression factors calculated at zero burnup are adequate at any stage of depletion. However, this is not the case for subregions containing burnable poison rods because the BP burns out rapidly with increasing burnup causing the flux depression factors to vary significantly with burnup. Since a large number of quarter-assembly PDQ runs would have to be made to calculate these factors for input to NULIF, it was decided to use only the quarter assembly runs themselves to generate the BP cross sections. This calculational technique was performed in accordance with the following procedure:

1) Set up a discrete quarter assembly PDQ07 calculation with a 12BP rod cluster inserted in an assembly of the desired fuel enrichment

2) Perform a depletion calculation at the core average power density

3) At appropriate burnup steps during the depletion, perform a calculation with the BP rods removed from the assembly

4) Determine the change in assembly average, flux-weighted, two-group macroscopic cross sections (or "delta" cross sections) resulting from the removal of the BP rods

5) These delta cross sections are then incorporated into the same cross section tablesets for a non-BP assembly to give HARMONY tablesets that are applicable to assemblies containing BP (as represented in the one zone model)

Even though the delta cross sections are actually macroscopic quantities, they are incorporated in the one zone HARMONY tables as "microscopic" cross sections. Then, a nuclide is defined to represent the presence or absence of BP. If a

given fuel assembly in the core has 12 BP rods then the BP nuclide concentration is set equal to unity in that assembly. Macroscopic cross sections for the BP are then determined during execution of the one zone calculation by taking the product of the "microscopic" delta cross sections and the BP nuclide concentration. Thus, the cross section contribution of the BP is effectively added to the total cross sections for each mesh block within the assembly. If no burnable poison is present in a fuel assembly then the BP concentration is set equal to zero so that the macroscopic cross section contribution is also zero.

BP clusters comprised of other than 12 rods are modeled by taking the ratio of the number of rods to 12 rods as the BP concentration. For instance, the BP concentration for 20 rods would be 20/12 x 1 or 1.667 while the BP concentration for 8 rods would be 8/12 x 1 or 0.667. This approach was adopted because it was found that the delta cross section values were directly proportional to the number of BP rods present in a fuel assembly to within an acceptable degree of accuracy.

3.2.5 GENERATION OF CONTROL ROD FEW-GROUP CROSS SECTIONS

The control rod cross sections are generated in a similar manner as the BP cross sections. The procedure for calculating the control rod cross sections is outlined in the following steps:

1) Set up a discrete quarter assembly PDQ07 calculation (for a given fuel enrichment) with no control rods inserted.

2) Perform a depletion calculation at the core average relative power density.

3) At appropriate burnup steps during the depletion, perform a calculation with the control rods inserted.

4) Determine the change in assembly average flux-weighted, two-group macroscopic cross sections resulting from the insertion of the control rods.

5) In the same manner as was done with the BP delta cross sections, the control rod delta cross sections are then incorporated into the appropriate HARMONY cross section tablesets.

3.2.6 GENERATION OF REFLECTOR FEW-GROUP CROSS SECTIONS

The cross sections in the reflector region are calculated by NULIF using the unit cell option. The reflector region extends from the outside of the core baffle to the reactor vessel wall, including the thermal shield and core barrel. The stainless steel and water in this region of the reactor are homogenized (volume-weighted) in NULIF, and a neutron spectrum and spectrum-weighted cross sections are calculated for this region. These calculations are performed for several soluble boron concentrations, and tables representing the cross sections of the reflector region as a function of soluble boron concentration are prepared by HAFIT for use in the HARMONY tablesets of the PDQ07 code.

3.2.7 GENERATION OF BAFFLE FEW-GROUP CROSS SECTIONS

The cross sections for the stainless steel baffle region were obtained from cross sections identical to those used in the discrete model. (See Reference 6). The discrete model baffle cross sections were generated by unit cell NULIF calculations for stainless steel. Then the cross sections were adjusted to provide a discrete model calculated radial power distribution that closely agreed with the corresponding measured power distribution for beginning of Cycle 1 of Surry Unit No. 1. However, since the baffle cannot be explicitly represented in the one zone model (i.e., the mesh spacing is about twice the actual baffle thickness) some of the reflector region must also be included. Therefore, cross section values for each of these material compositions (baffle and reflector) were appropriately volume weighted to produce cross sections to represent this region. Also, the standard modification of the macroscopic thermal absorption cross section was made to bring the one zone model calculated core power distribution into closer agreement with the discrete model calculated core power distribution. This was done by first performing a one zone calculation with the same cross sections (appropriately volume weighted) as used in the discrete model. If the one zone model then predicted a higher power on the core periphery, for instance,

then the macroscopic thermal absorption cross section was adjusted upward to force lower peripheral assembly powers. A trial and error process was employed until the one zone predicted overall radial power distribution was equivalent to the discrete model prediction. The need for the modification was not unexpected, since the relative overall radial power distribution is influenced by changes in mesh spacing and cross section differences. In addition, diffusion theory codes do not predict as well in regions of rapidly changing fluxes, such as is found near the core periphery, since the basic assumption in diffusion theory requires an essentially isotropic flux distribution.

3.2.8 FORMATING OF FEW-GROUP CROSS SECTIONS:

The HAFIT computer code is a data manipulation code which is used to prepare HARMONY cross section tablesets for input to PDQ07. The input to the HAFIT program consists of a magnetic computer tape containing the spectrum-weighted two-group cross sections calculated by the NULIF code, and a description of how these cross sections are to be used to create a set of HARMONY tables for input to PDQ07. An automated data processing code like HAFIT must be used to prepare the HARMONY tablesets for PDQ07 because of the substantial volume of data involved. More detailed information on HAFIT and the HARMONY system is contained in References 2, 4, and 6.

3.3 DIFFUSION THEORY CALCULATION

3.3.1 INTRODUCTION

The PDQ07 computer code, as used in the PDQ07 one zone calculational model, is a two-dimensional, two-group, diffusion-depletion program which is used to calculate the neutron flux, power, and nuclide concentrations as a function of radial (x-y) position and burnup. The PDQ07 computer code utilizes the appro-triate and properly formated cross sections along with the initial description of the reactor core (i.e., geometry and material composition description) to

calculate the neutron flux distribution at spatial mesh points (and for two energy groups) at the desired core power. The spatially dependent neutron flux is then combined with the appropriate nuclide concentrations and cross sections to obtain the spatially dependent power distribution. Once the initial spatially dependent flux and power distributions are obtained, the depletion of the nuclide concentrations is calculated.

3.3.2 GEOMETRY DESCRIPTION

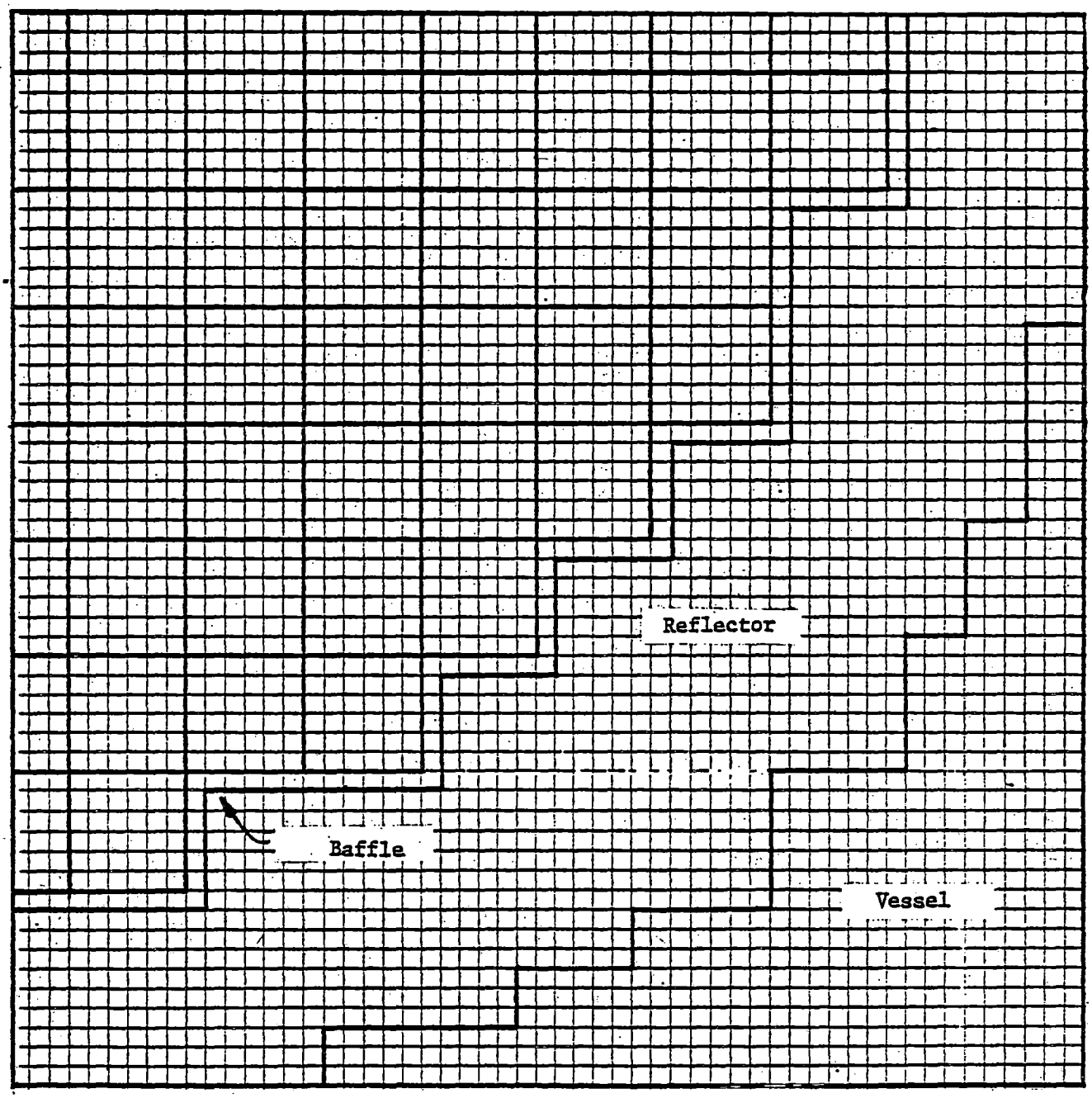
The size of the mesh spacing used in the one zone model to represent a fuel assembly may vary. The exact size of the mesh spacing selected depends on the type of calculation to be performed, the accuracy desired, and the computer resources available. In the one zone model, the material compositions of the fuel rod cells, instrument channel, control rod guide tubes and assembly water gap are all homogenized together in each assembly. A typical quarter core geometry representation showing the mesh spacing over the region of solution is shown in Figure 3-2. This representation depicts a 6 x 6 mesh spacing per assembly where there are 55 mesh lines (45 fuel, 1 baffle, 9 reflector) equally spaced in both x and y directions, resulting in a total of 3025 mesh points. By comparison, the number of mesh points used to represent the same core with a discrete model is 132 x 132 or 17,424 which explicitly represents each fuel rod, guide tube and water gap in the core. Since the computer calculational time is almost directly proportional to the number of mesh points, the result is that the running time can be reduced by a factor of 5 to 12 (depending on the particular type of calculation and the mesh spacing used) with the one zone relative to the discrete model.

The boundary conditions used in the quarter core solution of the two-dimensional diffusion theory equation are:

1) Zero current for the boundaries located along the core axis
2) Zero neutron flux for the boundaries located at the reactor vessel wall

FIGURE 3-2

Typical One Zone Quarter Core
Geometry Representation
(6 x 6 mesh/assembly)



The boundary conditions used in the full core geometry representation are zero flux at the boundaries located at the reactor vessel wall.

3.3.3 DEPLETION EQUATIONS:

Each mesh block in the PDQ07 code contains a single homogenous composition. The volume-weighted nuclide concentrations for each mesh block in the core are input to PDQ07 for beginning of life core conditions. In addition, a set of equations, which is used by PDQ07 to calculate the change in nuclide concentrations with burnup, is input to PDQ07 for each different composition in the core.

The appropriate set of material (nuclide) depletion equations is assigned in PDQ07 to each mesh block. These equations are used by PDQ07 to deplete the nuclide concentrations in each mesh block based on:

1) The average fast and thermal energy group neutron fluxes calculated by PDQ07 for the mesh block

2) The spectrum-weighted fast and thermal group absorption cross sections determined by PDQ07 from the cross section table-set assigned to the mesh block.

The depletion chains described with these depletion equations in the one zone model for each fuel cell type are summarized in Table 3-3. A detailed description of how the depletion equations are input to PDQ07 describing these depletion chains is given in Section 5 of Reference 2.

3.3.4 THERMAL-HYDRAULIC FEEDBACK PARAMETERS:

The input to PDQ07 required for thermal-hydraulic feedback consists of:

1) Coolant inlet enthalpy

2) Heated perimeter per unit area of flow

3) Hydraulic diameter of the channels

4) Flow area of the fuel assembly per total area of flow

5) System pressure

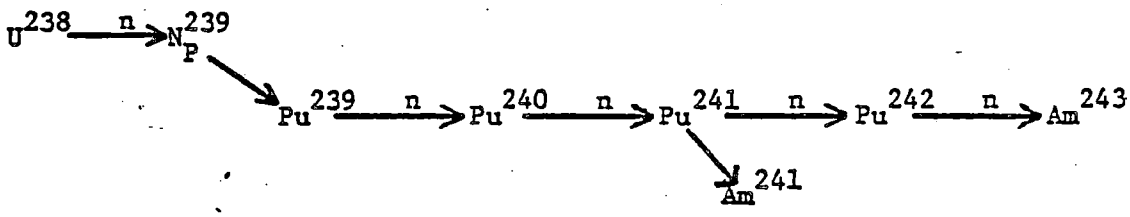
6) Difference between average fuel temperature and moderator temperature as a function of relative power density

Table 3-3

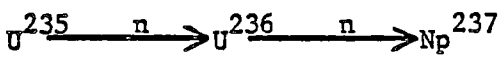
Depletion Equations used in PDQ07

1. Neutron Absorptions Not Leading to Fission

   a. U^{238}, Pu depletion chain



   b. U^{235} depletion chain



2. Neutron absorptions which produce fission are represented with the following fission products:

   a. I^{135}, Xe^{135}, Pm^{149}, and Sm^{149} which are represented explicitly

   b. Two groups of fission products which eventually build up to an equilibrium concentration (since they are created by fission reactions and destroyed by decay reactions). One group is characteristic of fission reactions by uranium isotopes and the other group is characteristic of fission reactions by plutonium isotopes.

   c. Two groups of non-saturating fission products which are either stable isotopes or have half-lives greater than a few years. Again, one group is characteristic of fission reactions by uranium isotopes and the other group is characteristic of fission reactions by plutonium isotopes.

The strategy used in the feedback calculation consists of first making an initial estimate of the fuel and moderator temperature for each coolant channel. Based on this initial estimate and the cross section tables for each fuel cell, the PDQ07 code calculates the two-group, spectrum-weighted cross sections for each mesh block. These cross sections are used in a diffusion theory calculation of power density in each fuel rod cell. This power density is then used in a calculation of the fuel and moderator temperature for each fuel cell. In turn, the new fuel and moderator temperatures are used to calculate new two-group, spectrum-weighted cross sections for another diffusion theory power distribution calculation. This process is continued until the power density for each fuel rod in the N^{th} iteration differs from the power density in the $N-1^{th}$ iteration by less than the convergence criterion.

Thermal-hydraulic feedback effects are represented in the PDQ07 model in order to more accurately calculate the power and burnup distributions.

SECTION 4 - COMPARISON OF ONE ZONE PREDICTIONS TO DISCRETE MODEL PREDICTIONS AND EXPERIMENTAL DATA

4.1 INTRODUCTION

The purpose of this section is to present a comparison of analytical predictions from the PDQ07 one zone model with PDQ07 discrete model predictions and experimental data obtained from Surry Units No. 1 and 2. These comparisons encompass both initial and reload cycles of operation in order to demonstrate both the accuracy and flexibility of the one zone model.

Since it is planned to perform calculations with the one zone model that would have otherwise been performed by the discrete model, it is important that the results compare well. Therefore, the accuracy of the one zone model is established by benchmarking against this model whose accuracy has been previously verified (see Reference 6). In areas where the discrete model has not been applied extensively, such as the calculations of temperature reactivity coefficients, the one zone model is compared to available measured data.

4.2 ANALYTICAL CALCULATIONS

The types of calculations described in this section fall into two general groups: power distribution calculations and reactivity calculations.

Power distribution calculations include:

1. Fuel assembly average relative radial power distributions as a function of cycle depletion.
2. Power distributions resulting from various symmetric and non-symmetric control rod bank positions.
3. Batch and assembly average burnup sharing.

Power distribution calculations as a function of burnup are performed to assure that the assemblywise relative powers are within acceptable limits for the entire cycle depletion (power distributions for various rodded configurations are typically performed at beginning of cycle (BOL) and end of cycle (EOC) at

hot full power (HFP) or hot zero power (HZP). Cycle depletions are run with all rods out (ARO) over a series of depletion intervals, typically 1000 or 2000 MWD/MTU increments. The flux and power distributions are calculated at the beginning of each interval and is assumed constant for the entire depletion interval. The change in nuclide concentrations over the depletion interval is calculated based on this flux distribution and provides the nuclide concentrations that are to be input to the next depletion step. Then the flux and power distributions for the next time step are determined based upon the nuclide concentrations input from the previous time step. This process is repeated until the end of cycle burnup is attained. From the last time step, the batch and assembly average burnups and the isotopic data for fuel assemblies which are to be either permanently discharged or shuffled to the next cycle of operation are obtained.

Reactivity calculations include:

1. Integral control rod bank worths
2. Stuck and ejected rod worths
3. Critical boron concentrations and differential boron worths
4. Isothermal temperature coefficients

Integral control rod bank worths are calculated by holding all reactor parameters constant (including soluble boron concentration) except for the rod bank(s) whose worth is to be determined. First, a calculation is performed for all rods out, then for D bank fully inserted, then C and D banks fully inserted, etc. The change in core reactivity resulting from each of the rod bank configuration changes is a direct measure of the control rod bank worth.

Stuck rod worths are calculated to assure that there is adequate shutdown margin with the most reactive control rod stuck out of the core. The method for calculating stuck rod worth is to perform two full core one zone calculations at hot zero power (HZP): one with all rods in (ARI), and the other with

ARI less one rod removed from the core. The change in reactivity resulting from these two cases is the stuck rod worth. Calculations for ejected rod worths are similar to stuck rod worth calculations except that the initial conditions are different, i.e., ejected rod worth calculations would typically be performed for HZP or HFP operation with the control rod bank(s) at approximate insertion limits. The ejected rod worth is then determined by calculating the reactivity change resulting from the removal of one control rod. Misaligned control rod conditions can also be simulated in a similar manner.

Core criticality is maintained by adjusting the boron concentration as a function of burnup, power level, etc. The boron concentration at which the reactor is just critical is called the critical boron concentration. This value is calculated by first using a best-estimate boron concentration to determine the core k_{eff} and then correcting this boron concentration to a value which corresponds to the just critical condition. Since the one zone model does not represent the core explicitly, it will not necessarily compute a core k_{eff} equal to one with the boron concentration set at the just critical condition (as would be determined from measured data). Therefore, it is necessary to establish a "target" k_{eff} at beginning of life (BOL) based on measured data or discrete model PDQ calculations for the critical boron concentration. This target k_{eff} is then used as a basis throughout the cycle depletion to predict the critical boron concentration. It has been found that the target k_{eff} does not deviate significantly during the cycle or from one cycle to another so that the need to update this value is not usually necessary.

The isothermal temperature coefficient is defined as the change in core reactivity per degree change in the moderator, clad, and fuel temperature (i.e., the sum of the moderator and Doppler temperature coefficients). The calculation of the isothermal temperature coefficient values at the hot zero power (HZP) condition is important because they can be compared to plant

measurements taken during startup physics testing and therefore, can provide a basis for evaluating the accuracy of isothermal temperature coefficient, moderator temperature coefficient, and Doppler coefficient design predictions. At HZP conditions (547°F for the Surry Units) the isothermal temperature coefficient is calculated by determining the change in reactivity, $\Delta\rho$, for +2°F and -2°F around 547°F. In other words, one calculation is performed at 545°F and another at 549°F, where these temperatures are held uniform across the core. Then the isothermal temperature coefficient is derived by dividing $\Delta\rho$ by ΔT. At part or full power operation the calculational method is different because the moderator and fuel temperature are not uniform over the core. For this situation, two calculations (e.g., at HFP) having different core inlet moderator enthalpies are made. The isothermal temperature coefficient is then $\Delta\rho$ divided by the change in core average moderator temperature resulting from the change in inlet enthalpy. In these calculations, the change in the core average fuel temperature is assumed to be the same as the change in moderator temperature.

4.3 MEASUREMENT DATA

Measurement data is obtained for the Surry Units from routine physics testing conducted during the startup of each cycle of operation as well as from routine core performance monitoring conducted during the depletion of each cycle. The methods used for measuring core power distribution, burnups, control rod bank worths and critical boron concentrations are described in Reference 6. The procedure for measuring stuck rod worths and isothermal temperature coefficients were not covered in Reference 6 and will be discribed briefly below.

Stuck rod worths were measured during the initial startup physics testing for Cycle 1 of each of the Surry Units. The value of the stuck rod worth was measured by manually tripping one of the control rods from step 228 (out of the core), observing the average flux level change indicated by the source range detectors, and calculating the resulting change in reactivity from the point-kinetics equation.

Isothermal temperature coefficients are measured during the startup of each cycle by adjusting the reactor coolant system (RCS) heat gains and losses through the use of condenser steam dump valves to establish uniform heatup and cooldown rates, and then monitoring the resulting reactivity changes. Specifically, the measurements are performed during approximately a 10°F/hr RCS heatup and/or cooldown ramp during which the RCS temperature changes up to 4°F. Reactivity is determined using the reactivity computer and is plotted against RCS temperature on an x-y recorder. The temperature coefficient is then determined from the slope of the plotted line. Normally, both positive and negative temperature swings are used to negate the effects of any inadvertant reactivity additions to the system (i.e., boron concentration mismatches between pressurizer and/or volume control tank and the main coolant). The measurements are done at very low power levels to insure that nuclear heat is not added, thereby maintaining the moderator and fuel at approximately the same temperature and minimizing Doppler feedback.

4.4 RESULTS

Representative results of the one zone model power distribution and reactivity predictions compared to the discrete model calculations and measurements obtained from the Surry Power Station are presented in this section. The specific types of results compared are summarized in Table 4-1. Table 4-2 presents the standard deviation between predicted (one zone and discrete models) and measured assembly average relative power distributions for representative reactor conditions for both initial and reload cycles of Surry Units 1 and 2. The standard deviation (σ) between predicted and measured power distribution is given by:

$$\sigma = \left(\frac{1}{156} \sum_{i=1}^{157} (X_i^c - X_i^m)^2\right)^{1/2}$$

where: X_i^c is the calculated assembly average power for the i^{th} assembly

X_i^m is the measured assembly average power for the i^{th} assembly

Figures 4-1 through 4-4 provide representative (i.e., of the cases delineated in Table 4-2) quarter core comparisons of individual assembly relative power distributions between measurement and one zone model predictions at various core conditions during Cycle 2 of Surry Unit No. 2. Also given for each fuel assembly is the percent difference which is defined as:

$$\% \text{ difference } (\%\Delta) = \frac{\text{One Zone Model - Measured}}{\text{Measured}} \times 100$$

Figures 4-5 and 4-6 provide a comparison between the one zone and discrete models for calculating full core power distributions. Figure 4-5 gives the one zone and discrete calculated assembly average relative power densities (and percent differences) for Surry Unit No. 1, Cycle 1 at a BOC, HZP, and ARO condition. Figure 4-6 gives the assembly average relative power distributions (and percent differences for the assembly relative powers greater than unity) for the same conditions except that all rods are inserted with rod H-14 assumed to be stuck out of the core.

A comparison of the assembly average end of cycle burnup distributions between the one zone model and measurement is given in Figures 4-7 and 4-8 for an initial and reload cycle, respectively. Also included is the corresponding comparison of the batch average burnup sharing.

The reactivity comparisons are given in Tables 4-3 through 4-9. All comparisons were made at HZP, BOC conditions with the exception of the critical boron concentration versus burnup given in Tables 4-5 and 4-6 which were calculated at HFP at the indicated burnup. Both one zone model and discrete model predictions were compared to the measured reactivity values except for the isothermal temperature coefficient predictions (Table 4-9) where only one zone model predictions and measured values were compared.

TABLE 4-1

ONE ZONE MODEL COMPARISONS

TYPE OF COMPARISON	REACTOR CONDITION AT WHICH COMPARISON IS MADE	REFERENCE FOR COMPARISON
Power Distribution		
Assembly Average	Units 1 & 2, Cycle 1 & 2 Operation	Table 4-2
	Unit 2, Cycle 2, HZP, ARO, BOC	Figure 4-1
	Unit 2, Cycle 2, HZP, D-bank in, BOC	Figure 4-2
	Unit 2, Cycle 2, HFP, ARO, 2790 MWD/MTU	Figure 4-3
	Unit 2, Cycle 2, HFP, ARO, 8850 MWD/MTU	Figure 4-4
	Unit 1, Cycle 1, HZP, ARO, BOC	Figure 4-5
Stuck Rod	Unit 1, Cycle 1, HZP, ARI, less rod H-14, BOC	Figure 4-6
Assemblywise Burnup, and	Unit 1, Cycle 1, EOC	Figure 4-7
Batch Burnup Sharing	Unit 1, Cycle 2, EOC	Figure 4-8
Reactivity		
D & C Bank Control Rod Worths	Unit 1 & 2, Cycles 1, 2, & 3, HZP, BOC	Table 4-3
Total Shutdown and Stuck Rod Worths	Unit 1 & 2, Cycle 1, HZP, BOC	Table 4-4
Critical Boron Concentration vs. Burnup	Unit 1, Cycles 1, 2, & 3	Table 4-5
	Unit 2, Cycles 1, 2, & 3	Table 4-6
Critical Boron Concentration for Various Control Rod Configurations	Unit 1, Cycles 1, 2, & 3, BOC	Table 4-7
Differential Boron Worth	Unit 1, Cycles 1, 2, & 3, BOC	Table 4-8
Isothermal Temperature Coefficients	Unit 1, Cycles 1, 2, & 3, BOC	Table 4-9

TABLE 4-2

COMPARISON OF PREDICTED AND MEASURED ASSEMBLY AVERAGE POWER DISTRIBUTIONS FOR SURRY UNITS 1 AND 2, CYCLES 1 AND 2

Unit	Cycle	M/D Map Number	Power Level (%)	Control Rod Configuration	Cycle Burnup (MWD/MTU)	Standard Deviation Between Measured And One-Zone	Discrete
1	1	40	75	ARO	1150	0.021	0.015
1	1	59	94	ARO	6790	0.022	0.019
1	1	73	60	ARO	13415	0.017	0.021
1	2	1	0	D-Bank In	0	0.048	0.030
1	2	2	0	ARO	0	0.058	0.043
1	2	16	99	ARO	3102	0.028	0.021
1	2	19	100	ARO	6569	0.028	0.019
2	1	38	92	ARO	2950	0.032	0.029
2	1	48	91	ARO	6780	0.017	0.015
2	1	65	100	ARO	14520	0.016	0.020
2	2	1	2	ARO	0	0.029	0.043
2	2	2	2	D-Bank In	0	0.032	0.042
2	2	12	97	ARO	2790	0.012	0.016
2	2	23	100	ARO	8850	0.024	0.015

FIGURE 4-1

ASSEMBLYWISE 1-ZONE VS INCORE REL. POWER DIST. FOR S2C2 HZP ARO AT 0 MWD/T
ONE-ZONE VS MEASUREMENT

	1	2	3	4	5	6	7	8
1	0.720 0.690 4.3	1.030 0.980 5.1	1.080 1.010 6.9	0.970 0.940 3.2	1.200 1.210 -0.8	0.840 0.850 -1.2	0.840 0.860 -2.3	0.750 0.770 -2.6
2	1.030 0.980 5.1	1.160 1.110 4.5	0.910 0.870 4.6	1.210 1.180 2.5	1.000 1.010 -1.0	1.050 1.070 -1.9	1.190 1.250 -4.8	0.620 0.650 -4.6
3	1.080 1.040 3.8	0.910 0.880 3.4	0.990 0.930 6.5	0.970 0.970 0.0	1.260 1.260 0.0	0.950 0.980 -3.1	0.980 1.020 -3.9	
4	0.970 0.950 2.1	1.210 1.200 0.8	0.970 0.950 2.1	1.270 1.260 0.8	1.060 1.070 -0.9	1.230 1.230 0.0	0.680 0.720 -5.6	
5	1.200 1.190 0.8	1.000 0.990 1.0	1.260 1.260 0.0	1.060 1.060 0.0	1.160 1.190 -2.5	0.800 0.820 -2.4		
6	0.840 0.810 3.7	1.050 1.040 1.0	0.950 0.960 -1.0	1.230 1.220 0.8	0.800 0.830 -3.6			
7	0.840 0.820 2.4	1.190 1.200 -0.8	0.980 1.010 -3.0	0.680 0.700 -2.9				
8	0.750 0.800 -6.3	0.620 0.650 -4.6						

ONE-ZONE MODEL
MEASUREMENT
PER CENT DIFF

FIGURE 4-2

ASSEMBLYWISE 1-ZONE VS INCORE REL. POWER DIST. FOR S2C2 HZP D-IN AT 0 MWD/T
ONE-ZONE VS MEASUREMENT

	1	2	3	4	5	6	7	8
1	0.710 0.690 2.9	1.000 0.960 4.2	1.040 0.980 6.1	0.970 0.950 2.1	1.220 1.250 -2.4	0.740 0.780 -5.1	0.420 0.440 -4.5	0.490 0.550 -10.9
2	1.000 0.970 3.1	1.070 1.060 0.9	0.790 0.780 1.3	1.180 1.170 0.9	1.050 1.050 0.0	1.020 1.060 -3.8	0.990 1.060 -6.6	0.480 0.530 -9.4
3	1.040 1.000 4.0	0.790 0.790 0.0	0.560 0.530 5.7	0.910 0.920 -1.1	1.390 1.370 1.5	1.050 1.050 0.0	1.030 1.070 -3.7	
4	0.970 0.950 2.1	1.180 1.180 0.0	0.910 0.920 -1.1	1.360 1.350 0.7	1.240 1.220 1.6	1.470 1.430 2.8	0.800 0.840 -4.8	
5	1.220 1.230 -0.8	1.050 1.040 1.0	1.390 1.390 0.0	1.240 1.220 1.6	1.430 1.410 1.4	0.990 1.000 -1.0		
6	0.740 0.760 -2.6	1.020 1.040 -1.9	1.050 1.050 0.0	1.470 1.430 2.8	0.990 1.000 -1.0			
7	0.420 0.430 -2.3	0.990 1.050 -5.7	1.030 1.050 -1.9	0.800 0.790 1.3				
8	0.490 0.600 -18.3	0.480 0.520 -7.7						

ONE-ZONE MODEL
MEASUREMENT
PER CENT DIFF

FIGURE 4-3

ASSEMBLYWISE 1-ZONE VS INCORE REL. POWER DIST. FOR S2C2 HFP ARO AT 3000 MWD/T
ONE-ZONE VS MEASUREMENT

	1	2	3	4	5	6	7	8
1	0.800 0.790 1.3	1.070 1.050 1.9	1.110 1.090 1.8	0.990 1.000 -1.0	1.210 1.220 -0.8	0.870 0.880 -1.1	0.860 0.860 0.0	0.770 0.760 1.3
2	1.070 1.050 1.9	1.210 1.190 1.7	0.960 0.950 1.1	1.230 1.220 0.8	1.010 1.010 0.0	1.040 1.030 1.0	1.160 1.150 0.9	0.640 0.650 -1.5
3	1.110 1.080 2.8	0.960 0.950 1.1	1.030 1.010 2.0	0.980 0.980 0.0	1.240 1.240 0.0	0.940 0.950 -1.1	0.960 0.960 0.0	
4	0.990 0.990 0.0	1.230 1.220 0.8	0.980 0.980 0.0	1.250 1.250 0.0	1.020 1.030 -1.0	1.160 1.160 0.0	0.670 0.690 -2.9	
5	1.210 1.200 0.8	1.010 1.010 0.0	1.240 1.250 -0.8	1.020 1.020 0.0	1.120 1.140 -1.8	0.770 0.800 -3.7		
6	0.870 0.860 1.2	1.040 1.040 0.0	0.940 0.960 -2.1	1.160 1.150 0.9	0.770 0.780 -1.3			
7	0.860 0.850 1.2	1.160 1.150 0.9	0.960 0.960 0.0	0.670 0.680 -1.5				
8	0.770 0.760 1.3	0.640 0.630 1.6						

ONE-ZONE MODEL
MEASUREMENT
PER CENT DIFF

FIGURE 4-4

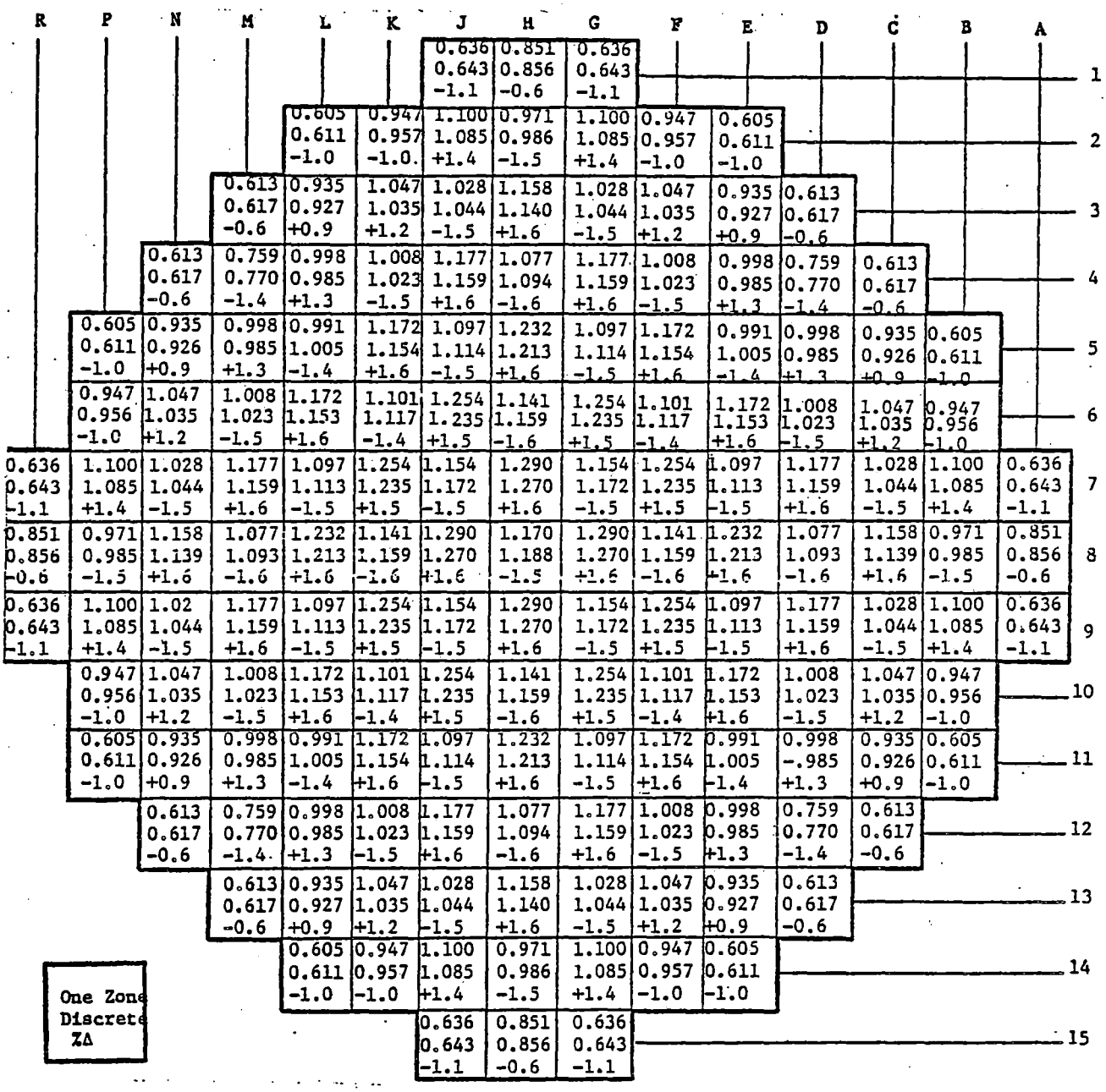
ASSEMBLYWISE 1-ZONE VS INCORE REL. POWER DIST. FOR S2C2 HFP ARO AT 9000 MWD/T
ONE-ZONE VS MEASUREMENT

	1	2	3	4	5	6	7	8
1	0.880 0.840 4.8	1.100 1.060 3.8	1.140 1.090 4.6	1.010 0.990 2.0	1.220 1.210 0.8	0.910 0.920 -1.1	0.880 0.910 -3.3	0.600 0.810 -1.2
2	1.100 1.060 3.8	1.230 1.180 4.2	0.990 0.970 2.1	1.230 1.200 2.5	1.020 1.020 0.0	1.040 1.050 -1.0	1.130 1.150 -1.7	0.660 0.680 -2.9
3	1.140 1.090 4.6	0.990 0.980 1.0	1.060 1.030 2.9	0.990 0.990 0.0	1.220 1.200 1.7	0.940 0.960 -2.1	0.940 0.970 -3.1	
4	1.010 0.990 2.0	1.230 1.190 3.4	0.990 0.990 0.0	1.220 1.200 1.7	1.000 1.010 -1.0	1.110 1.130 -1.8	0.670 0.700 -4.3	
5	1.220 1.170 4.3	1.020 1.010 1.0	1.220 1.220 0.0	1.000 1.020 -2.0	1.080 1.100 -1.8	0.760 0.790 -3.8		
6	0.910 0.900 1.1	1.040 1.050 -1.0	0.940 0.960 -2.1	1.110 1.130 -1.8	0.760 0.780 -2.6			
7	0.880 0.890 -1.1	1.130 1.150 -1.7	0.940 0.950 -1.1	0.670 0.700 -4.3				
8	0.800 0.810 -1.2	0.660 0.690 -4.3						

ONE-ZONE MODEL
MEASUREMENT
PER CENT DIFF

Figure 4-5

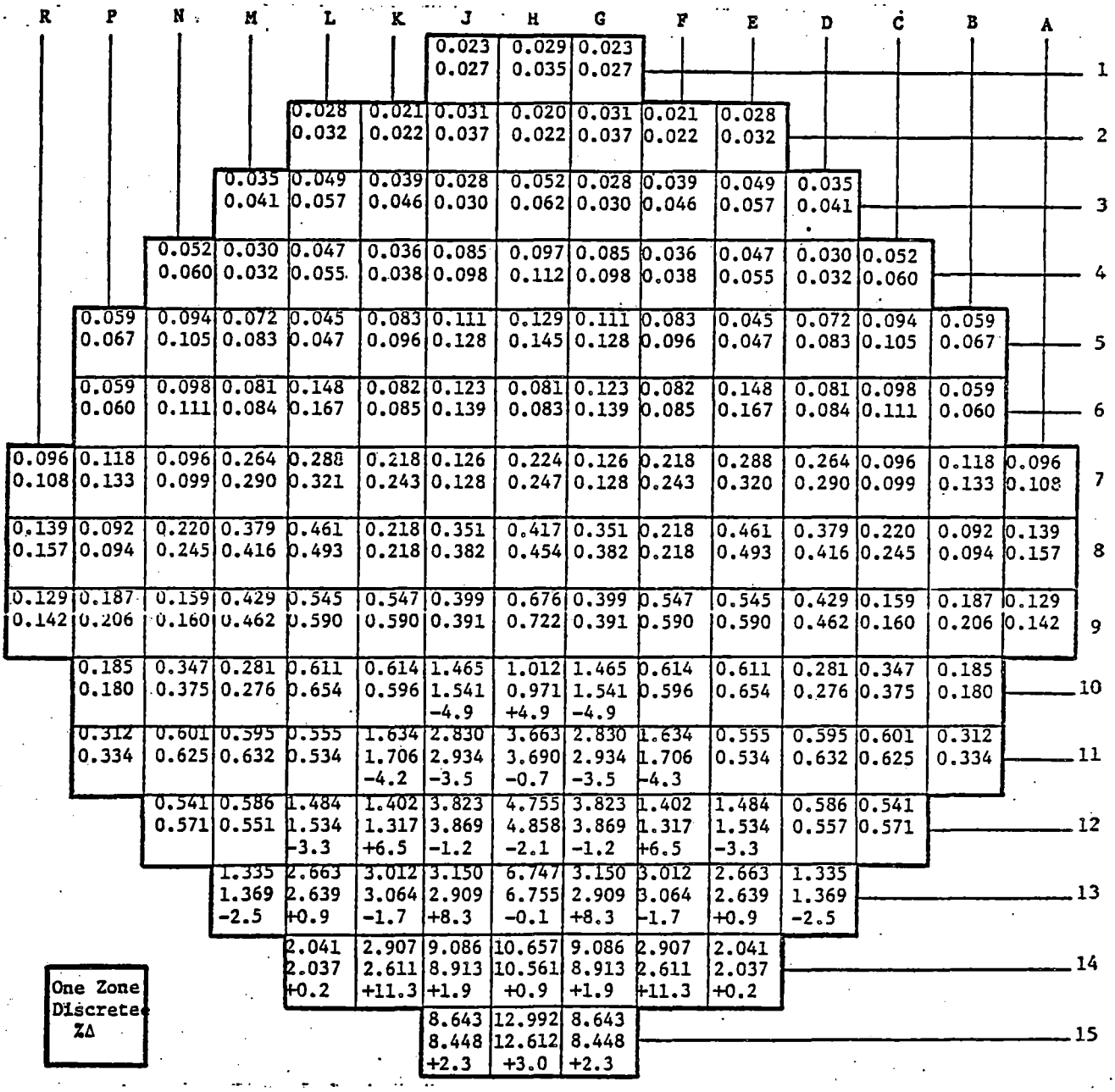
ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR HOT ZERO POWER, ALL RODS OUT AT BEGINNING
OF INITIAL CYCLE FOR SURRY UNIT 1.



AVERAGE PERCENT DIFFERENCE = 1.3

Figure 4-6

ASSEMBLYWISE AVERAGE POWER DISTRIBURION FOR
HOT ZERO POWER, ALL RODS IN WITH H-14 OUT AT BEGINNING
OF INITIAL CYCLE FOR SURRY UNIT 1

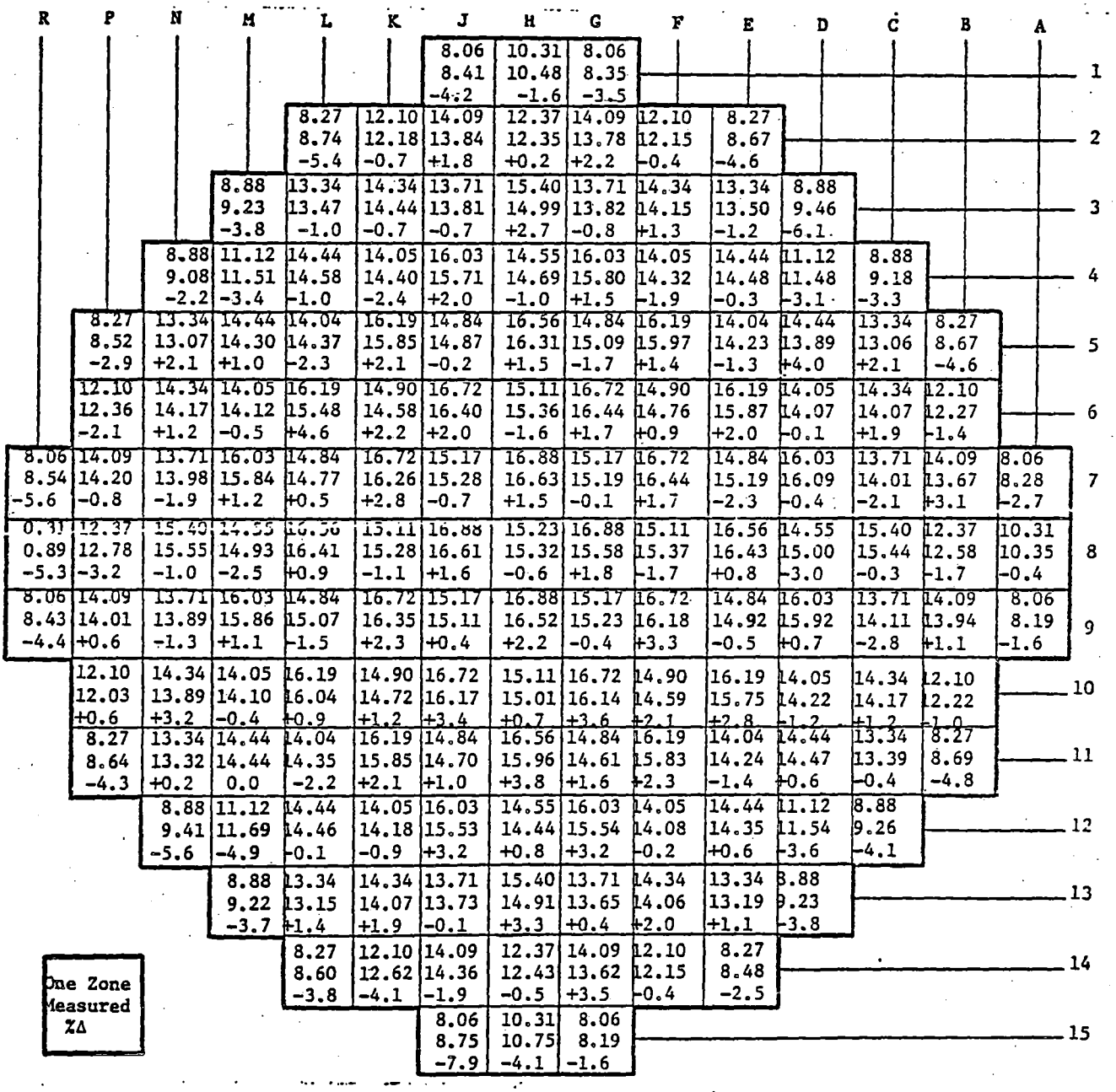


	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
1							0.023 0.027	0.029 0.035	0.023 0.027							1
2					0.028 0.032	0.021 0.022	0.031 0.037	0.020 0.022	0.031 0.037	0.021 0.022	0.028 0.032					2
3				0.035 0.041	0.049 0.057	0.039 0.046	0.028 0.030	0.052 0.062	0.028 0.030	0.039 0.046	0.049 0.057	0.035 0.041				3
4			0.052 0.060	0.030 0.032	0.047 0.055	0.036 0.038	0.085 0.098	0.097 0.112	0.085 0.098	0.036 0.038	0.047 0.055	0.030 0.032	0.052 0.060			4
5		0.059 0.067	0.094 0.105	0.072 0.083	0.045 0.047	0.083 0.096	0.111 0.128	0.129 0.145	0.111 0.128	0.083 0.096	0.045 0.047	0.072 0.083	0.094 0.105	0.059 0.067		5
6		0.059 0.060	0.098 0.111	0.081 0.084	0.148 0.167	0.082 0.085	0.123 0.139	0.081 0.083	0.123 0.139	0.082 0.085	0.148 0.167	0.081 0.084	0.098 0.111	0.059 0.060		6
7	0.096 0.108	0.118 0.133	0.096 0.099	0.264 0.290	0.288 0.321	0.218 0.243	0.126 0.128	0.224 0.247	0.126 0.128	0.218 0.243	0.288 0.320	0.264 0.290	0.096 0.099	0.118 0.133	0.096 0.108	7
8	0.139 0.157	0.092 0.094	0.220 0.245	0.379 0.416	0.461 0.493	0.218 0.218	0.351 0.382	0.417 0.454	0.351 0.382	0.218 0.218	0.461 0.493	0.379 0.416	0.220 0.245	0.092 0.094	0.139 0.157	8
9	0.129 0.142	0.187 0.206	0.159 0.160	0.429 0.462	0.545 0.590	0.547 0.590	0.399 0.391	0.676 0.722	0.399 0.391	0.547 0.590	0.545 0.590	0.429 0.462	0.159 0.160	0.187 0.206	0.129 0.142	9
10		0.185 0.180	0.347 0.375	0.281 0.276	0.611 0.654	0.614 0.596	1.465 1.541 -4.9	1.012 0.971 +4.9	1.465 1.541 -4.9	0.614 0.596	0.611 0.654	0.281 0.276	0.347 0.375	0.185 0.180		10
11		0.312 0.334	0.601 0.625	0.595 0.632	0.555 0.534	1.634 1.706 -4.2	2.830 2.934 -3.5	3.663 3.690 -0.7	2.830 2.934 -3.5	1.634 1.706 -4.3	0.555 0.534	0.595 0.632	0.601 0.625	0.312 0.334		11
12			0.541 0.571	0.586 0.551	1.484 1.534 -3.3	1.402 1.317 +6.5	3.823 3.869 -1.2	4.755 4.858 -2.1	3.823 3.869 -1.2	1.402 1.317 +6.5	1.484 1.534 -3.3	0.586 0.557	0.541 0.571			12
13				1.335 1.369 -2.5	2.663 2.639 +0.9	3.012 3.064 -1.7	3.150 2.909 +8.3	6.747 6.755 -0.1	3.150 2.909 +8.3	3.012 3.064 -1.7	2.663 2.639 +0.9	1.335 1.369 -2.5				13
14					2.041 2.037 +0.2	2.907 2.611 +11.3	9.086 8.913 +1.9	10.657 10.561 +0.9	9.086 8.913 +1.9	2.907 2.611 +11.3	2.041 2.037 +0.2					14
15							8.643 8.448 +2.3	12.992 12.612 +3.0	8.643 8.448 +2.3							15

AVERAGE PERCENT DIFFERENCE = 3.4

Figure 4-7

ASSEMBLYWISE ACCUMULATED BURNUP AND BATCH BURNUP SHARING (10^3 MWD/MTU) FOR THE CYCLE 1 OPERATION OF SURRY UNIT 1

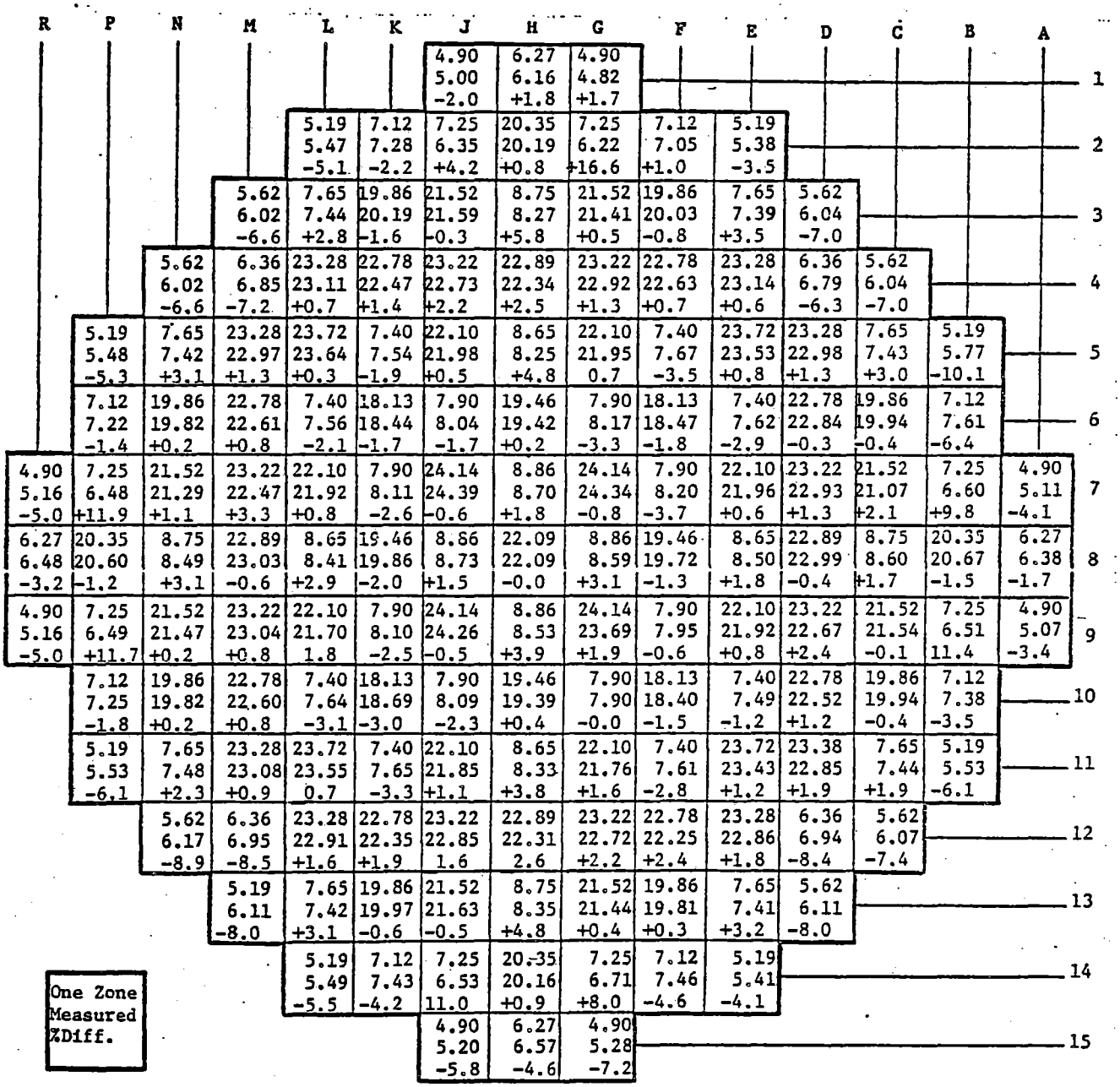


CORE AVERAGE BURNUP = I3,547 MWD/MTU

	Measured	One Zone Model	One Zone Model Percent Difference	Discrete Model	Discrete Model Percent Difference
Batch 1	14.25	14.06	-1.3	14.20	+0.4
Batch 2	15.46	15.71	+1.6	15.62	-1.0
Batch 3	10.93	10.75	-1.6	10.81	+1.1

Figure 4-8

ASSEMBLYWISE ACCUMULATED BURNUP AND BATCH BURNUP SHARING (10^3 MWD/MTU) FOR THE CYCLE 2 OPERATION OF SURRY UNIT 1



	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
							4.90 5.00 -2.0	6.27 6.16 +1.8	4.90 4.82 +1.7							1
					5.19 5.47 -5.1	7.12 7.28 -2.2	7.25 6.35 +4.2	20.35 20.19 +0.8	7.25 6.22 +16.6	7.12 7.05 +1.0	5.19 5.38 -3.5					2
				5.62 6.02 -6.6	7.65 7.44 +2.8	19.86 20.19 -1.6	21.52 21.59 -0.3	8.75 8.27 +5.8	21.52 21.41 +0.5	19.86 20.03 -0.8	7.65 7.39 +3.5	5.62 6.04 -7.0				3
			5.62 6.02 -6.6	6.36 6.85 -7.2	23.28 23.11 +0.7	22.78 22.47 +1.4	23.22 22.73 +2.2	22.89 22.34 +2.5	23.22 22.92 +1.3	22.78 22.63 +0.7	23.28 23.14 +0.6	6.36 6.79 -6.3	5.62 6.04 -7.0			4
		5.19 5.48 -5.3	7.65 7.42 +3.1	23.28 22.97 +1.3	23.72 23.64 +0.3	7.40 7.54 -1.9	22.10 21.98 +0.5	8.65 8.25 +4.8	22.10 21.95 0.7	7.40 7.67 -3.5	23.72 23.53 +0.8	23.28 22.98 +1.3	7.65 7.43 +3.0	5.19 5.77 -10.1		5
		7.12 7.22 -1.4	19.86 19.82 +0.2	22.78 22.61 +0.8	7.40 7.56 -2.1	18.13 18.44 -1.7	7.90 8.04 -1.7	19.46 19.42 +0.2	7.90 8.17 -3.3	18.13 18.47 -1.8	7.40 7.62 -2.9	22.78 22.84 -0.3	19.86 19.94 -0.4	7.12 7.61 -6.4		6
4.90 5.16 -5.0	7.25 6.48 +11.9	21.52 21.29 +1.1	23.22 22.47 +3.3	22.10 21.92 +0.8	7.90 8.11 -2.6	24.14 24.39 -0.6	8.86 8.70 +1.8	24.14 24.34 -0.8	7.90 8.20 -3.7	22.10 21.96 +0.6	23.22 22.93 +1.3	21.52 21.07 +2.1	7.25 6.60 +9.8	4.90 5.11 -4.1		7
6.27 6.48 -3.2	20.35 20.60 -1.2	8.75 8.49 +3.1	22.89 23.03 -0.6	8.65 8.41 +2.9	19.46 19.86 -2.0	8.86 8.73 +1.5	22.09 22.09 -0.0	8.86 8.59 +3.1	19.46 19.72 -1.3	8.65 8.50 +1.8	22.89 22.99 -0.4	8.75 8.60 +1.7	20.35 20.67 -1.5	6.27 6.38 -1.7		8
4.90 5.16 -5.0	7.25 6.49 +11.7	21.52 21.47 +0.2	23.22 23.04 +0.8	22.10 21.70 1.8	7.90 8.10 -2.5	24.14 24.26 -0.5	8.86 8.53 +3.9	24.14 23.69 +1.9	7.90 7.95 -0.6	22.10 21.92 +0.8	23.22 22.67 +2.4	21.52 21.54 -0.1	7.25 6.51 11.4	4.90 5.07 -3.4		9
	7.12 7.25 -1.8	19.86 19.82 +0.2	22.78 22.60 +0.8	7.40 7.64 -3.1	18.13 18.69 -3.0	7.90 8.09 -2.3	19.46 19.39 +0.4	7.90 7.90 -0.0	18.13 18.40 -1.5	7.40 7.49 -1.2	22.78 22.52 +1.2	19.86 19.94 -0.4	7.12 7.38 -3.5			10
	5.19 5.53 -6.1	7.65 7.48 +2.3	23.28 23.08 +0.9	23.72 23.55 0.7	7.40 7.65 -3.3	22.10 21.85 +1.1	8.65 8.33 +3.8	22.10 21.76 +1.6	7.40 7.61 -2.8	23.72 23.43 +1.2	23.38 22.85 +1.9	7.65 7.44 +1.9	5.19 5.53 -6.1			11
		5.62 6.17 -8.9	6.36 6.95 -8.5	23.28 22.91 +1.6	22.78 22.35 +1.9	23.22 22.85 1.6	22.89 22.31 2.6	23.22 22.72 +2.2	22.78 22.25 +2.4	23.28 22.86 +1.8	6.36 6.94 -8.4	5.62 6.07 -7.4				12
			5.19 6.11 -8.0	7.65 7.42 +3.1	19.86 19.97 -0.6	21.52 21.63 -0.5	8.75 8.35 +4.8	21.52 21.44 +0.4	19.86 19.81 +0.3	7.65 7.41 +3.2	5.62 6.11 -8.0					13
				5.19 5.49 -5.5	7.12 7.43 -4.2	7.25 6.53 11.0	20.35 20.16 +0.9	7.25 6.71 +8.0	7.12 7.46 -4.6	5.19 5.41 -4.1						14
						4.90 5.20 -5.8	6.27 6.57 -4.6	4.90 5.28 -7.2								15

CORE AVERAGE BURNUP = 6915 MWD/MTU

	Measured	One Zone Model	One Zone Model Percent Difference	Discrete Model	Discrete Model Percent Difference
Batch 1A	19.63	19.64	+0.1	19.82	+1.0
Batch 2	22.60	22.83	+1.0	22.80	+0.9
Batch 4A	7.60	7.39	-2.8	7.49	-1.4
Batch 4B	8.40	8.75	+4.2	8.19	-2.5
Batch 4C	6.25	6.29	+0.6	6.27	+0.3

TABLE 4-3

COMPARISON OF PREDICTED AND MEASURED D AND C BANK
CONTROL ROD WORTHS FOR BOC, HZP CONDITIONS

Surry Unit	Cycle No.	Control Bank	Rod Worths, pcm* One Zone Model	Measured	Percent Difference**	Discrete Model	Percent Difference**
1	1	D	1407	1480	-4.9	1379	-6.8
		C (with D in)	1278	1300	-1.7	1234	-5.1
1	2	D	1157	1051	10.1	1079	2.7
		C (with D in)	1216	1331	-8.6	1202	-9.7
1	3	D	1255	1129	11.2	1176	4.2
		C (with D in)	1159	1055	9.9	1068	1.2
2	1	D	1407	1435	-2.0	1379	-3.9
		C (with D in)	1278	1309	-2.4	1234	-5.7
2	2	D	935	880	6.3	931	5.8
		C (with D in)	1309	1244	5.2	1249	0.4
2	3	D	1082	1098	-1.5	1067	-2.8
		C (with D in)	1242	1213	2.4	1196	-1.4

*pcm = $\Delta\rho \times 10^{-5}$

**Percent Difference = $\frac{\text{Vepco Model - Measured}}{\text{Measured}} \times 100$

TABLE 4-4

COMPARISON OF PREDICTED AND MEASURED TOTAL SHUTDOWN WORTH AND STUCK ROD WORTH FOR CYCLE 1, SURRY UNITS 1 AND 2, BOC, HZP CONDITIONS

Surry Unit	Control Bank	Rod Worths, pcm* One Zone Model	Measured	Percent Difference**	Discrete Model	Percent Difference**
1	D	1407	1480	-4.9	1379	-6.8
	C (with D in)	1278	1300	-1.7	1234	-5.1
	B (with D + C in)	1943	1920	+1.2	***	-
	A (with D + C + B in)	1481	1440	+2.9	***	-
	ARI	10223	10460	-2.2	10051	-3.9
	Rod H-14 out (with ARI)	2336	2300	1.6	2215	-3.7
2	D	1407	1435	-2.0	1379	-3.9
	C (with D in)	1278	1309	-2.4	1234	-5.7
	B (with D + C in)	1943	1929	+0.7	***	-
	A (with D + C + B in)	1481	1508	-1.8	***	-
	ARI	10223	10440	-2.0	10051	-3.7
	Rod H-14 out (with ARI)	2336	2425	-3.7	2215	-8.7

*pcm $\equiv \Delta\rho \times 10^{-5}$

**Percent Difference $= \frac{\text{Vepco Model} - \text{Measured}}{\text{Measured}} \times 100$

***This data has not been calculated.

TABLE 4-5

REPRESENTATIVE CRITICAL BORON CONCENTRATION VS. BURNUP COMPARISONS FOR SURRY UNIT 1

Cycle	Burnup (MWD/MTU)	One Zone Model (PPM)	Discrete Model (PPM)	Measured (PPM)	Burnup at Measured (MWD/MTU)
1	BOC	1100	1094	*	-
	150	817	810	738	270
	7000	537	512	531	7000
	13000	154	131	102	13000
2	BOC	898	895	*	-
	150	626	626	636	145
	3000	421	421	425	3010
	6915	134	102	132	6810
3	BOC	1221	1202	*	-
	150	948	932	925	265
	3000	687	672	684	3065
	7000	327	316	318	7070

*No measurements are available at BOC non-equilibrium xenon, HFP

TABLE 4-6

REPRESENTATIVE CRITICAL BORON CONCENTRATION VS BURNUP
COMPARISONS FOR SURRY UNIT 2

Cycle	Burnup (MWD/MTU)	One Zone Model (PPM)	Discrete Model (PPM)	Measured (PPM)	Burnup At Measured (MWD/MTU)
1	BOC	1100	1094	*	-
	150	817	810	-	-
	7000	537	512	543	6995
	13000	154	131	131	13015
2	BOC	1309	1303	*	-
	150	1021	1017	970	160
	5000	588	587	593	5000
	9000	226	230	246	8985
3	BOC	1152	1113	*	-
	150	875	836	898	170
	5000	423	397	**	-
	8000	161	140	**	-

*No measurements are available at BOC non-equilibrium xenon, HFP
**S2C3 has not reached these burnup levels at the time of this writing.

TABLE 4-7

COMPARISON OF PREDICTED AND MEASURED CRITICAL BORON CONCENTRATION FOR VARIOUS CONTROL ROD CONFIGURATIONS FOR SURRY UNIT 1, CYCLES 1, 2, AND 3

Cycle	Control Rod Bank Position	Measured Critical Boron Concentration (PPM)	Discrete Model Critical Boron Concentration (PPM)	Discrete Model Percent Difference	One-Zone Model Critical Boron Concentration (PPM)	One-Zone Model Percent Difference
1	ARO	1196	1168	-2.3	1207	+0.9
1	D-Bank In	1077	1050	-2.5	1080	+0.3
1	C and D-Banks In	957	942	-1.6	967	+1.0
2	ARO	1033	997	-3.5	1032	-0.1
2	D-Bank In	917	899	-2.0	921	+0.4
2	C and D-Banks In	800	787	-1.6	799	-0.1
3	ARO	1311	1322	-0.1	1355	+3.4
3	D-Bank In	1196	1207	-0.1	1229	+2.8
3	C and D-Banks In	1095	1101	-0.1	1113	+1.6

TABLE 4-8

COMPARISON OF PREDICTED AND MEASURED DIFFERENTIAL BORON WORTH
FOR VARIOUS CONTROL ROD CONFIGURATIONS FOR SURRY UNIT 1, CYCLES 1, 2, AND 3

Cycle	Control Rod Bank Position	One Zone Model Boron Worth (PCM/PPM)	Measured Boron Worth (PCM/PPM)	Discrete Model Boron Worth (PCM/PPM)
1	ARO	11.3	12.1	11.9
1	D-Bank In	11.2	-	-
1	C and D-Banks In	11.1	-	-
2	ARO	10.5	10.2	10.8
2	D-Bank In	10.4	-	-
2	C and D-Banks In	10.2	-	-
3	ARO	9.9	11.0	10.2
3	D-Bank In	10.0	-	-
3	C and D-Banks In	10.0	-	-

TABLE 4-9

COMPARISON OF PREDICTED AND MEASURED ISOTHERMAL TEMPERATURE COEFFICIENTS FOR VARIOUS ROD CONFIGURATIONS FOR SURRY UNIT 1, CYCLES 1, 2, AND 3

Cycle	Control Rod Bank Position	Measured Isothermal Temperature Coefficient (PCM/°F)	One-Zone Model Isothermal Temperature Coefficient (PCM/°F)	Difference (PCM/°F)
1	ARO	-0.3	+0.2	-0.5
1	D-Bank In	-3.5	-2.60	-0.9
1	C and D-Banks In	-7.9	-6.88	-1.0
2	ARO	-2.88	-2.8	-0.1
2	D-Bank In	-5.27	-7.43	+2.2
2	C and D-Banks In	-9.61	-11.38	+1.8
3	ARO	+1.68	+0.6	+1.1
3	D-Bank In	-0.93	-2.05	+1.1
3	C and D-Banks In	-4.67	-5.30	+0.6

SECTION 5 - SUMMARY AND CONCLUSION

The Vepco PDQ07 one zone model has been developed from the Vepco PDQ07 discrete model for the purpose of performing two-dimensional (x-y) reactor physics calculations. The coarse (i.e., as opposed to discrete) mesh representation allows calculations to be performed both faster and with smaller computer memory requirements, since fewer mesh lines are required to represent the geometry of the reactor core. In general, it is intended to use the more efficient one zone model to perform fuel management and reactor physics scoping calculations, and then perform the final production calculations to support reactor operations with the discrete model. However, for the calculational evaluation of abnormal control rod positioning, assembly average radial power distributions, assemblywise and batch burnup distributions, and reactivity coefficients, it is intended to use the one zone model for final production calculations to support reactor operations.

The accuracy of the one zone model has been demonstrated for each of the above production calculations through comparisons with both analytical PDQ07 discrete model calculations and with measurements taken at Surry Units No. 1 and 2. The results of these comparisons are:

1. Assembly average power distributions are generally predicted within a 0.02 to 0.03 standard deviation range relative to measurement, with a maximum standard deviation of 0.058 for a low power measurement. For the stuck rod calculation at HZP, the one zone model predictes the peak assembly relative power density to within 3% of the discrete model predictions. Power distribution calculations to the above accuracies are acceptable for preduction use.

2. Batch burnup values are typically predicted within less than 2% of the measurement value with a maximum difference of 4.2%. Batch burnup distributions to the above accuracies are acceptable for production use.

3. The stuck rod worth values measured during initial startup of Cycle 1 of both Surry Unit 1 and 2 was adequately predicted by the one zone model (2336 pcm predicted vs. 2300 and 2425 pcm for Units 1 and 2, respectively). Control rods bank worths are typically within 5% of the measured values with the

maximum deviation being approximately 11% of the measured values. Total shutdown worth was predicted within approximately 2% of the measured value. The accuracy of the one zone model for use in normally positioned control rod worth production calculations are adequate (i.e., predicted average control rod worths are within 10% of the measured values) even though the primary application is for scoping calculations.

4. Critical boron concentrations as a function of burnup were typically predicted within 25 ppm with a maximum deviation of 51 ppm. The predicted critical boron concentration for various rodded configurations was generally within 1% of the measured values with a maximum deviation of approximately 3%. Differential boron worths were predicted within 1.1 pcm/ppm (approximately 10%) of measurement. The accuracies of the above critical boron concentrations and the differential boron worths are adequate (i.e., within 4% of measurement and conservative to the values assumed in appropriate accident analyses respectively) for production use even though the primary application is for scoping calculations.

5. Isothermal temperature coefficients are predicted within an average of 1.0 pcm and with a maximum difference of 2.2 pcm relative to the measured values. Temperature coefficients to the above accuracy are acceptable (i.e., within $\pm$3 pcm of the measured values) for production use.

Verification and model improvements will continue to be made as more experience is gained through continued application of the Vepco PDQ07 one zone model to the Surry and North Anna reactors.

SECTION 6 - REFERENCES

1. W. A. Wittkopf, et. al., NULIF - "Neutron Spectrum Generator, Few Group Constant Calculator, and Fuel Depletion Code", BAW-10115, June 1976.

2. H. H. Hassan, et. al., "Babcock and Wilcox Version of PDQ07 - User's Manual", BAW-10117P, December 1975.

3. H. H. Hassan, et. al., "SHUFFLE - Program to Perform Fuel Shuffle in Nuclear Reactor Core", BAW-422, Rev. 1, July 1975.

4. H. H. Hassan, W. A. Wittkopf, et. al., "HAFIT", BAW-425, July 1973.

5. Private contractual correspondence from the Babcock and Wilcox Company to the Virginia Electric and Power Company dated February 3, 1971, and October 6, 1971.

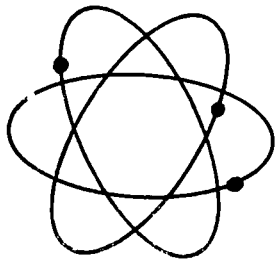
6. M. L. Smith, "The PDQ07 Discrete Model," Virginia Electric and Power Company, VEP-FRD-19, July 1976.

7. Final Safety Analysis Report - Surry Power Station Units 1 and 2, Virginia Electric and Power Company.

VEP-FRD-24A
JULY, 1981

Vepco

THE VEPCO FLAME MODEL



FUEL RESOURCES DEPARTMENT
VIRGINIA ELECTRIC AND POWER COMPANY

VEP-FRD-24 A

THE VEPCO FLAME MODEL

BY

W. C. BECK

NUCLEAR FUEL ENGINEERING GROUP
FUEL RESOURCES DEPARTMENT

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA

JULY, 1981

Recommended for Approval:

Marvin L Smith

M. L. Smith
Nuclear Fuel Engineer

Approved:

Martin L. Bowling

M. L. Bowling, Director
Nuclear Fuel Engineering

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 13 1981

Mr. W. N. Thomas, Vice President
Fuel Resources
Virginia Electric Power Company
Richmond, Virginia 23261

Dear Mr. Thomas:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT VEP-FRD-24 "THE VEPCO FLAME MODEL"

The Nuclear Regulatory Commission (NRC) staff has completed its review of the Virginia Electric and Power Company Topical Report number VEP-FRD-24 entitled "The VEPCO FLAME Model". This VEPCO developed model is a three dimensional, one energy group, modified diffusion theory calculational model. The model is used to perform three dimensional reactor physics analyses in support of reactor startup and cycle operations of the Surry and North Anna nuclear reactors. A summary of our evaluation of the licensing topical report is attached.

As a result of our review, we find that Licensing Topical Report VEP-FRD-24 entitled "The VEPCO FLAME Model" dated October 1978 as augmented by responses to NRC questions submitted in your November 1978 letter is acceptable for referencing in licensing applications to the extent specified and under the limitations in the report and the attached evaluation.

We do not intend to repeat the review of the safety features described in the report and its amendment as found acceptable herein. Our acceptance applies only to the use of features described in the topical report and its amendment as discussed in the attached safety evaluation.

In accordance with established requirements, it is requested that Virginia Electric and Power Company issue a revised version of this report within three months of the receipt of this letter. The revised version is to appropriately incorporate the information submitted in your October, 1978 letter. This evaluation letter and the attached safety evaluation is to be included in the revised version between the title page and the abstract and the approved report will carry the identifier VEP-FRD-24A.

Should Nuclear Regulatory Commission criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Virginia Electric and Power Company will be expected to revise and resubmit the topical report or submit justification for the continued effective applicability of the topical report without revision.

If you have any questions about the review or our conclusion, please contact us.

Sincerely,

R J Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

MAY 7 1981

REVIEW OF TOPICAL REPORT VEP-FRD-24, "VEPCO FLAME MODEL"

Report Number:	VEP-FRD-24
Report Title:	The VEPCO FLAME Model
Report Date:	October, 1979
Originating Organization:	Virginia Electric and Power Company
Reviewed By:	Walter L. Brooks /Core Performance Branch

Introduction

Virginia Electric and Power Company has submitted licensing topical report VEP-FRD-24 entitled "The VEPCO FLAME Model." FLAME is three-dimensional nodal analysis code which is capable of calculating:

1. Assembly and core average axial power distributions
2. Differential control rod bank worths
3. Integral control rod bank worths vs. bank positon
4. Control rod bank insertion limits
5. Axial burnup distribution
6. Axial offset
7. Peaking factors (F_Q^T, $F_{xy}(z)$, F_z)

However, the code is currently used to calculate only differential and integral control rod worths and core average axial peaking factors on a production basis. Our review of this report follows.

Summary of Report

This report presents a summary description of the FLAME code and several service codes which provide input to or handle output from the code. In addition the report includes descriptions of the cores of the Surry Nuclear Power Station, Units 1 and 2 and fuel loadings for the first four cycles of each reactor. Normalization of the FLAME results to higher order (e.g., PDQ07 discrete model) calculations is described. Axial and radial albedoes are chosen to provide

agreement at boundaries. The migration area values are adjusted for rodded nodes to provide agreement at interior nodes. Total rod bank worths calculated by the FLAME code are normalized to those calculated by PDQ07 in order to determine the differential and integral bank worths as a function of insertion.

Extensive comparisons are presented of radial power distributions, axial power distributions and differential and integral rod worths for both Surry reactors for the first four cycles. The radial power distribution comparisons are between FLAME and PDQ07 calculations. The other comparisons are between FLAME calculations and experiments. The results of these comparisons have been summarized to obtain the following uncertainties for the parameters calculated by the code on a preduction basis:

Core average axial peaking factor	-	8%
Differential Bank worth	-	2.0 PCM/step
Integral Bank Worth		
Individual	-	15%
Cumulative	-	10%

These uncertainties are the result of comparisons with experiment and represent the calculation-experiment differences. In effect the measurement uncertainty has been assumed to be nil. This is a conservative evaluation of calculational uncertainty.

Summary of Evaluation

We have reviewed the summary descriptions of the FLAME code and its associated service codes. The FLAME code and the input preparation codes, NULIF, PDQ07, and FLAFIT were purchased by VEPCO from Babcock and Wilcox, who have submitted

topical reports describing methods used in the design of Babcock and Wilcox reactors. These methods should be equally applicable to the Westinghouse supplied reactors for which VEPCO proposes to use them provided they are suitably modelled. We have reviewed the data in VEP-FRD-24 which is presented to support the conclusion that this code is suitable for VEPCO reactors. These data show that the VEPCO FLAME model (which includes normalization to the PDQ07 discrete model) provides the capability to predict axial peaking factors and axial power distributions and differential and integral rod worths. This conclusion is based on the review of the comparisons with measured data presented in the report.

VEPCO has analyzed the data to obtain calculational uncertainties for axial peaking factors and control rod worths. The analysis was performed in a conservative manner and the results are consistent with industry wide values. We conclude that they are acceptable.

Evaluation Procedure

The review of topical report VEP-FRD-24 has been conducted within the guidelines provided by the Standard Review Plan, Section 4.3. Enough information is provided, directly and by reference, to permit a knowledgeable person to conclude that the methods described are state-of-the-art.

Regulatory Position

Based on our review we conclude that the VEPCO FLAME code is a suitable method for calculating axial power distributions and differential and integral rod bank worths when the code is suitably normalized to PDQ07 calculations. In addition certain other parameters which depend on these calculations, e.g., axial

burnup distributions, rod insertion limits, axial offsets, etc., may also be obtained from the calculated results. We further conclude that the topical report VEP-FRD-24 may be used as a reference to the description of the code and its suitability for use in performing these calculations. In addition, the report may be used as a reference to support the use of the uncertainty values given therein for axial peaking factors and rod worths.

CLASSIFICATION/DISCLAIMER

The data and analytical techniques described in this report have been prepared specifically for application by the Virginia Electric and Power Company. The Virginia Electric and Power Company makes no claim as to the accuracy of the data or techniques contained in this report if used by other organizations. Any use of this report or any part thereof must have the prior written approval of the Virginia Electric and Power Company.

ABSTRACT

The Virginia Electric and Power Company (VEPCO) has developed a three-dimensional (x, y, z), one energy group, modified diffusion theory calculational model designated as the Vepco FLAME Model.

The model utilizes the NULIF, PDQ07, FLAFIT, and FLAME3 computer Codes which are part of the Fuel Utilization and Performance Analysis Code (FUPAC) system obtained from the Babcock and Wilcox Company. In addition, the EDITQAR, PICCOLO, and FLMSHUFL codes have been written by Vepco for use in the FLAME Model. The model is used to perform three-dimensional reactor physics analysis in support of reactor startup and cycle operation of the Vepco Surry and North Anna nuclear reactors. The accuracy of the FLAME model is demonstrated through comparisons with measurements taken at Surry Units No. 1 and 2.

ACKNOWLEDGEMENTS

The author would like to thank Messrs. C. B. Franklin, J. R. Rodes and M. L. Smith for their technical assistance during the development of the FLAME model and to Ms. Cathy Langston, Ms. Ivy Wilkerson, and Ms. Miranda Cooper for their typing of the draft and final manuscripts. The author would also like to express his appreciation to a number of people who reviewed and provided comments on this report.

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SECTION 1 - INTRODUCTION

The purposes of this report are to describe one of the computational models developed at Virginia Electric and Power Company (Vepco) and to demonstrate the accuracy of this model by comparing analytical results generated by the model to applicable measurements from Surry Units No. 1 and 2. The capabilities obtained with this model will be directly applicable to Surry Units 1 and 2 and generally applicable to all of the units at the North Anna Nuclear Power Station.

The model described herein is a three-dimensional (x, y, z), one energy group, modified diffusion theory (with thermal feedback) calculational package and is designated as the Vepco FLAME Model. The Vepco FLAME Model uses the NULIF[1], PDQ07[2], FLAME3[3], and FLAFIT[3] computer codes which are part of the Fuel Utilization and Performance Analysis Code[4] (FUPAC) system obtained from the Babcock and Wilcox Company. In addition, the EDITQAR[5], PICCOLO[5], and FLMSHUFL[6] computer codes have been written by Vepco for use in the Vepco FLAME Model. A detailed description of the input requirements, functioning, physical models and output capabilities of the above codes can be obtained from the referenced code manuals.

The types of reactor physics calculations which can be performed within the general capabilities of the Vepco FLAME Model include:

1. Assembly and core average axial power distribution
2. Differential control rod bank worths
3. Integral control rod bank worths as a function of rod bank position
4. Control rod bank insertion limits
5. Axial burnup distribution
6. Axial offset
7. Peaking factors (F_Q^T, $F_{xy}(Z)$, F_Z)

This report is concerned primarily with the documentation of the capability of the Vepco FLAME Model to accurately compute axial power distributions, axial offset, and differential and integral control rod bank worths.

The remainder of this report describes the Surry Units No. 1 and 2 reactor cores to be modeled, the purposes and interrelationships of the various computer codes which comprise the Vepco FLAME Model, the specific modeling of a reactor core with these codes, and comparison of calculated results with appropriate results obtained with the Vepco PDQ07 Discrete Model[7] and with core measurements obtained from Surry Units No. 1 and 2.

SECTION 2 - CORE DESCRIPTION

2.1 Introduction

The Surry Nuclear Power Station, which consists of two operating units, has been selected as the operating system to be modeled for verification of the Vepco FLAME Model. The Surry Units No. 1 and 2 are identical Westinghouse designed three coolant loop pressurized water reactors with thermal ratings of 2441 Mwt. The operating history of the Surry Power Station is summarized in Table 2-1.

2.2 Core Design

The Surry cores consist of 157 fuel assemblies surrounded by a core baffle, barrel, and thermal shield and enclosed in a steel pressure vessel. The pressure inside the vessel is maintained at a nominal 2250 psia. The coolant (and moderator) is pressurized water which enters the bottom of the core at a nominal 532°F and undergoes a nominal average rise in temperature of 65.5°F before exiting the core. The average coolant temperature is 566°F and the average linear power density of the core is 6.2 kw/ft.

Each of the 157 fuel assemblies consists of 204 fuel rods arranged in a 15 by 15 square array. The fuel used in the Surry cores consists of slightly enriched uranium dioxide fuel pellets contained within a Zircaloy-4 clad. A small gap containing pressurized helium exists between the pellets and the inner diameter of the clad. For the positions in the 15 by 15 array not occupied by fuel rods, there are 20 guide tube locations for either solid burnable poison rods or control rods and one centrally located instrumentation tube. (See Figure 2-1). The fuel rods in each fuel assembly are supported by seven Inconel-718 grids located along the length of the assembly. These grids are mechanically attached to the guide tubes, which are, in turn, fastened to the upper and lower nozzles, and thus provide for assembly structural support.

There are 48 full-length Rod Cluster Control Assemblies (referred to as control rods) used to control core reactivity as well as five part-length rods for axial power shaping. (It should be noted that the part-length control rods are physically present but are not currently allowed to be inserted into the core). The absorber material of the control rods is an alloy consisting of 80% silver, 15% indium, and 5% cadmium. The various control rods are arranged in and move in symmetrically located groups, or banks, as depicted in Figure 2-2. Banks D, C, B, and A are denoted as the control banks and are moved in a fixed sequential pattern to control the reactor over the power range of operation. The remaining rods, Banks SA and SB, are denoted as shutdown banks and are used to provide shutdown margin.

In addition to the control rods, a chemical (boric acid) shim is used to control excess core reactivity and to facilitate operational flexibility. Above certain concentrations of chemical shim, burnable poison rods are also used to control excess reactivity. Fresh and/or depleted burnable poison rods can also be used to shape (i.e., improve) the core power distribution. The burnable poison rods contain borosilicate in the form of Pyrex glass clad in a stainless steel tube. Burnable poison rods, which may be used in any fuel assembly not under a control rod bank location, consist of clusters of either 8, 12, 16, or 20 rods which are inserted into the Zircaloy-4 control rod guide tubes.

Specific values of the principal mechanical and thermal-hydraulic parameters for the Surry core are provided in Table 2-2. A complete description of the Surry units is given in Reference 8.

2.3 Fuel Loadings

The initial and reload quarter core fuel loadings (i.e., initial enrichments and density, previous cycle location if appropriate, beginning of cycle burnup predicted by the Vepco PDQ07 Discrete Model and number of

fresh or depleted burnable poison rods present) for both Surry units are provided in Figures 2-3 through 2-16. It should be noted that the fuel loadings for Cycle 1 of both Surry units are identical. The fuel management strategy employed in the initial cycle of operation of each unit was the checkerboard loading of the two lower enriched fuel batches in the center of the core and the highest enriched fuel batch around the periphery of the core. After the first cycle, the fuel management became more complicated as the result of the need to minimize the impact of fuel densification (which was most severe in the lower initial density, lower prepressurization Batches 1, 2, and 3). Generally, a modified out-in strategy was followed wherein higher enrichment fresh fuel was loaded on the core periphery with lower enrichment fresh fuel (and once-burned fuel and twice-burned fuel) checkerboard loaded in the inner region of the core. An exception to this was in the third cycle of Unit No. 1 where no fresh fuel was loaded on the periphery. The only fresh fuel was 16 lower enrichment assemblies loaded in the inner region of the core. Beginning in Cycle 4 of both units, a change was made from the typical 12 month operating cycle to an 18 month operating cycle. However, the basic fuel management strategy (i. e., modified out-in) was not changed for the Cycle 4 loading patterns.

TABLE 2-1

SURRY NUCLEAR POWER STATION OPERATING HISTORY

Surry Unit	Cycle No.	Beginning of Cycle	End of Cycle	Cycle Burnup (MWD/MTU)
1	1	July 1, 1972	October 24, 1974	13547
1	2	January 30, 1975	September 26, 1975	6915
1	3	December 6, 1975	October 17, 1976	8944
1	4	January 17, 1977	April 21, 1978	13107
2	1	March 7, 1973	April 26, 1975	14870
2	2	June 14, 1975	April 22, 1976	9054
2	3	June 1, 1976	September 10, 1977	9422
2	4	October 8, 1977	1st Qtr 1979*	14000*

*Projected

TABLE 2-2

SURRY CORE DESCRIPTION

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total core heat output, Mwt	2441
Heat generated in fuel, %	97.4
System operating pressure, psi	2250
Total coolant flow rate, lb/hr (gpm)	100.7×10^6 (265,500)
Coolant Temperatures, °F (@100% power)	
Nominal inlet	532
Average rise in the core	65.5
Average in the core	566
Nominal outlet of hot channel	642
Average linear power density, Kw/ft	6.2

MECHANICAL DESIGN PARAMETERS

Fuel Assemblies		
Design	Canless 15 x 15	
Number	157	
Rod pitch, inches	0.563	
Overall dimensions, inches	8.426 x 8.426	
Number of grids per assembly (material)	7 (Inconel-718)	
Number of instrumentation tubes	1	
Fuel Rods		
Number	32,028	
Number of rods/assembly	204	
	Batch 1,2,4,5,6	Batch 3
Outside diameter, inches	0.422	0.422
Diametrical gap, inches	0.0075	0.0085
Clad thickness, inches	0.0243	0.0243
Clad material	Zircaloy-4	
Fuel Pellets		
Material	Sintered UO_2	
Density (% of theoretical) and Enrichment (w/o U235)	See Figures 2-4 through 2-12	
	Batch 1,2,4,5,6	Batch 3
Outer diameter	0.3659	0.3649
Control Rod Assemblies		
Neutron absorber	5% Cd-15% In-80% Ag	
Cladding Material	Type 304 SS-Cold worked	
Clad thickness, inches	0.019	
Number (full length)	48	
Number of rods per assembly	20	

TABLE 2-2 (Continued)

Burnable Poison Rods	
Material	Pyrex glass
Content B_2O_3 (w/o)	12.5
Core Structure	
Core barrel I.D./O.D., inches	133.875/137.875
Thermal shield I.D./O.D., inches	142.625/148.000
Core diameter, inches (approximate)	119.5
Reflector thickness (approximate) and composition	
Top - Water plus steel, in.	10
Bottom - Water plus steel, in.	10
Side - Water plus steel, in.	15

FIGURE 2-1

CROSS SECTIONAL VIEW OF SURRY FUEL ASSEMBLY

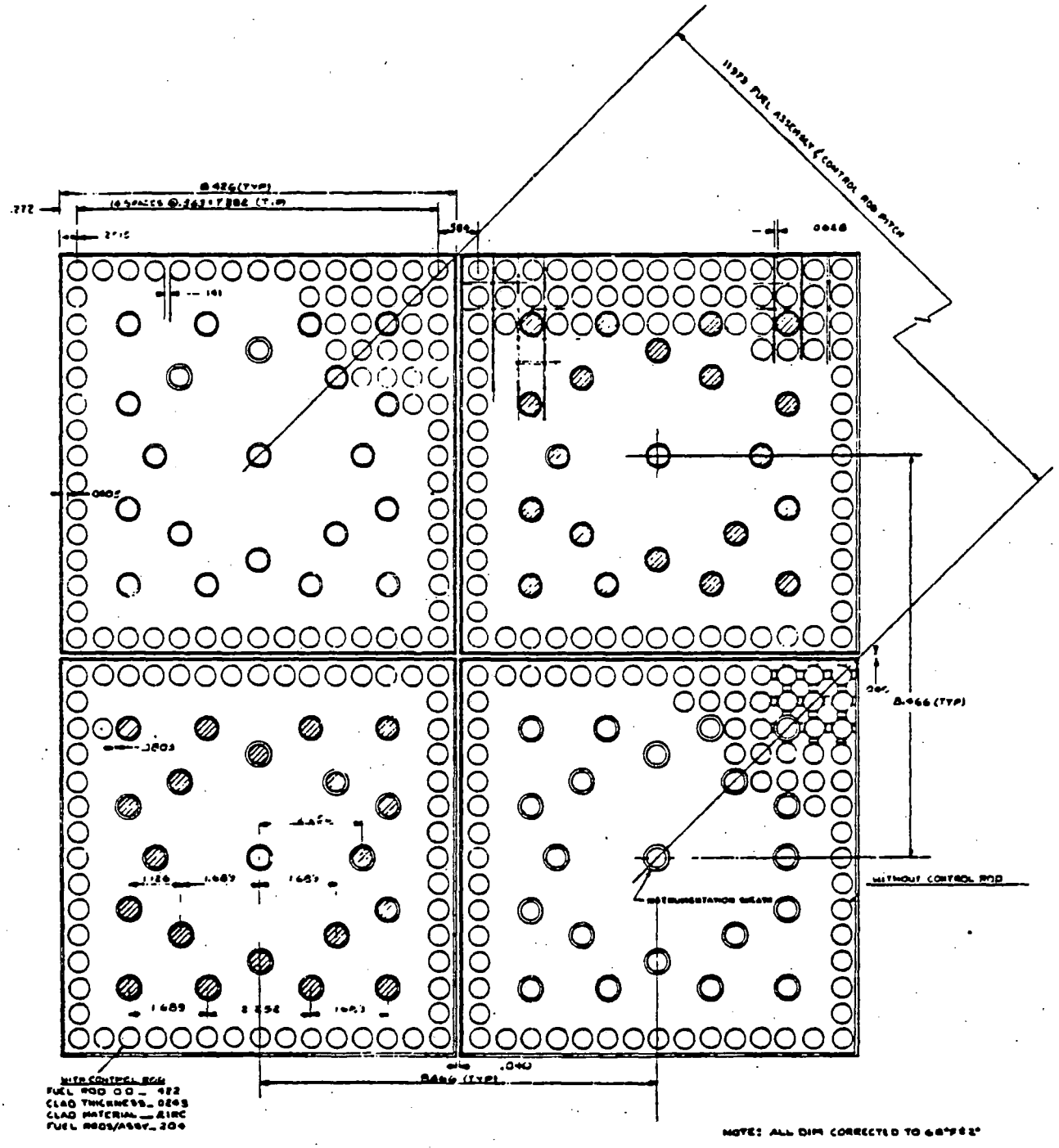
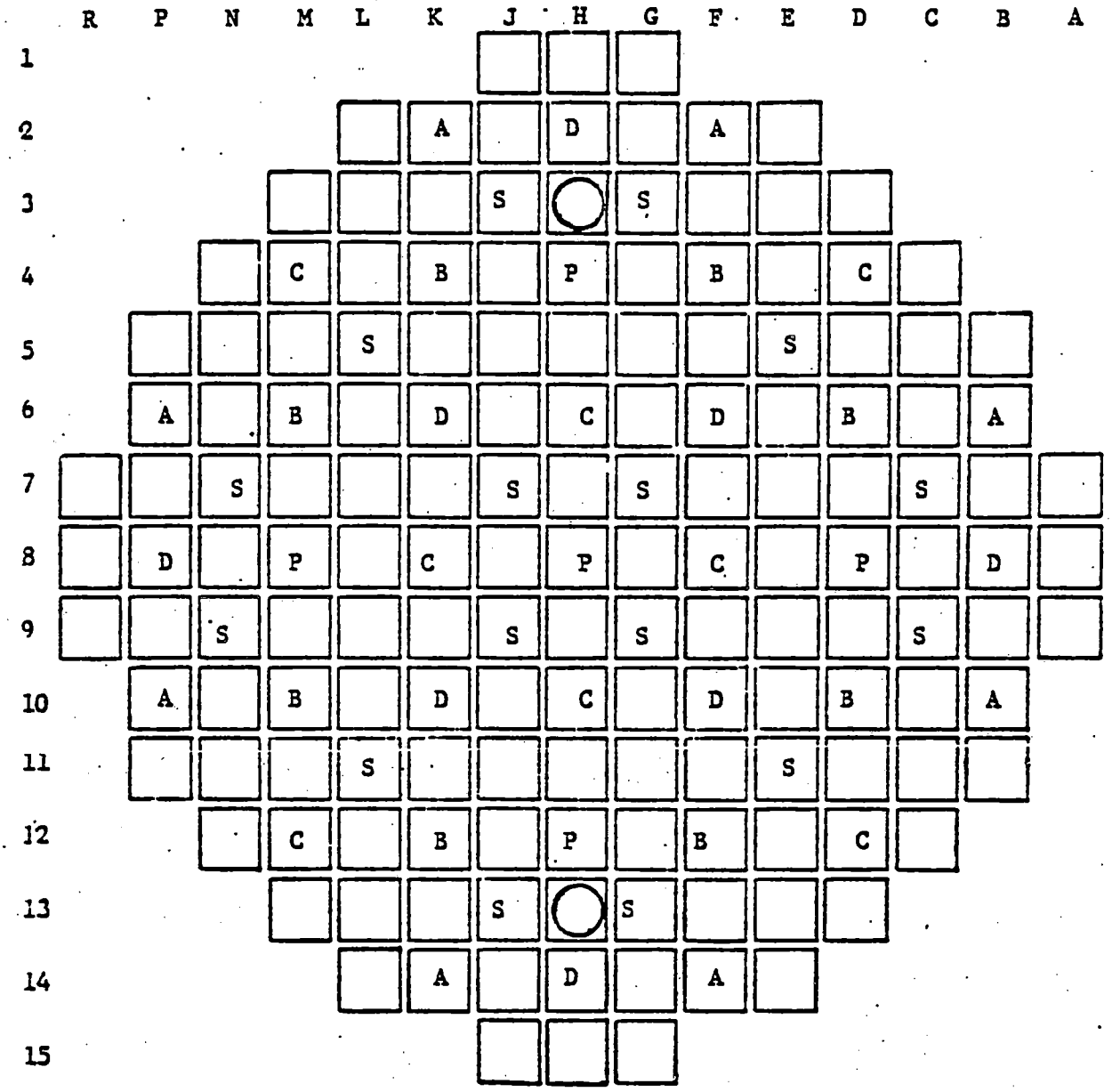


FIGURE 2-2

CONTROL ROD BANK LOCATIONS



CONTROL ROD ASSEMBLY BANKS

Function	Number of Assemblies
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown (S)	16
Part Length (P)	5
	53

◯ = SOURCE ASSEMBLY LOCATIONS

FIGURE 2-3

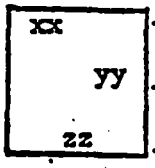
SURRY UNITS 1 AND 2 -- CYCLE 1

FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh
J	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh	3 0 Fresh
K	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	
L	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	3 0 Fresh	
M	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh		
N	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	3 0 Fresh			
P	1 0 Fresh	3 0 Fresh	3 0 Fresh	3 0 Fresh				
R	3 0 Fresh	3 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (53)	1.85	94
2 (52)	2.55	93
3 (52)	3.10	92

LEGEND



---Batch No.

---Initial Burnup (MWD/MTU)

---Previous Location (If applicable)

FIGURE 2-4

SURRY UNIT 1 - CYCLE 2
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 15358 H08	4B 0 Fresh	1 12536 H14	4B 0 Fresh	2 15298 H13	4B 0 Fresh	1 14680 H12	4C 0 Fresh
J	4B 0 Fresh	2 16435 L08	4A 0 Fresh	2 14299 K13	2 16075 K11	2 14378 L12	4C 0 Fresh	4C 0 Fresh
K	1 12536 P08	4A 0 Fresh	1 11276 M12	4A 0 Fresh	2 15918 J12	1 14191 K12	4C 0 Fresh	
L	4B 0 Fresh	2 14299 N10	4A 0 Fresh	2 16755 J08	2 16599 J10	4C 0 Fresh	4C 0 Fresh	
M	2 15298 N08	2 16075 L10	2 15918 M09	2 16599 K09	4A 0 Fresh	4C 0 Fresh		
N	4B 0 Fresh	2 14378 M11	1 14191 M10	4C 0 Fresh	4C 0 Fresh			
P	1 14680 M08	4C 0 Fresh	4C 0 Fresh	4C 0 Fresh				
R	4C 0 Fresh	4C 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (21)	1.85	94
2 (52)	2.55	93
4A(20)	1.85	95
4B(12)	2.60	95
4C(52)	3.35	95

LEGEND

xx	---Batch No.
yy	---Initial Burnup (MWD/MTU)
zz	---Previous Location (If applicable)

FIGURE 2-5

SURRY UNIT 2 - CYCLE 2
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 16693 J09	3 15422 G14	3 11414 H15	2 17992 H11	4A 0 Fresh	2 18295 H09	2 16805 C08	4B 0 Fresh
J	3 15422 P09	4A 0 Fresh	2 17635 K11	4A 0 Fresh	2 15755 K13	3 14725 N11	4B 0 Fresh	4B 0 Fresh
K	3 11414 R08	2 17635 L10	3 15422 J14	2 18151 J10	4A 0 Fresh	2 17470 J12	4B 0 Fresh	
L	2 17992 L08	4A 0 Fresh	2 18151 K09	4A 0 Fresh	2 15860 L12	4B 0 Fresh	4B 0 Fresh	
M	4A 0 Fresh	2 15755 N10	4A 0 Fresh	2 15860 M11	4A 0 Fresh	4B 0 Fresh		
N	2 18295 J08	3 14725 L13	2 17470 M09	4B 0 Fresh	4B 0 Fresh			
P	2 16805 H13	4B 0 Fresh	4B 0 Fresh	4B 0 Fresh				
R	4B 0 Fresh	4B 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (1)	1.85	94
2 (52)	2.55	93
3 (20)	3.10	92
4A(32)	2.60	94
4B(52)	3.10	95

LEGEND

xx	---Batch No.
yy	---Initial Burnup (MWD/MTU)
zz	---Previous Location (If applicable)

FIGURE 2-6

SURRY UNIT 1 - CYCLE 3
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 15236 Cy1 K08	3 10420 Cy1 H01	1 14182 Cy1 E11	3 14034 Cy1 G14	3 14034 Cy1 J14	3 12224 Cy1 K14	3 8387 Cy1 L14	4C 6239 Cy2 H01
J	3 10420 Cy1 A08	4B 8568 Cy2 L08	3 8973 Cy1 C12	4A 7946 Cy2 J10	3 13334 Cy1 E03	1 13859 Cy1 J13	4C 7142 Cy2 F14	4C 4903 Cy2 G01
K	1 14182 Cy1 L11	3 8973 Cy1 M03	4A 6462 Cy2 M12	5 0 Fresh	5 0 Fresh	3 8174 Cy1 J15	4C 6967 Cy2 G02	
L	3 14034 Cy1 P09	4A 7946 Cy2 K09	5 0 Fresh	3 12224 Cy1 P10	4A 7486 Cy2 K11	4C 7601 Cy2 E03	4C 5667 Cy2 D03	
M	3 14034 Cy1 P07	3 13334 Cy1 C05	5 0 Fresh	4A 7486 Cy2 L10	3 8377 Cy1 B05	4C 5230 Cy2 E02		
N	3 12224 Cy1 P06	1 13859 Cy1 N09	3 8174 Cy1 R09	4C 7601 Cy2 C05	4C 5230 Cy2 B05			
P	3 8377 Cy1 P05	4C 7142 Cy2 P06	4C 6967 Cy2 B07	4C 5667 Cy2 C04				
R	4C 6239 Cy2 A08	4C 4903 Cy2 A07						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (13)	1.85	94
3 (52)	3.10	92
4A(20)	1.85	95
4B(4)	2.60	95
4C(52)	3.35	95
5 (16)	2.10	95

LEGEND

xx	---Batch No.
yy	---Initial Burnup (MWD/MTU)
zz	---Previous Location (If applicable)
Previous cycle-- nn	

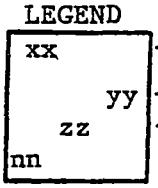
FIGURE 2-7

SURRY UNIT 2 - CYCLE 3
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 16635 Cy1 K08	4B 10342 Cy2 J14	4A 10943 Cy2 G09	3 13414 Cy1 K14	1 15214 Cy1 J13	4B 10342 Cy2 G14	1 13761 Cy2 H14	4B 6985 Cy2 H01
J	4B 10342 Cy2 P07	1 15214 Cy1 N09	4B 10076 Cy2 L13	4A 11093 Cy2 K12	4B 8505 Cy2 K14	4A 11101 Cy2 J11	4B 5739 Cy2 J15	5 0 Fresh
K	4A 10943 Cy2 J09	4B 10076 Cy2 N11	4A 10916 Cy2 M08	3 9228 Cy1 L14	1 15586 Cy1 K12	3 8978 Cy1 J15	4B 5972 Cy2 L14	
L	3 13414 Cy1 P06	4B 11093 Cy2 M10	3 9228 Cy1 P11	1 16377 Cy1 L09	3 9896 Cy1 M13	4B 6806 Cy2 M13	5 0 Fresh	
M	1 15214 Cy1 N07	4B 8505 Cy2 P10	1 15586 Cy1 M10	3 9896 Cy1 N12	3 13414 Cy1 P10	5 0 Fresh		
N	4B 10342 Cy2 P09	4A 11101 Cy2 L09	3 8978 Cy1 R09	4B 6806 Cy2 N12	5 0 Fresh			
P	1 13761 Cy1 P08	4B 5739 Cy2 R09	4B 5972 Cy2 P11	5 0 Fresh				
R	4B 6985 Cy2 A08	5 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (25)	1.85	94
3 (32)	3.10	92
4A(24)	2.60	94
4B(52)	3.10	95
5 (24)	3.10	95

LEGEND



---Batch No.
---Initial Burnup (MWD/MTU)
---Previous Location (If applicable)
Previous Cycle---

FIGURE 2-8

SURRY UNIT 1 - CYCLE 4
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 15236 Cy1 H06	6A 0 Fresh	5 10552 Cy3 F11	6A 0 Fresh	4B 8760 Cy2 H09	5 10552 Cy3 E10	4B 8615 CY2 H13	6C 0 Fresh
J	6A 0 Fresh	4C 17296 Cy3 N11	6A 0 Fresh	4C 10529 Cy3 A09	4C 11355 Cy3 N04	4C 14951 Cy3 F02	6C 0 Fresh	6C 0 Fresh
K	5 10552 Cy3 L10	6A 0 Fresh	2/4A 9890 S2C2 M12	6A 0 Fresh	4C 16875 Cy3 J14	6B 0 Fresh	6C 0 Fresh	
L	6A 0 Fresh	4C 10529 Cy3 J01	6A 0 Fresh	4C 17287 Cy3 L13	4C 11267 Cy3 L14	6C 0 Fresh	6C 0 Fresh	
M	4B 8760 Cy2 J08	4C 11355 Cy3 D13	4C 16875 Cy3 P09	4C 11267 Cy3 P11	4C 13240 Cy3 R08	6C 0 Fresh		
N	5 10552 Cy3 K11	4C 14951 Cy3 B06	6B 0 Fresh	6C 0 Fresh	6C 0 Fresh			
P	4B 8615 Cy2 N08	6C 0 Fresh	6C 0 Fresh	6C 0 Fresh				
R	6C 0 Fresh	6C 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (1)	1.85	94
2/4A (4)	2.60	95
4B (8)	2.60	95
4C (52)	3.35	95
5 (8)	2.10	95
6A (24)	2.60	95
6B (8)	2.60	95
6C (52)	2.90	95

LEGEND

xx	----Batch No.
yy	----Initial Burnup (MWD/MTU)
zz	----Previous Location (If applicable)

FIGURE 2-9

SURRY UNIT 2 - CYCLE 4
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 16635 Cy1 F08	6A 0 Fresh	4B 21458 Cy3 G10	4B 13392 Cy3 H15	4B 21240 Cy3 H09	4B 20813 Cy3 H13	4B 17227 Cy3 E13	6B 0 Fresh
J	6A 0 Fresh	4B 19642 Cy3 M09	6A 0 Fresh	5 5849 Cy3 J15	5 6678 Cy3 L14	4B 14467 Cy3 K14	6B 0 Fresh	6B 0 Fresh
K	4B 21458 Cy3 K09	6A 0 Fresh	4B 19630 J12	5 7340 Cy3 M13	4B 15745 Cy3 J14	6B 0 Fresh	6A 0 Fresh	
L	4B 13392 Cy3 R08	5 5849 Cy3 R09	5 7340 Cy3 N12	4B 21464 Cy3 J10	6A 0 Fresh	6B 0 Fresh	6B 0 Fresh	
M	4B 21240 Cy3 J08	5 6678 Cy3 P11	4B 15745 Cy3 P09	6A 0 Fresh	4B 17255 Cy3 L13	6B 0 Fresh		
N	4B 20813 Cy3 N08	4B 14467 Cy3 P10	6B 0 Fresh	6B 0 Fresh	6B 0 Fresh			
P	4B 17227 Cy3 N11	6B 0 Fresh	6A 0 Fresh	6B 0 Fresh				
R	6B 0 Fresh	6B 0 Fresh						

Batch (#F.A.'s)	Initial Enrichment w/o U235	Density %TD
1 (1)	1.85	94
4B (52)	3.10	95
5 (24)	3.10	95
6A (28)	2.90	95
6B (52)	3.20	95

Legend

xx ----Batch No.
yy ----Initial Burnup (MWD/MTU)
zz ----Previous Location (If applicable)

FIGURE 2-10

SURRY UNITS 1 AND 2 -- CYCLE 1
Burnable Poison Rod Loading

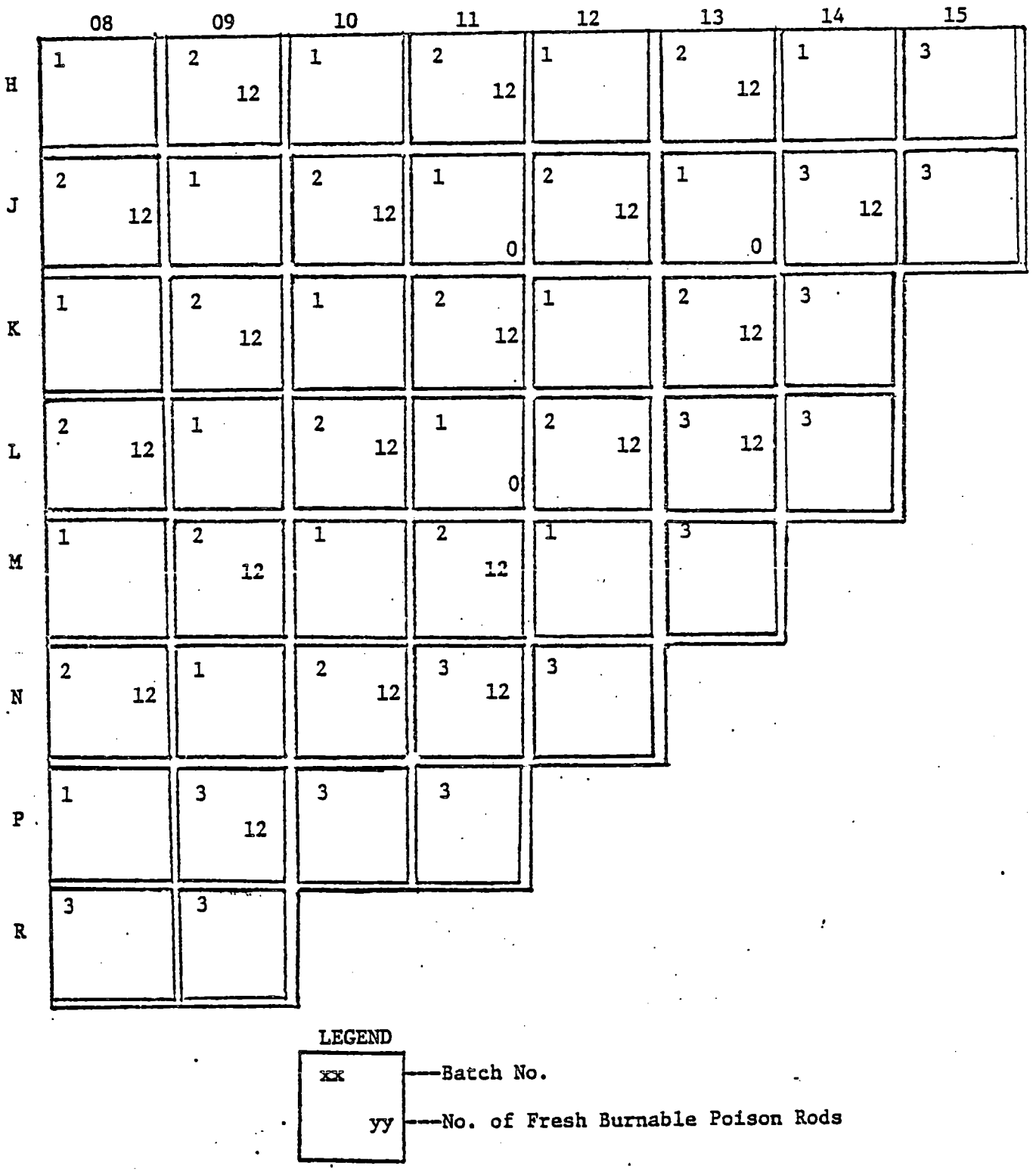


FIGURE 2-11

SURRY UNIT 1 -- CYCLE 2
BURNABLE POISON ROD LOADING

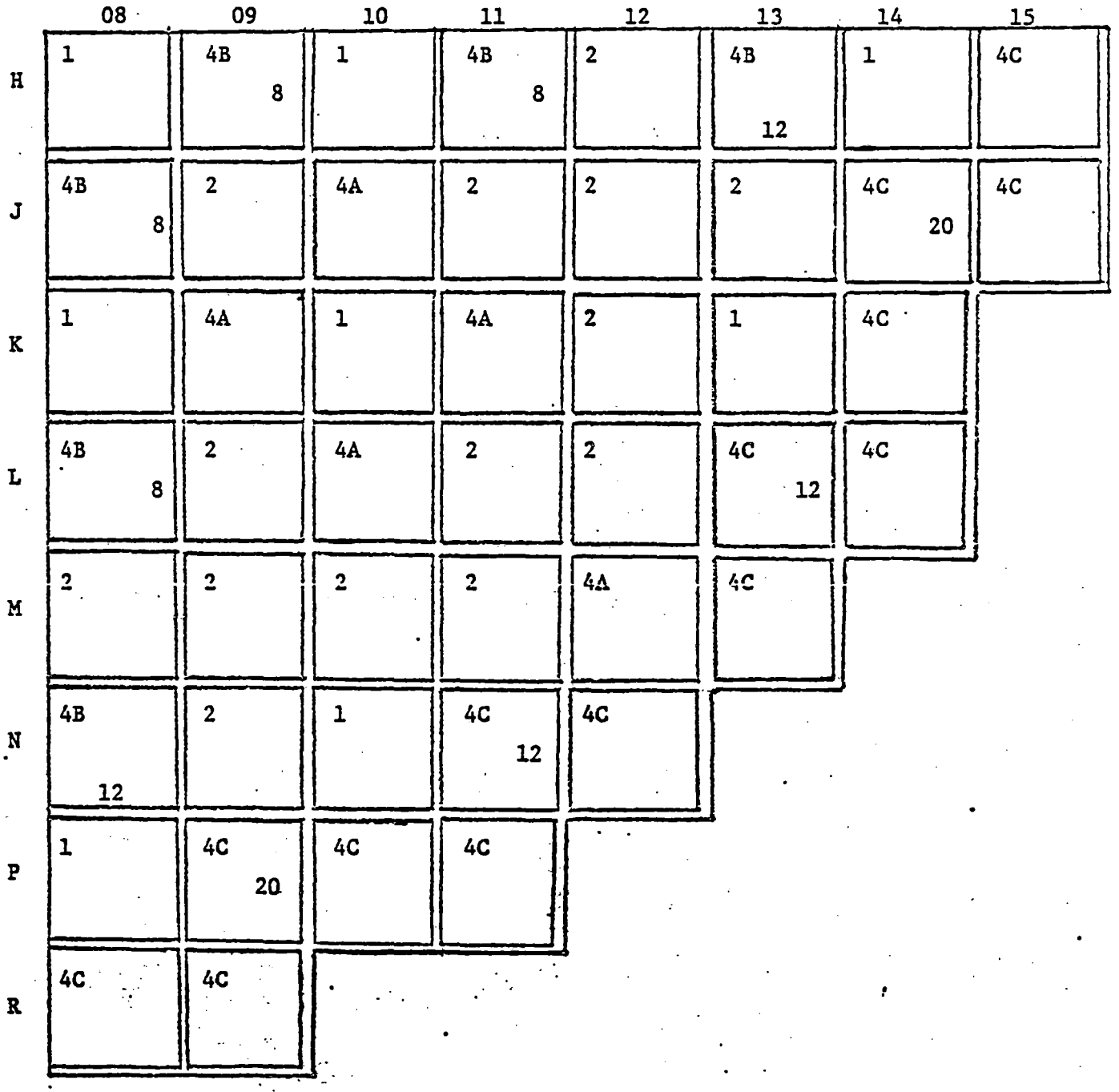
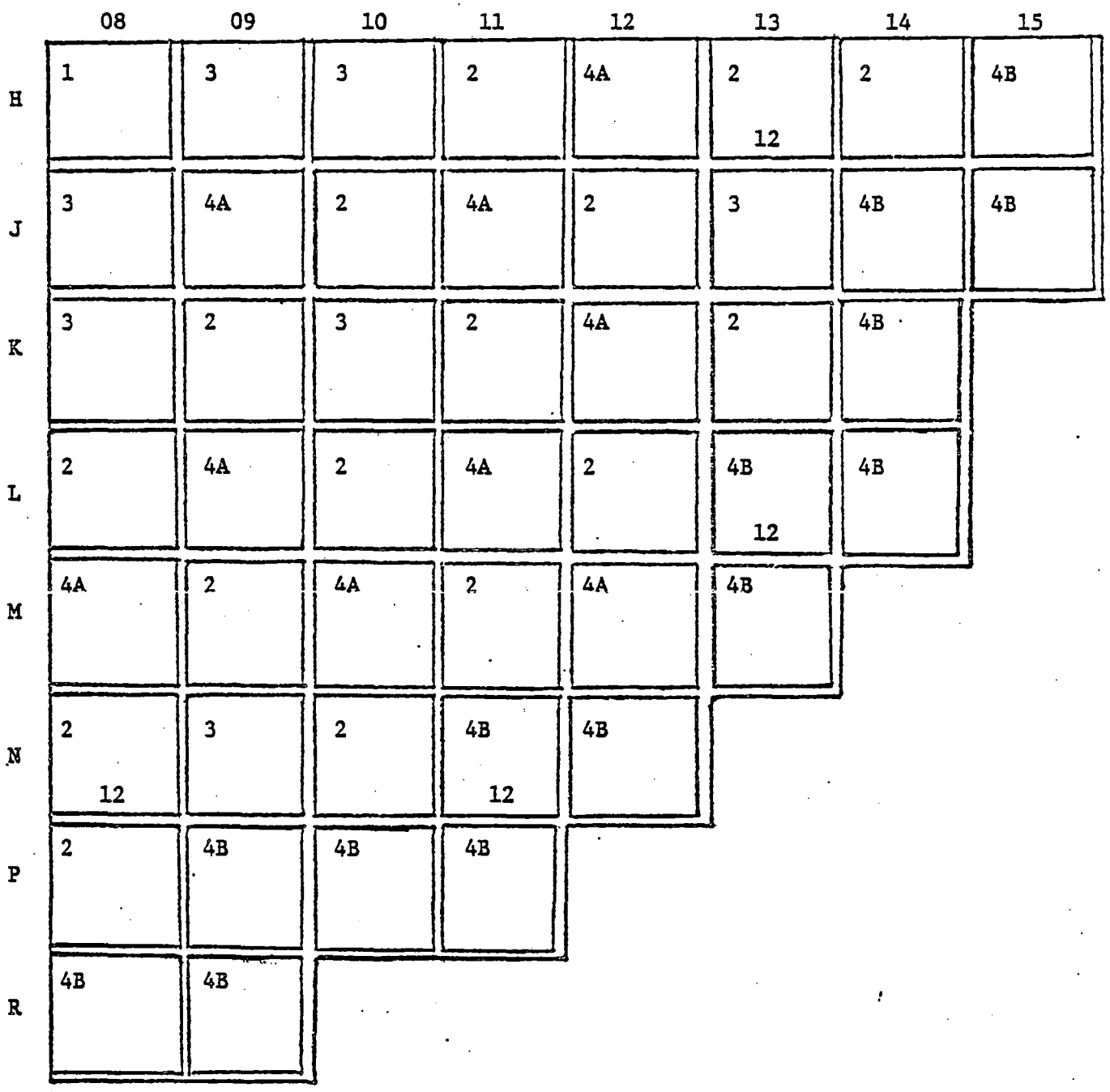


FIGURE 2-12

SURRY UNIT 2 - CYCLE 2
BURNABLE POISON ROD LOADING



LEGEND

xx ——Batch No.

zz ——No. of Depleted Burnable Poison Rods

FIGURE 2-13

SURRY UNIT 1 - CYCLE 3
BURNABLE POISON ROD LOADING

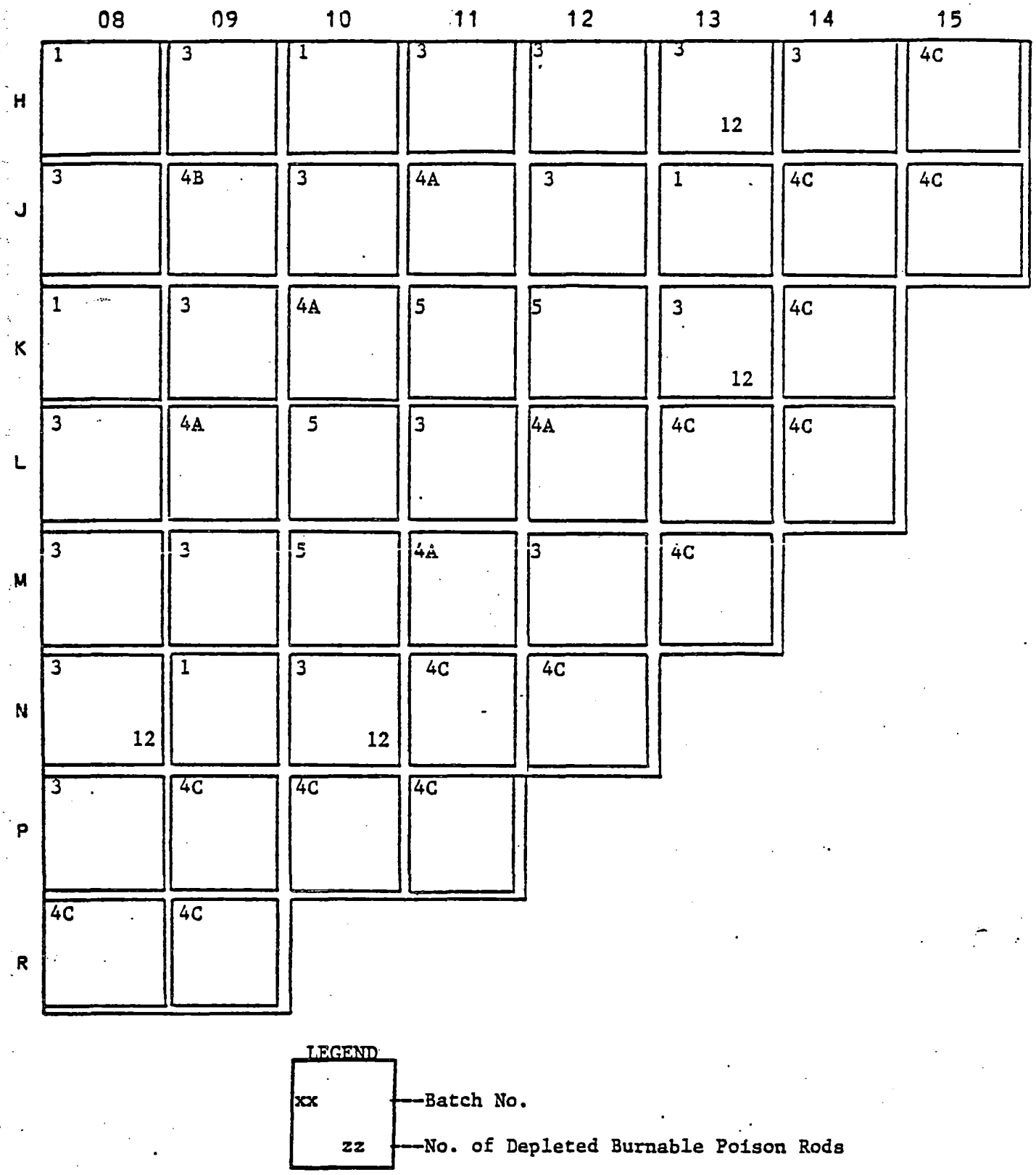
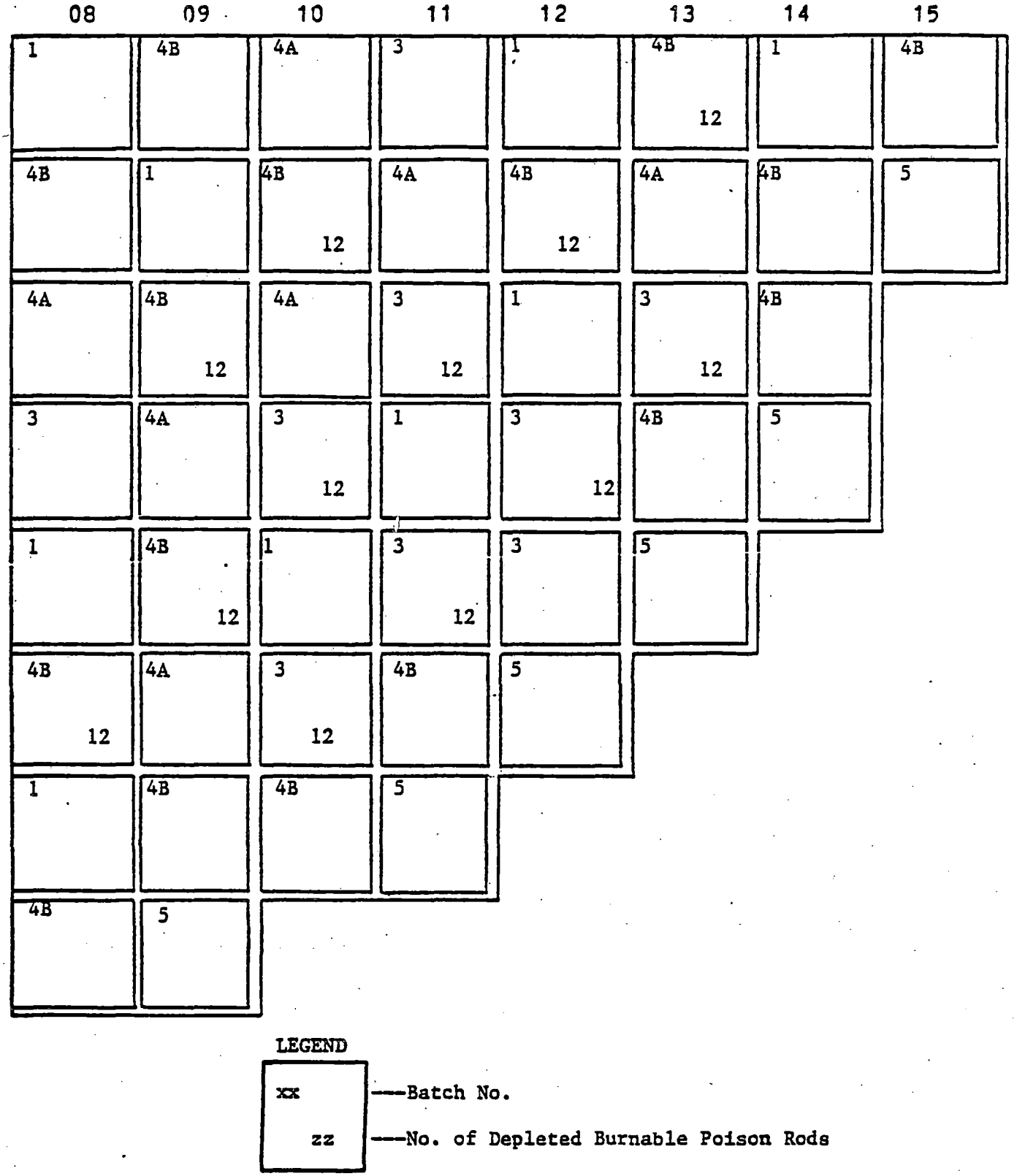


FIGURE 2-14

SURRY UNIT 2 - CYCLE 3
BURNABLE POISON ROD LOADING



LEGEND

xx ---Batch No.

zz ---No. of Depleted Burnable Poison Rods

FIGURE 2-15

SURRY UNIT 1 - CYCLE 4
BURNABLE POISON ROD LOADING

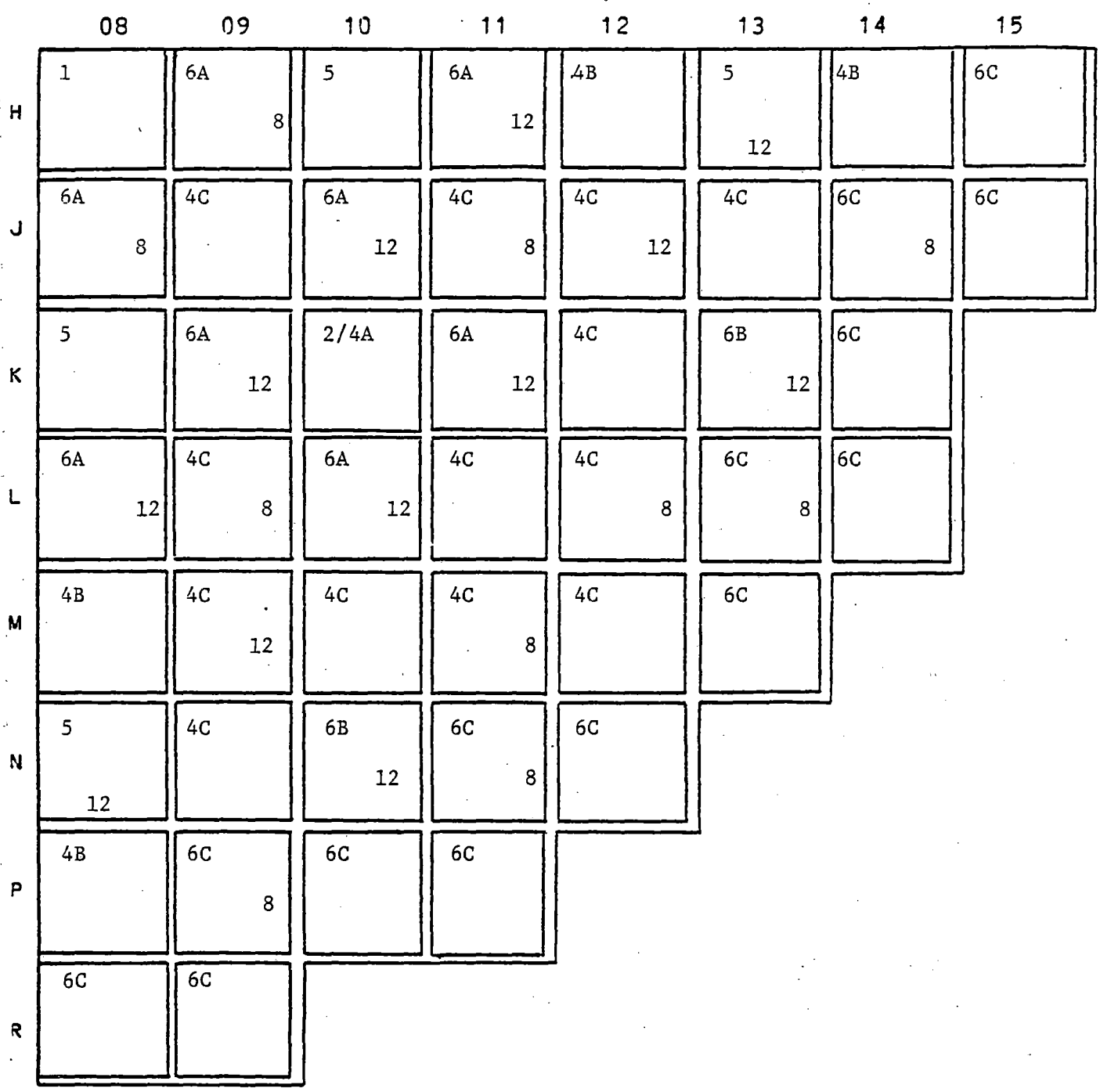
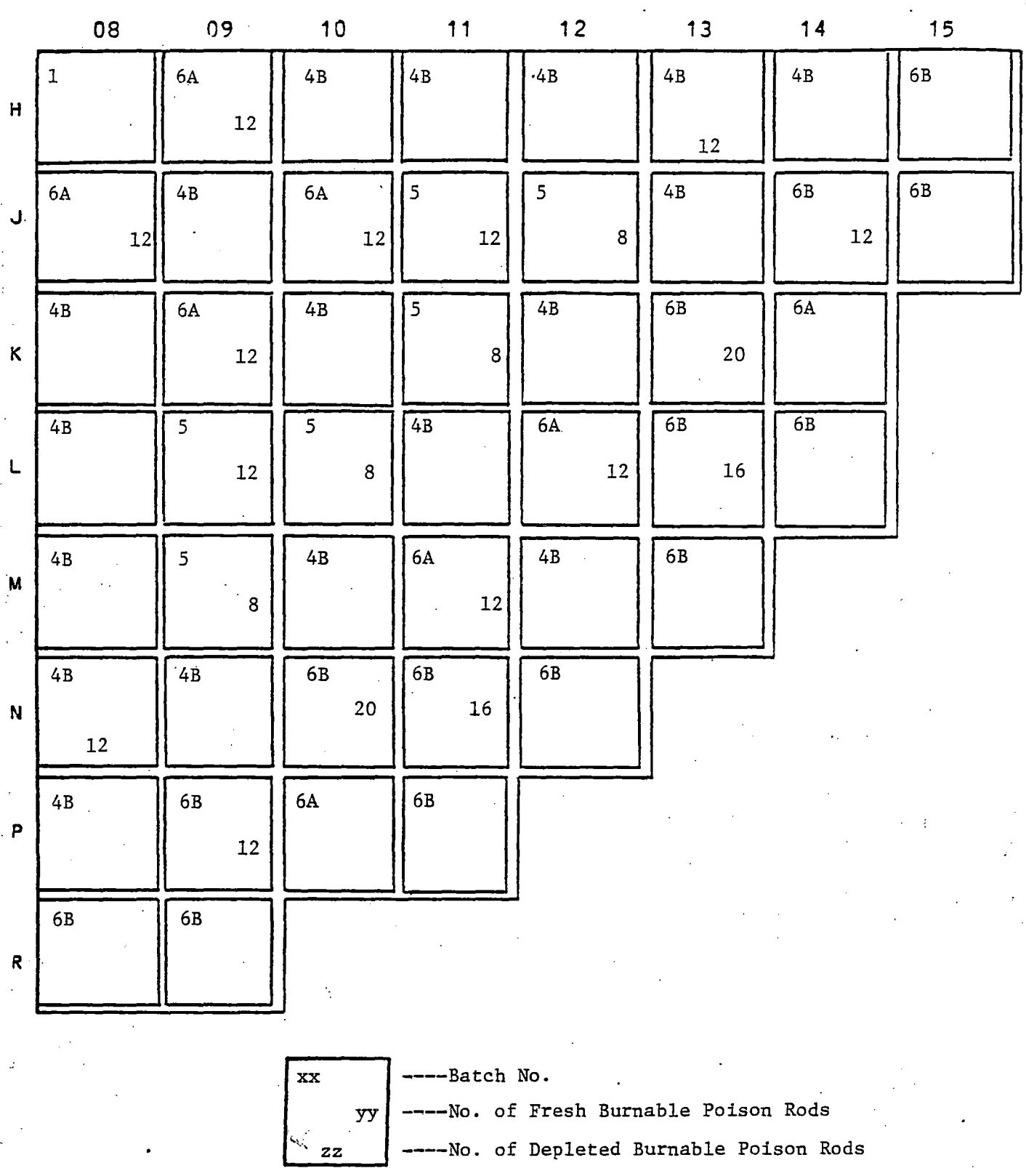


FIGURE 2-16

SURRY UNIT 2 - CYCLE 4
BURNABLE POISON ROD LOADING



SECTION 3 - MODEL DESCRIPTION

3.1 Introduction

The Vepco FLAME Model is used to calculate nodal power densities and core reactivity for three-dimensional geometries in which each fuel assembly is represented by one radial node and by up to 32 axial nodes. The method used by the Vepco FLAME Model to perform these calculations is based on the FLARE[9] technique which is derived from modified, one energy group diffusion theory. Effects of nonuniform moderator density and fuel temperature are accounted for by thermal-hydraulic feedback.

The Vepco FLAME Model performs calculations in several steps. First, a fine-group neutron flux spectrum and the appropriate cross sections as a function of neutron energy are calculated for each material composition in the fuel assembly by a cross section generating code (i.e., the NULIF[1] computer code). Then the fine-group flux spectrum is used to spectrum weight and collapse the fine-group cross sections into the one energy group parameters required (i.e., K_∞ and M^2) by the FLAME3[3] computer code. These parameters, as well as any appropriate normalization factors, are then used by FLAME3 to perform an iterative, modified diffusion theory calculation for the neutron production rate density as a function of position. The method of solution comprises two levels of iteration: source (or nuclear) and thermal-hydraulic feedback. The source calculation is performed first based on an initial guess for the thermal-hydraulic parameters. Then a new set of thermal-hydraulic parameters are calculated. Using these new thermal-hydraulic parameters, another source calculation is performed. This process is continued until both the source and thermal-hydraulic convergence criteria are satisfied.

Several interrelated computer codes are used to perform the calculations outlined above. The computer codes comprising the Vepco FLAME Model and their interrelationships are presented in the flow chart in Figure 3-1. The FLAME3 computer code is the principal analytical tool in the Vepco FLAME Model. FLAME3 is used to perform the three-dimensional, one group, modified diffusion theory calculations. The other codes provide either input data or data manipulation. As indicated in Figure 3-1, the NULIF code is used to generate the required one group data for the non-rodded fuel assembly. The FLAFIT[3] code formats these data for use by FLAME3. Data for burnable poison and control rods are generated with the Vepco PDQ07 Discrete Model[7] using a quarter assembly discrete representation. The in-house codes EDITQAR[5] and PICCOLO[5] format the data from the quarter assembly PDQ07 calculations for use by FLAME3. The FLMSHUFL[6] code is a data manipulation code that shuffles the appropriate end-of-cycle burnup distributions to duplicate the movement of fuel assemblies during refueling. The shuffled burnup distributions from FLMSHUFL are input to FLAME3 to begin reload cycle analyses.

The remainder of this chapter describes in greater detail the input to and functioning of the computer codes used in the Vepco FLAME Model.

3.2 Input Preparation

The Vepco FLAME Model uses the FLARE neutron source option for all calculations with the FLAME3 code. Two physics parameters are required at each node by the FLARE[9] option: the infinite multiplication factor (K_∞) and the migration area (M^2). At peripheral nodes, it may also be necessary to input a leakage parameter (or albedo).

The infinite multiplication factor, migration area, and albedoes for each node are usually functions of one or more variables. Among these are initial

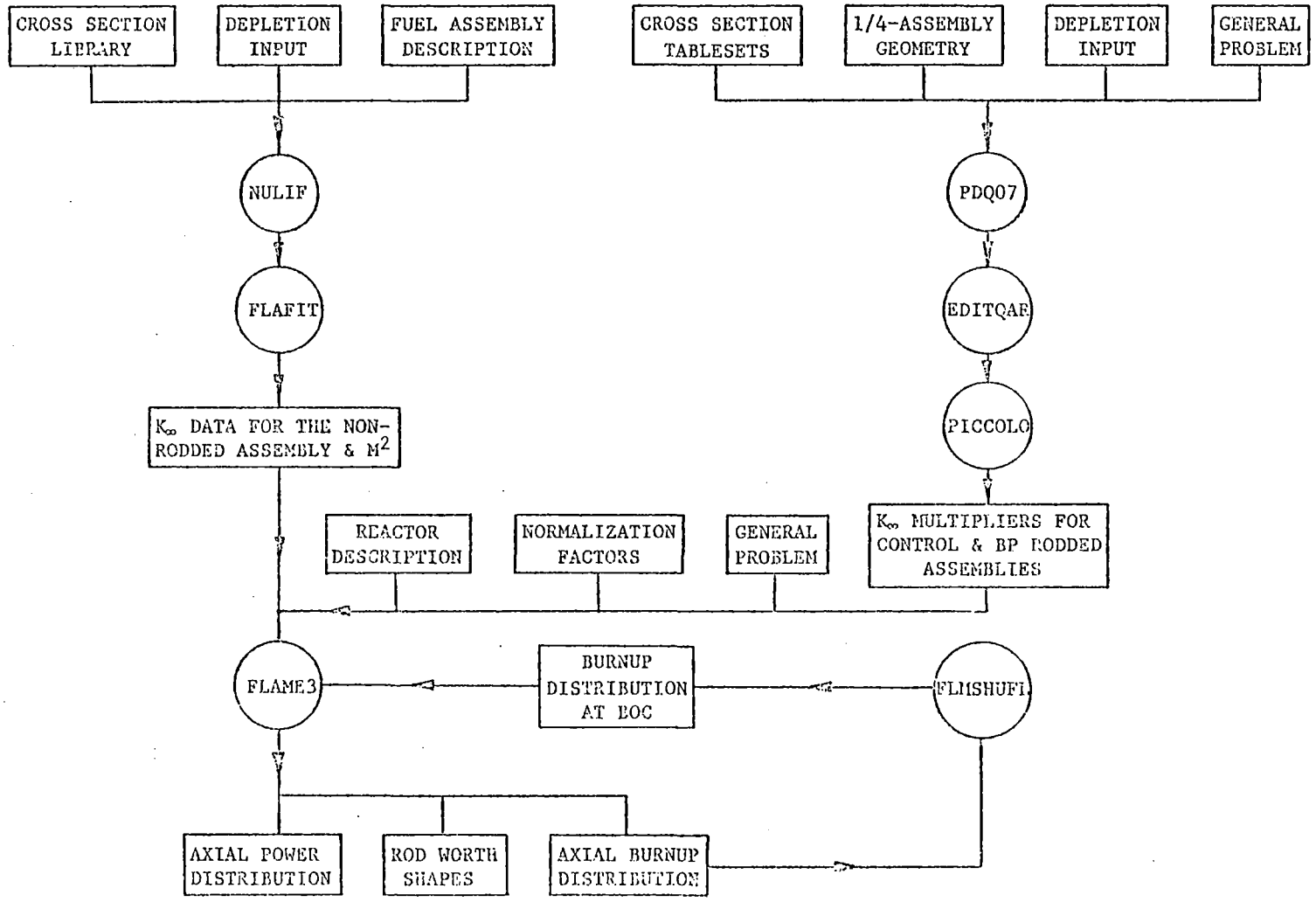


FIGURE 3-1
FLOW CHART FOR VEPCO
FLAME MODEL.

enrichment, burnup, soluble boron concentration, xenon concentration, fuel and moderator temperatures, and the presence or absence of burnable poison or control rods. To represent the functional dependencies, K_∞ and M^2 are calculated for various combinations of the appropriate variables. The resulting values of K_∞ (or M^2) are then tabulated for use in the FLAME3 code by using the FLAFIT and PICCOLO codes. The FLAME3 code uses a Lagrangian interpolation routine for selecting the appropriate input data from the tables constructed by FLAFIT and PICCOLO. This routine allows any input variable (e.g., K_∞) to be fit as a function of one, two, or three independent variables. Due to this constraint, it has been necessary to use several tables to properly represent all the functional dependencies of K_∞.

The leakage parameters cannot be input in tabular form. However, the leakage parameters may be changed to reflect changing reactor conditions. This is discussed further in Section 4.1.

The processes by which tabular input for K_∞ and M^2 are developed are described in this section.

3.2.1 Method of Calculating Input Data for the Non-Rodded Fuel Assembly

The NULIF code is used to generate the non-rodded (i.e., no control or burnable poison rods are present) fuel assembly input data for the FLAME3 code. The NULIF calculations are performed using the supercell option. A supercell is defined as a group of unit cells comprising, for example, a fuel assembly. For the supercell group, the fuel rod unit cell is designated as the central cell. Any other unit cell types (e.g., control rod guide tube cell) present in an assembly are designated as subregion cells (subcells). The supercell option is used to represent the fuel assembly in the Vepco FLAME Model because only homogenized groups of unit cells can be represented due the larger-than-discrete mesh description implicit in a nodal representation of the reactor core. The homogenization is performed in a manner that

results in one node representing each fuel assembly.

The calculation of the neutron energy spectrum and the spectrum-weighted two-group cross sections for each supercell is described in detail in Reference 1. Those aspects which involve calculations using the supercell option are described below.

With the supercell option of the NULIF code, the material compositions of the central cell and the various subcells must be homogenized together before the fine-group neutron flux is calculated, since the NULIF code does not perform a spatial calculation for the various subregions. Because of this, a method must be employed to represent the heterogeneous nature of the supercell. This is done by inputting appropriate thermal flux depression factors for each subcell relative to the supercell. These flux depression factors are generated by a detailed spatial calculation (i.e., a quarter assembly discrete PDQ07 calculation where each fuel rod, thimble cell, and water channel associated with the fuel assembly is explicitly represented). From this detailed spatial calculation, the ratios of the thermal flux in the average fuel cell, thimble cell, and water gap relative to the thermal flux in the entire assembly are determined. The above flux depression factors are combined in NULIF with those normally calculated by NULIF for the central cell (i.e., the flux distribution in the fuel pellet, clad, and moderator regions of the central cell) to give the overall flux depression factors to be applied over each of the 80 thermal fine-groups for each nuclide in the supercell.

NULIF calculates the neutron flux in the supercell for each of 31 fast and 80 thermal energy fine groups. The macroscopic, one-energy group parameters needed as input to the FLAME3 code are then determined from the neutron flux and cross sections for each fine group.

Assembly (supercell) calculations of K_∞ and M^2 with the NULIF code are performed as a function of:

1. Initial enrichment
2. Burnup
3. Soluble boron concentration
4. Moderator specific volume
5. Fuel temperature
6. Xenon concentration

An assembly of each enrichment used in the core is depleted with NULIF while maintaining constant soluble boron and xenon concentrations, moderator specific volume and fuel temperature. Recovery cases are then performed at selected burnups where these parameters are varied. Care has been taken to vary each parameter over the range of values expected to occur during the types of operation to be modeled with the Vepco FLAME Model.

3.2.1.1 K-Infinity

The Vepco FLAME Model uses for each enrichment a basic K_∞ which is a function of burnup, soluble boron concentration, and xenon concentration. This combination of independent variables was chosen for two reasons. First, these variables have the largest effects on K_∞ for the non-rodded assembly (aside from enrichment which, as stated above, is represented by generating separate tablesets for each initial enrichment). Second, the effects of these variables are relatively insensitive to changes in other parameters. For these reasons, the functional dependencies of K_∞ on burnup, boron, and xenon are represented together in the basic K_∞.

Since K_∞ cannot be adequately represented as a function of only three independent variables (the limit for any table used in the FLAME3 code), other tables (i.e., K_∞ multipliers) are developed to allow the incorporation of additional functional dependencies. The K_∞ multiplier tables contain

factors which are used to modify the values contained in the basic K_∞ tables. These factors are arrived at by taking the ratio of K_∞ at some nominal condition to the K_∞ generated by the NULIF code after a change in one or two independent variables. For the non-rodded fuel assembly, two K_∞ multipliers are used to represent the effects of fuel and moderator temparatures.

The first K_∞ multiplier represents the effects on K_∞ of changing moderator temperature. Changes in moderator temperature result in two important phenomena: 1) change in moderation (neutron spectrum) and 2) change in parasitic absorption in the moderator. For these reasons, the first K_∞ multiplier is fit as a function of moderator specific volume, burnup, and soluble boron concentration.

The second K_∞ multiplier incorporates the dependence of K_∞ on fuel temperature. Changing fuel temperature causes a change in resonance absorption. This effect is sensitive only to burnup. Therefore, the second K_∞ multiplier is fit as a function of fuel temperature and burnup.

3.2.1.2 Migration Area

The migration area, M^2, is used by the FLAME3 code as a measure of the leakage from an assembly. As such, M^2 is sensitive only to moderator specific volume, burnup, and enrichment. Therefore, M^2 can be adequately represented by a single table fit as a function of these three parameters.

In addition, the FLAME3 code permits the use of a M^2 multiplier as a normalization factor. The use of M^2 multipliers is discussed in Section 4.1.

3.2.1.3 Input Parameters for Calculating Xenon Concentration

The FLARE neutron source option does not allow the explicit calculation of any material concentrations. However, to accurately represent K_∞ it is necessary to know the xenon concentration at each node in the core. Once the nodal xenon concentrations are found, then K_∞ for each node may be obtained from the appropriate tables.

To find the nodal xenon concentrations, the FLAME3 code solves the differential equations for xenon and iodine based on the assumption of constant power throughout a burnup step (see Section 3.4). This solution requires that several parameters be input to the FLAME3 code in tabular form as functions of the appropriate independent variables. These input parameters are listed below:

$$A = (K_1\Sigma_{f1}\phi_1 + K_2\Sigma_{f2}\phi_2)(1/\phi_2)$$

$$YI = (\gamma_I\Sigma_{f1}\phi_1 + \gamma_I\Sigma_{f2}\phi_2)(1/\phi_2)$$

$$YX = (\gamma_X\Sigma_{f1}\phi_1 + \gamma_X\Sigma_{f2}\phi_2)(1/\phi_2)$$

σ_{a2}^{Xe} = thermal microscopic absorption cross section of xenon,

SIGM = multiplier for σ_{a2}^{Xe}

where: K = energy released per fission

Σ_f = macroscopic fission cross section

ϕ = neutron flux

γ_I = yield per fission of Iodine-135

γ_X = yield per fission of Xenon-135

and subscripts 1 and 2 refer to the fast and thermal energy groups respectively. The parameters are all generated by the NULIF code and are tabulated by FLAFIT for use in the FLAME3 code.

The parameters A, YI, and YX are strongly dependent on burnup and enrichment. However, variations in other independent variables do not significantly influence these parameters. For this reason, A, YI, and YX are fit as a function of burnup and enrichment only.

The thermal microscopic absorption cross section of xenon is dependent upon the neutron energy spectrum in the fuel pellet. The neutron spectrum is, in turn, dependent upon the amounts of absorbing and fissile material in the fuel and the moderation which occurs outside the fuel pellet. Therefore, σ_{a2}^{Xe} is fit as a function of burnup, xenon concentration and enrichment. To account for the spectral effects of the moderator, the multiplier

for σ_{a2}^{Xe}, (SIGM), is fit as a function of soluble boron concentration and moderator specific volume.

3.2.2 Method of Calculating Input Data for the Burnable Poison Rodded Assembly

3.2.2.1 Introduction

The input data for an assembly containing burnable poison (BP) rods are calculated with the Vepco PDQ07 Discrete Model using a 2-D, quarter-assembly representation. Theoretically, these data could be obtained by NULIF supercell calculations (as described in Section 3.2.3 above) in which one of the subcells contains the appropriate amount of BP. However, the fact that the hetergeneous effect of the various subcells are accounted for by applying thermal flux depression factors determined from quarter assembly discrete calculations must be considered. In general, for assemblies containing no BP, these flux depression factors do not change significantly as a function of burnup, so that the flux depression factors calculated at zero burnup are adequate at any stage of depletion. However, this is not the case for subcells containing BP because the BP depletes rapidly with increasing burnup causing the flux depression factors to vary significantly with burnup. Since a large number of quarter-assembly discrete calculations would have to be made to calculate these factors for input to NULIF, it has been decided to use the quarter assembly runs directly to generate K_∞ multipliers to account for BP reactivity effects. These multipliers (in conjunction with the K_∞ data for the non-rodded fuel assembly discussed in Section 3.2.1) are applied to nodes which contain (or have contained) BP rods.

Four K_∞ multipliers are used in the Vepco FLAME Model to represent BP rods. Two multipliers represent the presence of fresh BP in a fuel assembly. The third multiplier accounts for the presence of depleted BP. The fourth K_∞

multiplier is used to represent the reactivity "after-effect" resulting from the depletion of a fuel assembly containing fresh BP rods.

3.2.2.2 K-Infinity Multipliers For Fuel Containing Fresh BP

The presence of fresh BP rods in a fuel assembly changes that assembly's characteristics in two ways. First, a large concentration of non-fissile, neutron absorbing material is added to the assembly. Second, moderator is displaced from the assembly. The reactivity effects associated with these changes depend on initial assembly enrichment, cumulative burnup of the assembly while BP rods are present, soluble boron concentration, and moderator temperature. To represent these effects, two K_∞ multipliers are used.

The first K_∞ multiplier table for BP accounts for the dependence on enrichment, cummulative burnup, and soluble boron concentration. To generate this table, assemblies of various enrichments are depleted at nominal moderator and fuel temperatures with and without BP rods using a quarter assembly representation of the Vepco PDQ07 Discrete Model. At a number of burnup points, recovery cases are performed during which the soluble boron concentration is changed. The values of the first K_∞ multiplier are then obtained by ratioing K_∞ in the rodded state to K_∞ in the non-rodded state. The PICCOLO code is used to tabulate these values as a function of cumulative burnup, initial enrichment, and soluble boron concentration. This procedure is followed to construct separate tables for configurations of 8, 12, 16, and 20 BP rods per assembly.

The second K_∞ multiplier accounts for the variation in the reactivity effect of fresh BP with changes in moderator temperature. Sensitivity calculations have shown that this multiplier is not significantly affected by enrichment or number of BP rods per assembly.[10] Therefore, the second K_∞ multiplier is generated for an assembly of average enrichment (i.e., 2.90 w/o U235) containing

the most commonly used configuration of BP rods (i.e., 12 fresh BP rods per assembly). Again using the Vepco PDQ07 Discrete Model, a 2.90 w/o enriched assembly is first depleted with 12 fresh BP rods present and then depleted without the presence of any BP. At several burnup points during the above two depletions, additional cases are performed in which moderator temperature and soluble boron concentration are varied simultaneously. The values of the second K_∞ multiplier are obtained in two steps. First, for each set of conditions, the value of K_∞ obtained from the depletion case with BP rods is divided by the value of K_∞ obtained from the depletion case without BP rods. Then the ratios determined in the first step are divided by the ratio of K_∞ with BP rods to K_∞ without BP rods obtained for the nominal moderator temperature (i.e., 566°F). The PICCOLO code is then used to tabulate these values in a form acceptable to the FLAME3 code as a function of cumulative burnup, soluble boron concentration, and moderator temperature.

3.2.2.3 K-Infinity Multipliers for Fuel Containing Depleted BP

To account for the presence of depleted BP in a fuel assembly, a third K_∞ multiplier is used. As the BP rods in an assembly are depleted, the boron concentration decreases rapidly. At the end of one cycle of burnup, the boron concentration in the BP rods may be assumed to be zero. When these depleted BP rods are inserted in other assemblies, their principal effects are the displacement of water and absorption of neutrons in stainless steel. Sensitivity calculations have shown that this multiplier is not significantly affected by enrichment.[10] The only configuration of depleted BP rods used in Cycles 1 through 4 of Surry Units 1 and 2 is 12 rods per assembly. Therefore, this multiplier is generated for an assembly of average enrichment (i.e., 2.90 w/o U235) containing 12 depleted BP rods. Using the Vepco PDQ07 Discrete Model, a 2.90 w/o enriched assembly is depleted both with and without 12 depleted BP rods. At several burnup points additional

cases are performed for which moderator temperature and soluble boron concentration are varied simultaneously. The values of the third K_∞ multiplier are obtained by dividing K_∞ with depleted BP by K_∞ without BP. The PICCOLO code is then used to tabulate these values for input to the FLAME3 code as a function of total burnup, moderator temperature, and soluble boron concentration.

3.2.2.4 K-Infinity Multiplier for the "After Effect" of Fresh BP

The presence of fresh BP rods in a fuel assembly during depletion also affects the reactivity in that assembly after the BP rods are removed. This BP "after-effect" arises from the change in isotopics accompanying depletion of the fuel assembly in the harder neutron spectrum resulting from the presence of fresh BP rods. Once the BP rods are removed, the after-effect diminishes with continued depletion of the assembly. To represent this effect, assemblies of various enrichments are depleted in two ways using a quarter assembly representation of the Vepco PDQ07 Discrete Model. First, an assembly is depleted without BP rods. Second, an assembly of the same enrichment containing 12 BP rods is depleted to various burnups. Then, the BP rods are removed, and the assembly is depleted further. The values of the fourth K_∞ multiplier for BP are obtained by dividing K_∞ from the assembly which has had BP by K_∞ of the assembly which has not had BP. The PICCOLO code is used to tabulate these values as a function of cumulative burnup experienced by the assembly with BP rods present, total burnup, and enrichment.

3.2.3 Method of Calculating Input Data for the Control Rodded Assembly

3.2.3.1 Introduction

The input data for assemblies containing control rods are generated in a manner similar to that used to generate the BP data (Section 3.2.2). Again, a quarter-assembly representation of the Vepco PDQ07 Discrete Model is used to perform all the basic calculations, while the PICCOLO code is

used to construct the tables for input to the FLAME3 code.

Four K_∞ multipliers are used in the Vepco FLAME Model to represent control rods. Three multipliers represent the presence of control rods in a fuel assembly. The fourth K_∞ multiplier is used to account for the reactivity "after-effect" resulting from the depletion of a fuel assembly containing control rods.

3.2.3.2 K-Infinity Multipliers for Fuel Containing Control Rods

The presence of control rods in a fuel assembly has much the same effect as the presence of fresh BP in an assembly. First, a large concentration of non-fissile, neutron absorbing material is added to the assembly. Second, moderator is displaced from the assembly. The reactivity effects associated with the presence of control rods depend on initial assembly enrichment, cumulative burnup of the assembly while the rods are present, total assembly burnup, soluble boron concentration, and moderator temperature. To represent these effects, two K_∞ multipliers are used.

The first K_∞ multiplier accounts for the dependence on enrichment, moderator temperature, and soluble boron concentration. To generate the values for this multiplier, calculations are performed for assemblies of various enrichments with and without control rods using the Vepco PDQ07 Discrete Model. These calculations are done at zero burnup and with different values of soluble boron concentration and moderator temperature. The values of the first K_∞ multiplier are obtained by ratioing K_∞ in the rodded condition to K_∞ in the non-rodded condition.

The second K_∞ multiplier accounts for the variation in the reactivity effect of control rods with changes in assembly burnup. The values of the second K_∞ multiplier are generated by depleting assemblies of various enrichments at constant soluble boron concentration and moderator temperature using the Vepco PDQ07 Discrete Model. At various burnup points, separate recovery cases

are performed during which control rods are inserted. To calculate the values of the multiplier, the ratio of K_∞ with the control rods to K_∞ without control rods at each burnup point is divided by the ratio of K_∞ with control rods to K_∞ without control rods at zero burnup.

The third K_∞ multiplier accounts for the dependence on cumulative burnup of fuel while control rods are present. The change in isotopics accompanying depletion of an assembly containing control rods can result in significant reactivity variations while the control rods are still in the assembly. While the input data for BP include this effect in the first K_∞ multiplier for BP (Section 3.2.2.2), a separate multiplier is used to represent this effect for control rods. To generate the values for this multiplier, assemblies of various enrichments are depleted with constant moderator temperature and soluble boron concentration in the following manner:

1) an assembly containing control rods is depleted to various burnups,
2) at these burnups, the control rods are removed and the assembly is depleted further,
3) recovery cases are performed at various burnup points in the non-rodded portion of the depletion for which the control rods are reinserted.

The values of the third K_∞ multiplier for control rods are obtained in two steps. First, for a particular combination of cumulative and total burnup, the value of K_∞ with control rods is divided by the value of K_∞ without control rods. Second, these values are divided by the ratio of K_∞ with and without control rods obtained at zero cumulative and zero total burnup.

3.2.3.3 K-Infinity Multiplier for the "After Effect" of Control Rods

The method by which the fourth K_∞ multiplier for control rods is generated is the same as the method described in Section 3.2.1.4 for the fourth

K_∞ multiplier for BP. First, an assembly is depleted without control rods using the Vepco PDQ07 Discrete Model. Second, an assembly of the same enrichment containing control rods is depleted to various burnups. Then the rods are removed, and the assembly is depleted further. The values of the fourth K_∞ multiplier for control rods are obtained by dividing K_∞ from the assembly which has had control rods by K_∞ of the assembly which has not had control rods present.

3.3 Thermal-Hydraulic Feedback Parameters

Thermal-hydraulic feedback effects are represented in the Vepco FLAME Model in order to more accurately calculate the power distribution. The thermal-hydraulic feedback model incorporated in the FLAME3 code is the same as the feedback model used in the PDQ07 code portion of the Vepco PDQ07 Discrete Model.[7] The input required consists of:

1) Coolant inlet enthalpy,
2) Heated perimeter per unit area of flow,
3) Flow area of the fuel assembly per total cross-sectional area of the assembly,
4) System pressure,
5) Difference between average fuel temperature and moderator temperature as a function of relative power density.

The feedback calculation is performed in the following manner. First, an initial estimate of the fuel and moderator temperatures is made for each coolant channel. Based on this estimate and the function tables supplied, the FLAME3 code calculates the K-infinity and the migration area for each node. These parameters are then used to calculate the power density in each node. This power distribution is used to compute new fuel and moderator temperatures for each channel. In turn, the new fuel and moderator temperatures are used to

calculate new K-infinities and migration areas for another power distribution calculation. This process is continued until the power density for each node in the Nth iteration differs from the power density in the N-1th iteration by less than the specified convergence criterion.

3.4 Xenon Concentration Calculation

The Vepco FLAME Model represents the xenon reactivity effect as a functional dependence of the basic K_∞ on xenon concentration (refer to Section 3.2.1.1). The xenon and iodine concentrations at each node are calculated using the parameters described in Section 3.2.1.3 and the assumption of constant flux and power throughout the duration of a time/burnup step. With the assumption of constant power and flux, the differential equations which describe the dependence of xenon and iodine concentration are solved by integration over the time step interval, Δt.

The nodal iodine concentration, $I_\ell(t)$ (atoms/barn-cm), is calculated with Equation 3.1:

$$I_\ell(t) = \frac{YI_\ell \cdot G1}{\lambda_I} - \left[\frac{YI_\ell \cdot G1}{\lambda_I} - I_\ell(t-\Delta t)\right] \cdot \exp(-\lambda_I \Delta t) \qquad (3.1)$$

where

$$G1 = \left(\frac{S_\ell \cdot P_{core} \cdot 10^{-24}}{A_\ell \cdot V_{core}}\right)\Bigg|_{t=t-\Delta t}$$

A_ℓ, YI_ℓ are given in Section 3.2.1.3,

P_{core} = core power (watts),

V_{core} = core volume (cm^3),

S_ℓ = relative power density at node ℓ,

λ_I = decay constant of Iodine-135.

The nodal xenon concentration, $X_\ell(t)$ (atoms/barn-cm), is calculated with Equation 3.2:

$$X_\ell(t) = \frac{(YI_\ell+YX_\ell)\cdot G1}{G2} - \frac{\lambda_I}{G2-\lambda_I}\cdot\left[\frac{YI_\ell\cdot G1}{\lambda_I} - I_\ell(t-\Delta t)\right]\cdot \exp(-\lambda_I\Delta t)$$
$$- \left\{\frac{(YI_\ell+YX_\ell)\cdot G1}{G2} - X_\ell(t-\Delta t)\right.$$
$$\left. - \frac{\lambda_I}{G2-\lambda_I}\cdot\left[\frac{YI_\ell\cdot G1}{\lambda_I} - I_\ell(t-\Delta t)\right]\right\}\cdot \exp(-G2\cdot\Delta t) \qquad (3.2)$$

where $G2 = \lambda_x + \sigma_{a\ell}^{Xe}(t-\Delta t)\cdot G1$,

$\sigma_{a\ell}^{Xe} = \sigma_{a2\ell}^{Xe}\cdot SIGM_\ell$ = thermal microscopic absorption cross section of Xenon-135 (barns),

λ_x = decay constant of Xenon-135,

YX_ℓ, $\sigma_{a2\ell}^{Xe}$, $SIGM_\ell$ are given in Section 3.2.1.3, and other parameters are as previously defined for Equation 3.1.

SECTION 4 - CALCULATIONAL TECHNIQUES

4.1 Power Distribution Normalization

The FLAME3 code provides two means by which the predicted power distribution may be normalized to the results of either measurements or another calculational model (e.g., Vepco PDQ07 Discrete Model). The two normalization factors which may be used are 1) leakage parameters (radial and/or axial albedoes) and 2) migration area multipliers.

For FLAME3 calculations, all the nodes modeled are inside the core. The presence of the reflector is represented by leakage parameters (albedoes) which may be input for peripheral nodes.

The migration area multiplier is intended to correct a limitation in the FLARE neutron source option. This limitation arises from the inability of the FLARE option to account for spectral effects in assemblies containing burnable poison or control rods. The result of this limitation is that the power may be somewhat underpredicted in these assemblies. The migration area multiplier may also be used to "fine-tune" the radial power distribution of interior located assemblies that do not necessarily contain burnable poison.

4.1.1 Radial Power Distribution

The radial power distribution calculated with the Vepco FLAME Model is normalized to the radial power distribution calculated with the Vepco PDQ07 Discrete Model. This normalization is generally accomplished by using a combination of radial albedoes and migration area multipliers. Radial albedoes are chosen so that the desired in-out power sharing and peripheral powers are given by the FLAME3 code. The migration area multipliers are used to "fine tune" the radial power distribution at interior assembly locations.

The radial power distribution is normalized for two sets of conditions. The first set of conditions consists of cycle depletion calculations where normalization is performed for the all rods out (ARO), hot full power (HFP) condition. The design objective is to maintain less than a 5% difference between the assemblywise radial powers computed by the Vepco FLAME and PDQ07 Discrete Models throughout each cycle. The reason for this 5% limit is to insure that acceptable burnup distributions will be calculated during depletion. For "short" (e.g., annual) cycles in which relatively little fresh BP is used, the normalization factors for beginning of cycle (BOC) generally yield acceptable radial power distributions throughout the cycle. However for "long" cycles (i.e., 18 month cycles) which contain large numbers of fresh fuel assemblies and BP rods, it is necessary to renormalize the in-out power sharing (i.e., change radial albedoes) at approximately the middle of cycle (MOC) to maintain acceptable agreement in radial power distribution.

The second set of conditions at which the radial power distribution is normalized is BOC, HZP with control banks C and D inserted in the core. The primary purpose for this normalization is to find the proper M^2 multiplier for control rodded assemblies. For cases at less than full power and with control rods inserted, the radial albedoes for HFP, ARO provide acceptable peripheral powers so that changing the peripheral albedoes at HZP is usually not required. The acceptance criterion of 5% difference between the Vepco FLAME and PDQ07 Discrete Models is relaxed for rodded cases in which no depletion is required (e.g., HZP and HFP rod worths), because variations in radial power distribution have been demonstrated to have little affect on the axial information calculated with the FLAME model.[11]

4.1.2 Axial Power Distribution

Normalization of the axial power distribution calculated by the FLAME model is required only for zero or part power cases of annual cycles containing little fresh BP. In these cases, a generic value of the axial albedo gives close agreement between measured and calculated axial power distributions for HZP. For part power situations, the axial albedo is taken to be a percentage of the HZP value.

4.2 Differential and Integral Control Rod Worths as a Function of Bank Position

Differential and integral control rod worths are calculated with the Vepco FLAME Model in two steps. First, a series of cases is run with the FLAME3 code in which all reactor parameters are held constant except for the position of the rod bank(s) whose worth is to be determined. For example, if the differential and integral worths of control bank D are required, the following set of cases is analyzed:

1) all rods out

2) D bank inserted in the top node of the appropriate assemblies

3) D bank inserted in the top 2 nodes of the appropriate assemblies

n+1) D bank inserted in the top n nodes of the appropriate assemblies

last) D bank fully inserted into the core.

The change in core reactivity resulting from each movement of the rod bank(s) is a direct measure of the control rod bank(s) differential worth. The second step in the process of calculating bank worths is normalization to results from the Vepco PDQ07 Discrete Model. The discrete model is the production calculational model for determining total integral control rod worths. There-

fore, a normalization factor is applied to the results obtained with the FLAME3 code to assure that the total integral worth predicted by the Vepco FLAME Model is the same as that predicted by the Vepco PDQ07 Discrete Model.

Based on the methodology outlined above, the following equations are used to compute the rod worths:

$$\text{Differential Worth at Node 'i' (pcm/step)} = \frac{k^{i-1} - k^{i}}{k^{i-1} \times k^{i}} \times \frac{10^5}{SPN} \times N \qquad (4.1)$$

$$\text{Integral Worth at Node 'i' (pcm)} = \frac{k^{o} - k^{i}}{k^{o} \times k^{i}} \times 10^5 \times N \qquad (4.2)$$

where k^{o} = eigenvalue given by FLAME3 for the control bank out,

k^{i} = eigenvalue given by FLAME3 for the control bank inserted in the i^{th} node,

SPN = number of steps of control rod movement per node,

N = total integral worth from Vepco PDQ07 Discrete Model divided by total integral worth from Vepco FLAME Model.

SECTION 5 - RESULTS

5.1 Introduction

The purpose of this section is to demonstrate the predictive capability of the Vepco FLAME Model for calculations of axial power distributions, axial offsets, and differential and integral control rod worths. As explained in the previous section, the radial power distribution calculated with the Vepco FLAME Model is normalized to a 2-D (x, y) discrete power distribution calculated with the Vepco PDQ07 Discrete Model prior to initiating axially dependent calculations. Therefore, this section presents 1) typical radial power distribution comparisons between the Vepco FLAME Model (after normalization) and the PDQ07 Discrete Model, and 2) comparisons to measured data taken at the Surry Nuclear Power Stations for axial power distribution, axial offset, and differential and integral rod worths. The specific types of results compared are presented in Table 5-1.

5.2 Radial Power Distribution

As discussed in Section 4.1.1, radial power distributions calculated with the Vepco FLAME Model are normalized to 2-D (x, y) discrete model power distributions calculated at the ARO, HFP condition as a function of burnup. The power distribution comparisons given in Figures 5-1 through 5-4 for Surry 1, Cycle 1 and in Figures 5-5 through 5-8 for Surry 1, Cycle 4 are representative of the agreement obtained between the Vepco FLAME Model (after normalization) and the Vepco PDQ07 Discrete Model.

Normally, it is desirable to maintain the assemblywise agreement in radial power distribution to less than a 5% difference, but this "administrative" limit is usually relaxed if violations only occur infrequently in a small number of fuel assemblies. The reason for this limit is to ensure that the accumulated assemblywise burnup obtained during a cycle depletion will be acceptable. This limit may also be relaxed for power distributions associated

with non-depletion types of calculations (e.g., rod worths), since it has been demonstrated to have little impact based on sensitivity studies. (11)

5.3 Axial Power Distribution

Axial power distribution and axial offset comparisons between the Vepco FLAME Model and measurements are presented in this section for core conditions ranging from beginning-of-cycle (BOC) to end-of-cycle (EOC) at hot zero power (HZP) and hot full power (HFP).

In general, the Vepco Flame Model predictions attempted to simulate the actual core conditions in terms of power level, core burnup, and control rod position. However, for reasons of increased calculational efficiency, the Vepco FLAME Model does not explicitly represent the effect of spacer grids, nor can the control rod position be exactly specified (due to the limitations of the axial mesh spacing). However, the accuracy that is compromised for the increased calculational efficiency is not significant.

Representative axial power distribution comparisons are presented for Surry 1, Cycle 1, Surry 2, Cycle 3, Surry 1, Cycle 4 and Surry 2, Cycle 4. Figures 5-9 and 5-10 give the Surry 1, Cycle 1 core average axial power distribution at BOC, HZP for D-Bank partially inserted and near fully inserted, respectively. The axial power distribution of an individual assembly (B-8) containing a partially inserted control rod (same core conditions as for Figure 5-10) is provided in Figure 5-11. Surry 2, Cycle 3 axial power distribution comparisons are presented in Figures 5-12 and 5-13 at BOC, HZP conditions for ARO and D-bank inserted. Additionally, comparisons at HFP, ARO at approximately BOC, MOC, and EOC conditions are given in Figures 5-14 through 5-16. Similar comparisons for Surry 1 and 2 Cycles 4 are provided in Figures 5-17 through 5-20 and 5-21 through 5-23, respectively. Measured and predicted axial

offset comparisons are also indicated on Figures 5-9 through 5-23, as appropriate.

5.4 Differential And Integral Rod Worths

Results of the Vepco FLAME Model predictions and startup physics measurements for differential and integral control rod bank worths for the first four cycles of Surry Units No. 1 and 2 are presented in this section. It should be noted that Surry 2, Cycle 1 results are not given, since they are essentially identical to Surry 1, Cycle 1.

Figures 5-24 through 5-27 provide the Surry 1, Cycle 1 individual differential and integral bank worths for Banks D through A, respectively. Also worths for Banks D through B moving in 100 step overlap are given in Figure 5-28. For Cycles 2 and 3 of both Surry Units 1 and 2, only banks D and C control rod worths were measured during startup physics testing. These results for Surry 1, Cycle 2 and Surry 2, Cycle 3 are compared to Vepco FLAME Model predictions in Figures 5-29 through 5-32. Measurements taken for Banks D through A during the startup of Surry 1, Cycle 4 are compared with Vepco FLAME Model predictions in Figures 5-33 through 5-36 and in Figure 5-37 for the 100 step overlap mode. Similar comparisons for Surry 2, Cycle 4 are displayed in Figures 5-38 through 5-42.

TABLE 5-1
SUMMARY OF COMPARISONS

Type of Comparison	Reactor Condition At Which Comparision Is Made	Reference To Figure
Radial Power Distribution		
	Unit 1, Cycle 1, HZP, ARO, BOC	5-1
	Unit 1, Cycle 1, HFP, ARO, BOC	5-2
	Unit 1, Cycle 1, HFP, ARO, MOC	5-3
	Unit 1, Cycle 1, HFP, ARO, EOC	5-4
	Unit 1, Cycle 4, HZP, ARO, BOC	5-5
	Unit 1, Cycle 4, HFP, ARO, BOC	5-6
	Unit 1, Cycle 4, HFP, ARO, MOC	5-7
	Unit 1, Cycle 4, HFP, ARO, EOC	5-8
Axial Power Distribution		
	Unit 1, Cycle 1, HZP, D@ 204, 70 HWD/MTU	5-9
	Unit 1, Cycle 1, HZP, D@ 26, 70 MWD/MTU	5-10
	Unit 1, Cycle 1, HZP, D@ 26, 70 MWD/MTU Assembly B-8	5-11
	Unit 2, Cycle 3, HZP, D@ 202, 0 MWD/MTU	5-12
	Unit 2, Cycle 3, HZP, D in, 0 MWD/MTU	5-13
	Unit 2, Cycle 3, HFP, D@ 219, 150 MWD/MTU	5-14
	Unit 2, Cycle 3, HFP, D@ 221, 3920 MWD/MTU	5-15
	Unit 2, Cycle 3, HFP, ARO, 7740 MWD/MTU	5-16
	Unit 1, Cycle 4, HZP, D@ 220, 0 MWD/MTU	5-17
	Unit 1, Cycle 4, HFP, D@ 215, 150 MWD/MTU	5-18
	Unit 1, Cycle 4, HFP, D@ 206, 7715 MWD/MTU	5-19
	Unit 1, Cycle 4, HFP, D@ 220, 12585 MWD/MTU	5-20
	Unit 2, Cycle 4, HZP, ARO, 0 MWD/MTU	5-21
	Unit 2, Cycle 4, HFP, ARO, 204 MWD/MTU	5-22
	Unit 2, Cycle 4, HFP, D@ 210, 6986 MWD/MTU	5-23
Differential and Integral Rod Worths		
	Unit 1, Cycle 1, HZP, BOC, D-Bank	5-24
	Unit 1, Cycle 1, HZP, BOC, C-Bank	5-25
	Unit 1, Cycle 1, HZP, BOC, B-Bank	5-26
	Unit 1, Cycle 1, HZP, BOC, A-Bank	5-27
	Unit 1, Cycle 1, HZP, BOC, B through D in 100 step overlap mode	5-28
	Unit 1, Cycle 2, HZP, BOC, D-Bank	5-29
	Unit 1, Cycle 2, HZP, BOC, C-Bank	5-30
	Unit 2, Cycle 3, HZP, BOC, D-Bank	5-31
	Unit 2, Cycle 3, HZP, BOC, C-Bank	5-32
	Unit 1, Cycle 4, HZP, BOC, D-Bank	5-33
	Unit 1, Cycle 4, HZP, BOC, C-Bank	5-34
	Unit 1, Cycle 4, HZP, BOC, B-Bank	5-35
	Unit 1, Cycle 4, HZP, BOC, A-Bank	5-36

TABLE 5-1
SUMMARY OF COMPARISONS (Continued)

Type of Comparison	Reactor Condition At Which Comparison is Made	Reference To Figures
Differential and Integral Rod Worths (cont.)		
	Unit 1, Cycle 4, HZP, BOC, A through D in 100 step overlap mode	5-37
	Unit 2, Cycle 4, HZP, BOC, D-Bank	5-38
	Unit 2, Cycle 4, HZP, BOC, C-Bank	5-39
	Unit 2, Cycle 4, HZP, BOC, B-Bank	5-40
	Unit 2, Cycle 4, HZP, BOC, A-Bank	5-41
	Unit 2, Cycle 4, HZP, BOC, A through D in 100 step overlap mode	5-42

FIGURE 5-1

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 1, HZP, ARO, BOC

	08	09	10	11	12	13	14	15
H	1.158 1.1895 +2.7							
J	1.243 1.2971 +4.4	1.144 1.1708 +2.3						
K	1.134 1.1550 +1.9	1.214 1.2563 +3.5	1.098 1.1098 +1.1					
L	1.197 1.2279 +2.6	1.097 1.1031 +0.6	1.143 1.1640 +1.8	0.998 0.9896 -0.8				
M	1.084 1.0748 -0.8	1.154 1.1630 +0.8	1.019 1.0027 -1.6	0.990 0.9941 +0.4	0.774 0.7630 +1.4			
N	1.144 1.1342 -0.9	1.047 1.0208 -2.5	1.047 1.0525 +0.5	0.943 0.9230 -2.1	0.626 0.6109 -2.4			
P	0.995 0.9809 -1.4	1.103 1.0929 -0.9	0.975 0.9580 -1.7	0.625 0.6108 -2.3				
R	0.872 0.8564 -1.8	0.657 0.6445 -1.9						

PDQ07
FLAME
Δ%

FIGURE 5-2

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 1, HFP, ARO, BOC

08	09	10	11	12	13	14	15
1.1641 1.1699 +0.5							
1.2425 1.2630 +1.6	1.1519 1.1558 +0.3						
1.1417 1.1434 +0.2	1.2158 1.2323 +1.4	1.1090 1.1079 -0.1					
1.1982 1.2092 +0.9	1.1057 1.1015 -0.4	1.1485 1.1589 +0.3	1.0141 1.0067 -0.7				
1.0878 1.0751 -1.2	1.1522 1.1539 +0.1	1.0276 1.0143 -1.3	0.9981 1.0113 +1.3	0.7937 0.7988 +0.6			
1.1322 1.1226 -0.8	1.0413 1.0212 -1.9	1.0356 1.0499 +1.4	0.9393 0.9355 -0.4	0.6414 0.6442 +0.4			
0.9835 0.9767 -0.7	1.0783 1.0696 -0.8	0.9578 0.9453 -1.3	0.6270 0.6293 +0.4				
0.8605 0.8537 -0.8	0.6526 0.6553 +0.4						

PDQ07
FLAME
Δ%

FIGURE 5-3

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 1, HFP, ARO, MOC

08	09	10	11	12	13	14	15
1.1290 1.1330 +0.4							
1.2249 1.2288 +0.3	1.1273 1.1320 +0.4						
1.1248 1.1301 +0.5	1.2186 1.2238 +0.4	1.1155 1.1219 +0.6					
1.2095 1.2149 +0.4	1.1115 1.1174 +0.5	1.1901 1.1956 +0.5	1.0646 1.0710 +0.6				
1.0920 1.0961 +0.4	1.1773 1.1801 +0.2	1.0639 1.0647 +0.1	1.0772 1.0859 +0.8	0.8549 0.8639 +1.1			
1.1295 1.1271 -0.2	1.0332 1.0323 -0.1	1.0649 1.0733 +0.8	0.9825 0.9630 -2.0	0.6601 0.6571 -0.5			
0.9288 0.9309 +0.2	1.0140 0.9963 -1.7	0.8825 0.8747 -0.9	0.6152 0.6070 -0.9				
0.7427 0.7348 -1.1	0.5884 0.5865 -0.3						

PDQ07
FLAME
Δ %

FIGURE 5-4

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 1, HFP, ARO, EOC

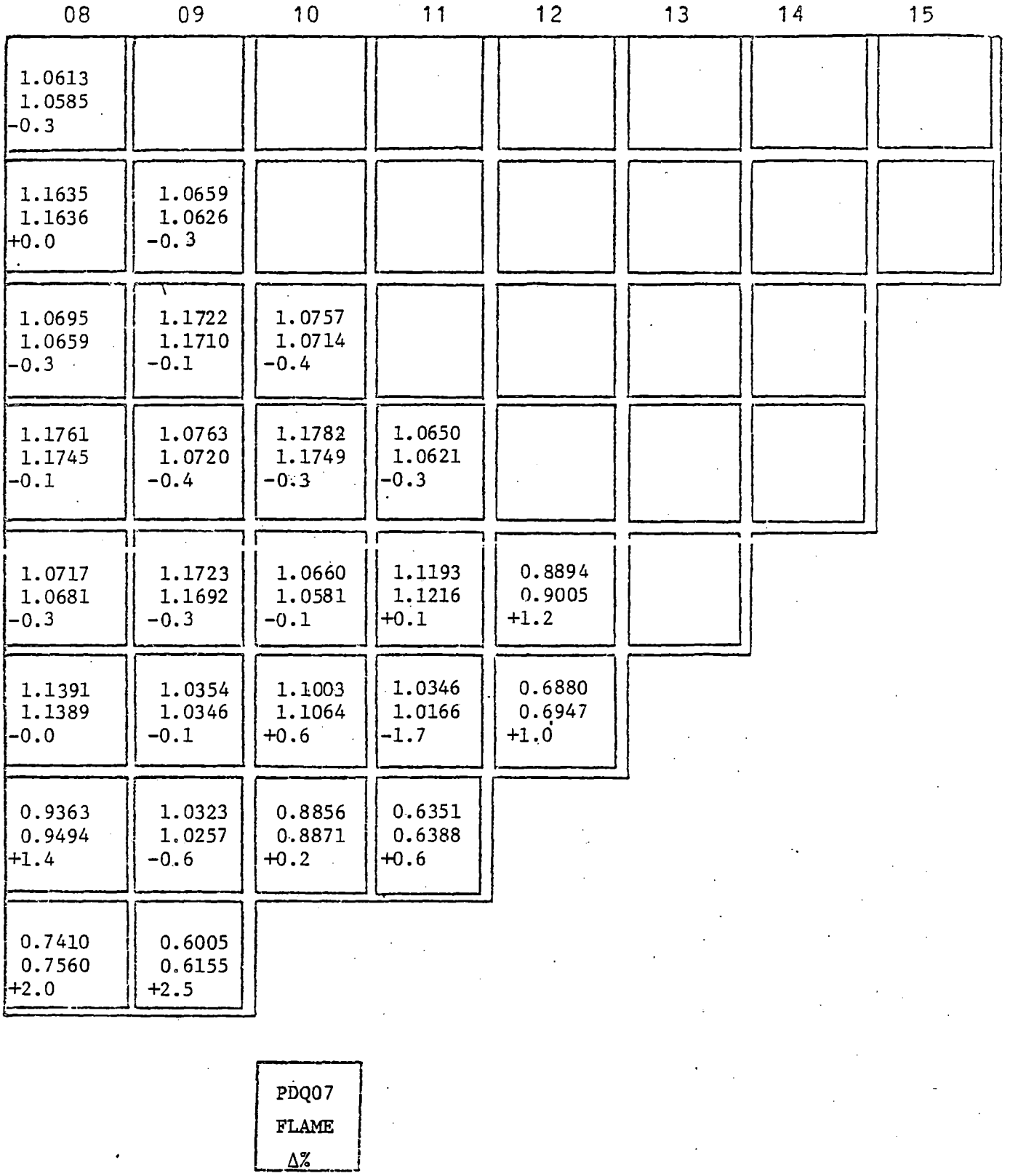


FIGURE 5-5

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 4, HZP, ARO, BOC

08	09	10	11	12	13	14	15
0.783 0.7635 -2.5							
1.039 1.0722 +3.2	1.088 1.1422 +5.0						
0.907 0.8911 -1.8	1.028 1.0454 +1.7	1.096 1.0651 -2.8					
1.036 1.0229 -1.3	1.101 1.1211 +1.8	1.083 1.0873 +0.4	1.157 1.1509 -0.5				
1.113 1.0890 -2.2	1.078 1.0606 -1.6	1.153 1.1539 +0.1	1.108 1.1145 +0.6	1.029 1.0391 +1.0			
0.954 0.9242 -3.1	1.208 1.1702 -3.1	1.078 1.0577 -1.9	1.030 1.0339 +0.4	0.689 0.6967 +1.0			
1.075 1.0302 -4.2	1.091 1.1194 +2.6	0.967 0.9837 +1.7	0.634 0.6421 +1.3				
0.810 0.8057 -0.5	0.627 0.6268 -0.0						

PDQ07
FLAME
Δ%

FIGURE 5-6

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 4, HFP, ARO, BOC

	08	09	10	11	12	13	14	15
H	0.820 0.8139 -0.7							
J	1.067 1.0962 +2.7	1.108 1.1489 +3.7						
K	0.930 0.9300 0.0	1.040 1.0616 +2.1	1.097 1.0737 -2.1					
L	1.037 1.0347 -0.2	1.093 1.1097 +1.5	1.071 1.0785 +0.7	1.132 1.1194 -1.1				
M	1.103 1.0886 -1.3	1.061 1.0478 -1.2	1.130 1.1246 -0.5	1.085 1.0837 -0.1	1.018 1.0197 +0.2			
N	0.954 0.9419 -1.3	1.190 1.1470 -3.6	1.068 1.0494 -1.7	1.027 1.0242 -0.3	0.701 0.7132 +1.7			
P	1.082 1.0394 -3.9	1.093 1.1087 +1.4	0.976 0.9839 +0.8	0.652 0.6656 +2.1				
	0.837 0.8379 +0.1	0.653 0.6604 +1.1						

PDQ07
FLAME
Δ%

FIGURE 5-7

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 4, HFP, ARO, MOC

08	09	10	11	12	13	14	15
0.937 0.9228 -1.5							
1.233 1.1850 -2.9	1.198 1.2102 +1.0						
1.057 1.0261 -2.9	1.238 1.1947 -3.5	1.175 1.1356 -3.4					
1.212 1.1649 -3.9	1.212 1.2233 +0.9	1.202 1.1796 -1.9	1.117 1.1374 +1.8				
1.085 1.0793 -0.5	1.103 1.1046 +0.1	1.106 1.1332 +2.5	1.067 1.0890 +2.1	0.938 0.9728 +3.7			
0.877 0.8779 +0.1	1.069 1.0868 +1.7	1.070 1.0641 -0.6	1.001 1.0087 +0.8	0.662 0.6742 +1.8			
0.913 0.9004 -1.4	0.979 0.9940 +1.5	0.868 0.8812 +1.5	0.613 0.6169 +0.6				
0.701 0.7012 +0.0	0.567 0.5640 -0.5						

PDQ07
FLAME
Δ%

FIGURE 5-8

RADIAL POWER DISTRIBUTION COMPARISON
FOR SURRY 1, CYCLE 4, HFP, ARO, EOC

	08	09	10	11	12	13	14	15
H	0.908 0.8966 -1.3							
J	1.179 1.1291 -4.2	1.132 1.1400 +0.7						
K	1.037 1.0136 -2.3	1.227 1.1757 -4.2	1.140 1.1028 -3.3					
L	1.231 1.1708 -4.9	1.225 1.2222 -0.2	1.214 1.1638 -4.1	1.100 1.0939 -0.6				
M	1.078 1.0678 -0.9	1.137 1.1179 -1.7	1.100 1.0981 -0.2	1.083 1.0838 +0.1	0.934 0.9584 +2.6			
N	0.878 0.8793 +0.1	1.054 1.0604 +0.6	1.096 1.0789 -1.6	1.021 1.0319 +1.1	0.669 0.6932 +3.6			
P	0.899 0.9220 +2.6	0.982 1.0189 +3.8	0.858 0.8905 +3.8	0.621 0.6490 +4.5				
R	0.696 0.7454 +7.1	0.573 0.6171 +7.7						

PDQ07
FLAME
Δ%

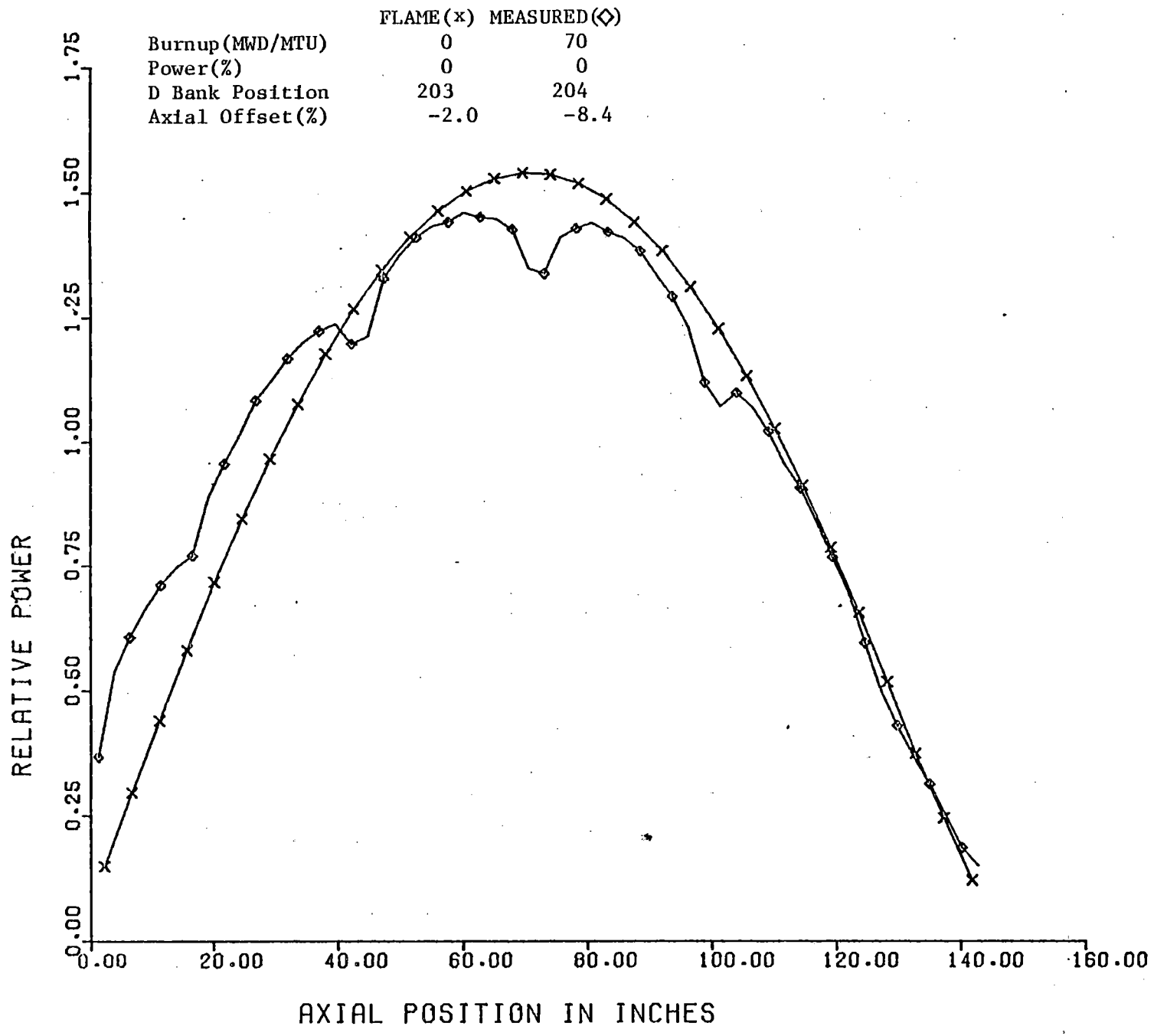


Figure 5-9

Core Average Axial Power Distribution Comparison
for Surry 1, Cycle 1, BOC,
HZP, ARO

FIGURE 5-10

Core Average Axial Power Distribution Comparison
for Surry 1, Cycle 1, BOC,
HZP, D Bank In

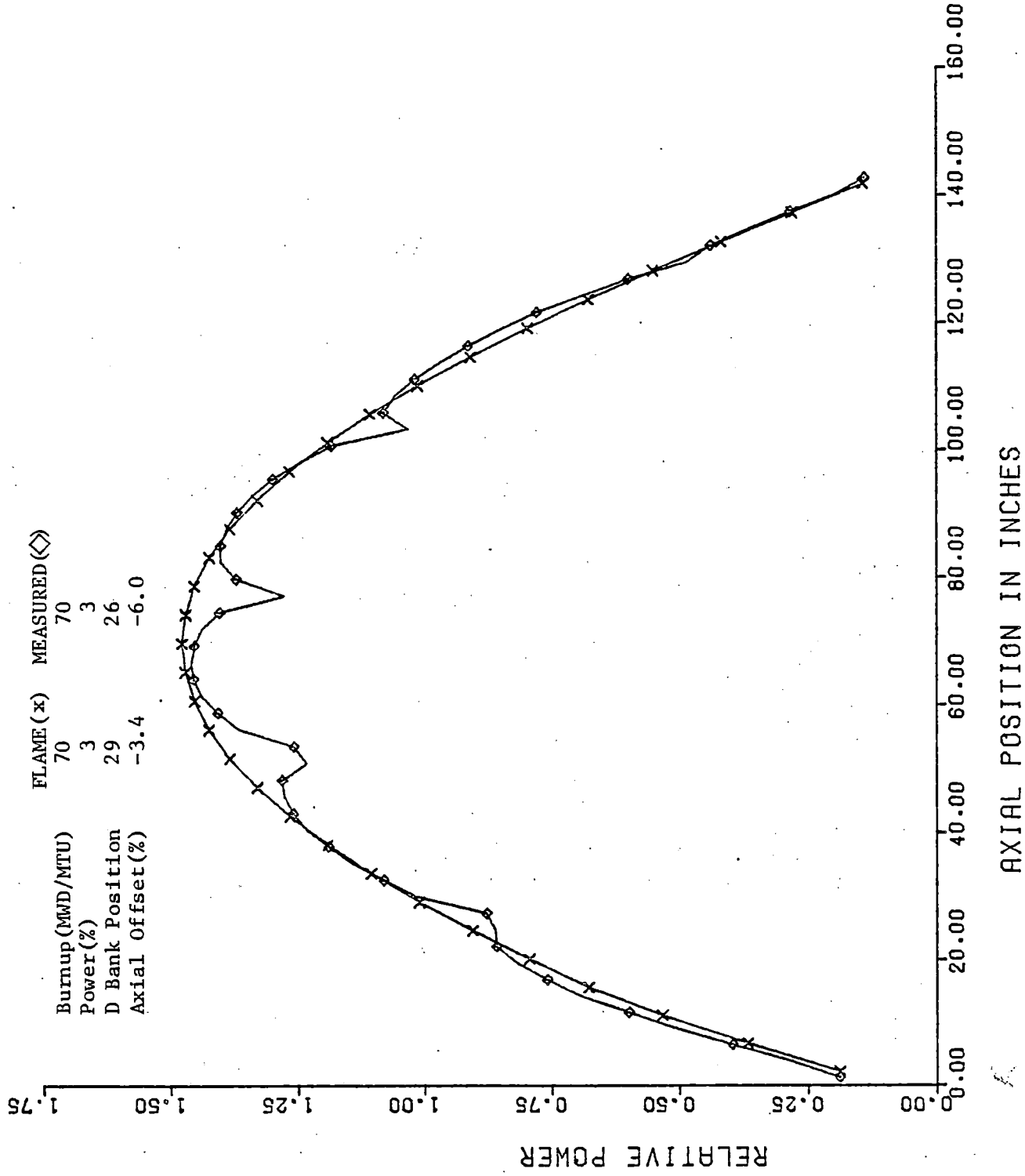
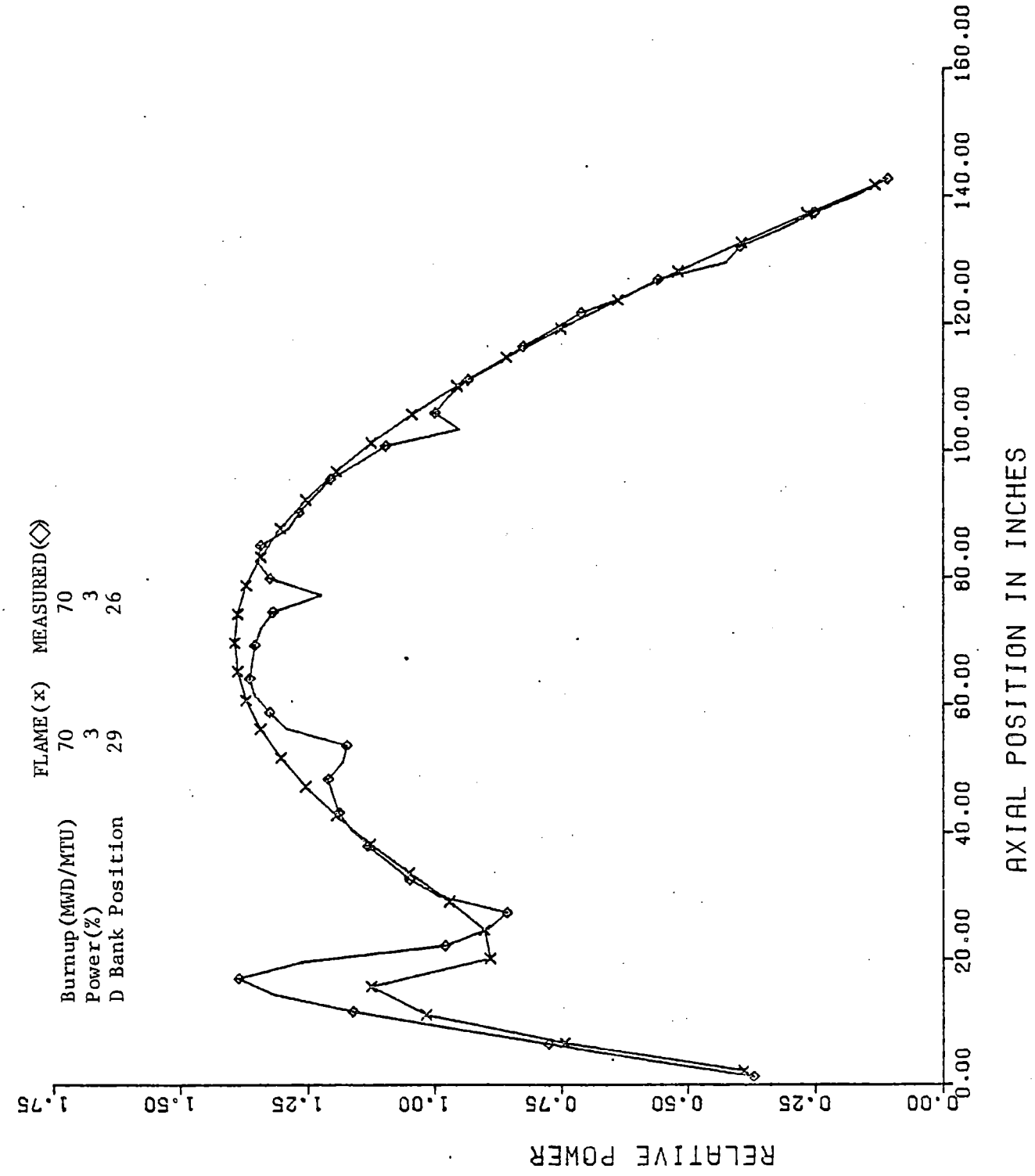
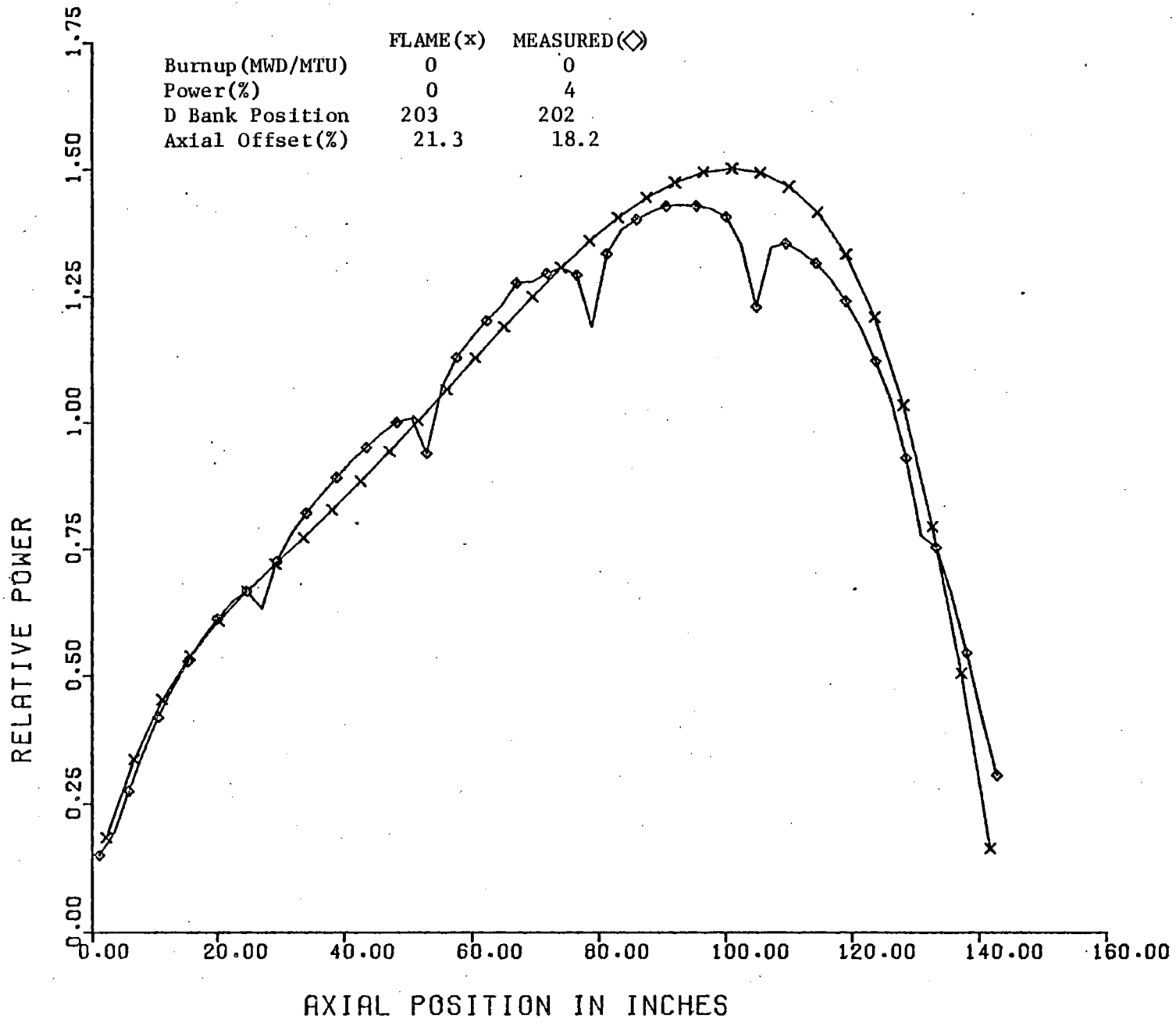


FIGURE 5-11

Assembly B-8 Axial Power Distribution Comparison
Surry 1, Cycle 1, BOC,
HZP, D Bank In





Core Average Axial Power Distribution Comparison
Surry 2, Cycle 3, BOC,
HZP, ARO

FIGURE 5-12

FIGURE 5-13

Core Average Axial Power Distribution Comparison
Surry 2, Cycle 3, BOC,
HZP, D Bank In

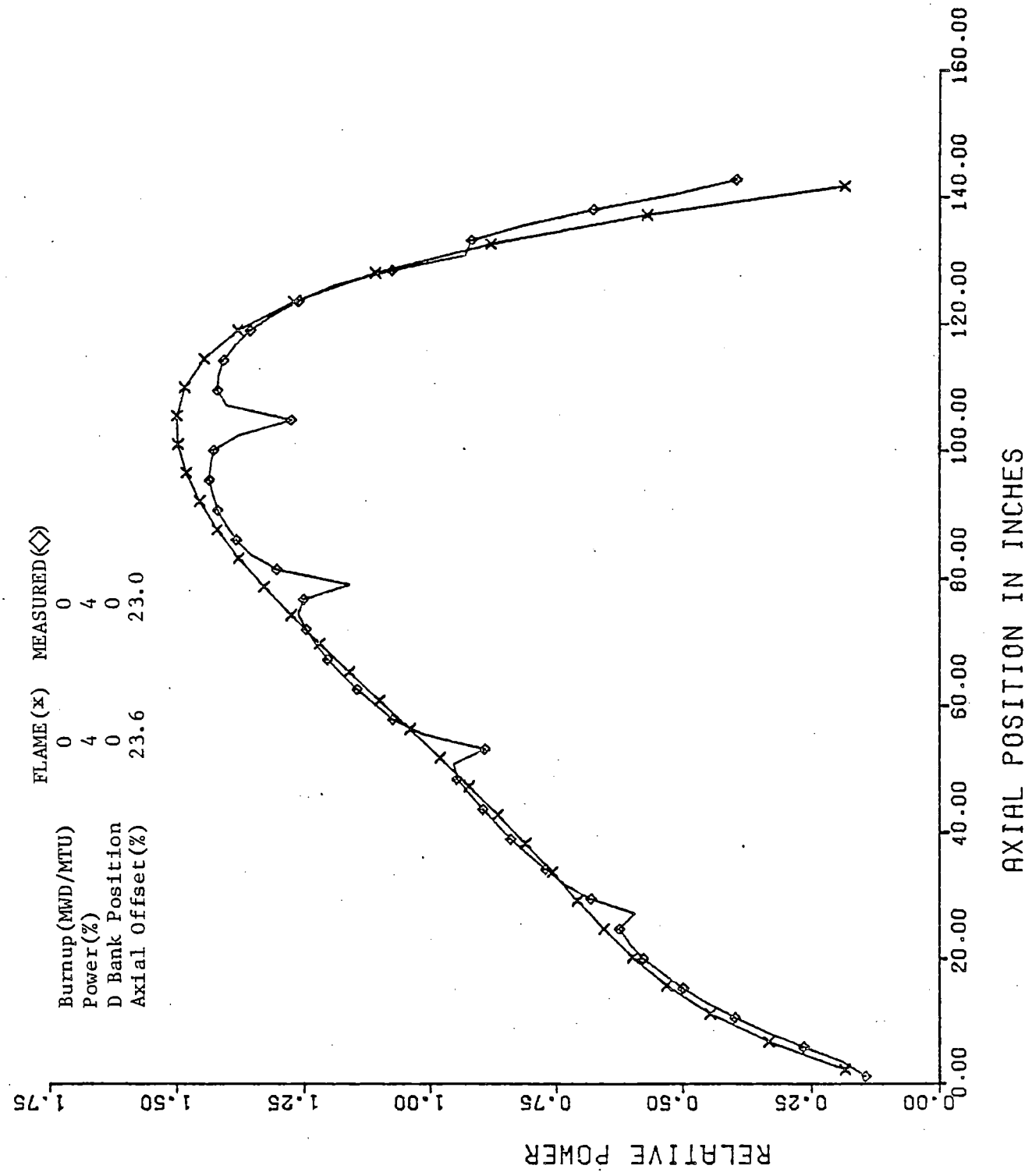


FIGURE 5-14

Core Average Axial Power Distribution Comparison
Surry 2, Cycle 3, BOC,
HFP, ARO

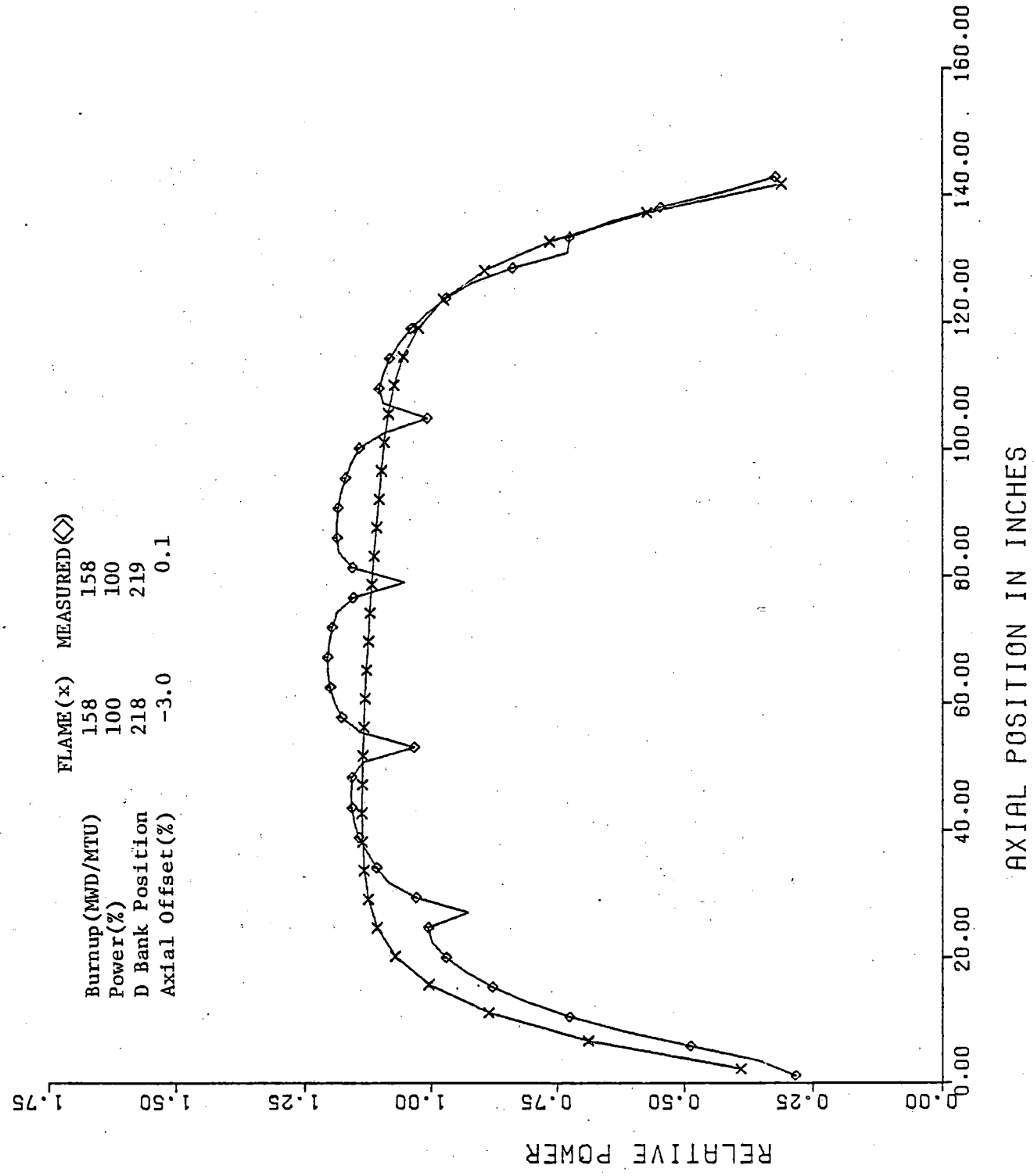


FIGURE 5-15

Core Average Axial Power Distribution Comparison
for Surry 2, Cycle 3, MOC,
HFP, ARO

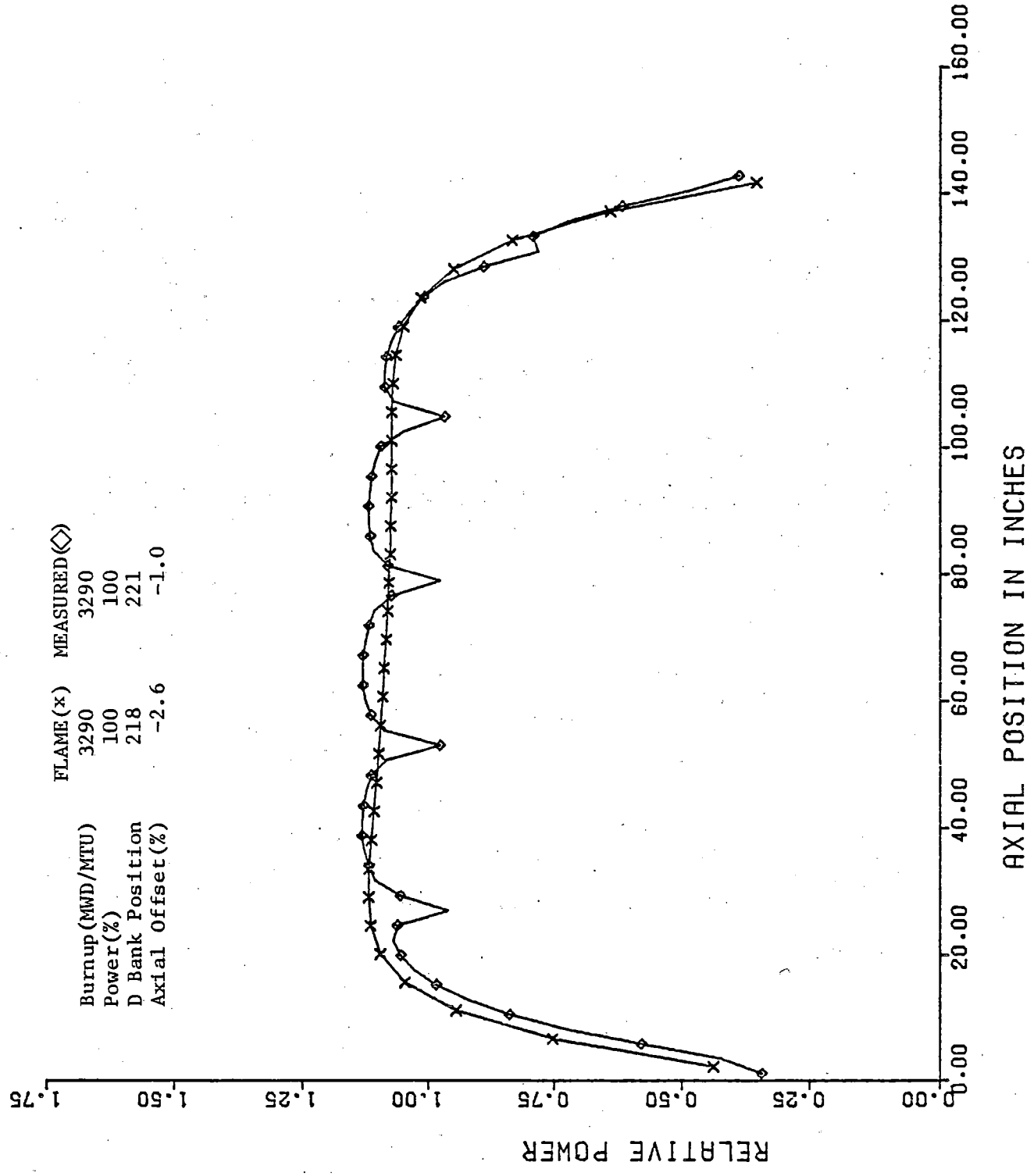


FIGURE 5-16

Core Average Axial Power Distribution Comparison
for Surry 2, Cycle 3, EOC,
HFP, ARO

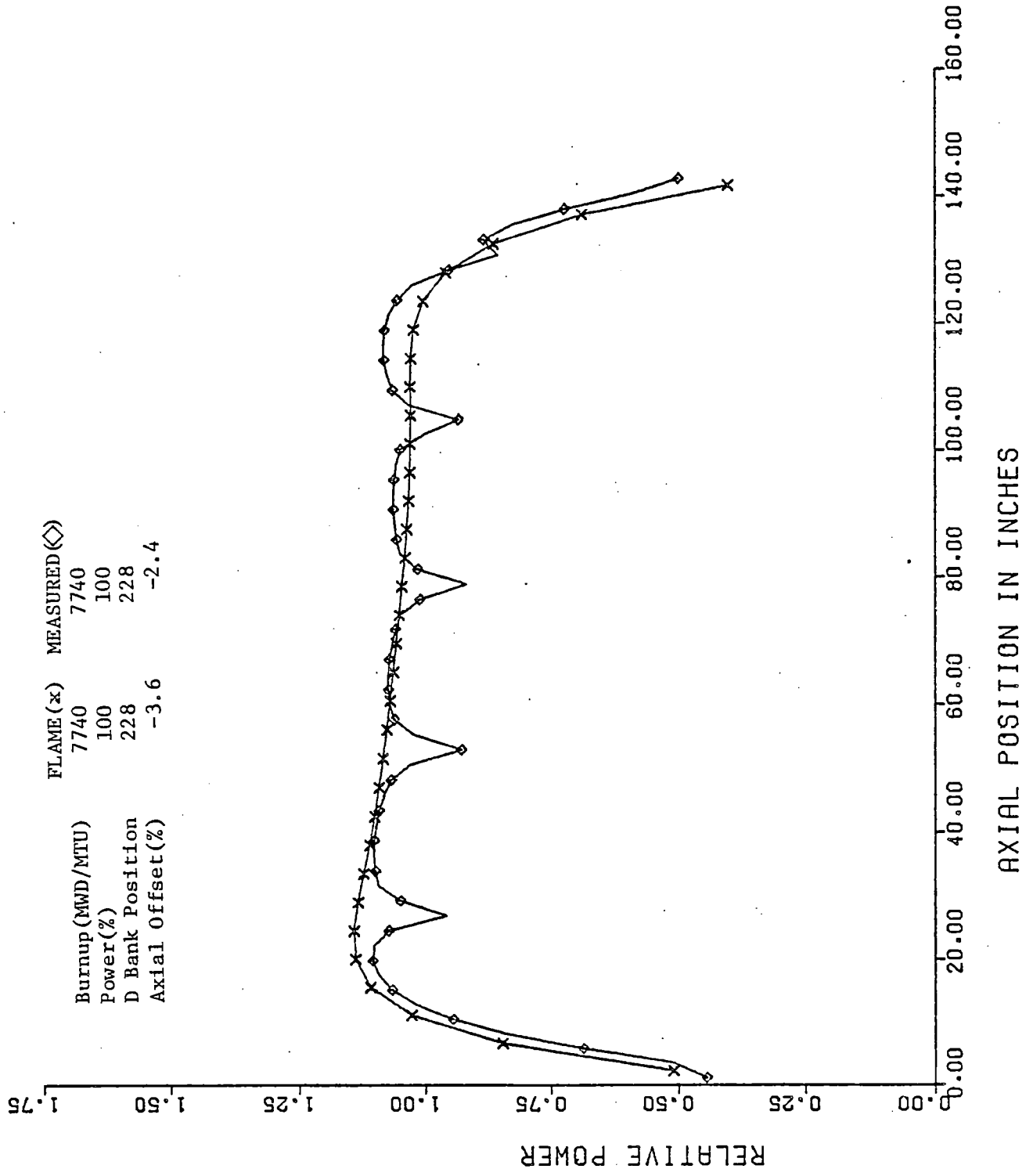


FIGURE 5-17

Core Average Axial Power Distribution Comparison
Surry 1, Cycle 4, BOC,
HZP, ARO

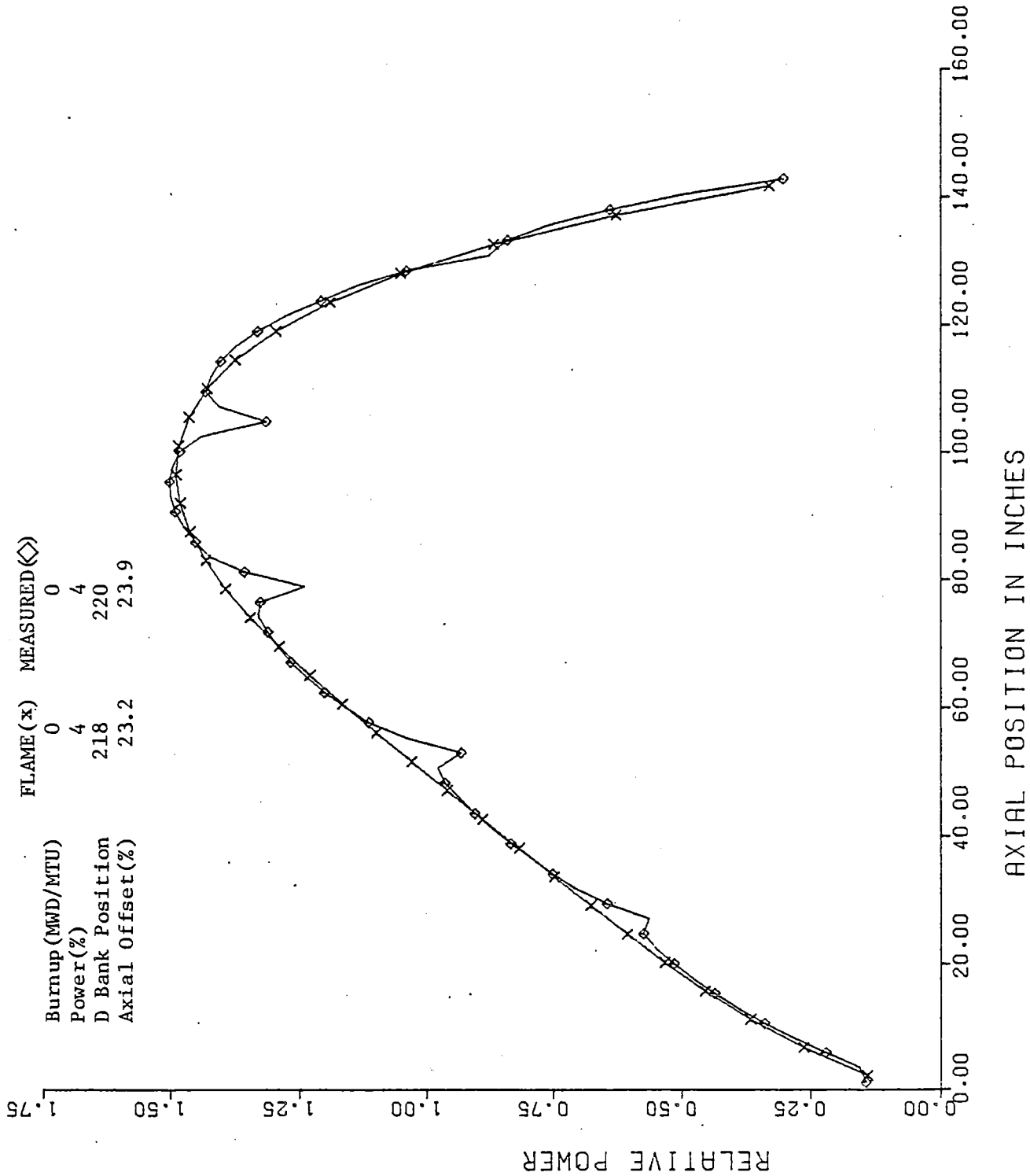
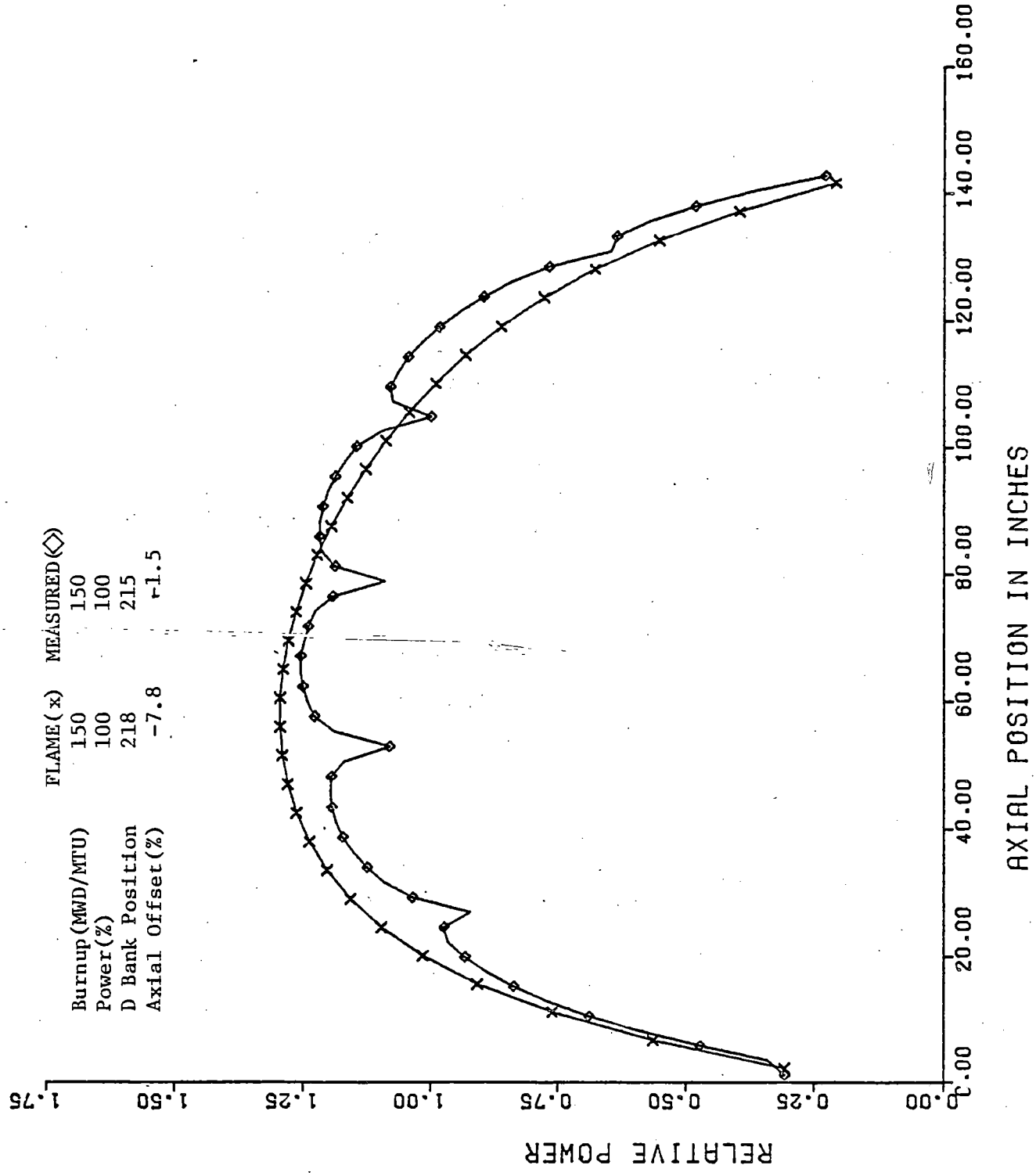


FIGURE 5-18

Core Average Axial Power Distribution Comparison
for Surry 1, Cycle 4, BOC,
HFP, ARO



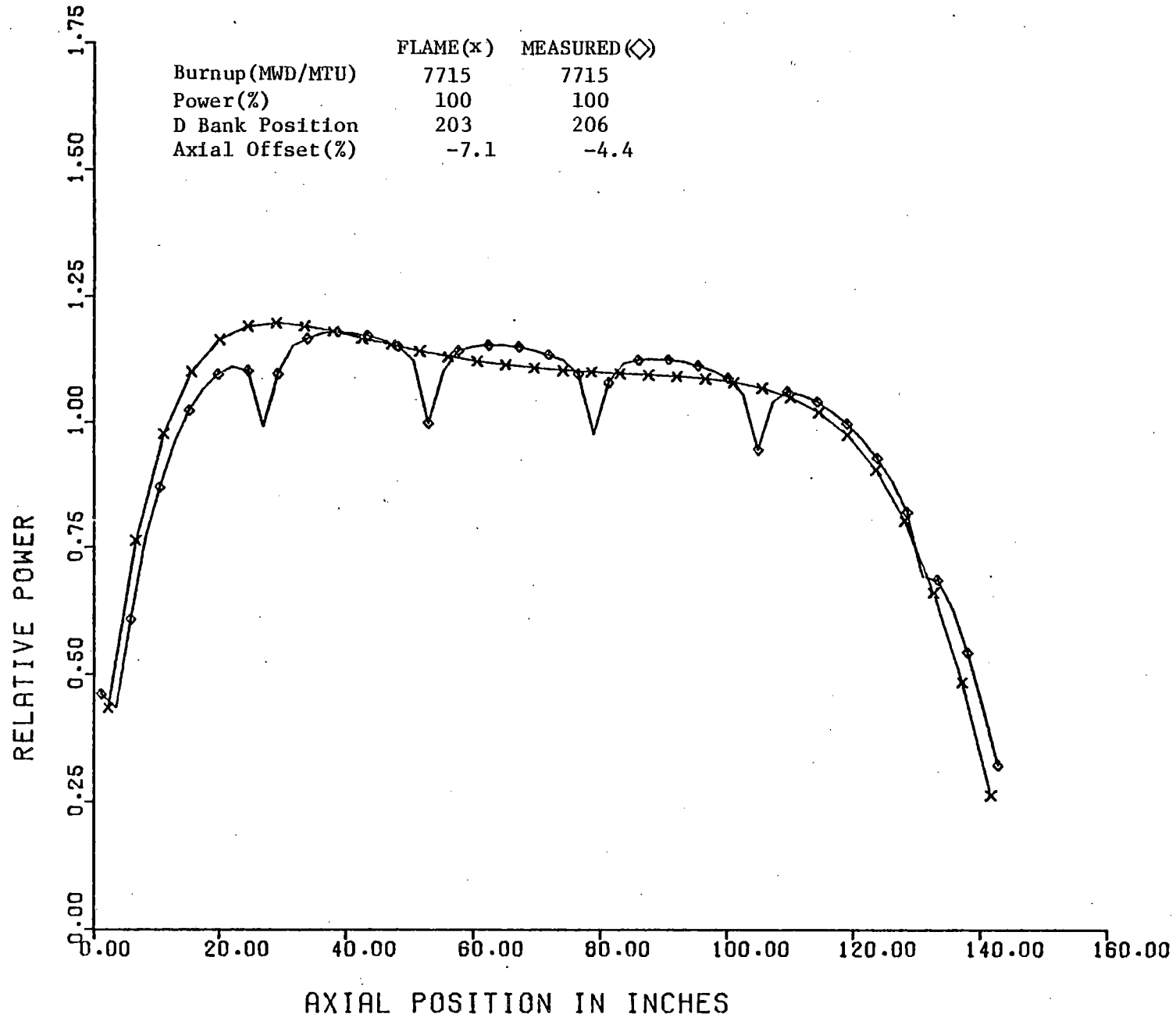
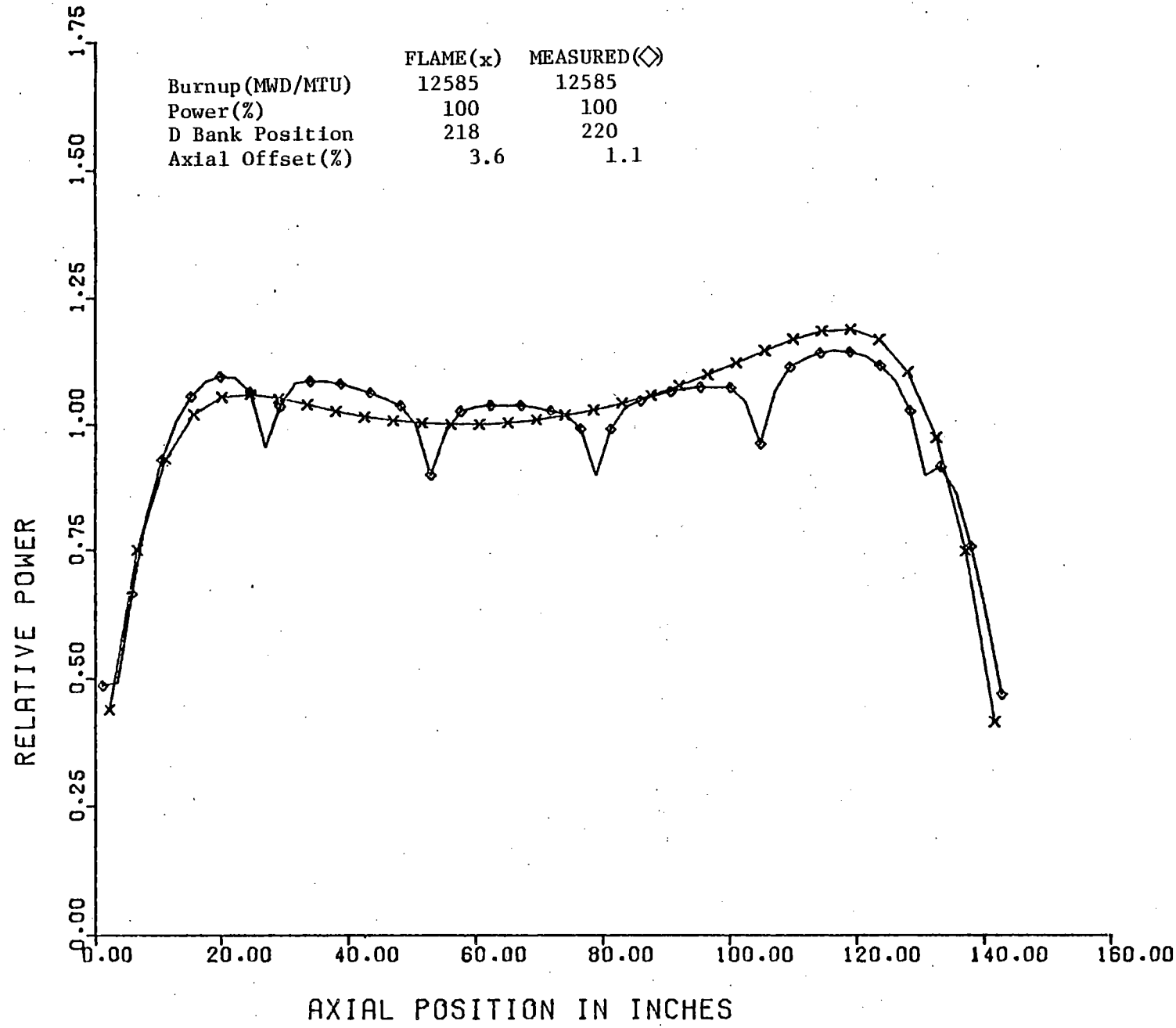


FIGURE 5-19

Core Average Axial Power Distribution Comparison
for Surry 1, Cycle 4, MOC,
HFP, ARO



Core Average Axial Power Distribution Comparison
for Surry 1, Cycle 4, EOC,
HFP, ARO

FIGURE 5-20

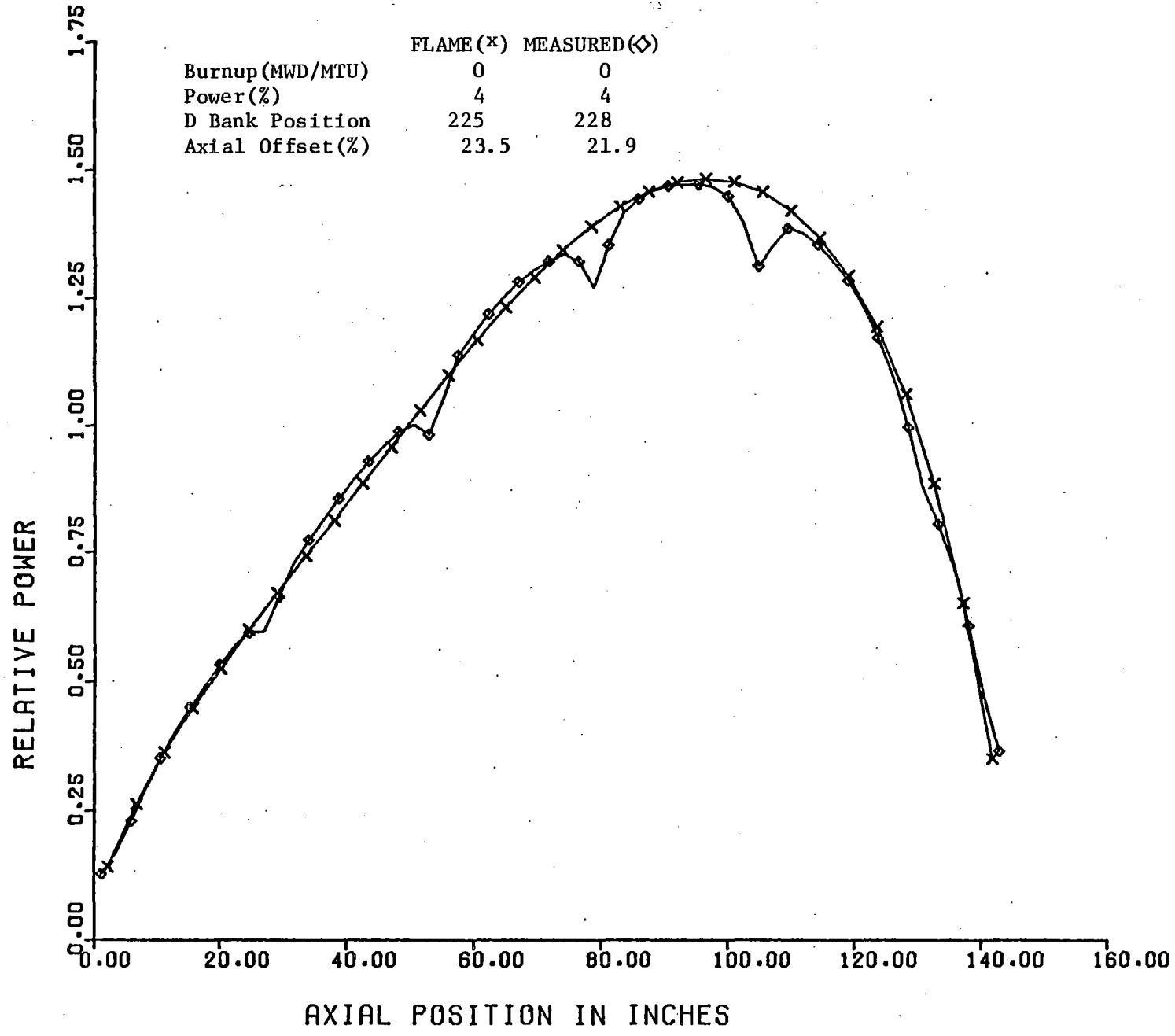


FIGURE 5-21

Core Average Axial Power Distribution Comparison
for Surry 2, Cycle 4, BOC,
HZP, ARO

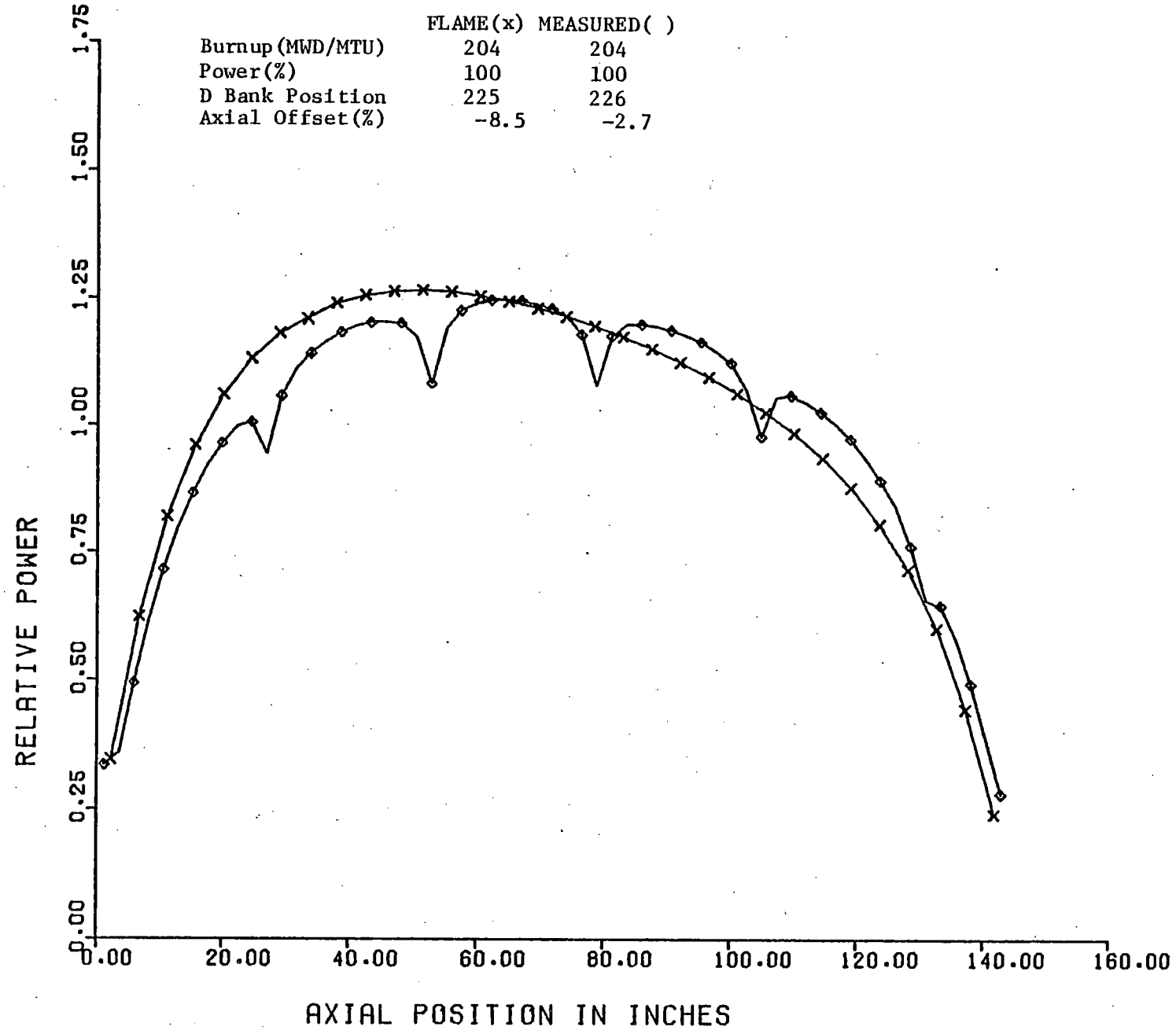


FIGURE 5-22

Core Average Axial Power Distribution Comparison
for Surry 2, Cycle 4, BOC,
HFP, ARO

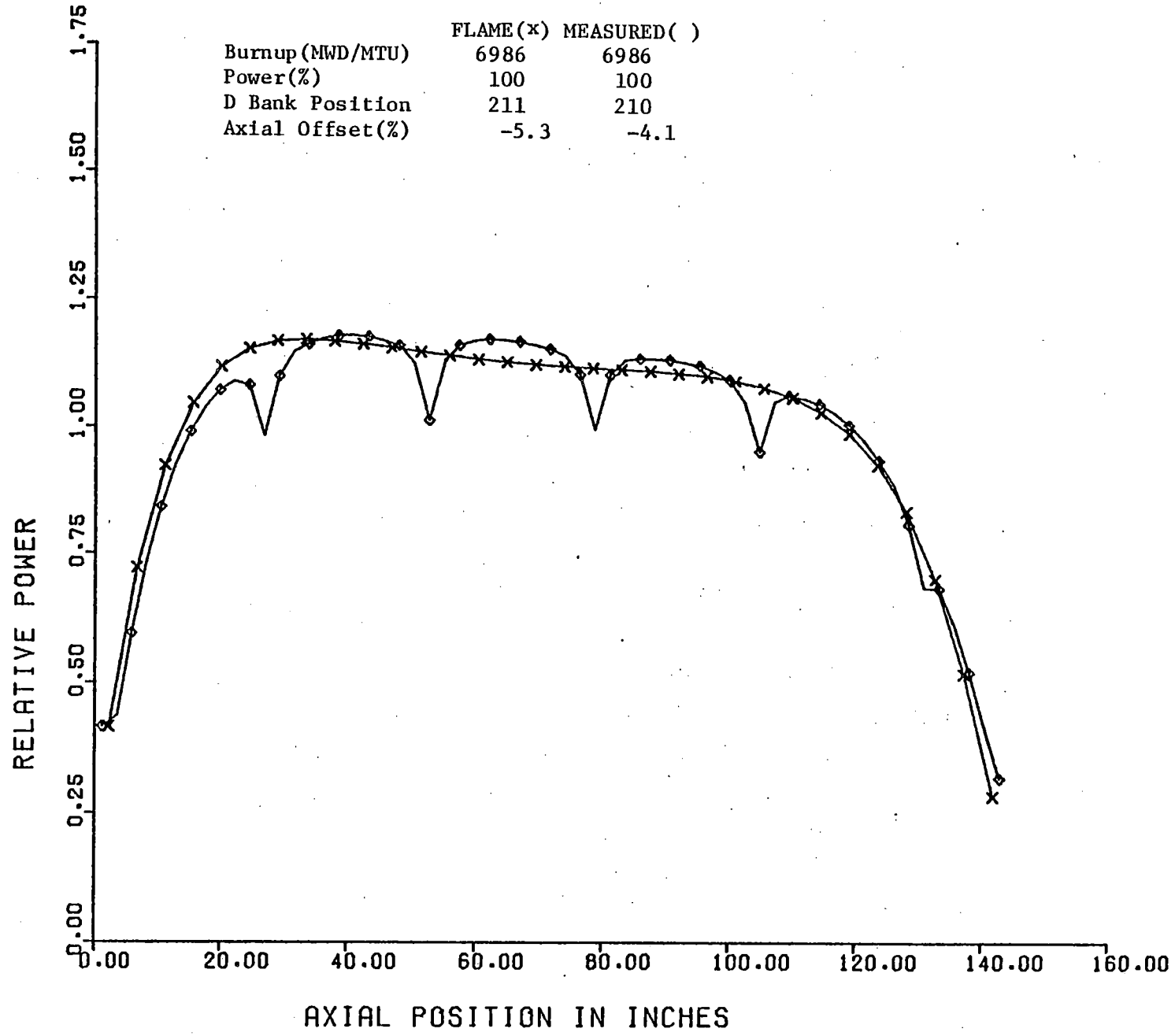


FIGURE 5-23

Core Average Axial Power Distribution Comparison
for Surry 2, Cycle 4, MOC,
HFP, ARO

FIGURE 5-24

Control Rod Worth Comparison
for Surry 1, Cycle 1, BOC,
HZP, D Bank

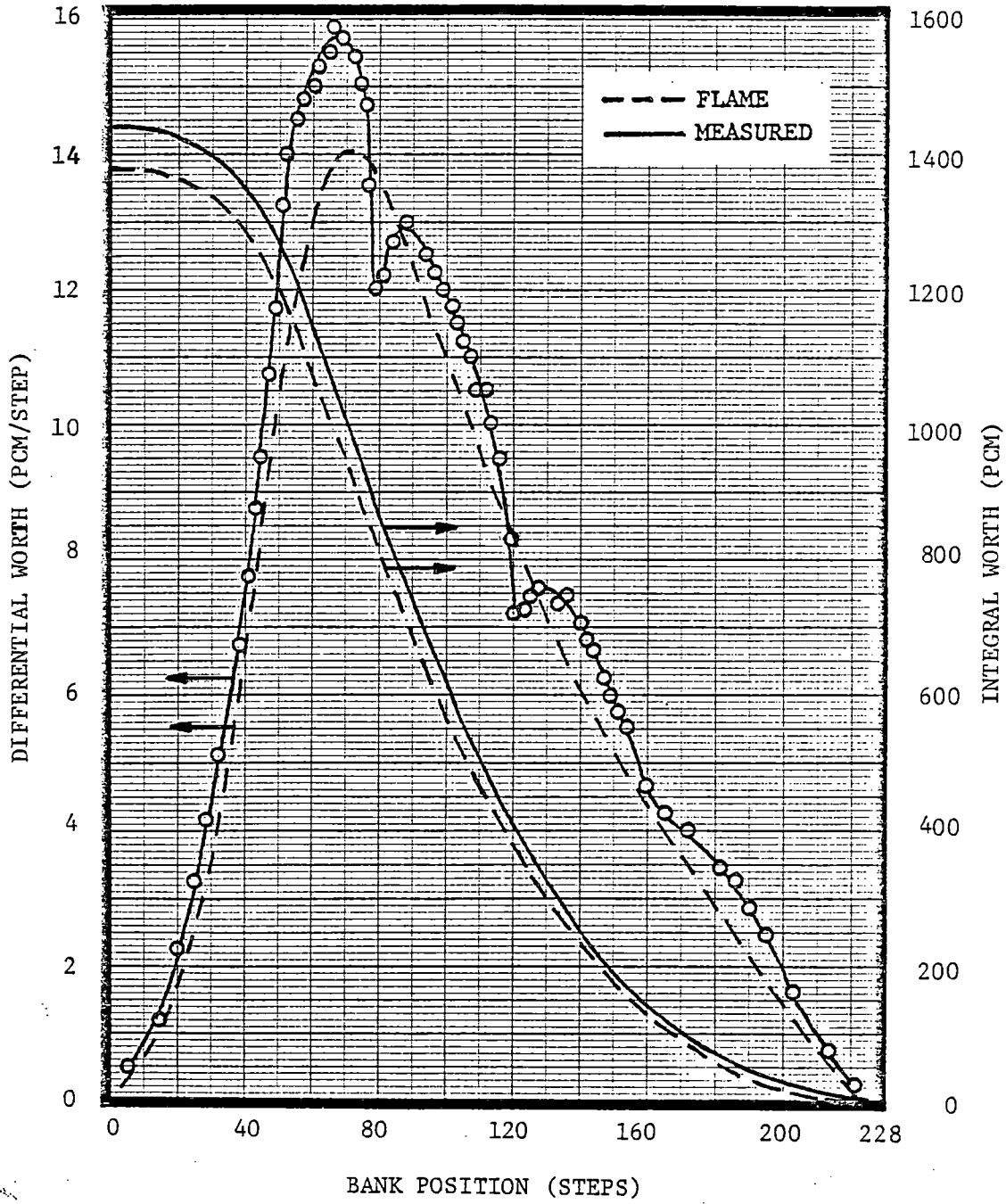


FIGURE 5-25

Control Rod Worth Comparison
for Surry 1, Cycle 1, BOC,
HZP, C Bank

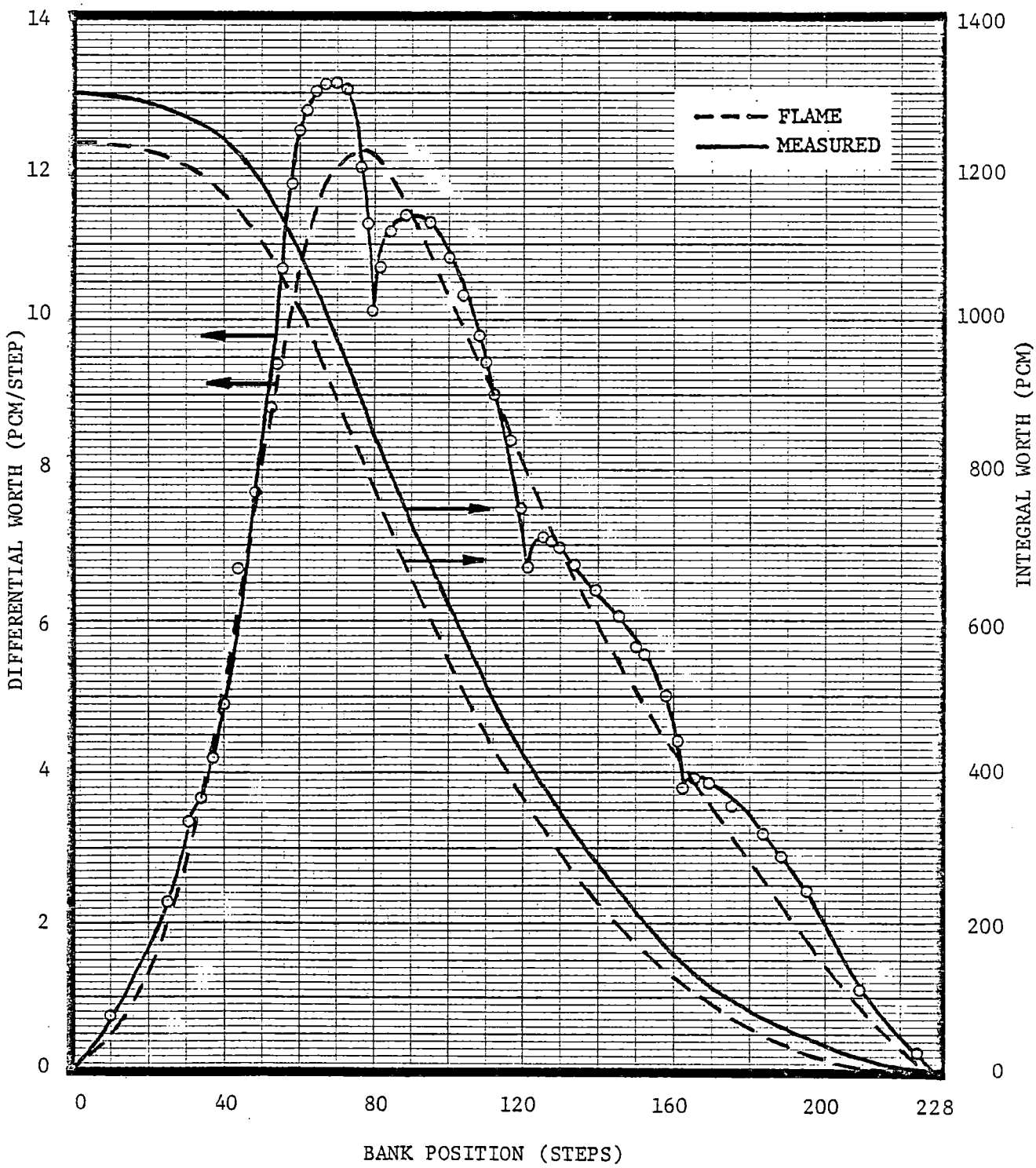


FIGURE 5-26

Control Rod Worth Comparison
for Surry 1, Cycle 1, BOC,
HZP, B Bank

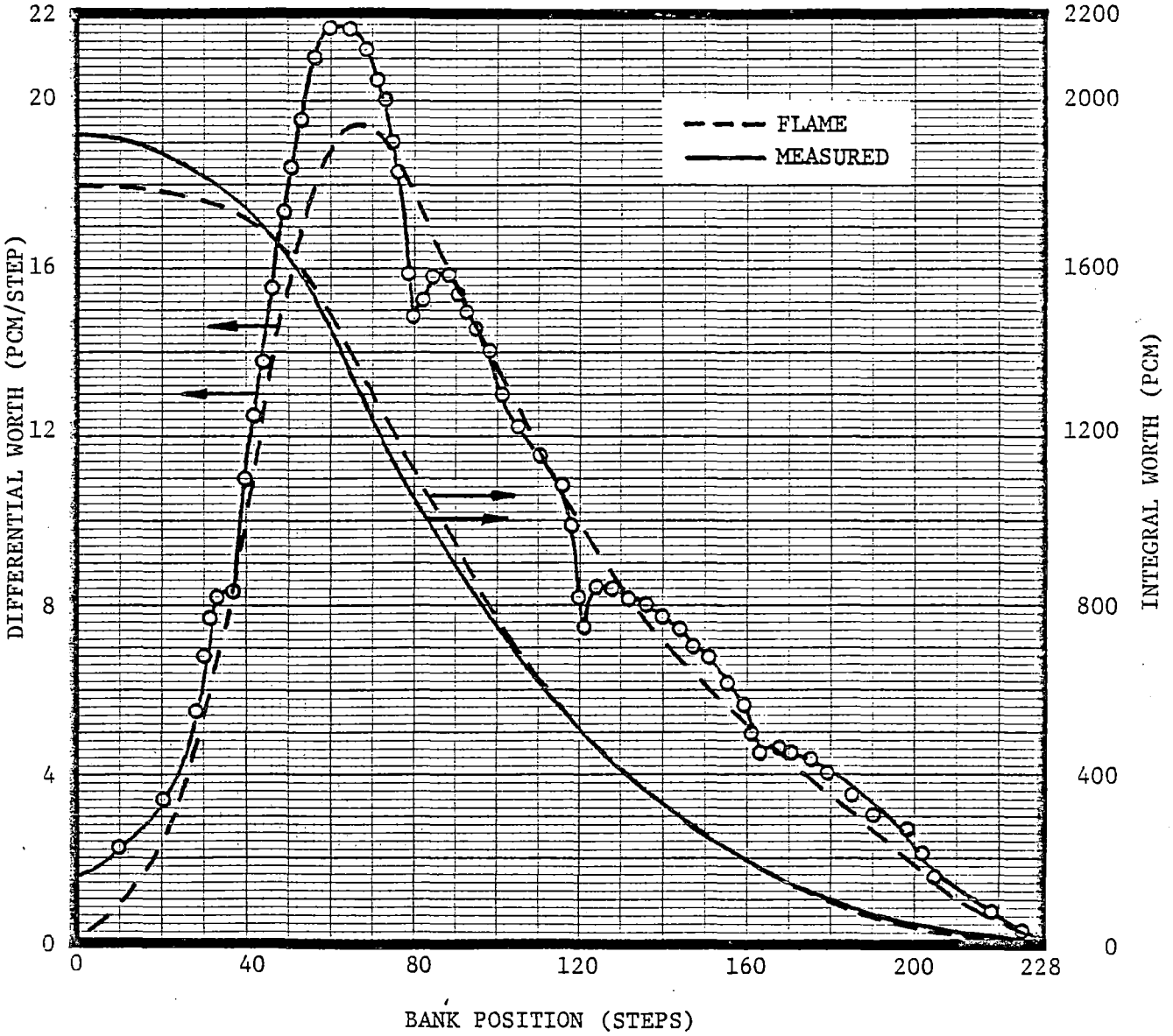


FIGURE 5-27

Control Rod Worth Comparison
for Surry 1, Cycle 1, BOC,
HZP, A Bank

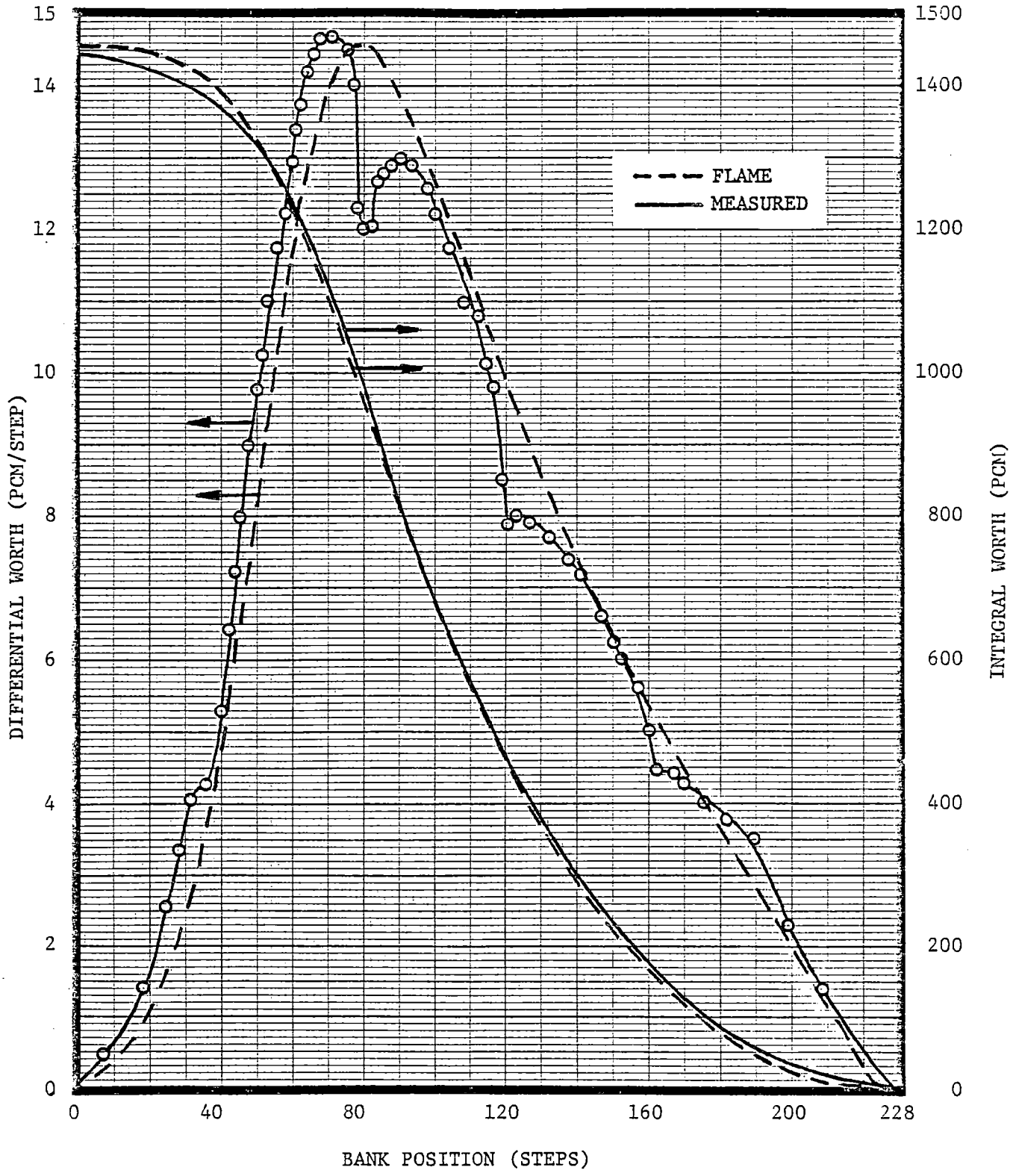


Figure 5-28
Control Rod Worth Comparison
for Surry 1, Cycle 1, BOC,
HZP, Banks B - D moving in
100 Step Overlap

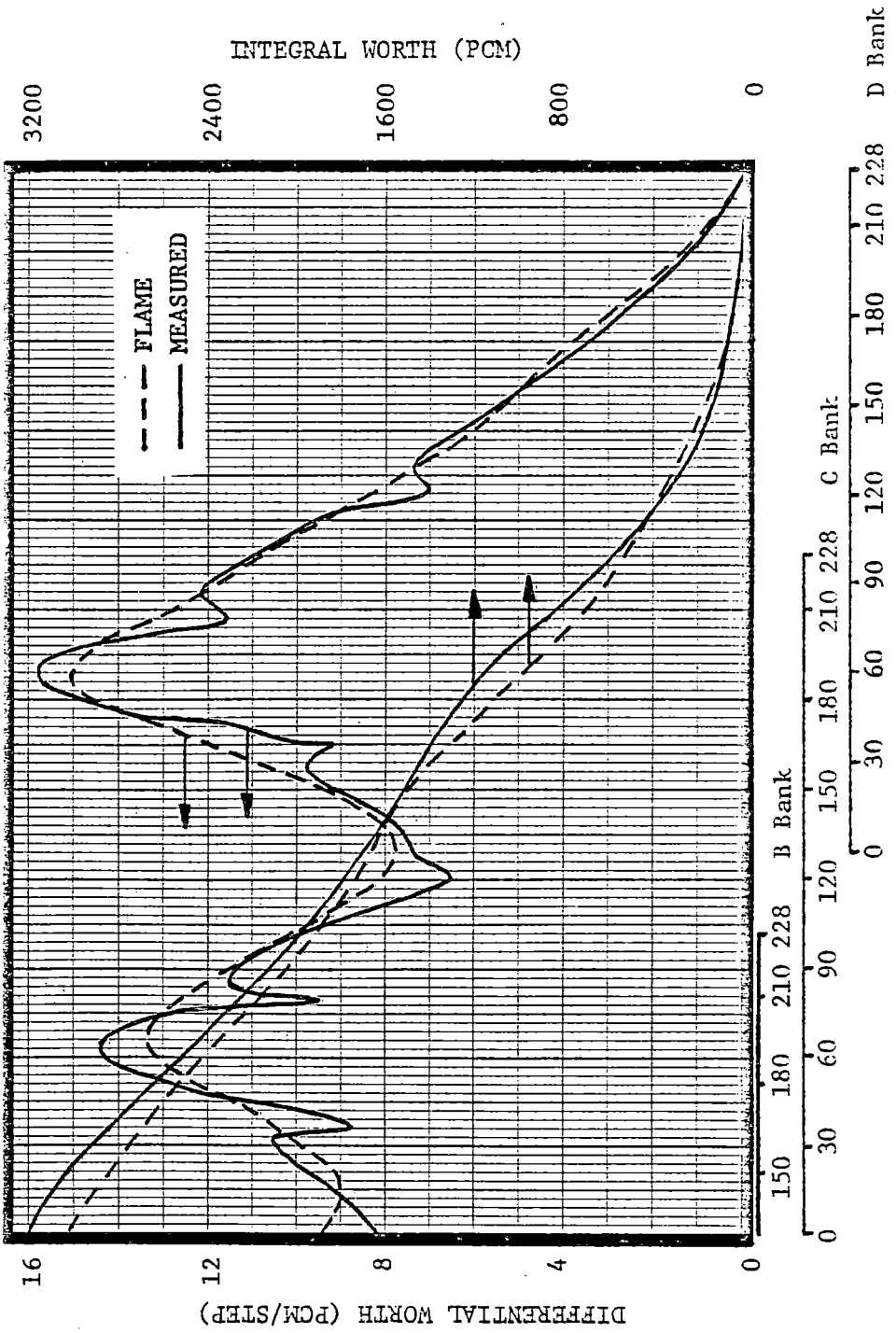


FIGURE 5-29

Control Rod Worth Comparison
for Surry 1, Cycle 2, BOC,
HZP, D Bank

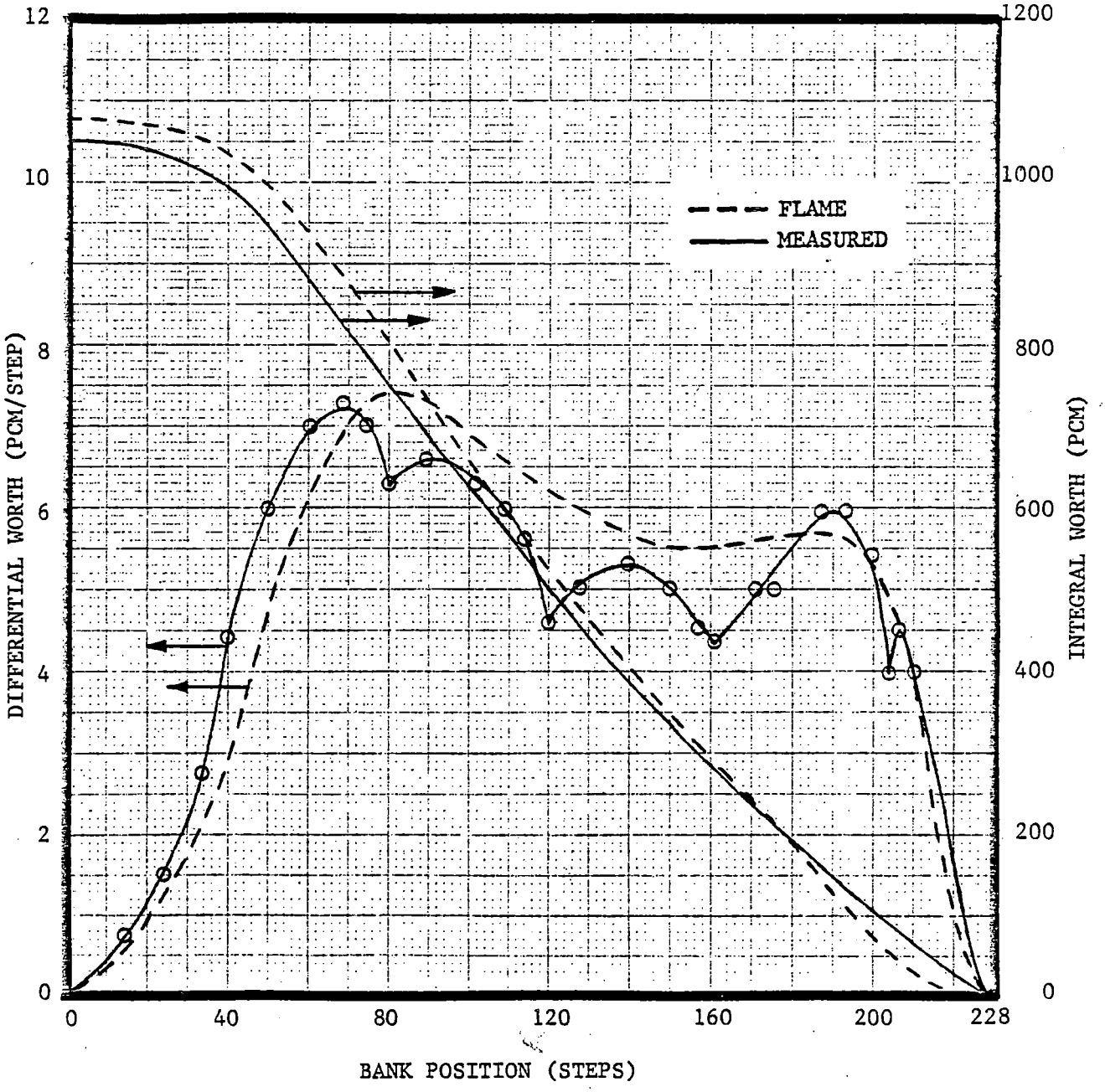


FIGURE 5-30

Control Rod Worth Comparison
for Surry 1, Cycle 2, BOC,
HZP, C Bank

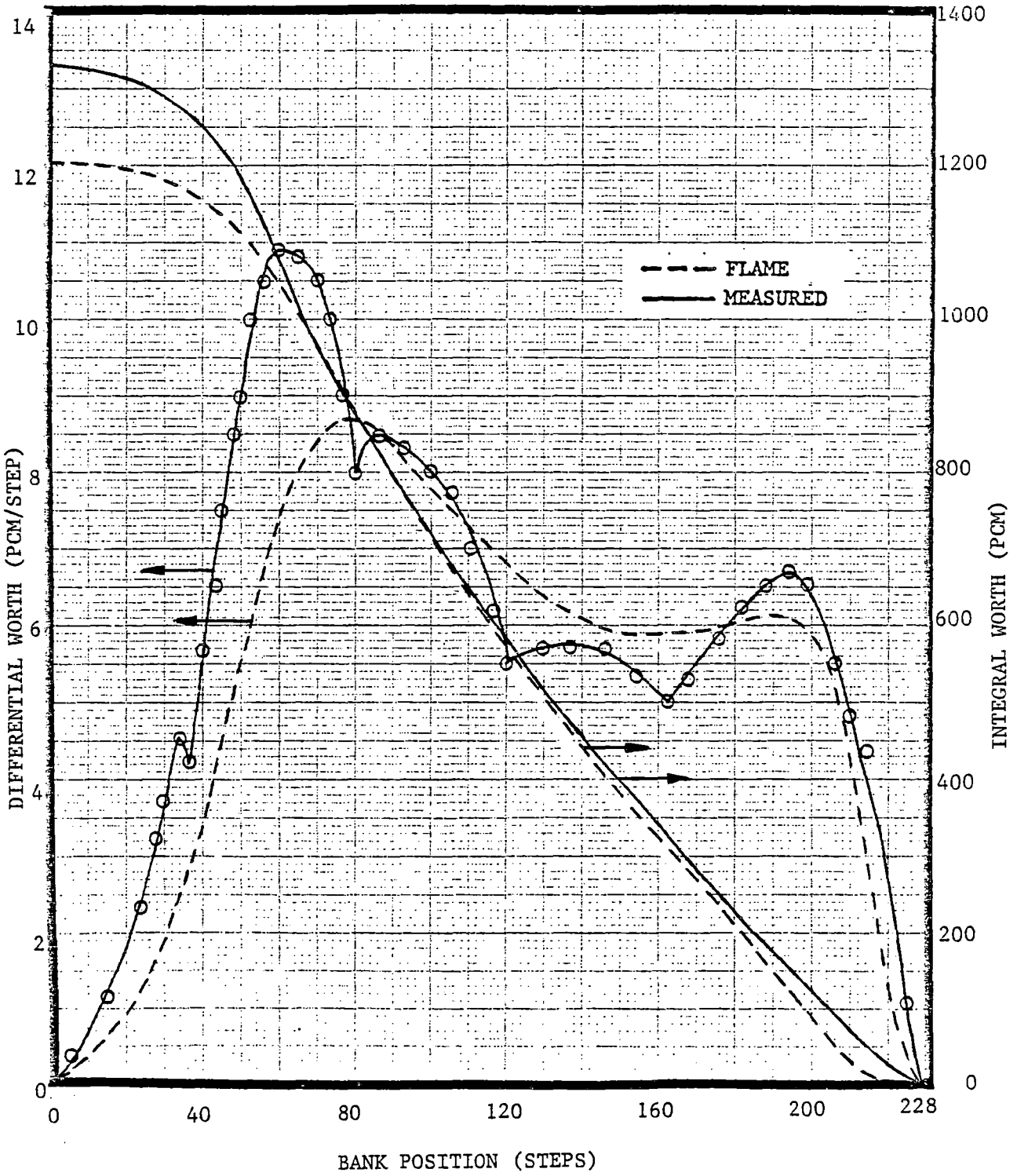


FIGURE 5-31

Control Rod Worth Comparison
for Surry 2, Cycle 3, BOC,
HZP, D Bank

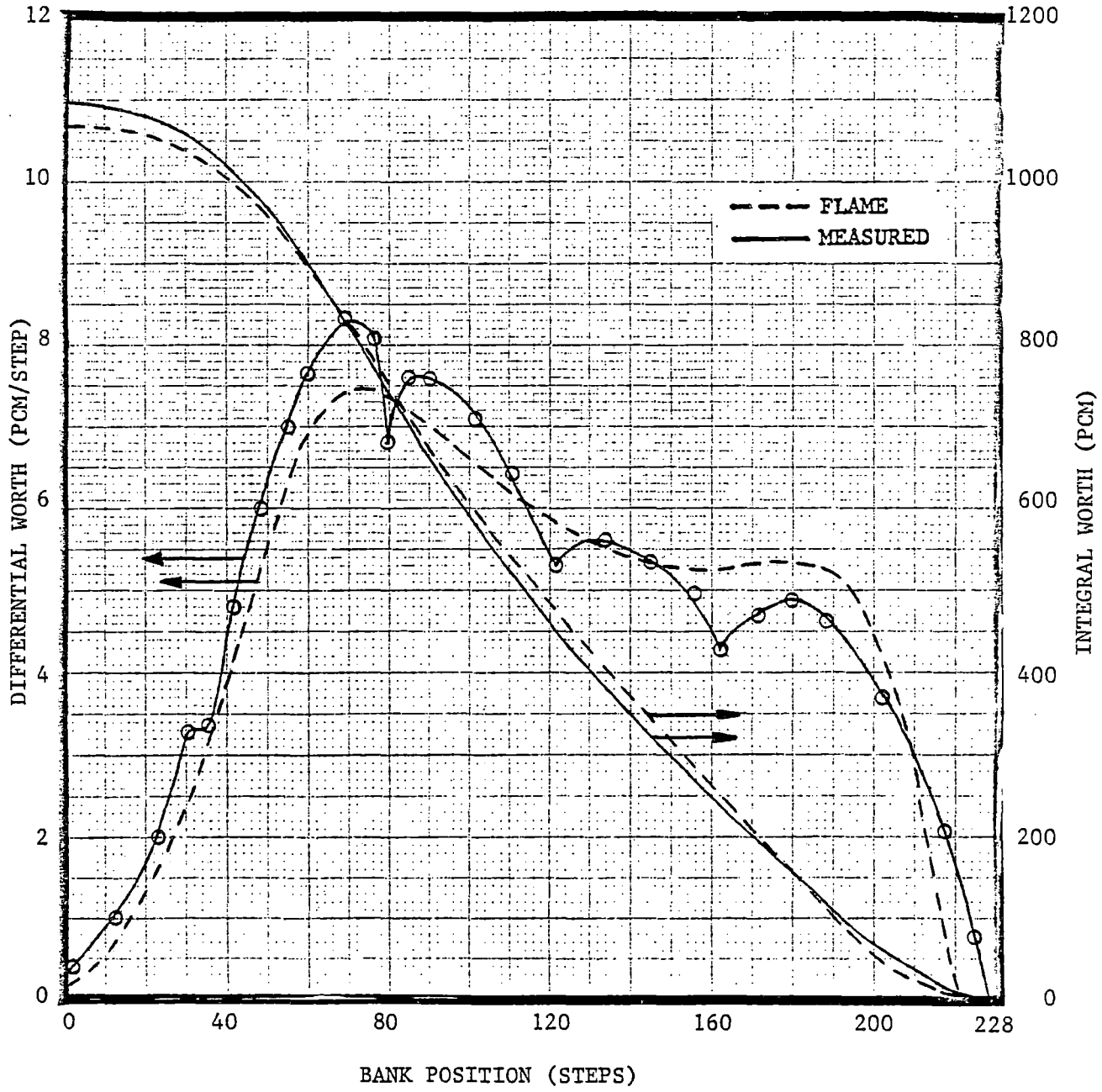


FIGURE 5-32

Control Rod Worth Comparison
for Surry 2, Cycle 3, BOC,
HZP, C Bank

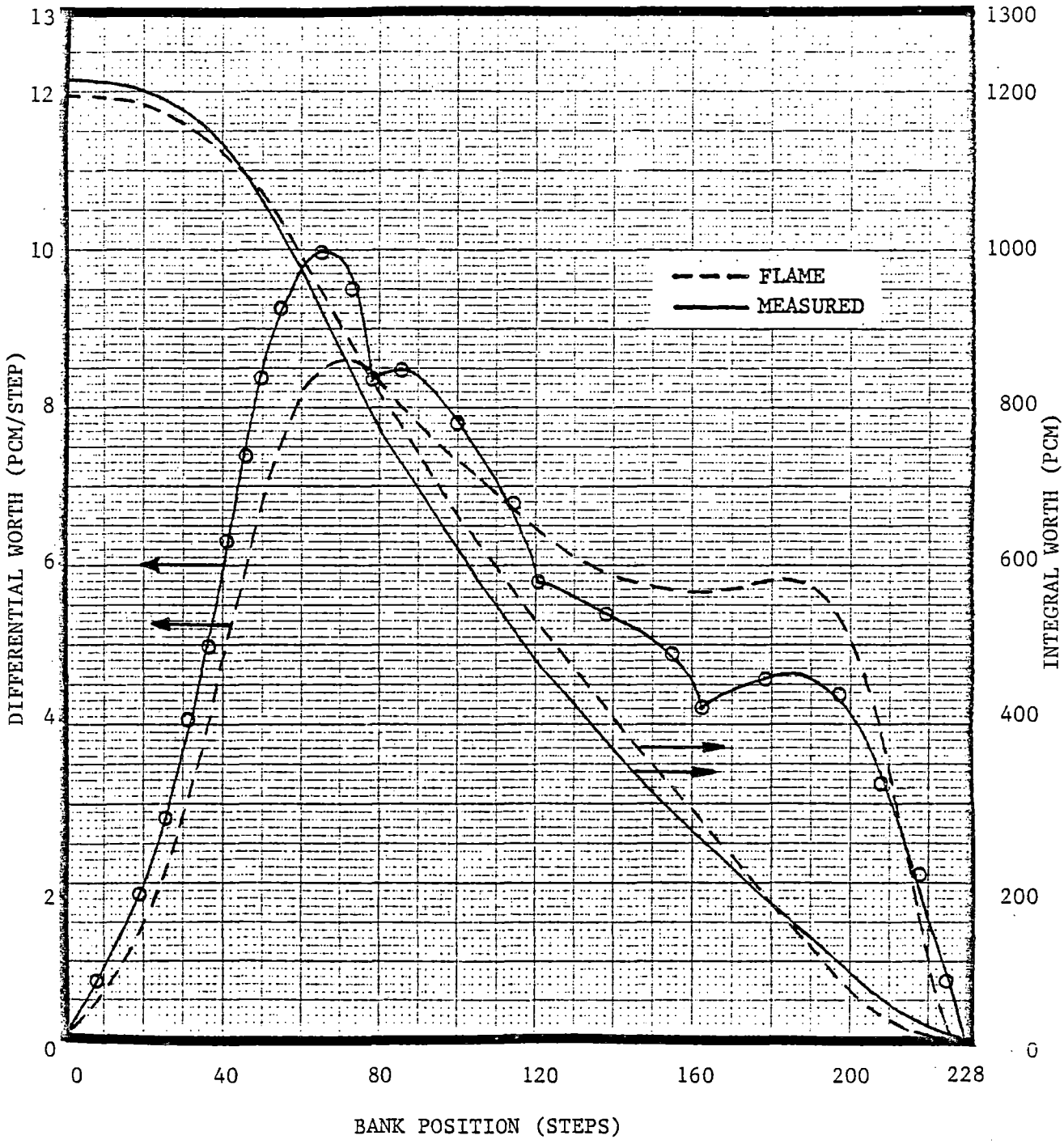


Figure 5-33
Control Rod Worth Comparison
for Surry 1, Cycle 4, BOC,
HZP, D Bank

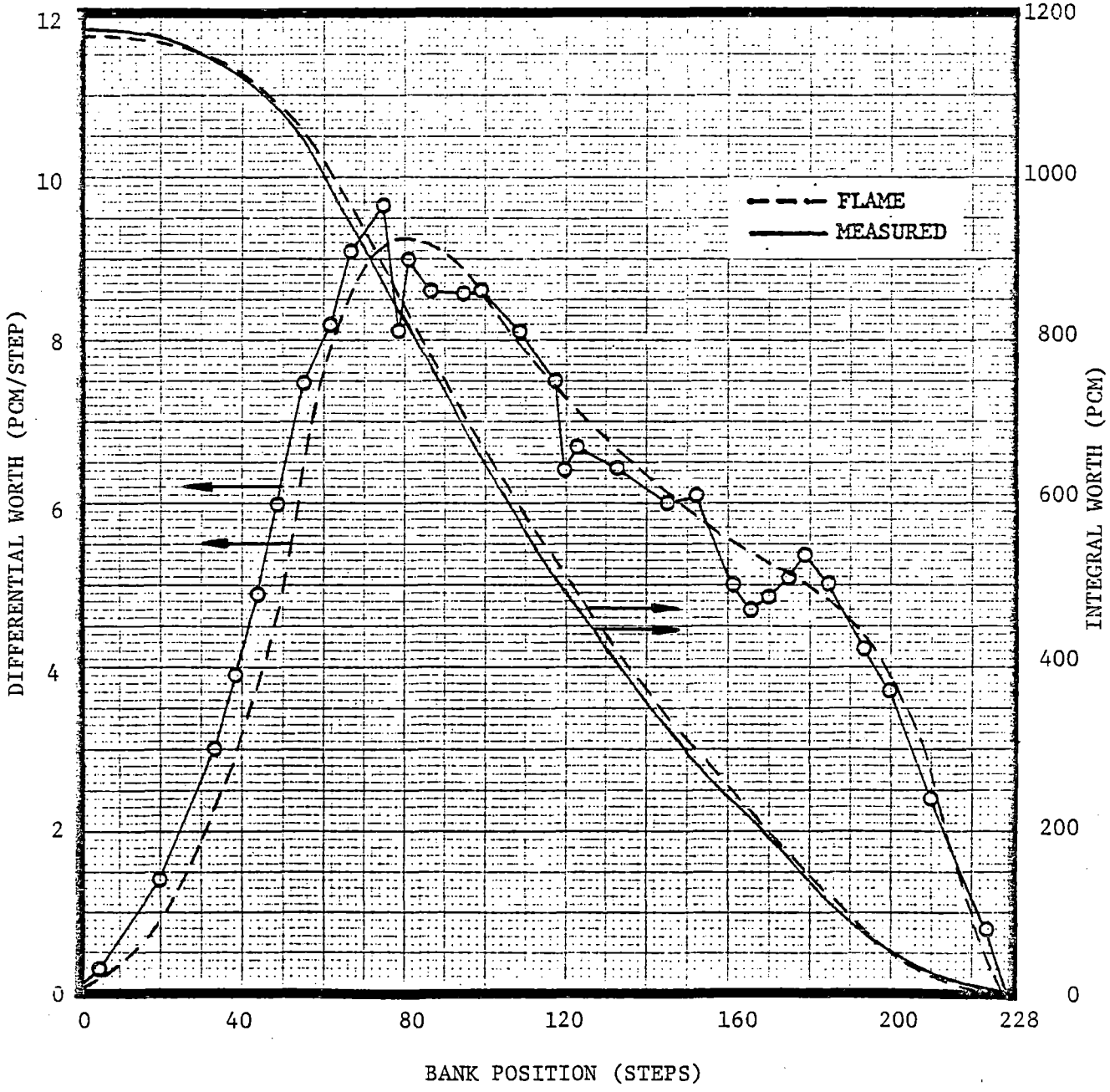


Figure 5-34
Control Rod Worth Comparison
for Surry 1, Cycle 4, BOC,
HZP, C Bank

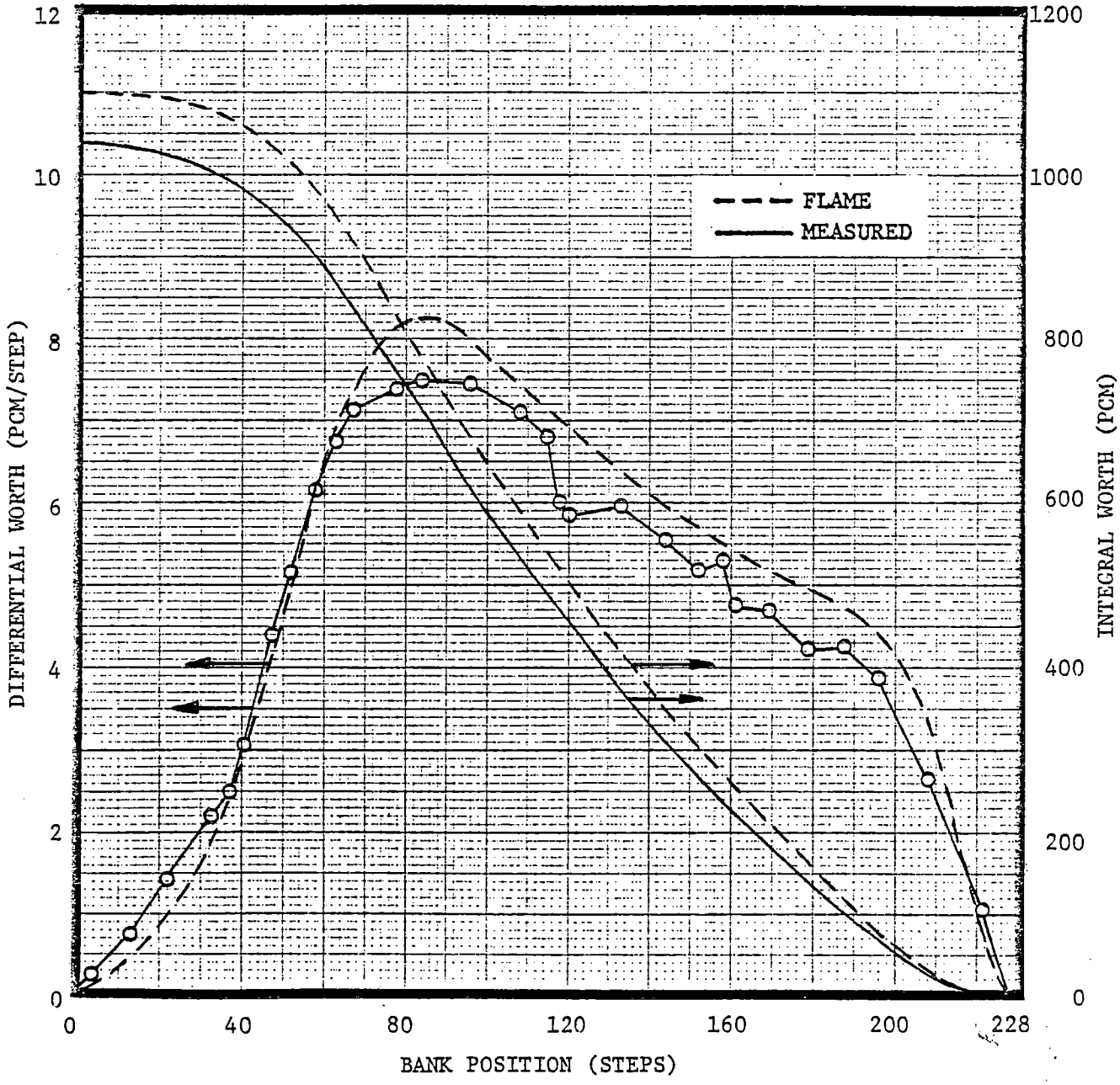


Figure 5-35
Control Rod Worth Comparison
for Surry 1, Cycle 4, BOC,
HZP, B Bank

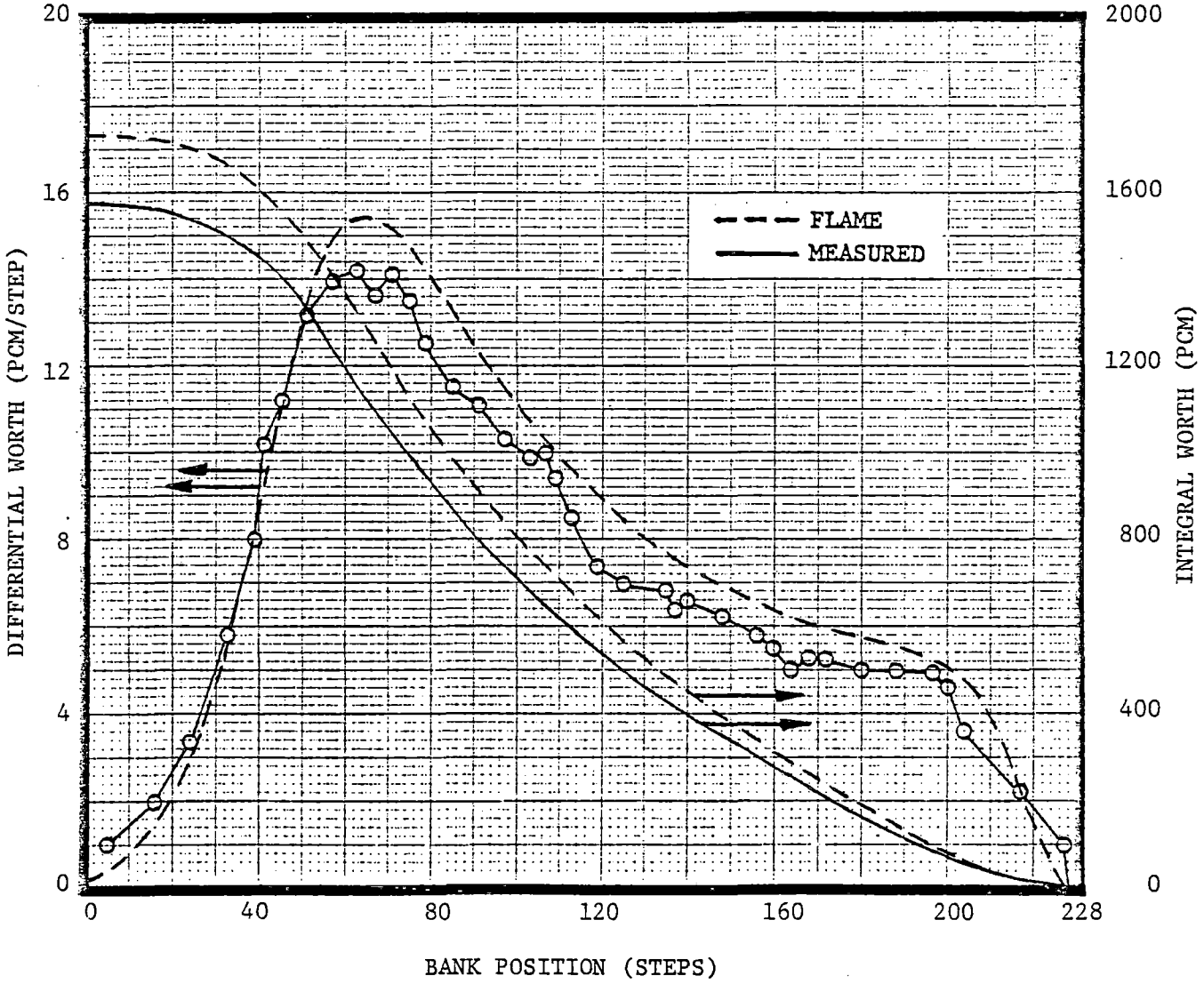


Figure 5-36
Control Rod Worth Comparison
for Surry 1, Cycle 4, BOC,
HZP, A Bank

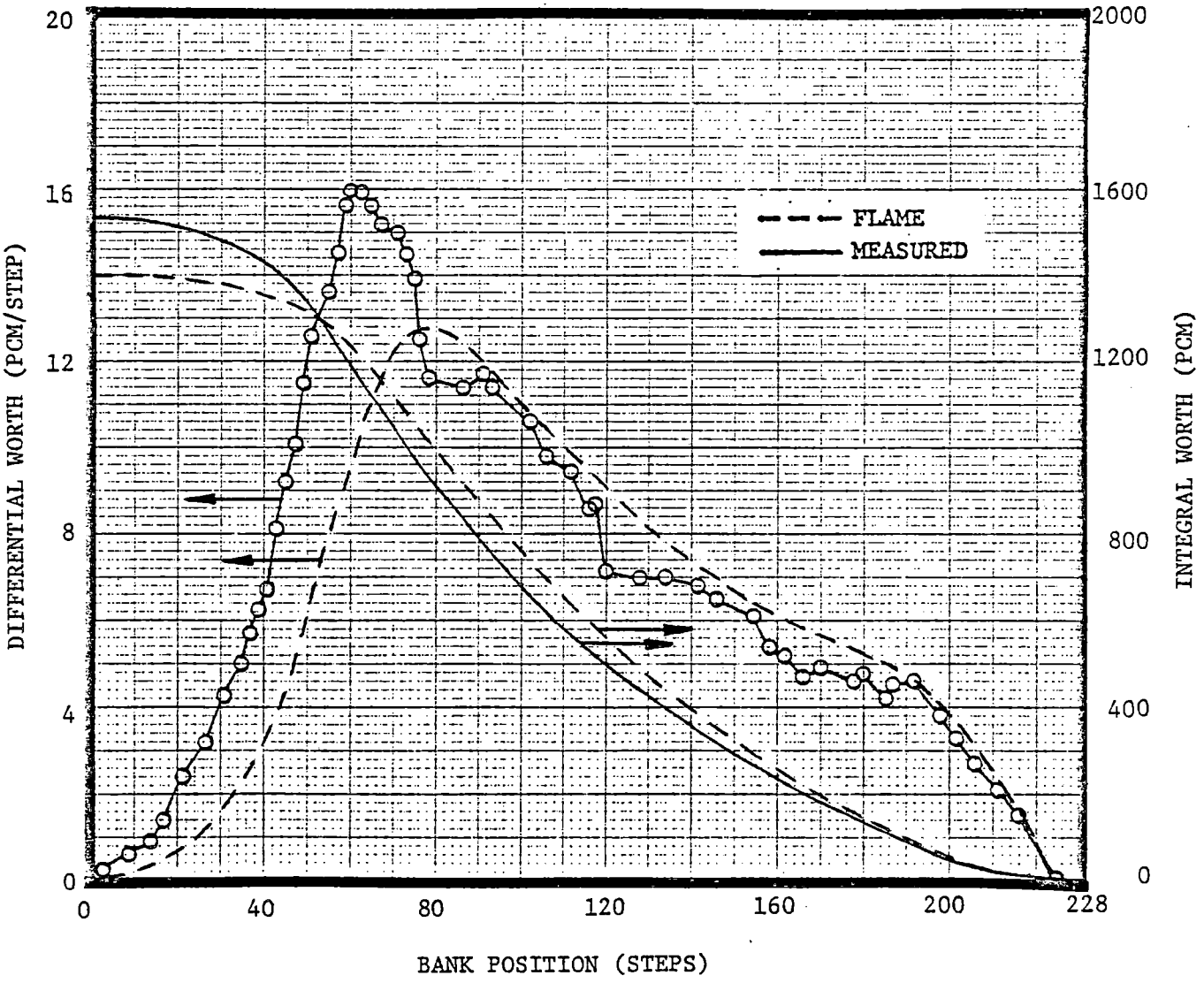


Figure 5-37
Control Rod Worth Comparison
for Surry 1, Cycle 4, BOC,
HZP, Banks A - D moving in
100 Step Overlap

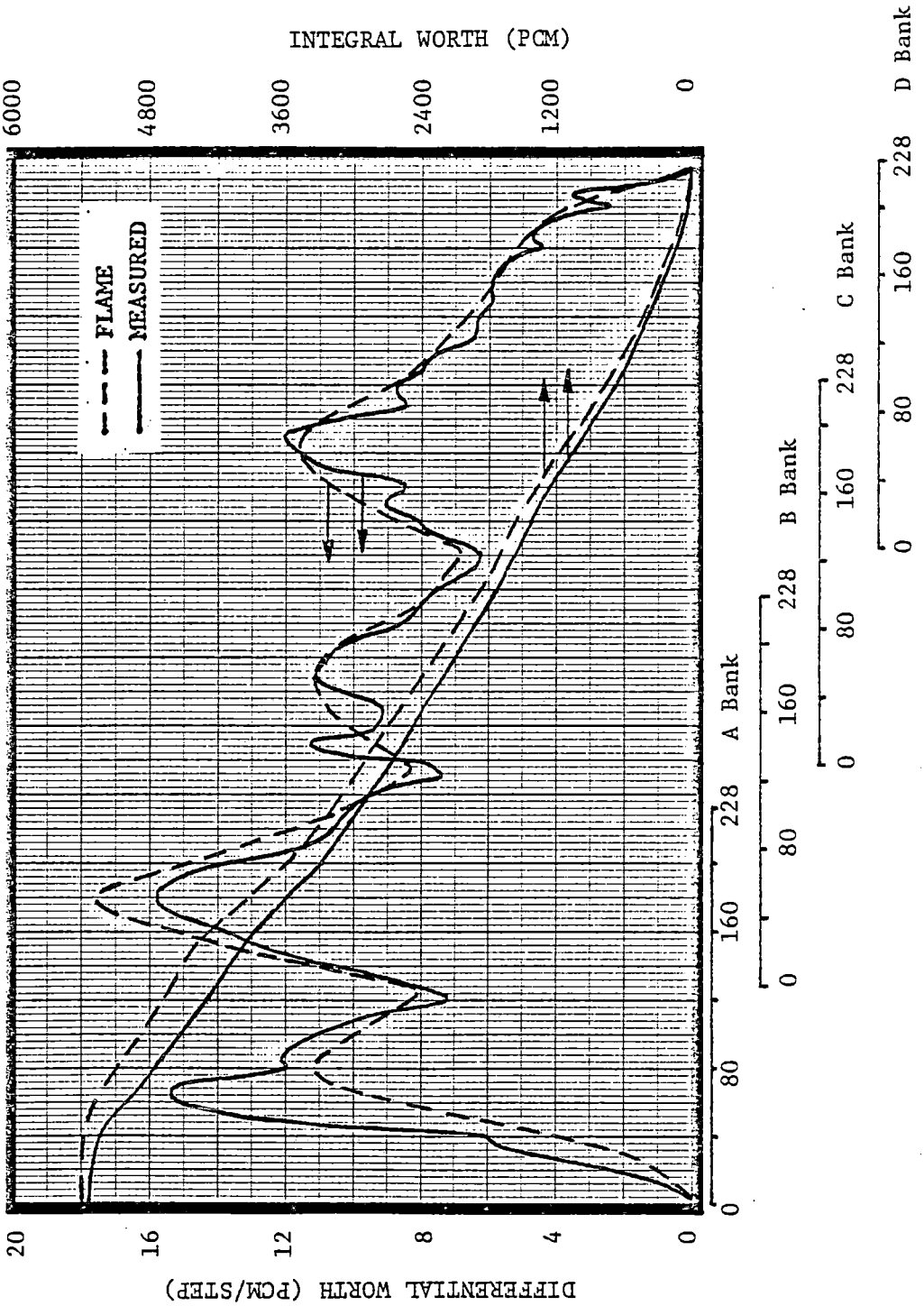


Figure 5-38
Control Rod Worth Comparison
for Surry 2, Cycle 4, BOC,
HZP, D Bank

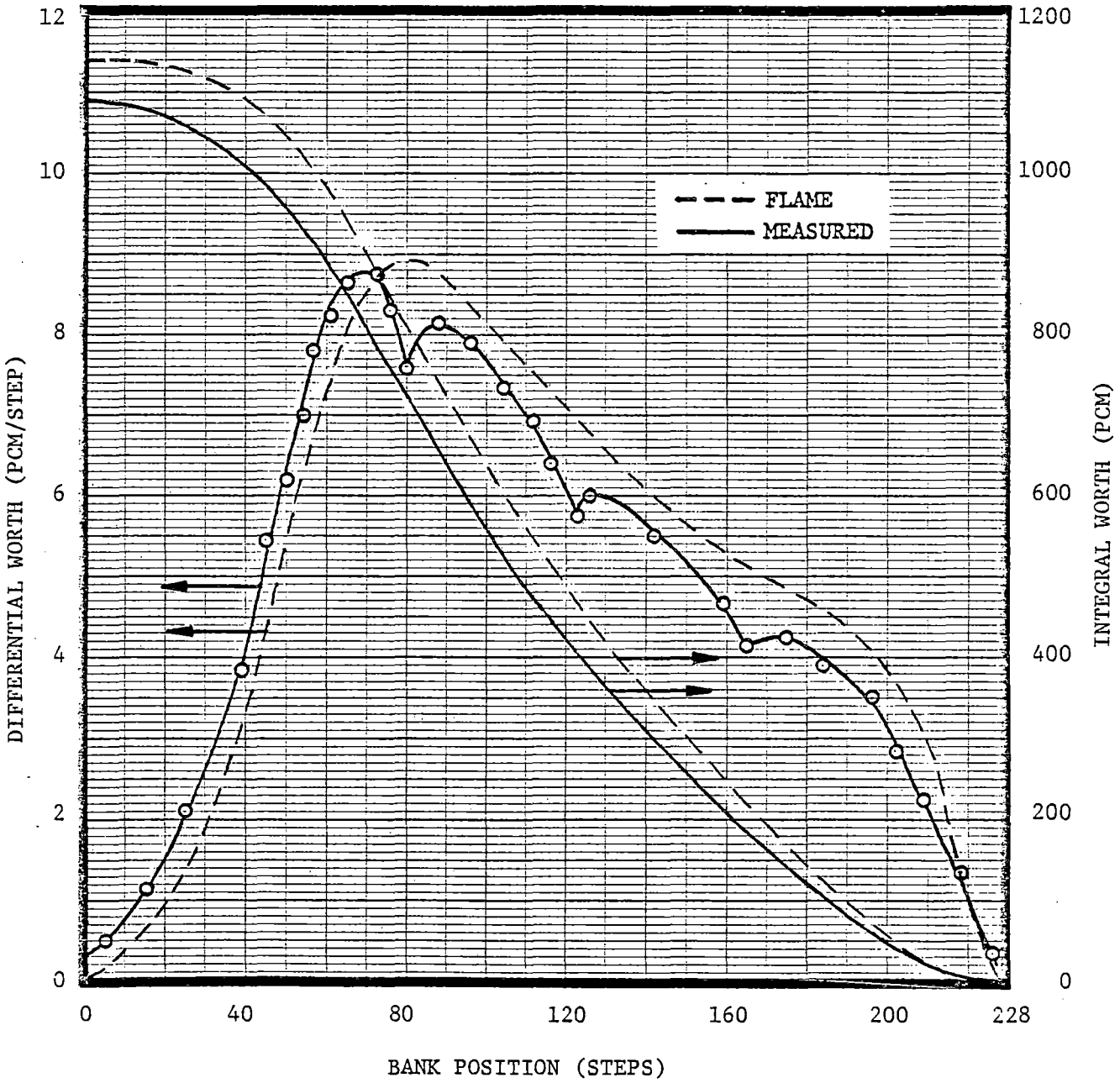


Figure 5-39
Control Rod Worth Comparison
for Surry 2, Cycle 4, BOC,
HZP, C Bank

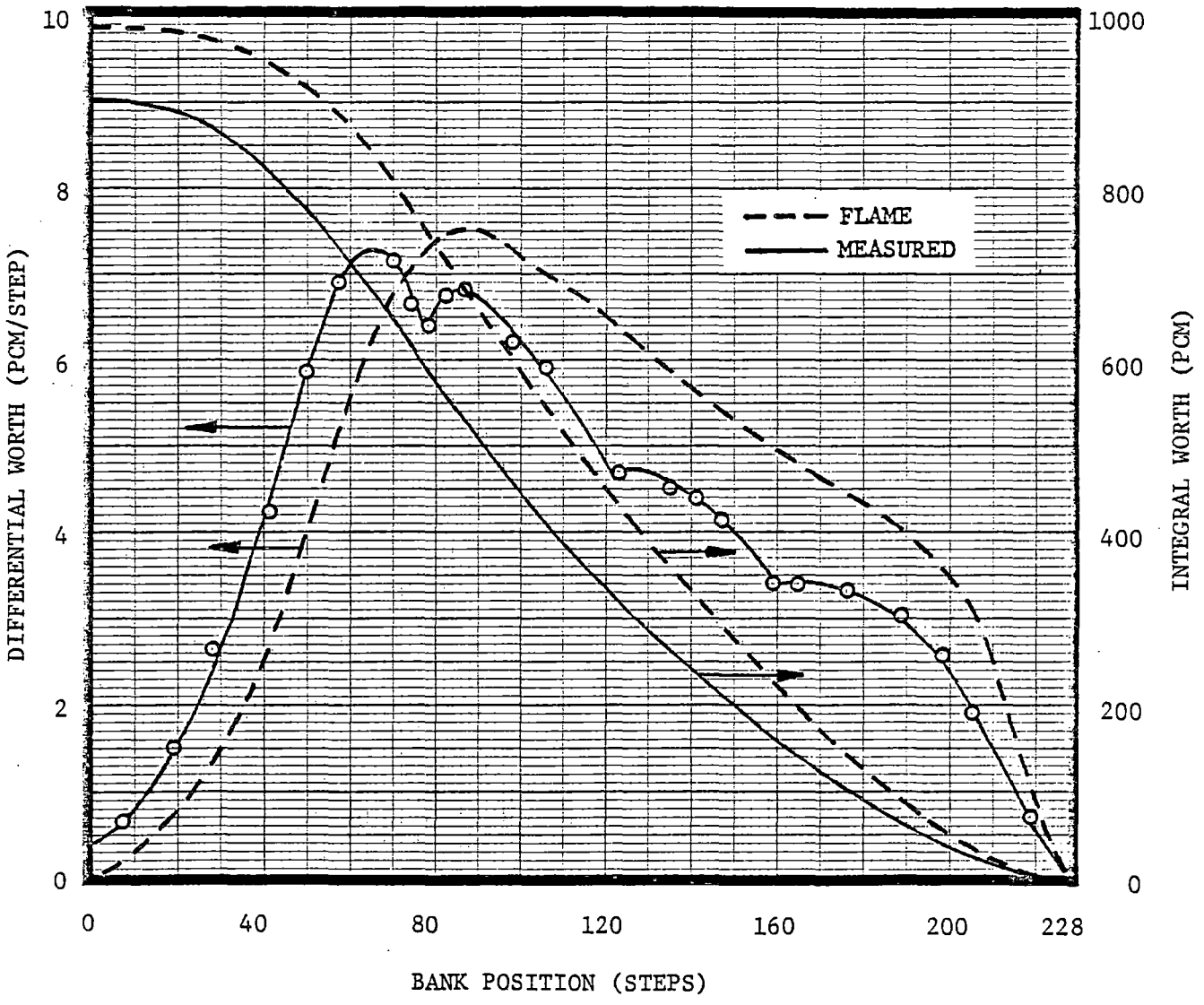


Figure 5-40
Control Rod Worth Comparison
for Surry 2, Cycle 4, BOC,
HZP, B Bank

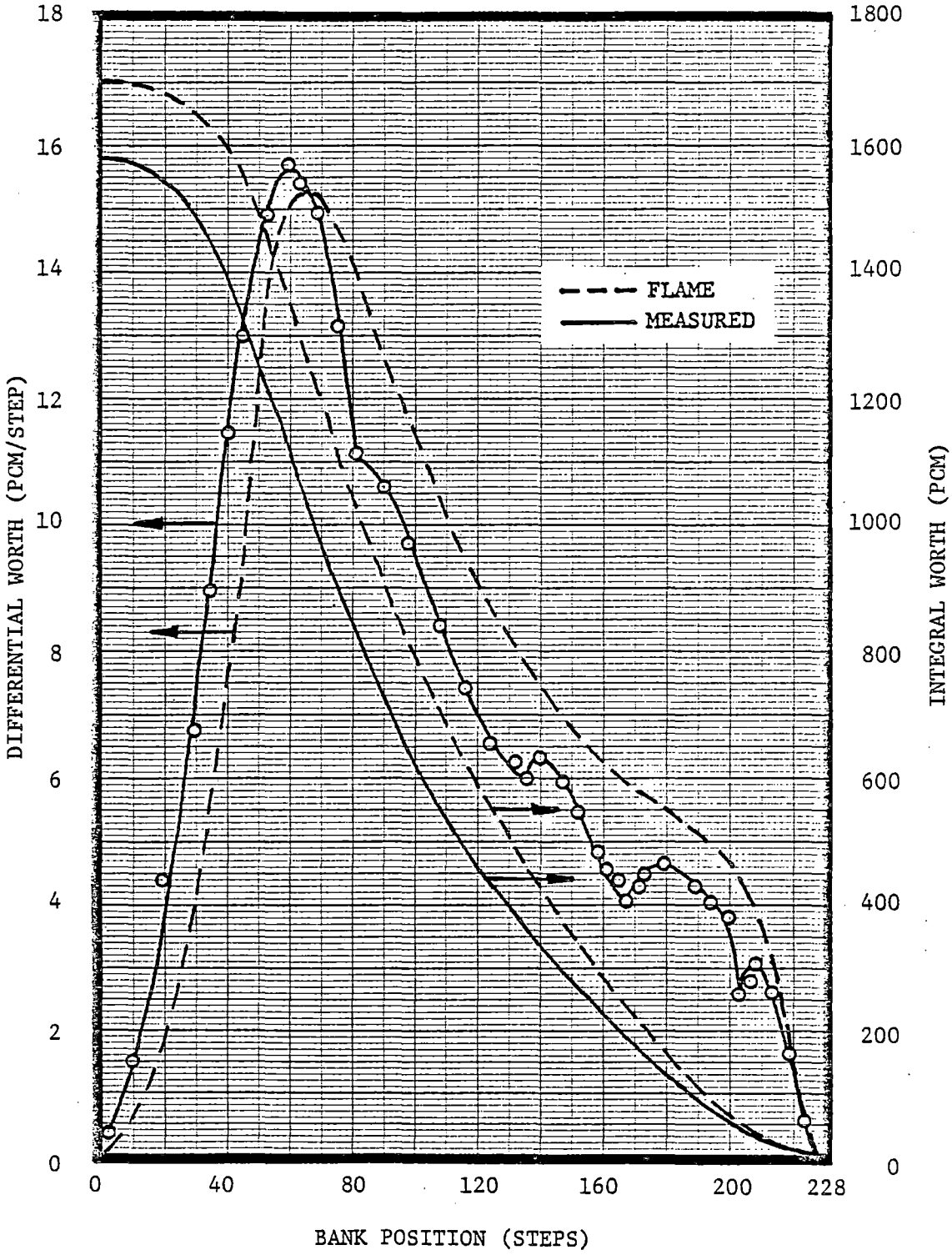


Figure 5-41
Control Rod Worth Comparison
for Surry 2, Cycle 4, BOC,
HZP, A Bank

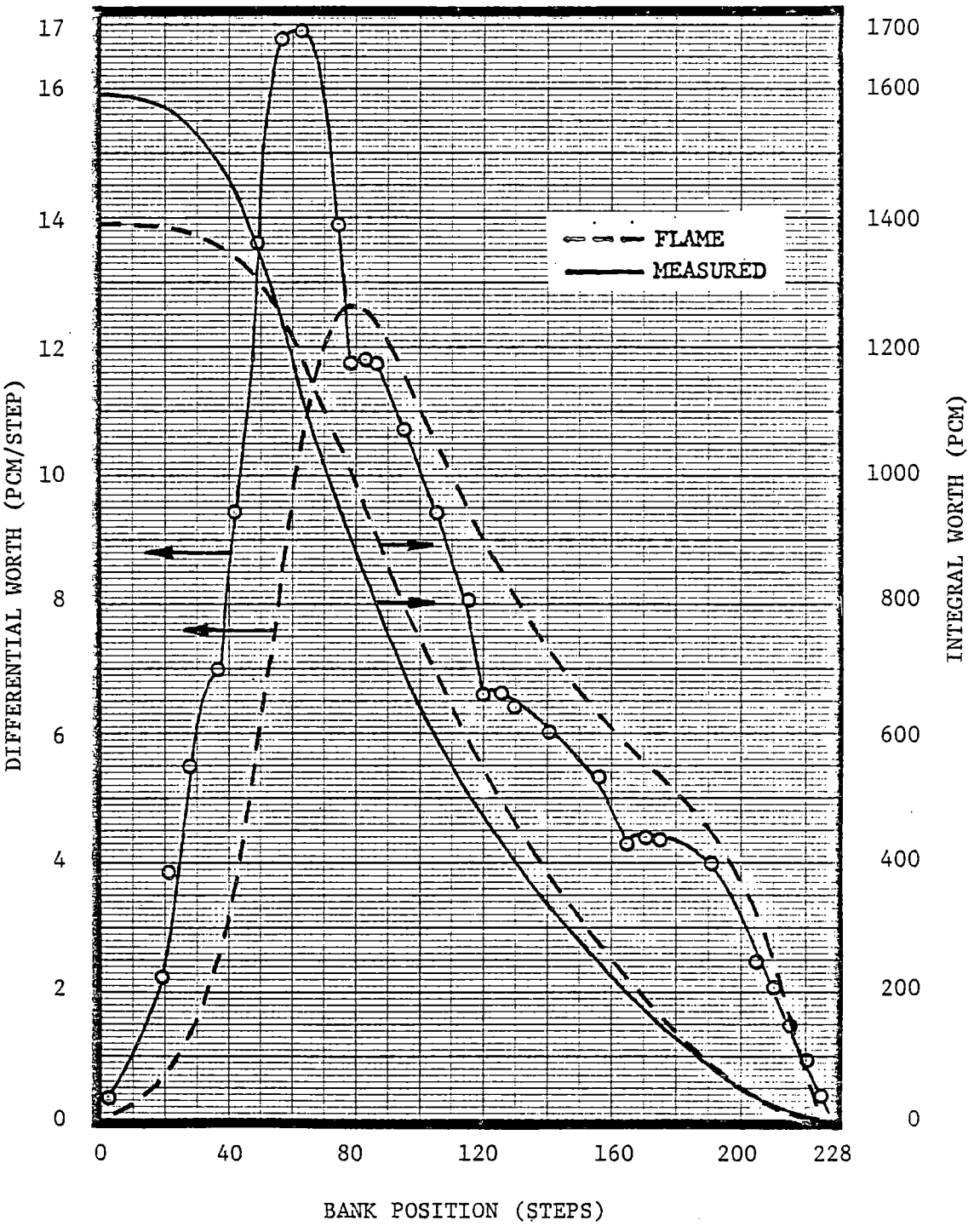
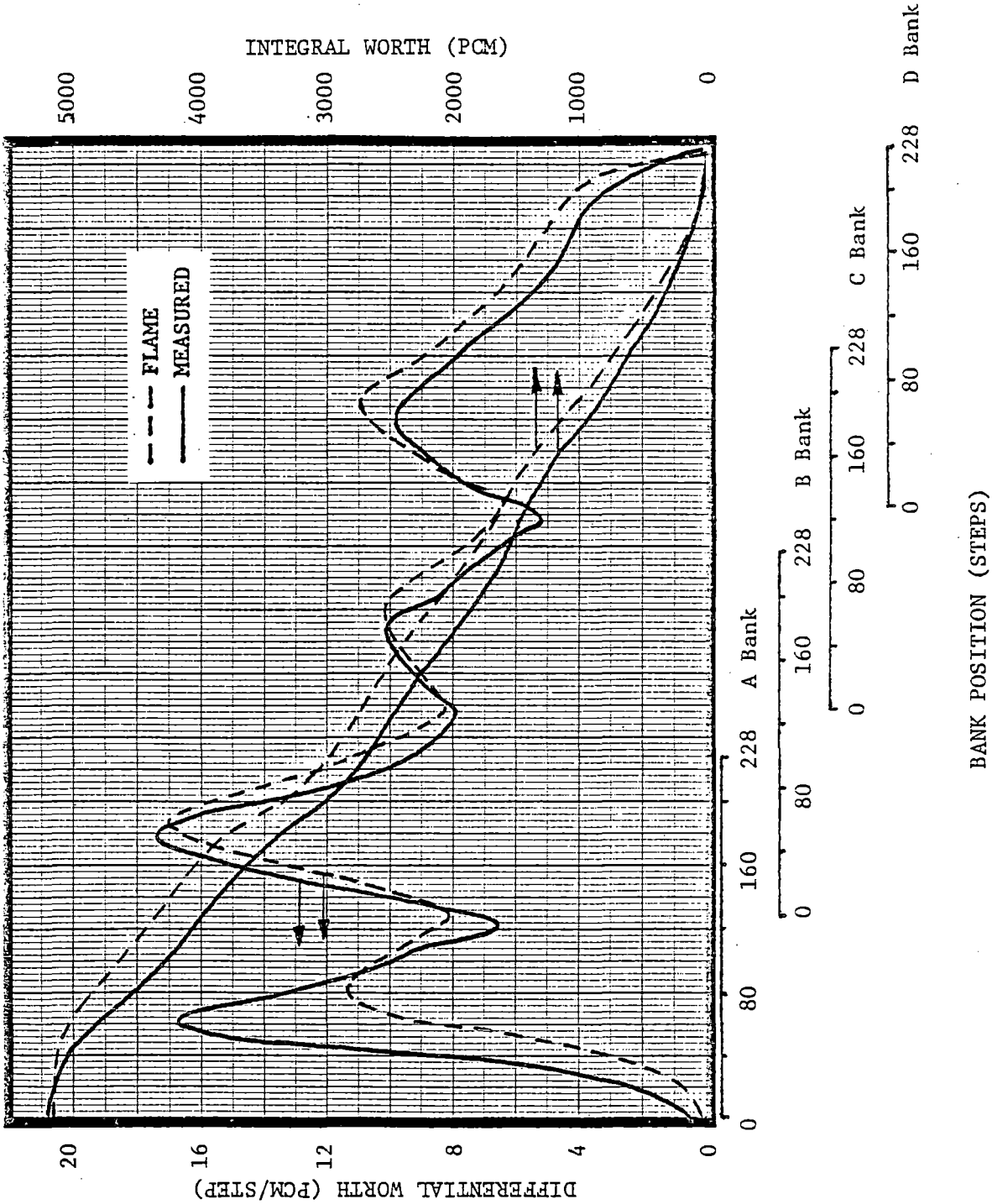


Figure 5-42

Control Rod Worth Comparison
for Surry 2, Cycle 4, BOC,
HZP, Banks A - D moving in
100 Step Overlap



Section 6 - Summary and Conclusions

The Vepco FLAME Model is operational at Vepco for the purpose of performing three-dimensional reactor physics analyses and supporting the evaluation of core performance. The model consists of the FLAME3 code with the NULIF, PDQ07, FLAFIT, EDITQAR, PICCOLO, and FLMSHUFL codes being used to provide either input or data manipulation. The accuracy of the Vepco FLAME Model has been established through extensive comparision of calculations with measurements from the Surry Units No. 1 and 2 over eight cycles of operation. The results of these comparisions indicate that the Vepco FLAME Model (which includes normalization to the Vepco PDQ07 Discrete Model) provides the capability to predict axial peaking factors and axial power distributions as well as differential and integral control rod worths. The comparisions also indicate an acceptable capability to predict axial offset trends as a function of changing reactor core conditions.

Verification, as well as improvements to the Vepco FLAME Model, will continue to be made as more experience is obtained through the application of the model to the units at the Surry and North Anna Nuclear Power Stations.

SECTION 7 - REFERENCES

1. W. A. Wittkopf, et. al., "NULIF - Neutron Spectrum Generator, Few Group Constant Calculator, and Fuel Depletion Code," BAW-10115, June 1976 (Babcock and Wilcox).

2. H. H. Hassan, et. al., "Babcock and Wilcox Version of PDQ07 - User's Manual," BAW-10117P, December 1975 (Babcock and Wilcox).

3. C. W. Mays and M. Furtney, "FLAME3 - A Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions," BAW-10124A, August 1976 (Babcock and Wilcox).

4. Private correspondence from the Babcock and Wilcox Company to the Virginia Electric and Power Company dated February 3, 1971, and October 6, 1971.

5. C. B. Franklin, "EDITQAR and PICCOLO - Auxiliary Fortran Computer Codes for Use with FLAME3," NFE Technical Report No. 27, April 1977 (Virginia Electric and Power Co.).

6. W. C. Beck, "FLMSHUFL - An Auxiliary Fortran Computer Code for Use with FLAME3," NFE Technical Report No. 55, October 1977 (Virginia Electric and Power Co.).

7. M. L. Smith, "The PDQ07 Discrete Model," VEP-FRD-19, July 1976 (Virginia Electric and Power Co.).

8. Final Safety Analysis Report - Surry Power Station Units 1 and 2, Virginia Electric and Power Company, December 1969.

9. D. L. Delp, et. al., "FLARE - A Three-Dimensional Boiling Water Reactor Simulator," GEAP-4598, July 1964 (General Electric).

10. C. B. Franklin, "Sensitivity Study of K-Infinity Multipliers for Depleted BP for the FLAME Model," NFE Calculational Note PM-4, July 1978 (Virginia Electric and Power Co.).

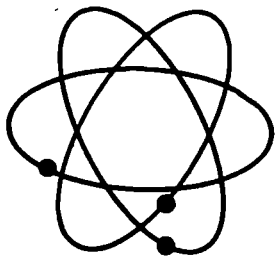
11. W. C. Beck, "Sensitivity of Control Rod Worth Shapes to Variation in Radial Power Distribution," NFE Calculational Note PM-3, July 1978 (Virginia Electric and Power Co.).

VEP-FRD-19A
JULY, 1981

Vepco

THE PDQ 07

DISCRETE MODEL



FUEL RESOURCES DEPARTMENT
VIRGINIA ELECTRIC AND POWER COMPANY

THE PDQ07 DISCRETE MODEL

by

M. L. Smith

Nuclear Fuel Engineering Group
Fuel Resources Department

Virginia Electric and Power Company
Richmond, Virginia

July, 1981

Recommended for Approval:

Martin L. Bowling

M. L. Bowling, Supervisor
Nuclear Fuel Engineering Group

Approved:

J. T. Rhodes

J. T. Rhodes, Director
Nuclear Fuel Engineering and
Operation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 18 1981

Mr. W. N. Thomas, Vice President
Fuel Resources
Virginia Electric Power Company
Richmond, Virginia 23261

Dear Mr. Thomas:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT VEP-FRD-19 "THE PDQ 07 DISCRETE MODEL"

The Nuclear Regulatory Commission (NRC) staff has completed its review of the Virginia Electric and Power Company Topical Report number VEP-FRD-19 entitled "The PDQ 07 Discrete Model". The PDQ 07 Discrete Model is two-dimensional (x-y geometry), two neutron energy group diffusion-depletion model which explicitly represents each fuel rod in the reactor. It was developed by the Virginia Electric and Power Company (VEPCO) and utilizes the Babcock and Wilcox developed NULIF, HAFIT, PDQ07, PAPDQ, and SHUFFLE codes. It is used specifically to perform reactor physics analysis, fuel management analysis, and to support the reactor startup and cycle operation of the Vepco Surry and North Anna nuclear reactors. The accuracy of the model predictions is demonstrated through comparison with measurement data obtained from the Surry reactors. Our summary of the evaluation is attached.

As the result of our reviews we conclude that the Virginia Electric and Power Company Licensing Topical Report VEP-FRD-19 entitled "The PDQ 07 Discrete Model" dated July 1976 is acceptable for referencing in licensing actions by VEPCO to the extent specified and under the limitations in the report and the attached evaluation.

We do not intend to repeat the review of the safety features described in the report and as found acceptable herein. Our acceptance applies only to the use of features described in the topical report and as discussed herein.

In accordance with established requirements, it is requested that Virginia Electric and Power Company issue a revised version of this report within three months of the receipt of this letter. This evaluation letter is to be included in the revised version between the title page and the abstract and the approved report will carry the identifier VEP-FRD-19A.

Should Nuclear Regulatory Commission criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Virginia Electric and Power Company will be expected to revise and resubmit the topical report or submit justification for the continue effective applicability of the topical report without revision.

If you have any questions about the review or our conclusion, please contact us.

Sincerely,

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

MAY 7 1981

EVALUATION OF VEPCO TOPICAL REPORT VEP-FRD-19

Report Number:	VEP-FRD-19
Report Title:	The PDQ07 Discrete Model
Report Date:	July 1976
Originating Organization:	Virginia Electric and Power Company
Reviewed by:	Core Performance Branch

The Virginia Electric Power Company has notified the NRC, by letter dated October 17, 1980, of their intention to perform their reload licensing in-house after late 1981. In support of this intention they have submitted and will submit a number of topical reports for our review. Report number VEP-FRD-19 is one of those reports. The Reactor Physics Section of the Core Performance Branch has reviewed this report. Our evaluation follows.

1. Summary of Report

The report VEP-FRD-19 is primarily a document which presents data to qualify the code PDQ07 in the discrete mode for use by VEPCO personnel to perform reload analyses for VEPCO-owned and operated reactors. Brief descriptions of the PDQ07 discrete model and its satellite codes are given. Since VEPCO purchased these codes from Babcock and Wilcox (B&W) they reference reports published by B&W for details of the various codes.

A detailed description of the Surry Unit 1 and Unit 2 reactor cores are presented. Included are thermal-hydraulic design parameters, and mechanical design parameters for the fuel assemblies, control rod assemblies, burnable poison rods and core structure including core barrel, thermal shield and side, and axial reflectors. Details of the first and second cycle loadings of both Unit 1 and Unit 2 are given.

A description of the manner in which the cores are modeled is given as well as a discussion of the manner in which the various satellite codes are used to provide input to and process the output from PDQ07. Comparisons of calculated values of certain parameters with measured values and with values calculated by vendor (Westinghouse) methods are given. Comparisons are made for critical boron concentrations, core radial power distributions, control bank worth, differential boron worth, and, at end of cycle, the burnup distributions. Calculations and power distribution measurements are compared in two dimensions only since computer size limits the discrete model calculations to two dimensions. At beginning of cycle comparisons are made for both hot zero power and hot full power conditions. At end of life only hot full power conditions are considered. Comparison are also presented between the measured data and the vendors calculations which have been used to perform reload analyses up to this time.

The results of the comparisons between calculation and measurement may be summarized as follow:

1. Assembly average power distributions are typically predicted to within a two percent standard deviation. The maximum standard deviation obtained for the first and second cycles of Unit 1 and Unit 2 was 4.3 percent at zero power and 3.6 percent a powers greater than 10 percent of full power.

2. Peak rod $F_{\Delta H}^{N}$ values are predicted typically within 2.5 percent with the maximum difference being 4.3 percent. The VEPCO model tends to underpredict this quantity.

3. Assembly average burnups are predicted to within 2.5 percent and batch average burnup to within 1.5 percent.

4. Typically, critical soluble boron concentration are predicted to within 30 parts per million and boron worth to within 3 percent.

5. Control rod bank integral worths are predicted typically within 6 percent with a maximum deviation of 9.7 percent.

Comparison of the results of the VEPCO model to those of the vendor for the same quantities shows that the average absolute deviations for the two models are similar with the VEPCO model tending to have smaller average deviations.

2. Summary of Evaluation

We have reviewed the information presented in licensing topical report, VEP-FRD-19, "The PDQ07 Discrete Model." The following comments summarize our evaluation.

The various codes that comprise the model are described by reference to documents that have been submitted by their developer, the B&W Company. These reports have been previously reviewed and accepted by the staff for reference by B&W. Their use as references for code descriptions by VEPCO is acceptable inasmuch as VEPCO purchased the codes from B&W. The procedures used by VEPCO to implement the codes are described in the report. These are standard procedures in industry-wide use and are acceptable.

We have reviewed the data presented to support the conclusions regarding the uncertainties in the calculated results. We conclude the sufficient examples of comparisons between calculation and measurement to permit the evaluation of calculational uncertainties. We concur with the particular values of uncertainties given in the topical report and repeated in Section 1 above. We further concur with the conclusion that the VEPCO model is an acceptable replacement for the vendor models currently in use. These conclusions apply to the calculation of

- assembly average radial power distributions;
- $F_{\Delta H}^{N}$ values;
- assembly average burnups;
- critical boron concentrations and boron worths; and
- control rod integral bank worths.

3. Evaluation Procedure

The review of topical report VEP-FRD-19 has been conducted within the guidelines provided for analytical methods in the Standard Review Plan, Section 4.3. Sufficient information is provided to permit a knowledgeable person to conclude that the VEPCO model described in this report is state-of-the-art and is acceptable. Sufficient data are presented to permit the conclusion that the derived uncertainties are reasonable and are acceptable.

4. Regulatory Position

Based on our review of licensing topical report VEP-FRD-19 we conclude that it is acceptable for reference in licensing actions by VEPCO. Such reference may be made for purposes of describing the code and for citing uncertainties in the following quantities:

- assembly average radial power distributions;
- $F_{\Delta H}^{N}$ values;
- assembly average burnups;
- critical boron concentrations and boron worths; and
- control rod integral bank worths.

We further conclude that this model is an acceptable substitute for vendor calculations of the above named quantities.

We endorse VEPCO's commitment in the report to continue verification and improvements to the PDQ07 discrete model as more data are obtained from the Surry and North Anna reactors.

CLASSIFICATION/DISCLAIMER

The data and analytical techniques described in this report have been prepared for specific application by the Virginia Electric and Power Company. The Virginia Electric and Power Company makes no claim as to the accuracy of the data or technique contained in this report if used by other organizations. In addition, any use of this report or any part thereof must have the prior written approval of the Virginia Electric and Power Company.

ABSTRACT

A two-dimensional (x-y geometry), two neutron energy group diffusion-depletion model which explicitly represents each fuel rod in the reactor has been developed by the Virginia Electric and Power Company (Vepco). The model, which is designated as the PDQ07 discrete model, utilizes the Babcock and Wilcox developed NULIF, HAFIT, PDQ07, PAPDQ, and SHUFFLE codes. It is used specifically to perform reactor physics analysis, fuel management analysis, and to support the reactor startup and cycle operation of the Vepco Surry and North Anna nuclear reactors. The accuracy of the model predictions is demonstrated through comparison with measurement data obtained from the Surry reactors.

ACKNOWLEDGEMENTS

The author would like to thank Mr. S. A. Ahmed for his assistance in data analysis and preparation of this report and Ms. Carolyn Watson for her typing of the final manuscript. The author would also like to express his appreciation for the contribution of a number of people who were responsible for reviewing this report.

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SECTION 1 - INTRODUCTION

The Virginia Electric and Power Company (Vepco) is currently developing the capability to perform nuclear reactor analyses for the Surry and North Anna nuclear power stations. The objective of this topical report is (1) to describe one of the computational models developed at Vepco for the purposes of reactor physics analyses, fuel management evaluation, and core follow support and (2) to demonstrate the accuracy of this model by comparing analytical results generated with the model with actual measurements from Surry Units No. 1 and 2.

The computational model to be described is a discrete (one mesh line per rod), two-dimensional, two neutron energy group, diffusion-depletion (with thermal-hydraulic feedback) calculational package and is designated as the PDQ07 discrete calculational model. The PDQ07 discrete model uses the NULIF[1], PDQ07[2], SHUFFLE[3], HAFIT[4], and PAPDQ[5] computer codes which are part of the Fuel Utilization and Performance Analysis Code[6] (FUPAC) system obtained from the Babcock and Wilcox Company. The FUPAC system is currently used by Babcock and Wilcox to perform production reactor analysis and design. A detailed description of the input/output, functioning, and physical model of the above computer codes can be obtained from the referenced Babcock and Wilcox computer code manuals. The FUPAC system is maintained by Vepco and updated through contractual arrangements between Vepco and Babcock and Wilcox.[6]

The types of calculations that can be performed with the PDQ07 discrete model include:

1. <u>REACTOR PHYSICS ANALYSIS</u>:

    a. Two-dimensional radial power distributions, including

relative radial peaking factors (F_{xy}) and enthalpy rise hot channel factors ($F^N_{\Delta H}$), for both rodded and unrodded planes as a function of burnup.

b. Critical soluble boron concentrations as a function of burnup.

c. Integral control rod bank worths.

d. Nuclide concentrations for the fuel and burnable poison rods as a function of burnup.

e. Fuel rod and assembly average burnup distributions.

2. FUEL MANAGEMENT:

a. Batch power and burnup sharing.

b. Fuel isotopics as a function of burnup.

c. Evaluation of alternative fuel loading patterns.

3. CORE FOLLOW:

a. Input constants for determining measured core power distributions through use of the INCORE Code.

b. Critical boron concentrations and control rod bank worths.

To date the PDQ07 discrete model has been used at Vepco (on the Vepco IBM System 370 computers) to calculate best estimates of the power and burnup sharing for future cycles in order to determine and optimize nuclear fuel requirements. In addition, substantial effort has gone into the evaluation of alternative fuel loading patterns for specific operating cycles in order to optimize fuel utilization and unit operating flexibility. The benefits obtained from this limited application have been significant. As a result, it is now intended to use the PDQ07 discrete model to perform the complete reactor physics analysis associated with specific operating cycles and to verify (through core follow analysis) that the reactor is operating in accordance with these design predictions. It is anticipated that the extended use of this model will provide additional benefits to

Vepco through increased design flexibility, technical support of licensing positions, and greater operational freedom.

The remainder of this report describes the Surry Units No. 1 and 2 reactor core to be modeled, the purpose and interrelationships of the various computer codes which comprise the PDQ07 discrete model, the processes through which these codes function, the specific modeling of these codes to represent the reactor core, and the comparison of calculated results with reactor measurements from Cycles 1 and 2 of Surry Units No. 1 and 2.

SECTION 2 - CORE DESCRIPTION

2.1 INTRODUCTION

The Surry Nuclear Power Station, which currently consists of two operating units, has been selected as the operating system to be modeled for verification of the PDQ07 discrete model. The Surry Units No. 1 and 2 are identical Westinghouse designed three coolant loop pressurized water reactors with thermal ratings of 2441 Mwt. Initial criticality was achieved for Surry Unit No. 1 on July 1, 1972 and for Surry Unit No. 2 on March 7, 1973. The initial cycle for Surry Unit No. 1 was completed on October 24, 1974 and for Surry Unit No. 2 on April 26, 1975. Second cycle operation commenced on January 30, 1975 and June 14, 1975 and was completed on September 26, 1975 and April 22, 1976 for Surry Units No. 1 and 2, respectively.

2.2 CORE DESIGN

The Surry cores consist of 157 fuel assemblies surrounded by a core baffle, barrel, and thermal shield and enclosed in a steel pressure vessel. The pressure inside the vessel is maintained at a nominal 2250 psia. The coolant (and moderator) is pressurized water which enters the bottom of the core at 532°F and undergos an average rise in temperature of 65.5°F before exiting the core. The average coolant temperature is 566°F and the average linear power density of the core is 6.2 kw/ft.

Each of the 157 fuel assemblies consists of 204 fuel rods arranged in a 15 by 15 square array. The fuel used in the Surry cores consist of slightly enriched uranium dioxide fuel pellets contained within a Zircaloy-4 clad. A small gap containing pressurized helium exists between the pellets and the inner diameter of the clad. For the positions in the 15 x 15 array not occupied by fuel rods, there are 20 guide tube locations for either

solid burnable poison rods or control rods and one centrally located instrumentation tube. (See Figure 2-1.) The fuel rods in each fuel assembly are supported by seven Inconel-718 grids located along the length of the assembly. These grids are mechanically attached to the guide tubes, which are, in turn, welded to the upper and lower nozzles, and thus provide for assembly structural support.

There are 48 full-length Rod Cluster Control Assemblies (referred to as control rods) used to control core reactivity as well as five part-length rods for axial power shaping. (It should be noted that the part-length control rods are physically present but are not currently allowed to be inserted into the core). The absorber material of the control rods is an alloy consisting of 80% silver, 15% indium, and 5% cadmium. The various control rods are arranged in and moved in symmetrically located groups, or banks, as depicted in Figure 2-2. Banks D, C, B, and A are denoted as the control banks and are moved in a fixed sequential pattern to control the reactor over the power range of operation. The remaining rods, Banks SA and SB, are denoted as shutdown banks and are used to provide shutdown margin.

In addition to the control rods, a chemical (boric acid) shim is used to control excess core reactivity and to facilitate operational flexibility. Above certain concentrations of chemical shim, burnable poison rods are also used to control excess reactivity. Fresh and/or depleted burnable poison rods can also be used to shape (i.e., improve) the core power distribution. The burnable poison rods contain borosilicate in the form of Pyrex glass clad in a stainless steel tube. Burnable poison rods, which may be used in any fuel assembly not under a control rod bank location, consist of clusters of either 8, 12, 16, or 20 rods which are inserted into the Zircaloy-4 control rod guide tubes.

Specific values of the principal mechanical and thermal-hydraulic parameters of the Surry core are provided in Table 2-1. A complete description of the Surry units is given in Reference 7.

2.3 FUEL LOADING

The initial and reload quarter core fuel loadings (i.e., initial enrichments and density, previous cycle location if appropriate, beginning of cycle burnup, and number of fresh or depleted burnable poison rods present) for both Surry units are provided in Figures 2-3 through 2-8. It should be noted that the fuel loadings for Cycle 1 of both Surry units are identical. The fuel management strategy employed in the initial cycle of operation of each unit was the checkerboard loading of the two lower enriched fuel batches in the center of the core and the highest enriched fuel batch around the periphery of the core. In the second cycle of operation, this strategy was adhered to by loading fresh fuel around the periphery of the core.

Table 2-1

SURRY CORE DESCRIPTION

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total core heat output, Mwt	2441
Heat generated in fuel, %	97.4
System operating pressure, psi	2250
Total coolant flow rate, lb/hr (gpm)	100.7×10^6 (265,500)
Coolant Temperatures, °F (@100% power)	
Nominal inlet	532
Average rise in the core	65.5
Average in the core	566
Nominal outlet of hot channel	642
Average linear power density, Kw/ft	6.2

MECHANICAL DESIGN PARAMETERS

Fuel Assemblies		
Design	Canless 15 x 15	
Number	157	
Rod pitch, inches	0.563	
Overall dimensions, inches	8.426 x 8.426	
Number of grids per assembly (material)	7 (Inconel-718)	
Number of instrumentation tubes	1	
Fuel Rods		
Number	32,028	
Number of rods/assembly	204	
	Batch 1, 2, 4	Batch 3
Outside diameter, inches	0.422	0.422
Diametrical gap, inches	0.0075	0.0085
Clad thickness, inches	0.0243	0.0243
Clad material	Zircaloy-4	
Fuel Pellets		
Material	Sintered UO_2	
Density (% of theoretical)	See Figures 3-4 through 3-	
	Batch 1, 2, 4	Batch 3
Outer diameter	0.3659	0.3649
Control Rod Assemblies		
Neutron absorber	5% Cd-15% In-80% Ag	
Cladding Material	Type 304 SS-Cold worked	
Clad thickness, inches	0.019	
Number (full length)	48	
Number of rods per assembly	20	

Table 2-1
(Continued)

Burnable Poison Rods	
Material	Pyrex glass
Content B_2O_3 (w/o)	12.5
Core Structure	
Core barrel I.D./O.D., inches	133.875/137.875
Thermal shield I.D./O.D., inches	142.625/148.000
Core diameter, inches (approximate)	119.5
Reflector thickness (approximate) and composition	
Top - Water plus steel, in.	10
Bottom - Water plus steel, in.	10
Side - Water plus steel, in.	15

FIGURE 2-1

CROSS SECTIONAL VIEW OF SURRY FUEL ASSEMBLIES

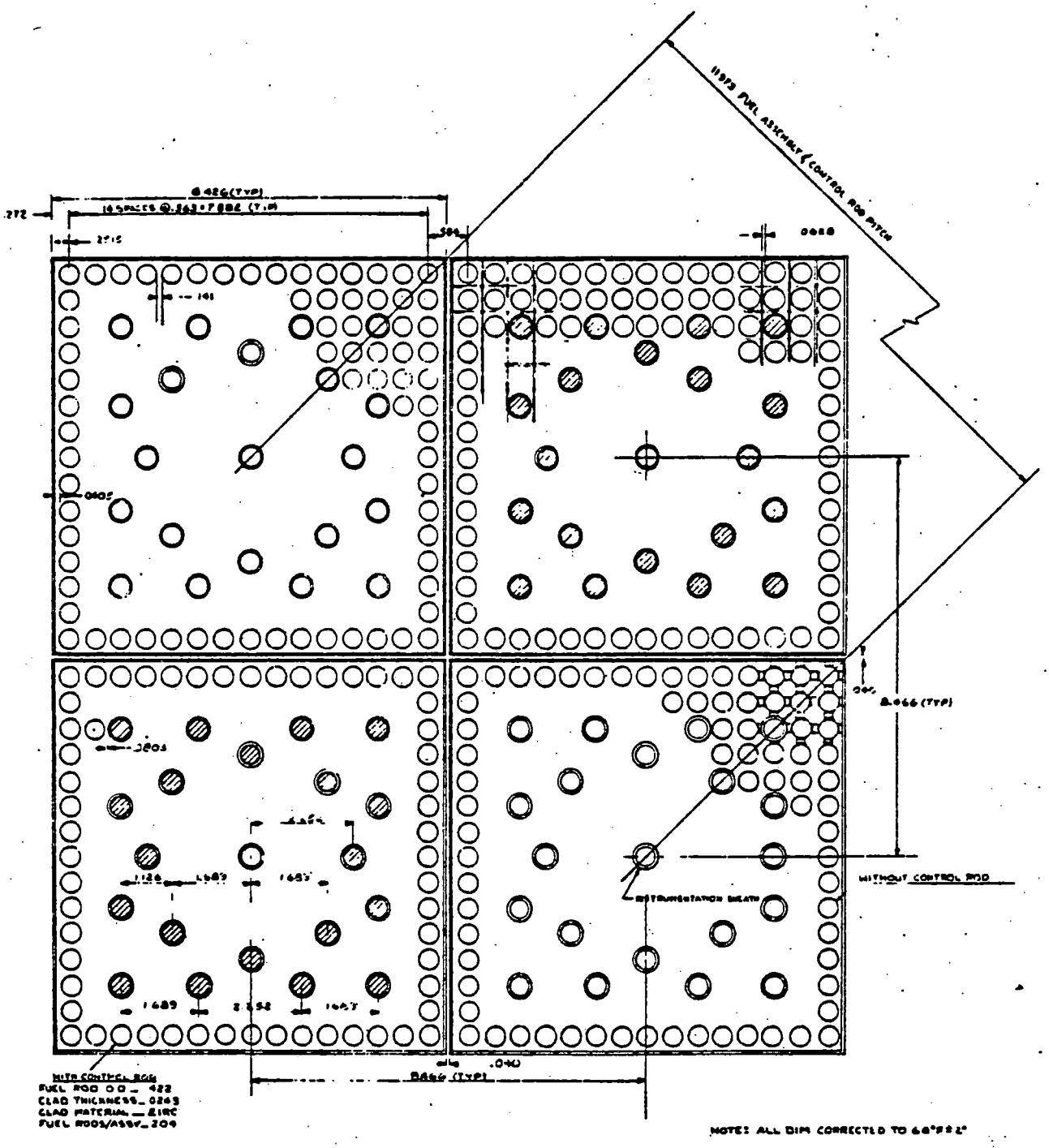
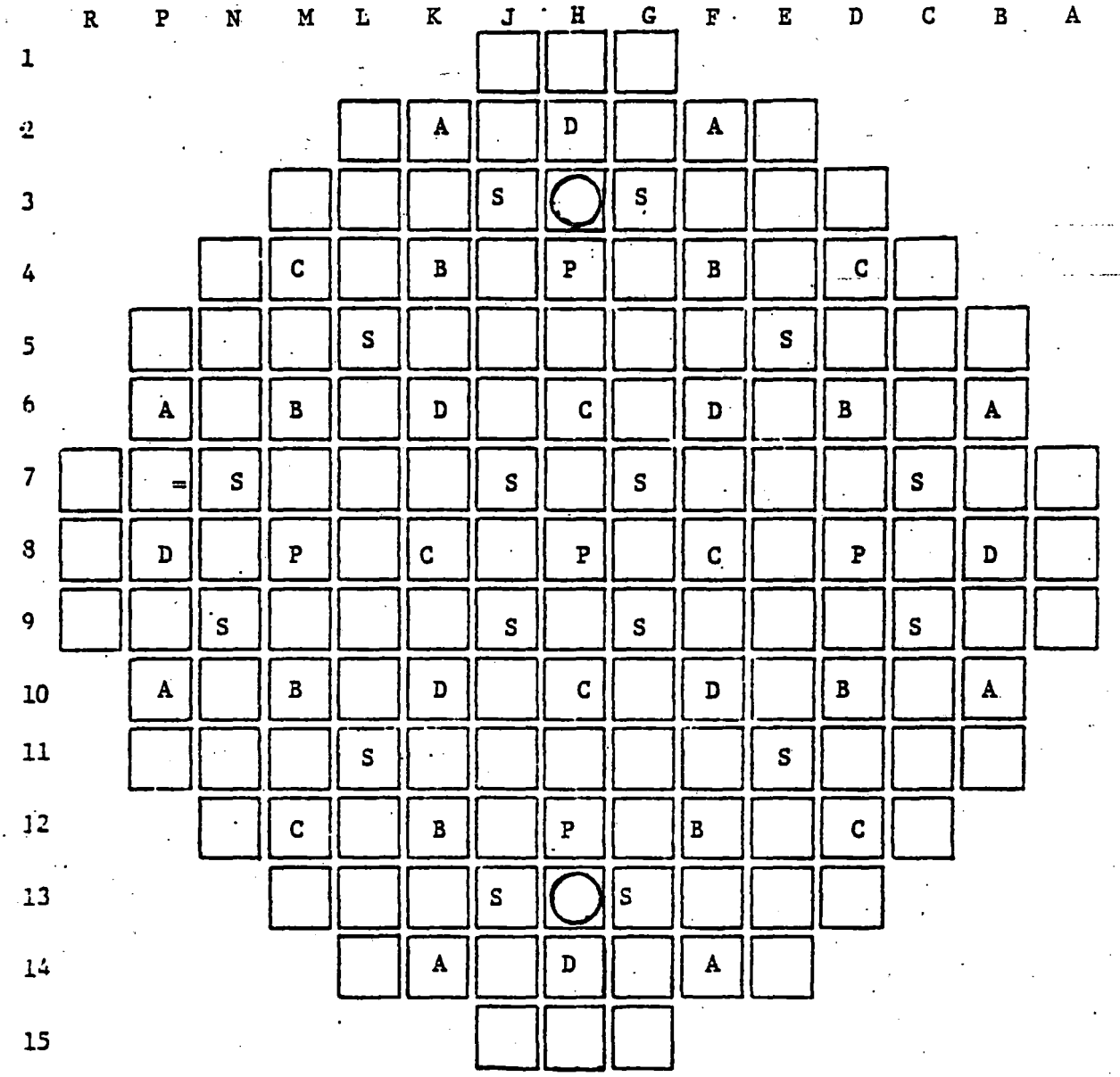


FIGURE 2-2

CONTROL ROD BANK LOCATIONS



CONTROL ROD ASSEMBLY BANKS

Function	Number of Assemblies
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown (S)	16
Part Length (P)	5
	53

◯ = SOURCE ASSEMBLY LOCATIONS

Figure 2-3

SURRY UNITS 1 AND 2 -- CYCLE 1

FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh
J	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh	3 0 Fresh
K	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	
L	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	3 0 Fresh	
M	1 0 Fresh	2 0 Fresh	1 0 Fresh	2 0 Fresh	1 0 Fresh	3 0 Fresh		
N	2 0 Fresh	1 0 Fresh	2 0 Fresh	3 0 Fresh	3 0 Fresh			
P	1 0 Fresh	3 0 Fresh	3 0 Fresh	3 0 Fresh				
R	3 0 Fresh	3 0 Fresh						

Batch No.	Enrichment w/o	Density %
1	1.85	94
2	2.55	93
3	3.10	92

LEGEND

xx	---Batch No.
yy	---Initial Burnup (MWD/MTU)
zz	---Previous Location (If applicable)

Figure 2-4

SURRY UNIT 1 -- CYCLE 2
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 15360 H08	4B 0 Fresh	1 12540 H14	4B 0 Fresh	2 15300 H13	4B 0 Fresh	1 14680 H12	4C 0 Fresh
J	4B 0 Fresh	2 16430 L08	4A 0 Fresh	2 14300 K13	2 16070 K11	2 14380 L12	4C 0 Fresh	4C 0 Fresh
K	1 12540 P08	4A 0 Fresh	1 11280 M12	4A 0 Fresh	2 15920 J12	1 14190 K12	4C 0 Fresh	
L	4B 0 Fresh	2 14300 N10	4A 0 Fresh	2 16760 J08	2 16600 J10	4C 0 Fresh	4C 0 Fresh	
M	2 15300 N08	2 16070 L10	2 15920 M09	2 16600 K09	4A 0 Fresh	4C 0 Fresh		
N	4B 0 Fresh	2 14380 M11	1 14190 M10	4C 0 Fresh	4C 0 Fresh			
P	1 14680 M08	4C 0 Fresh	4C 0 Fresh	4C 0 Fresh				
R	4C 0 Fresh	4C 0 Fresh						

Batch No.	Enrichment w/o	Density %
1	1.85	94
2	2.55	93
4A	1.85	95
4B	2.60	95
4C	3.35	95

LEGEND

xx	---Batch No.
yy	---Initial Burnup (MWD/MTU)
zz	---Previous Location (If applicable)

Figure 2-5

SURRY UNIT 2 — CYCLE 2
FUEL LOADING

	08	09	10	11	12	13	14	15
H	1 16690 H08	3 15420 G14	3 11410 H15	2 17990 H11	4A 0 Fresh	2 18300 H09	2 16810 H13	4B 0 Fresh
J	3 15420 P09	4A 0 Fresh	2 17630 K11	4A 0 Fresh	2 15750 K13	3 14730 N11	4B 0 Fresh	4B 0 Fresh
K	3 11410 R08	2 17630 L10	3 15420 J14	2 18150 J10	4A 0 Fresh	2 17470 J12	4B 0 Fresh	
L	2 17990 L08	4A 0 Fresh	2 18150 K09	4A 0 Fresh	2 15860 L12	4B 0 Fresh	4B 0 Fresh	
M	4A 0 Fresh	2 15750 N10	4A 0 Fresh	2 15860 M11	4A 0 Fresh	4B 0 Fresh		
N	2 18300 J08	3 14730 L13	2 17470 M09	4B 0 Fresh	4B 0 Fresh			
P	2 16810 N08	4B 0 Fresh	4B 0 Fresh	4B 0 Fresh				
R	4B 0 Fresh	4B 0 Fresh						

Batch No.	Enrichment w/o	Density %
1	1.85	94
2	2.55	93
3	3.10	92
4A	2.60	94
4B	3.10	95

LEGEND

xx	---Batch No.
yy	---Initial Burnup (MWD/MTU)
zz	---Previous Location (If applicable)

Figure 2-6

SURRY UNITS 1 AND 2 — CYCLE 1
Burnable Poison Rod Loading

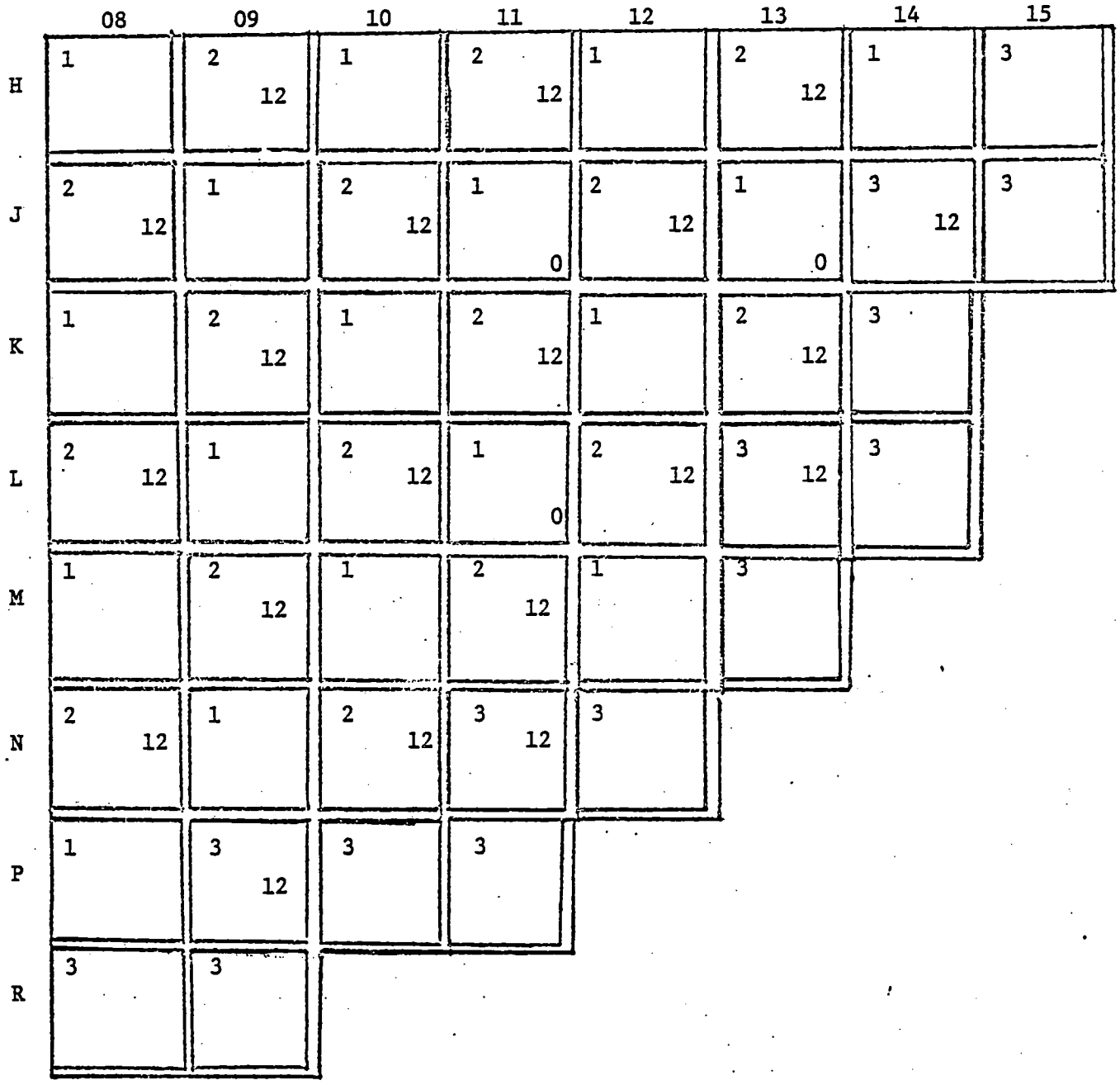


Figure 2-7

SURRY UNIT 1 -- CYCLE 2
BURNABLE POISON ROD LOADING

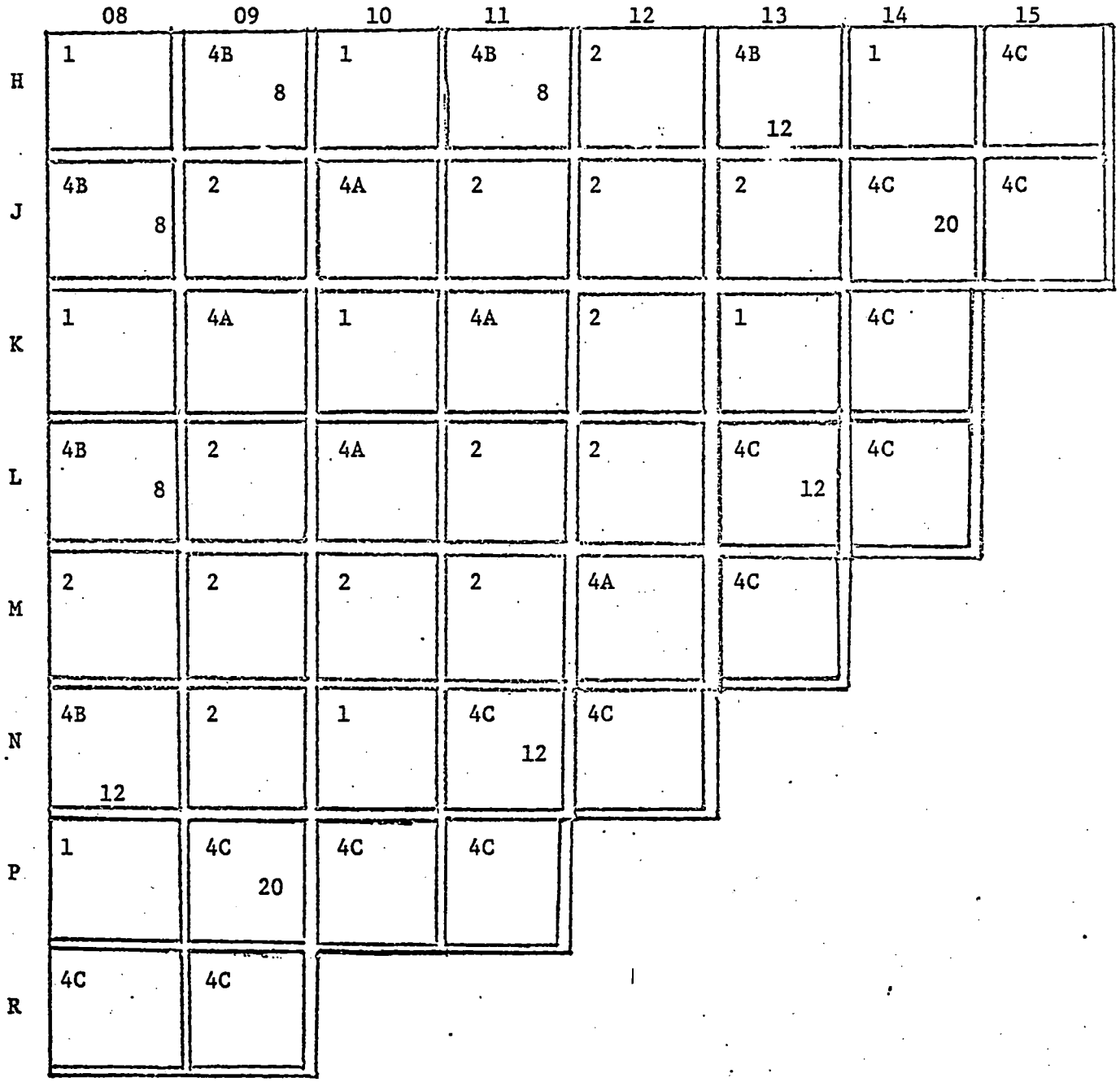
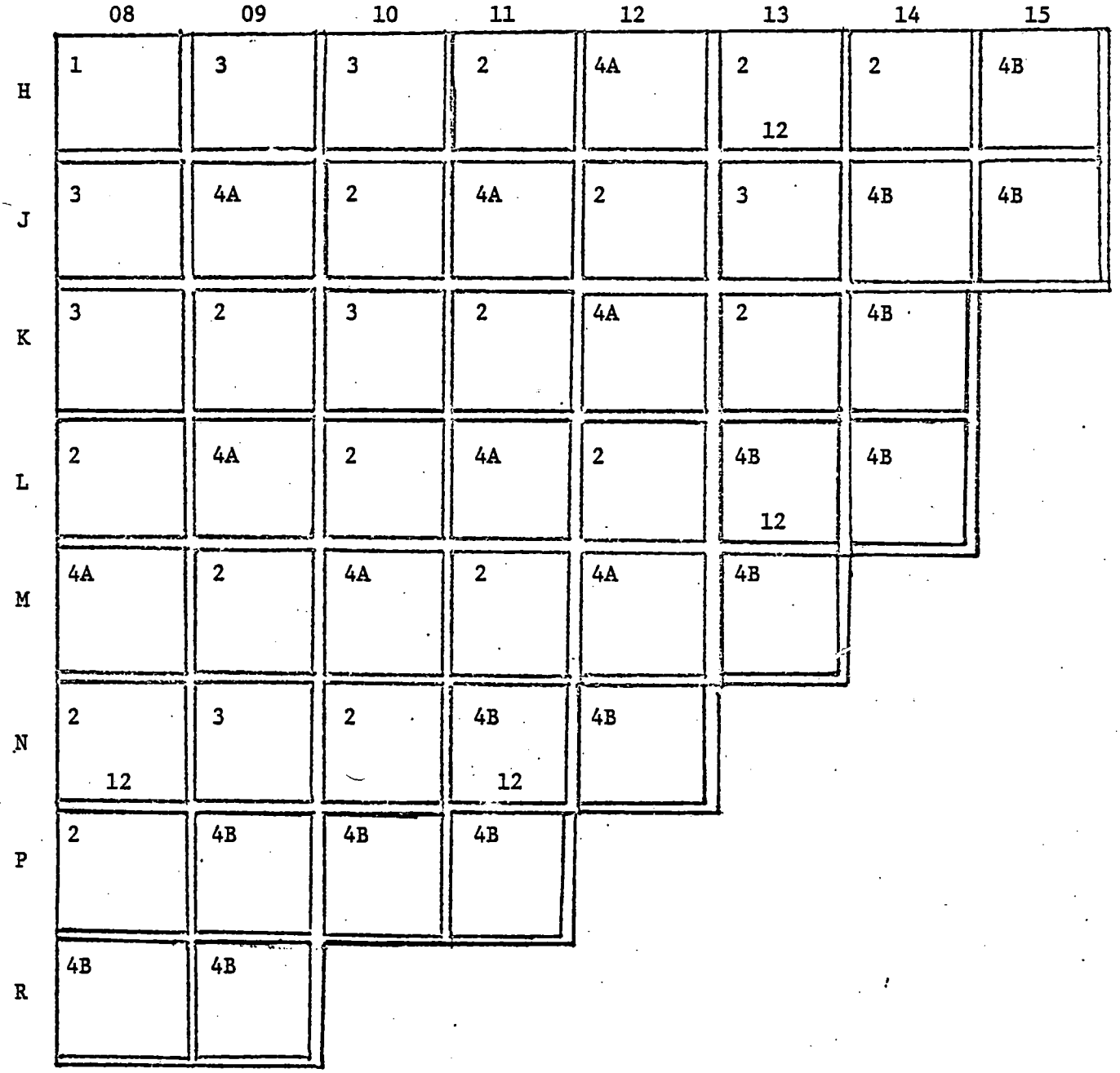


Figure 2-8

SURRY UNIT 2 - CYCLE 2
BURNABLE POISON ROD LOADING



SECTION 3 - MODEL DESCRIPTION

3.1 INTRODUCTION

The PDQ07 discrete model performs a two-dimensional (x-y) diffusion-depletion calculation in which each fuel rod, burnable poison rod, control rod, control rod guide tube and the instrument tube within each fuel assembly as well as the core baffle and radial reflector in one quarter of the reactor core are represented explicitly, and in which thermal-hydraulic feedback effects are considered. Since a detailed simultaneous calculation of the neutron flux as a function of both energy and position is impractical for large power reactor cores (due to the computational time that would be required), the neutron flux for each material composition in the reactor is first calculated as a function of neutron energy, and then this flux spectrum is used to prepare diffusion theory cross sections by collapsing the spectrum-weighted cross sections into two neutron energy groups (denoted as the fast and thermal energy groups or simply two groups). The material compositions represented in these calculations consist primarily of either a fuel rod, a control rod guide or instrumentation tube, a control rod, or a burnable poison rod, and the moderator-coolant associated with the rod. These material compositions are denoted as unit cells, since a lattice of appropriately chosen unit cells can be used to represent the geometry and material composition of the reactor core. The heterogeneous effects of the unit cell on the neutron flux are represented through the use of Dresner's method in the fast energy range and the ABH (Amouyal, Benoist, and Horowitz) approximation to transport theory in the thermal range. Fine energy multi-group cross sections (31 epithermal and 80 thermal) are required for the calculation of neutron flux in the unit cell as a function of energy prior to collapsing into two-group cross sections.

The spectrum-weighted two-group cross sections associated with each unit cell as well as the core baffle and reflector are then used in an

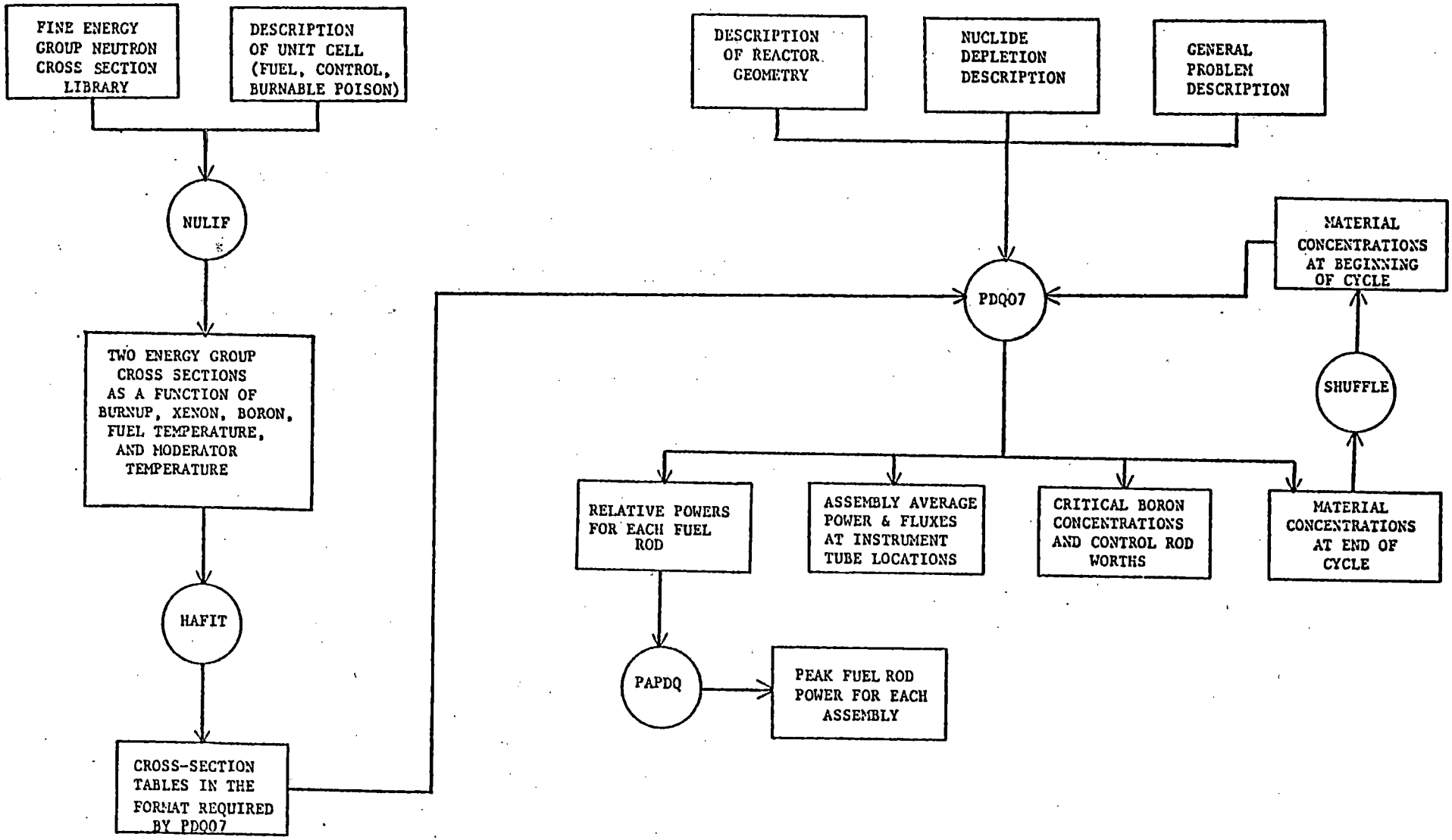
iterative diffusion theory calculation of the neutron flux as a function of radial position. From the neutron flux and cross sections the core power distribution is determined, and subsequently, the fuel and moderator temperature distributions are calculated. Thermal feedback effects are included in the diffusion theory calculation by recalculating the neutron cross sections, power distribution, and fuel and moderator temperature distribution iteratively until both the required nuclear and thermal convergences are achieved.

The neutron flux in a reactor is not only a function of energy and position but is also a function of changes in the nuclide concentrations which vary with burnup. A nuclide depletion calculation is performed based on the two-group fluxes and microscopic absorption and fission cross sections representing each fuel rod in the reactor core. The neutron flux is then recalculated in the diffusion theory calculation for the new nuclide concentrations.

Several interrelated computer codes are used to perform the calculations outlined above. The computer codes comprising the PDQ07 discrete model and their interrelatonships are presented in the flow chart of Figure 3-1. The PDQ07 computer code itself is the principal reactor analysis calculational tool in the PDQ07 discrete model and is used to perform the two-group, two-dimensional diffusion theory calculations. The other codes provide either input data, data manipulation, or use the PDQ07 code output. As indicated in Figure 3-1, the NULIF computer code is used to calculate the required two-group spectrum-weighted cross sections. The HAFIT computer code formats these cross sections for use in the PDQ07 code (as HARMONY table sets). The PDQ07 code calculates relative radial power distributions based on the neutron fluxes at each corner of each fuel rod. These values

Figure 3-1

FLOWCHART FOR THE PDQ07 DISCRETE MODEL



are averaged by the PAPDQ code to determine the relative power density in each fuel rod. PAPDQ is then used to survey the relative power density in each fuel rod within the fuel assembly to determine the peak fuel rod relative power density. The SHUFFLE computer code is a data manipulation code that takes appropriate end of cycle nuclide concentrations from the PDQ07 computer code and shuffles this data in the reactor core according to a specified scheme which duplicates calculationally the actual replacement and movement of fuel assemblies in the reactor core as the result of a refueling.

The details of the calculations performed by each of the computer codes in the PDQ07 discrete model are described in the remainder of this section.

3.2 NULIF AND HAFIT

3.2.1 FINE ENERGY GROUP CROSS SECTION DATA:

The source of basic nuclear cross section data for the NULIF computer code calculations is the standard fine-group cross section library used at Babcock and Wilcox (see Reference 1). This cross section library was supplied by Babcock and Wilcox as part of the FUPAC system.

The library contains cross sections for 31 fast and 80 thermal energy groups with a thermal energy cutoff of 1.85 eV. The fast library contains smooth cross sections, resonance parameters, and an (n, 2n) inelastic scattering matrix for each nuclide. The thermal library contains temperature-dependent cross sections for each thermal energy group and temperature-dependent thermal scattering kernels (both isotropic and anisotropic kernels for the bound atom model). The contents of the files in the cross section library are listed in Table 3-1.

The standard fine-group cross section library contains cross section data for all structural materials, fissionable isotopes, fission pro-

Table 3-1

CONTENTS OF FINE-ENERGY GROUP CROSS SECTION LIBRARY

FILE 1 GENERAL LIBRARY DATA

TAPE LABEL
MATERIAL CONTENTS
EPITHERMAL GROUP STRUCTURE
THERMAL GROUP STRUCTURE
DELAYED NEUTRON PRECURSOR DATA
FISSION SOURCE DISTRIBUTION DATA
GENERAL MATERIAL PARAMETERS
TEMPERATURE LIST
FISSION PRODUCT YIELDS
RESONANCE ISOTOPE DATA
FISSION SPECTRUM DATA
DELAYED NEUTRON DATA

FILE 2 FAST CROSS SECTION DATA

GROUP DATA
GENERAL MATERIAL PARAMETERS
GENERAL UNRESOLVED RESONANCE DATA
UNRESOLVED RESONANCE PARAMETERS
RESOLVED RESONANCE PARAMETERS
SMOOTH CROSS SECTION DATA

FILE 3 THERMAL CROSS SECTION DATA

GENERAL MATERIAL PARAMETERS
SLOWING-DOWN SOURCE DATA
SMOOTH CROSS SECTION DATA
ISOTROPIC SCATTERING KERNEL
ANISOTROPIC SCATTERING KERNEL

ducts, and moderator-coolant (water) used in the reactor core. The constituents of the library are listed in Table 3-2.

The NULIF code is used to calculate composition-dependent energy spectra and then collapse the fine-energy group cross sections to produce two-group cross sections for each unit cell.

3.2.2 NULIF:

The NULIF computer code is used to calculate two-group spectrum-weighted neutron cross sections for each unit cell type in the reactor core. A unit cell consists of either a fuel rod, a control rod guide tube, a control rod, or a burnable poison (BP) rod, and the moderator-coolant (water) associated with the rod.

The neutron energy spectrum for each unit cell is computed using a P1 multi-group approximation to the neutron transport equation. The slowing down treatment for hydrogen is exact, the Fermi age model is used for heavy elements, the Grueling-Guertzel model is used for light elements, and (n, n), (n, 2n), and (n, 3n) reaction effects are included. Resonance absorption is computed by Dresner's method using Sauer's approximation to compute the Dancoff correction for close-packed pin lattices; spectrum reduction corrections are applied for groups containing multiple resonances. The ABH approximation to transport theory is used to calculate the thermal (below 1.85 eV) neutron flux spatial distribution. In the upscattering or thermal energy range, bound atom scattering kernels are used for the principal scatterers and Doppler-broadened group cross sections are used for resonance absorbers. Two-group cross sections for each isotope of each unit cell are obtained by spectrum-weighted integrals. (See Reference 1 for a more detailed discussion of the above assumptions.)

The 1.85 eV energy cutoff is used in NULIF for the thermal energy group so that the low energy resonances of the plutonium isotopes will be

Table 3-2

FINE ENERGY GROUP CROSS SECTION LIBRARY CONSTITUENTS

HYDROGEN-1	PROMETHIUM-149
BORON-10	SAMARIUM-149
BORON-11	URANIUM-234
CARBON-12	URANIUM-235
NITROGEN-14	URANIUM-236
OXYGEN-16	URANIUM-238
SODIUM-23	NEPTUNIUM-237
NATURAL MAGNESIUM	NEPTUNIUM-239
ALUMINUM-27	PLUTONIUM-239
NATURAL SILICON	PLUTONIUM-240
NATURAL CHLORINE	PLUTONIUM-241
NATURAL POTASSIUM	PLUTONIUM-242
NATURAL CALCIUM	AMERICIUM-241
NATURAL CHROMIUM	AMERICIUM-243
MANGANESE-55	BURNABLE POISON (B10)
NATURAL IRON	NON-SAT U233 FISSION PRODUCTS
NATURAL NICKEL	RAP-SAT U233 FISSION PRODUCTS
NATURAL ZIRCONIUM	SLOW-SAT U233 FISSION PRODUCTS
NATURAL MOLYBDENUM	NON-SAT U235 FISSION PRODUCTS
SILVER-107	RAP-SAT U235 FISSION PRODUCTS
SILVER-109	SLOW-SAT U235 FISSION PRODUCTS
CADMIUM-113	NON-SAT PU239 FISSION PRODUCTS
IODINE-135	RAP-SAT PU239 FISSION PRODUCTS
XENON-135	SLOW-SAT PU239 FISSION PRODUCTS

included in the thermal group. Reactor cores containing significant quantities of plutonium are represented more accurately when these low energy resonances are included in the thermal group.

NULIF calculates the neutron flux in the unit cell for each of 31 fast and 80 thermal energy fine-groups. Macroscopic and microscopic cross sections are then determined for one fast and one thermal energy group by collapsing these 111 fine-groups based on the neutron flux and cross sections calculated for each fine-group. Cross sections are collapsed into two groups for use in PDQ07 calculations because it has been determined that the use of two groups is adequate for large thermal reactors (such as the Surry reactors) and the use of more energy groups in PDQ07 would result in substantially longer computer execution times.

The neutron energy spectrum calculated by NULIF for a unit cell depends on the material concentrations (i.e., the nuclide concentration or number density) in the unit cell. The material concentrations change during the operation of the reactor as a result of:

1) Depletion of the material
2) Changes in the soluble boron (chemical shim) and xenon concentrations
3) Changes in material temperature

The neutron spectrum is also dependent on the temperature of the fuel due to Doppler broadening of the resonance absorption peaks. The NULIF code is used to calculate the effect of both changes in material concentrations and in the fuel and moderator temperatures on the neutron spectrum and spectrum-weighted two-group cross section.

NULIF calculates the depletion of unit cells based on the spectrum-weighted neutron cross sections and the neutron flux. As the material is depleted, the material concentrations change. This change in concentrations affects both the neutron flux and the neutron spectrum and, therefore, requires the frequent recalculation of the spectrum-weighted cross sections.

3.2.3 HAFIT:

The HAFIT computer code is a data manipulation code which is used to prepare HARMONY cross section table sets for input to PDQ07. The HARMONY cross section interpolation and depletion routines have been incorporated into the PDQ07 computer code to represent spectrum-weighted cross sections and to perform material depletion calculations. The input to the HAFIT program consists of a magnetic computer tape containing the spectrum-weighted cross sections calculated by the NULIF code, and a description of how these cross sections are to be used to create a set of HARMONY tables for input to PDQ07. An automated data processing code like HAFIT must be used to prepare the HARMONY table sets for PDQ07 because of the substantial volume of data involved.

The HARMONY system used in PDQ07 for representation of neutron cross sections is based on the following equation:

$$\Sigma_{t_g} = \Sigma'_{t_g} + \sum_{i=1}^{m} \{N_i G_{t_{i,g}} \sigma_{t_{i,g}}\}$$

where t = type (transport, absorption, removal, fission) of cross section

g is the neutron energy group

N is the material concentration for nuclide i

G is the self-shielding factor

m is the total number of nuclides

Σ and σ are the macroscopic and microscopic cross sections

Σ'_{t_g} is either a fixed value or interpolating table for the macroscopic cross section

The HARMONY system is designed to represent cross sections as a function of several independent variables. The macroscopic, microscopic, and

G-factor terms in the above equation can be input either as fixed values or in tabular form as a function of up to three independent variables, such as fuel burnup, xenon concentration, and soluble boron concentration. The G-factors are not currently used in the PDQ07 discrete model, since the self-shielding effect is already represented in the cross sections obtained from the NULIF code.

For fuel rod cells, it is necessary to represent cross sections as a function of five independent variables - burnup, boron concentration, xenon concentration, average fuel temperature, and average moderator temperature. Since only three variables (i.e., burnup, xenon concentration, and boron concentration) can be represented in the HARMONY system, a mechanism for extending this representation (i.e., to include fuel and moderator temperature) is necessary. This is accomplished through the use of "pseudo" cross sections which are actually partial derivatives of the macroscopic cross section as a function of either moderator temperature or the square root of fuel temperature. These "pseudo" cross sections are treated in the same way as normal cross sections except that the nuclide number density for these cross sections is either the difference between the moderator temperature for the fuel cell and the nominal average moderator temperature or the difference between the square root of the average fuel temperature for the fuel cell and the square root of the nominal average fuel temperature. The macroscopic cross sections are determined for the nominal average fuel and moderator temperature and then corrected to the actual values through use of the appropriate "pseudo" cross sections.

For each type of table used in the HARMONY system, a table (referred to as a "mask") must be generated which gives values of the independent variables (for example - burnup, boron, and xenon concentration)

for each dependent variable data entry. The dependent variables in the HARMONY system, which are the macroscopic and microscopic cross sections and G-factors, are determined for independent variable values which do not appear in the mask by interpolation between the values which are represented in the mask. A detailed description of the HARMONY system can be found in Reference 2.

3.2.4 GENERATION OF FUEL UNIT CELL CROSS SECTIONS:

The input to the NULIF computer code for a fuel unit cell consists of:

1) Fuel cell dimensions (pellet diameter, clad inside diameter, clad outside diameter, and fuel rod pitch)

2) Material concentration for the fuel pellet, clad, gap, and moderator

3) Average temperatures for the fuel, clad, and moderator

4) Average power density

5) Depletion description including the burnup values at which NULIF calculates the neutron spectrum

NULIF calculations are then made for the unit fuel cell to determine the dependence of the two-group, spectrum-weighted cross sections for each fuel enrichment on:

1) Burnup

2) Soluble boron concentration

3) Xenon concentration

4) Moderator temperature

5) Average fuel temperature

Sets of HARMONY cross section tables based on these NULIF calculations are prepared by the HAFIT code. These tables represent:

1) Microscopic fast and thermal energy group absorption and fission cross sections as a function of burnup, soluble boron concentration, and xenon concentration

2) Macroscopic fast transport and removal, and thermal transport cross sections as a function of burnup, soluble boron concentration, and xenon concentration

3) The effect of fuel and moderator temperature changes on the macroscopic cross sections

3.2.5 GENERATION OF BURNABLE POISON (BP) UNIT CELL CROSS SECTIONS:

The input of the NULIF computer code for the burnable poison (BP) cell consists of:

1) BP cell dimensions

2) Material concentrations for the stainless steel clad, Zircaloy-4 guide tube, pyrex glass, and water

3) Average temperatures for each of the materials above

The dependence of the neutron spectrum of the BP unit cell on the depletion of the Boron-10 in the pyrex glass is determined by performing calculations for a number of Boron-10 concentrations. A set of cross section tables based on these NULIF calculations is prepared by the HAFIT computer code for use by the PDQ07 (HARMONY) code. These tables represent:

1) Microscopic fast and thermal energy group absorption cross sections as a function of Boron-10 concentration

2) Macroscopic fast transport and removal, and thermal transport cross sections as a function of Boron-10 concentration

An adjustment to the cross sections input to PDQ07 for the BP unit cell is made to correct for an under-prediction of the flux depression in the BP unit cell (i.e., over-prediction of the flux) in the PDQ07 computer code calculations. The high thermal absorption cross section for a BP unit cell compared to a fuel unit cell causes a depression of the thermal flux in the BP unit cell and consequently, diffusion theory under-predicts this flux depression when only one mesh block is used to represent each BP unit cell.

To correct for this, an adjustment factor is determined by (1) comparing PDQ07 quarter assembly calculations with 36 mesh blocks (6 x 6) for each unit cell with calculations using one mesh block per unit cell and (2) correcting the under-prediction of the flux depression with the one mesh block per unit cell representation by reducing the thermal absorption cross sections in the BP unit cell, so that the thermal neutron absorption rate in the BP cell is the same for both the one mesh block per unit cell and the 36 mesh blocks per unit cell representations.

3.2.6 GENERATION OF CONTROL ROD UNIT CELL CROSS SECTIONS:

The input to the NULIF computer code for a control rod unit cell consists of:

1) Control rod cell dimensions

2) Material concentrations for the control rod (Ag, In, Cd), stainless steel clad, Zircaloy-4 guide tube, and water

3) Average temperatures for each of the above materials

The NULIF calculations for a control rod determine the macroscopic two-group cross sections for the control rod cell which are input to the PDQ07 computer code.

An adjustment to the control rod unit cell cross sections is calculated in the same way as the adjustment to the BP unit cell cross sections in order to account for a similar under-prediction of the flux depression.

3.2.7 GENERATION OF CONTROL ROD GUIDE TUBE UNIT CELL CROSS SECTIONS:

The input to the NULIF computer code for a control rod guide tube unit cell consists of:

1) Control rod guide tube cell dimensions

2) Material concentrations for Zircaloy-4 guide tube, and water

3) Average temperatures for each of the above materials

The neutron spectrum for the control rod guide tube unit cell is calculated for several soluble boron concentrations to determine the spectrum-weighted cross sections as a function of soluble boron concentration.

A set of cross section tables based on these NULIF calculations is prepared by the HAFIT code to represent the control rod guide tube unit cell cross sections as a function of soluble boron concentration.

3.2.8 GENERATION OF BAFFLE CROSS SECTIONS

The cross sections for the stainless steel baffle surrounding the core are calculated by NULIF. The neutron spectrum and spectrum-weighted cross sections for the baffle are more difficult to calculate due to the three very different material compositions in the baffle area (i.e., fuel cells, stainless steel, and water). In addition, the baffle has a relatively high thermal absorption cross section, and as previously discussed, diffusion theory will not accurately calculate the neutron flux at material interfaces or in highly absorbing media even with correct spectrum-weighted cross sections. Therefore, the cross sections calculated by NULIF for the baffle have to be adjusted to give more accurate calculations of the power density for peripheral assemblies.

The adjustment to the cross sections for the baffle is based on calculations of relative power distributions with the PDQ07 computer code. The power distribution (as calculated by the PDQ07 computer code for the beginning of life condition of Cycle 1 for both Surry units) was compared with the corresponding measured power distribution. The cross sections for the baffle were then adjusted to improve the power distribution comparison for this situation. These adjusted baffle cross sections have provided accurate power distribution calculations for Surry Units 1 and 2, Cycle 1 and 2 as demonstrated in Section 4.

3.2.9 GENERATION OF REFLECTOR CROSS SECTIONS

The cross sections in the reflector region for the Surry reactors are calculated by NULIF. The reflector region of the Surry cores extends from the outside of the core baffle to the reactor vessel wall, including the thermal shield and core barrel. The stainless steel and water in this region of the reactor are homogenized (volume-weighted) in NULIF, and a neutron spectrum and spectrum-weighted cross sections are calculated for this region. These calculations are performed for several soluble boron concentrations, and tables representing the cross sections of the reflector region as a function of soluble boron concentration are prepared for use in PDQ07.

3.3 PDQ07

3.3.1 INTRODUCTION:

The PDQ07 computer code, as used in the PDQ07 discrete calculational model, is a two-dimensional, two-group diffusion-depletion program which is used to calculate the neutron flux, power, and nuclide concentrations as a function of radial (x-y) position and burnup. The PDQ07 computer code utilizes the appropriate microscopic and macroscopic cross sections calculated by NULIF (and as properly formated by HAFIT) along with the initial description of the reactor (i.e., geometry and material composition description) to calculate the neutron flux distribution at discrete spatial mesh points (and for two energy groups) at the desired core power. The spatially dependent neutron flux is then combined with the appropriate nuclide concentrations and cross sections to obtain the spatially dependent power distribution. Once the initial spatially dependent flux and power distributions are obtained, the depletion of the nuclide concentrations is calculated.

3.3.2 GEOMETRY DESCRIPTION:

A detailed description of the geometry input and terminology for the PDQ07 code is given in Section 3 of Reference 2. The geometry description for the discrete PDQ07 computational model is summarized in this section.

One quarter of the Surry reactor core is represented in two dimensions in the discrete PDQ07 model. Only one quarter of the core has to be represented, because of the quarter core symmetry of the fuel loading for the Surry core.

The boundary conditions used in the solution of the diffusion equation in the PDQ07 code are:

1) Zero neutron current for the boundaries located along the core axes

2) Zero neutron flux for the boundaries located at the reactor vessel wall.

The PDQ07 code uses one mesh line and therefore, one mesh block, for each unit cell (i.e., fuel rod, burnable poison (BP) rod, control rod, and control rod guide tube) in the Surry core. The small water gap between assemblies is also represented with one mesh line, so that the increased flux peaking that takes place in the water gap will be calculated correctly. The stainless steel baffle at the outside edge of the fuel in the Surry core is represented with two mesh lines, because the baffle is approximately two fuel rod pitches in thickness. The reflector region outside the baffle of the Surry core is represented with nine mesh lines which extend the region of solution of PDQ07 out to the reactor vessel wall where the zero neutron flux boundary condition is applied. This geometric description results in a total of 17,424 mesh blocks for the quarter core representation, including one for each fuel rod, control rod,

BP rod, and control rod guide tube.

This detailed two-dimensional representation is required so that the relative radial power density of each fuel rod can be calculated in order to determine the radial peaking factor and the enthalpy rise hot channel factor.

3.3.3 DEPLETION EQUATIONS:

Each mesh block in the PDQ07 code contains a single homogenous composition. The volume-weighted nuclide concentrations for each mesh block in the Surry core are input to PDQ07 for beginning of life core conditions. In addition, a set of equations, which is used by PDQ07 to calculate the change in nuclide concentrations with burnup, is input to PDQ07 for each different composition in the Surry core.

The appropriate set of material (nuclide) depletion equations is assigned in PDQ07 to each mesh block. These equations are used by PDQ07 to deplete the nuclide concentrations in each mesh block based on:

1) The average fast and thermal energy group neutron fluxes calculated by PDQ07 for the mesh block

2) The spectrum-weighted fast and thermal group absorption cross sections determined by PDQ07 from the cross section table set assigned to the mesh block

The depletion chains described with these depletion equations in the discrete model for each unit cell type are shown in Table 3-3. A detailed description of how the depletion equations are input to PDQ07 describing these depletion chains is given in Section 5 of Reference 2.

3.3.4 THERMAL-HYDRAULIC FEEDBACK PARAMETERS:

The input to PDQ07 required for thermal-hydraulic feedback consists of:

1) Coolant inlet enthalpy

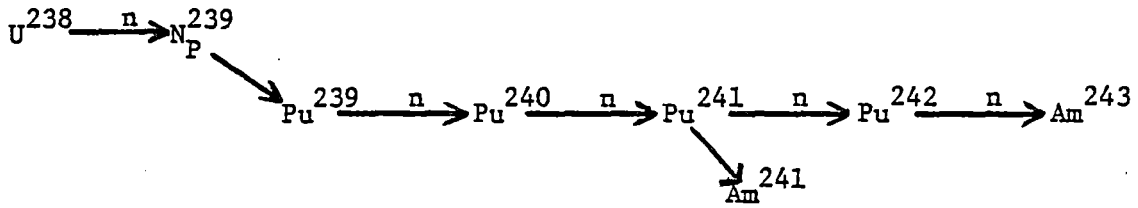
2) Heated perimeter per unit area of flow

Table 3-3

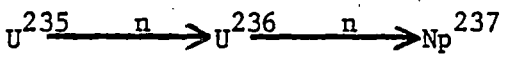
Depletion Equations used in PDQ07

1. Neutron Absorptions Not Leading to Fission

   a. U^{238}, Pu depletion chain



   b. U^{235} depletion chain



2. Neutron absorptions which produce fission are represented with the following fission products:

   a. I^{135}, Xe^{135}, Pm^{149}, and Sm^{149} which are represented explicitly

   b. Two groups of fission products which eventually build up to an equilibrium concentration (since they are created by fission reactions and destroyed by decay reactions). One group is characteristic of fission reactions by uranium isotopes and the other group is characteristic of fission reactions by plutonium isotopes.

   c. Two groups of non-saturating fission products which are either stable isotopes or have half-lives greater than a few years. Again, one group is characteristic of fission reactions by uranium isotopes and the other group is characteristic of fission reactions by plutonium isotopes.

3) Hydraulic diameter of the channels

4) Flow area of the fuel assembly per total area of flow

5) System pressure

6) Difference between average fuel temperature and moderator temperature as a function of relative power denisty

The strategy used in the feedback calculation consists of first making an initial estimate of the fuel and moderator temperature for each coolant channel. Based on this initial estimate and the cross section tables for each fuel cell, the PDQ07 code calculates the two-group, spectrum-weighted cross sections for each mesh block. These cross sections are used in a diffusion theory calculation of power density in each fuel rod cell. This power density is then used in a calculation of the fuel and moderator temperature for each fuel cell. In turn, the new fuel and moderator temperatures are used to calculate new two-group, spectrum-weighted cross sections for another diffusion theory power distribution calculation. This process is continued until the power density for each fuel rod in the N^{th} iteration differs from the power density in the $N-1^{th}$ iteration by less than the convergence criterion.

Thermal-hydraulic feedback effects are represented in the PDQ07 model in order to more accurately calculate the power and burnup distributions.

SECTION 4 - COMPARISON OF MODEL PREDICTIONS WITH MEASUREMENT DATA

4.1 INTRODUCTION

The purpose of this section is to present a comparison between the analytical predictions determined with the PDQ07 discrete model and measurement data obtained from the Surry reactors. Measured reactor data encompassing an initial and reload cycle operation are used in these comparisons to demonstrate both the accuracy and flexibility of the PDQ07 discrete model. A summary of the comparisons is given in Table 4-1.

In addition to comparisons between the Vepco model and reactor measurements, comparisons are made between vendor model predictions and the same measurement data. For these comparisons, only the average absolute value of the percentage difference between the vendor predicted and measured data is given. The comparisons are given so that the accuracy of the Vepco predictions can also be compared with the accuracy of the predictions from an accepted and verified vendor model which has been used in the design, licensing, and core-follow of the Surry reactors.

4.2 ANALYTICAL CALCULATIONS

Calculations presented in this section can be conveniently divided into power distribution calculations and reactivity calculations.

The power distribution calculations include fuel assembly average and peak fuel rod (within each fuel assembly) relative radial power distributions representative of both initial and reload cycle operation, batch burnup sharing, and assemblywise burnups during cycle operation. (it should be noted that the peak fuel rod relative power distribution in the core is considered to be equivalent to the enthalpy rise hot channel factor ($F^N_{\Delta H}$) for

Table 4-1

SUMMARY OF COMPARISONS FOR BOTH THE INITIAL AND RELOAD CYCLES

Core Condition*	Parameter Compared
HZP-BOL	Critical boron concentrations for various control rod bank configurations (i.e., inserted or not inserted)
	Core radial power distributions for various control rod bank configurations
	Control rod bank worths
	Differential Boron Worth
HFP-BOL-ARO	Critical boron concentrations
	Core average radial power distributions
HFP-BOL to EOL-ARO	Critical boron concentration
	Core radial power distributions
	Burnup sharing distribution

*HZP-BOL: Hot zero power - beginning of life
HFP-BOL-ARO: Hot full power - beginning of life - all rods out
HFP-BOL to EOL-ARO: Hot full power - depletion from beginning of life to end of life - all rods out - equilibrium xenon

average core conditions.) The power distribution calculations for cycle operation are calculated at various core operating conditions and burnup intervals. Between these intervals, the core is depleted using the power distribution calculated at the beginning of the depletion step.

Reactivity calculations presented in this section include critical soluble boron concentrations, differential boron worth, and integral control rod bank worth. The critical soluble boron concentration is that concentration of boron which maintains the reactor just critical and is obtained by determining the core eigenvalue (or K_{eff}) from a calculation using a best-estimate boron concentration and then correcting this boron concentration to a value which corresponds to the just critical condition. Differential boron worth is obtained by varying the soluble boron concentration about the critical boron concentration and then determining the resultant impact on K_{eff}. Since the only change made between calculations is in the soluble boron concentration, the change in reactivity due to changes in boron concentration (i.e., the differential boron worth) can be directly calculated. Control rod worths are calculated in similar manner except that the soluble boron concentration is held constant while specific control rod bank positions (e.g., all rods out, D bank fully inserted, C and D bank fully inserted) are changed. The control rod bank change produces a corresponding change in core reactivity which is directly correlated to control rod worth.

The reactivity and power distribution calculations were performed for the following reactor conditions:

1) Hot (547°F), zero power with all rods out, no xenon, D-Bank in, and C and D-Bank in

2) Hot (566°F), full power (2441 Mwt), no xenon and equilibrium xenon, zero and 150 MWD/MTU burnup with all rods out, D-Bank in, and C and D-Bank in

3) Hot (566°F), full power (2441 Mwt), equilibrium xenon, depletion from 150 MWD/MTU to end of life with all control rods out.

4.3 MEASUREMENT DATA

Measurement data is obtained for the Surry units from routine physics testing conducted during the startup of each cycle of operation and from routine core performance monitoring conducted during the depletion of each cycle.

Differential and integral control bank worths are measured by maintaining the reactor approximately critical and monitoring reactivity changes during exchanges of boron concentration with control rod bank position. Specifically, following the establishment of a constant boron dilution/boration rate, one of the control rod banks is periodically inserted/withdrawn in order to provide reactivity compensation for the changing primary coolant system boron concentration. The reactivity changes resulting from the control bank movements are recorded on a continuous basis by the reactivity computer at the reactor. The differential control rod bank reactivity worth is defined as the ratio of the change in reactivity to the corresponding change in bank position about an average bank position, and the integral worth is then obtained by summing the individual reactivity changes between measurement endpoints.

The primary coolant critical boron concentration is monitored during startup physics testing when the control rod banks reach their fully inserted (or withdrawn) endpoints during the control rod bank worth measurements. For this measurement, the reactor conditions are stabilized, and the base just critical boron concentration is determined. To this base value, a slight adjustment for control rod position is made in order to obtain a just critical boron endpoint at the exactly desired control rod bank

configuration. The critical boron concentration is also monitored frequently (i.e., several times a day) during cycle operation. The FOLLOW 8 computer code is used to normalize measured critical boron concentrations obtained during cycle operation to design conditions. (See Appendix A for a more detailed discussion of the FOLLOW computer code.) The normalization process takes into consideration actual control rod configurations, xenon and samarium concentrations, reactivity coefficients, and power levels.

The differential boron worth measurement is also made concurrent with the control rod bank worth measurements. For this measurement, frequent (i.e., every fifteen minutes) primary coolant boron concentrations are obtained and control rod bank positions are noted during the dilution and boration phases of the control rod bank worth measurement. Since the control rod bank positions as a funtion of time can be related to integrated reactivity, a relationship (graph) can be constructed of boron concentration as a function of integrated reactivity. The slope of this relationship is the differential boron worth.

The core power distributions are measured through the use of the movable detector flux mapping system. This system consists of five fission detectors which can traverse assembly instrumentation thimbles in 50 core locations. For each traverse, the detector output is continuously recorded on a strip chart and is also scanned for 61 discrete axial points by the on-site process computer. Two and three-dimensional core power distributions are then determined using the INCORE 9 computer code. (See Appendix A for a more detailed discussion of the INCORE code.) INCORE couples the experimental flux map measurements from the on-site process computer with analytical power-to-flux ratios and power distribution data for each measured and unmeasured location to determine the power distribution for the entire

core, including assembly average and peak rod ($F^N_{\Delta H}$) relative power distribution for each assembly.

The analytical power-to-flux ratio and power distribution data used in the INCORE computer code are determined by the PDQ07 discrete model. The PDQ07 discrete model calculates radial power distribution on a quarter core basis and stores the results of these calculations on magnetic tape or disk (in addition to printing out these results) for use by several data handling codes which prepare the analytical data decks for the INCORE computer code. These data handling codes:

1) Expand the radial power distribution calculations for each rodded configuration from quarter core to full core representation

2) Determine the relative power density of the peak rod in each fuel assembly

3) Prepare input for the INCORE code by giving the predicted radial power for each fuel assembly for each rodded configuration used in normal plant operation

4) Prepare input for the INCORE code by giving the predicted relative radial power for the peak rod in each assembly. When the peak rod changes with control rod bank configuration, the peak rod for each configuration is represented

5) Prepare input for the INCORE code by giving the predicted fast and thermal group neutron fluxes for each instrument tube location

The INCORE code then uses the above analytical data and the incore flux maps obtained from the movable detector flux mapping system to determine power distributions and peaking factors.

The TOTE[10] code is used along with the INCORE code to burnup distributions. (See Appendix A for a more detailed discussion of the TOTE code.) Burnup rate information for each fuel assembly, which is obtained from the INCORE code, and core energy generation rates, which are obtained from heat balances between the hot and cold legs of the secondary cooling system, are inputed into the TOTE code for the burnup

distribution calculation.

4.4 RESULTS

The results of the comparison between the power distribution and reactivity predictions and measurements obtained from the Surry units are presented below. In addition, comparisons between the vendor model predictions and the same reactor measurements are presented.

For all of the power distribution comparisons, which are obtained from the incore movable detector flux mapping system, analytical INCORE data decks have been prepared with both the Vepco PDQ07 discrete model and the vendor model. The accuracy of calculated assembly average power distribution compared to the measured distribution is determined with the INCORE code by calculating the standard deviation (σ) between measured and predicted assembly average radial power distributions with the following equation:

$$\sigma = \{\frac{1}{156} \sum_{i=1}^{157} (X_i^C - X_i^M)^2\}^{1/2}$$

Where: X_i^C is the calculated assembly average power for the i^{th} assembly

and

X_i^M is the measured assembly average power for the i^{th} assembly

The standard deviation of the assembly average power distribution provides a mathematical basis for evaluating the accuracy of the model.

The specific types of results compared are delineated in Table 4-2. For all comparisons, both the measured and the Vepco model predicted values are presented in addition to the standard deviation or percentage difference in these values. For comparisons with the vendor model, only the average absolute value of the standard deviation or percentage difference for all measurements is given. Representative INCORE output comparing the Vepco PDQ07 discrete model predictions and measurements is provided in Appendix B.

Table 4-2

TYPE OF COMPARISONS	REACTOR CONDITION	REFERENCE FOR COMPARISONS
Power Distribution:		
Assembly Average	Cycle Operation for Unit 1, Initial Cycle	Table 4-3
	Cycle Operation for Unit 1, Reload Cycle	Table 4-4
	Cycle Operation for Unit 2, Initial Cycle	Table 4-5
	Cycle Operation for Unit 2, Reload Cycle	Table 4-6
Peak Rod $F^N_{\Delta H}$	Cycle Operation for Unit 1, Initial Cycle	Table 4-7
	Cycle Operation for Unit 1, Reload Cycle	Table 4-8
	Cycle Operation for Unit 2, Initial Cycle	Table 4-9
	Cycle Operation for Unit 2, Reload Cycle	Table 4-10
Assemblywise Burnup and Batch Burnup Sharing	Cycle Operation for Unit 1, Initial Cycle	Table 4-11
	Cycle Operation for Unit 1, Reload Cycle	Table 4-12
	Cycle Operation for Unit 2, Initial Cycle	Table 4-13
	Cycle Operation for Unit 2, Reload Cycle	Table 4-14
Reactivity:		
Critical Boron Concentration For Various Control Rod Configurations	Startup for Units 1 & 2, Initial Cycle	Table 4-15
	Startup for Units 1 & 2, Reload Cycle	Table 4-16
Critical Boron Concentration vs. Burnup	Cycle Operation for Unit 1, Initial Cycle	Figure 4-1
	Cycle Operation for Unit 1, Reload Cycle	Figure 4-2
	Cycle Operation for Unit 2, Initial Cycle	Figure 4-3
	Cycle Operation for Unit 2, Reload Cycle	Figure 4-4
Differential Boron Worth	Startup for Units 1 & 2, Initial Cycle	Table 4-17
	Startup for Units 1 & 2, Reload Cycle	Table 4-18
Control Rod Bank Worths	Startup for Units 1 & 2, Initial Cycle	Table 4-19
	Startup for Units 1 & 2, Reload Cycle	Table 4-20

Table 4-3

COMPARISON OF PREDICTED AND MEASURED ASSEMBLY AVERAGE POWER DISTRIBUTIONS
FOR SURRY UNIT 1, CYCLE 1

M/D Map Number	Power Level (%)	Control Rod Configuration	Cycle Burnup (MWD/MTU)	Vepco Model σ (%)	Vendor Model σ (%)
6	0	ARO	71	2.4	
7	0	D-Bank In	71	2.1	
11	0	C and D-Bank In	71	2.3	
28	75	ARO	339	2.1	
35	75	ARO	620	1.5	
40	75	ARO	1150	1.5	
43	92	ARO	1900	1.8	
45	88	ARO	2380	1.5	
48	90	ARO	3175	1.6	
49	90	ARO	3550	2.1	
51	91	ARO	4190	1.8	
55	95	ARO	5275	2.4	
57	95	ARO	5975	1.9	
59	94	ARO	6790	1.9	
62	95	ARO	7540	1.2	
65	94	ARO	8435	1.4	
66	100	ARO	9150	1.3	
68	100	ARO	10100	1.3	
69	92	ARO	10820	1.3	
70	100	ARO	11875	2.8	
71	100	ARO	12670	2.5	
73	60	ARO	13415	2.1	

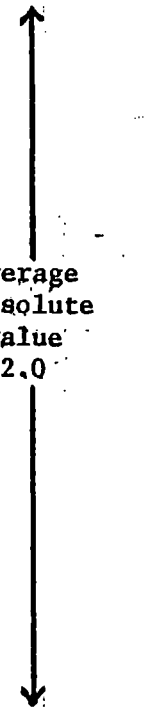


Table 4-4

COMPARISON OF PREDICTED AND MEASURED ASSEMBLY AVERAGE POWER DISTRIBUTIONS
FOR SURRY UNIT 1, CYCLE 2

M/D Map Number	Power Level (%)	Control Rod Configuration	Cycle Burnup (MWD/MTU)	Vepco Model σ (%)	Vendor Model σ (%)
1	3	D-Bank In	0	3.0	
2	3	ARO	0	4.3	
9	99	ARO	127	3.1	
10	100	ARO	307	3.6	
12	98	ARO	1103	3.0	
13	100	ARO	2043	2.8	
16	99	ARO	3102	2.1	
17	99	ARO	4015	2.1	
18	100	ARO	4899	1.8	
19	100	ARO	5612	1.5	
20	100	ARO	6569	1.4	

Average absolute value 2.8

Table 4-5

COMPARISON OF PREDICTED AND MEASURED ASSEMBLY AVERAGE POWER DISTRIBUTIONS
FOR SURRY UNIT 2, CYCLE 1

M/D Map Number	Power Level (%)	Control Rod Configuration	Cycle Burnup (MWD/MTU)	Vepco Model σ (%)	Vendor Model σ (%)
1	0	ARO	0	2.7	
2	0	D-Bank In	0	2.1	
9	0	C and D-Banks In	0	2.2	
23	90	ARO	365	2.2	
27	88	ARO	630	3.0	
31	90	ARO	1300	1.3	
34	88	ARO	2030	1.3	
38	92	ARO	2950	2.9	
40	92	ARO	3780	1.0	
42	94	ARO	4670	1.8	
43	92	ARO	5240	1.7	
45	83	ARO	5940	1.2	
48	91	ARO	6780	1.5	
50	91	ARO	7725	1.9	
52	98	ARO	8580	1.4	
54	98	ARO	9310	1.4	
59	100	ARO	9890	1.1	
61	100	ARO	11025	1.3	
62	100	ARO	11740	1.5	
63	100	ARO	12770	2.1	
64	98	ARO	13650	1.6	
65	100	ARO	14520	2.0	

Average absolute value 2.0

Table 4-6

COMPARISON OF PREDICTED AND MEASURED ASSEMBLY AVERAGE POWER DISTRIBUTIONS
FOR SURRY UNIT 2, CYCLE 2

M/D Map Number	Power Level (%)	Control Rod Configuration	Cycle Burnup (MWD/MTU)	Vepco Model σ (%)	Vendor Model σ (%)
1	2	ARO	0	4.3	
2	2	D-Bank In	0	4.2	
6	100	ARO	170	1.7	
7	100	ARO	730	2.2	
11	100	ARO	1875	2.5	
12	97	ARO	2790	1.6	
15	100	ARO	3750	1.2	
16	100	ARO	4520	1.4	
17	100	ARO	5650	1.4	
20	100	ARO	6875	1.3	
21	100	ARO	7125	1.5	
22	100	ARO	8040	1.6	
23	99	ARO	8850	1.5	

Average absolute value 1.7

Table 4-7

COMPARISON OF PREDICTED AND MEASURED $F^N_{\Delta H}$
FOR SURRY UNIT 1, CYCLE 1

M/D Map Number	Power Level (%)	Control Rod Configuration	Cycle Burnup (MWD/MTU)	Measured $F^N_{\Delta H}$	Vepco Model Predicted $F^N_{\Delta H}$	Vepco Model % Difference	Vendor Model % Difference
6	0	ARO	71	1.363	1.331	-2.3	
7	0	D-Bank In	71	1.418	1.406	-0.8	
11	0	C and D-Bank In	71	1.752	1.734	-1.0	
28	75	ARO	339	1.362	1.329	-2.4	
35	75	ARO	620	1.373	1.345	-2.0	
40	75	ARO	1150	1.377	1.364	-0.9	
43	92	ARO	1900	1.389	1.354	-2.5	
45	88	ARO	2380	1.375	1.345	-1.5	
48	90	ARO	3175	1.360	1.325	-2.6	
49	90	ARO	3550	1.367	1.317	-3.7	
51	91	ARO	4190	1.347	1.304	-3.2	
55	95	ARO	5275	1.313	1.285	-2.1	
57	95	ARO	5975	1.315	1.273	-3.2	
59	94	ARO	6720	1.271	1.260	-0.9	
62	95	ARO	7540	1.277	1.251	-2.0	
65	94	ARO	8435	1.256	1.241	-1.2	
66	100	ARO	9150	1.254	1.233	-1.7	
68	100	ARO	10100	1.228	1.277	-0.1	
69	92	ARO	10820	1.232	1.223	-0.7	
70	100	ARO	11875	1.250	1.214	-2.9	
71	100	ARO	12670	1.218	1.207	-0.9	
73	60	ARO	13415	1.208	1.200	-0.7	

Average absolute value 3.6

Table 4-8

COMPARISON OF PREDICTED AND MEASURED $F^N_{\Delta H}$
FOR SURRY 1, CYCLE 2

M/D Map Number	Power Level (%)	Control Rod Bank Location	Cycle Burnup (MWD/MTU)	Measured $F^N_{\Delta H}$	Vepco Model Predicted $F^N_{\Delta H}$	Vepco Model % Difference	Vendor Model % Difference
1	3	D-Bank In	0	1.653	1.675	1.3	
2	3	ARO	0	1.439	1.480	2.9	
9	99	ARO	127	1.394	1.386	-0.6	
10	100	ARO	307	1.408	1.383	-1.8	
12	98	ARO	1103	1.430	1.368	-4.3	
13	100	ARO	2043	1.416	1.371	-3.2	
16	99	ARO	3102	1.417	1.361	-4.0	
17	99	ARO	4015	1.390	1.350	-2.9	
18	100	ARO	4899	1.390	1.340	-3.6	
19	100	ARO	5612	1.373	1.337	-2.6	
20	100	ARO	6569	1.363	1.332	-2.3	

Average absolute value 1.4

Table 4-9

COMPARISON OF PREDICTED AND MEASURED $F^N_{\Delta H}$
FOR SURRY UNIT 2, CYCLE 1

M/D Map Number	Power Level (%)	Control Rod Bank Location	Cycle Burnup (MWD/MTU)	Measured $F^N_{\Delta H}$	Vepco Model Predicted $F^N_{\Delta H}$	Vepco Model % Difference	Vendor Model % Difference
1	0	ARO	0	1.350	1.331	-1.4	
2	0	D-Bank In	0	1.406	1.405	0.0	
9	0	C and D-Bank In	0	1.704	1.734	+1.8	
23	90	ARO	365	1.366	1.330	-2.6	
27	88	ARO	630	1.354	1.345	-0.7	
31	90	ARO	1300	1.367	1.363	-0.3	
34	88	ARO	2030	1.379	1.355	-1.7	
38	92	ARO	2950	1.350	1.329	-1.6	
40	92	ARO	3780	1.341	1.312	-2.2	
42	94	ARO	4670	1.331	1.295	-2.7	
43	92	ARO	5240	1.331	1.285	-3.5	Average absolute value 3.5
45	83	ARO	5940	1.302	1.274	-2.2	
48	91	ARO	6780	1.288	1.261	-2.1	
50	91	ARO	7725	1.290	1.249	-3.2	
52	98	ARO	8580	1.245	1.239	-0.5	
54	98	ARO	9310	1.238	1.232	-0.5	
59	100	ARO	9890	1.227	1.229	-0.2	
61	100	ARO	11025	1.227	1.222	-0.4	
62	100	ARO	11740	1.220	1.215	-0.4	
63	100	ARO	12770	1.210	1.206	-0.3	
64	98	ARO	13650	1.205	1.198	-0.6	
65	100	ARO	14520	1.200	1.190	-0.8	

Table 4-10

COMPARISON OF PREDICTED AND MEASURED $F^N_{\Delta H}$
FOR SURRY UNIT 2, CYCLE 2

M/D Map Number	Power Level (%)	Control Rod Bank Location	Cycle Burnup (MWD/MTU)	Measured $F^N_{\Delta H}$	Vepco Model Predicted $F^N_{\Delta H}$	Vepco Model % Difference	Vendor Model % Difference
1	2	ARO	0	1.426	1.374	-3.7	
2	2	D-Bank In	0	1.610	1.550	-3.7	
6	100	ARO	170	1.357	1.351	-0.4	
7	100	ARO	730	1.372	1.352	-1.5	
11	100	ARO	1875	1.361	1.341	-1.5	Average absolute value 1.2
12	97	ARO	2790	1.331	1.332	+0.1	
15	100	ARO	3750	1.325	1.321	-0.3	
16	100	ARO	4520	1.304	1.311	+0.5	
17	100	ARO	5650	1.306	1.300	-0.5	
20	100	ARO	6875	1.284	1.291	-0.5	
21	100	ARO	7125	1.297	1.289	-0.6	
22	100	ARO	8040	1.295	1.282	-1.0	
23	99	ARO	8850	1.276	1.275	-0.1	

TABLE 4-11

ASSEMBLYWISE ACCUMULATED BURNUP AND BATCH BURNUP SHARING (10^3 MWD/MTU) FOR THE CYCLE 1 OPERATION OF SURRY UNIT 1

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						8.17 8.41 +2.9	10.42 10.48 +0.6	8.17 8.35 +2.2							1
				8.37 8.74 +4.4	12.22 12.18 -0.3	14.03 13.84 -1.4	12.53 12.35 -1.4	14.03 13.78 -1.8	12.22 12.15 -0.6	8.37 8.67 +3.6					2
			8.97 9.23 +2.9	13.33 13.47 +1.1	14.29 14.44 +1.0	13.86 13.81 -0.4	15.30 14.99 -2.0	13.86 13.82 -0.3	14.29 14.15 -1.0	13.33 13.50 +1.0	8.97 9.46 +5.5				3
		8.97 9.08 +1.2	11.27 11.51 +2.1	14.37 14.58 +1.5	14.19 14.40 +1.5	15.92 15.71 -1.3	14.69 14.69 0.0	15.92 15.80 -0.8	14.19 14.32 +0.9	14.37 14.48 +0.8	11.27 11.48 +1.9	8.97 9.18 +2.3			4
	8.37 8.52 +1.8	13.32 13.07 -1.9	14.37 14.30 -0.5	14.18 14.37 +1.3	16.08 15.85 -1.4	14.98 14.87 -0.7	16.45 16.31 -0.9	14.98 15.09 +0.7	16.08 15.97 -0.7	14.18 14.23 +0.4	14.37 13.89 -3.3	13.32 13.06 -2.0	8.37 8.67 +3.6		5
	12.22 12.36 +1.1	14.29 14.17 -0.8	14.19 14.12 -0.5	16.08 15.48 -3.7	15.04 14.58 -3.1	16.61 16.40 -1.3	15.25 15.36 +0.7	16.61 16.44 -1.0	15.04 14.76 -1.9	16.08 15.87 -1.3	14.19 14.07 -0.9	14.29 14.07 -1.5	12.22 12.27 +0.4		6
8.17 8.54 +4.5	14.03 14.20 +1.2	13.86 13.98 +0.9	15.92 15.84 -0.5	14.98 14.77 -1.4	16.61 16.26 -2.1	15.31 15.28 -0.2	16.77 16.63 -0.8	15.31 15.19 -0.8	16.61 16.44 -1.0	14.98 15.19 +1.4	15.92 16.09 +1.1	13.86 14.01 +1.1	14.03 13.67 -2.6	8.17 8.28 +1.4	7
10.42 10.89 +4.5	12.53 12.78 +2.0	15.30 15.55 +1.6	14.69 14.93 +1.6	16.45 16.41 -0.2	15.25 15.28 +0.2	16.77 16.61 -1.0	15.37 15.32 -0.3	16.77 16.58 -1.1	15.25 15.37 -0.8	16.45 16.43 -0.1	14.69 15.00 +2.1	15.30 15.44 +0.9	12.53 12.58 +0.4	10.42 10.35 -0.7	8
8.17 8.43 +3.2	14.03 14.01 -0.1	13.86 13.89 +0.2	15.92 15.86 -0.4	14.98 15.07 +0.6	16.61 16.35 -1.6	15.31 15.11 -1.3	16.77 16.52 -1.5	15.31 15.23 -0.5	16.61 16.18 -2.6	14.98 14.92 -0.4	15.92 15.92 0.0	13.86 14.11 +1.8	14.03 13.94 -0.6	8.17 8.19 +0.2	9
	12.22 12.03 -1.6	14.29 13.89 -2.8	14.19 14.10 -0.6	16.08 16.04 -0.3	15.04 14.72 -2.1	16.61 16.17 -2.7	15.25 15.01 -1.6	16.61 16.14 -2.8	15.04 14.59 -3.0	16.08 15.75 -2.1	14.19 14.22 +0.2	14.29 14.17 -0.8	12.22 12.22 0.0		10
	8.37 8.64 +3.2	13.32 13.32 0.0	14.37 14.44 +0.5	14.18 14.35 +1.2	16.08 15.85 -1.4	14.98 14.70 -1.9	16.45 15.96 -3.0	14.98 14.61 -2.5	16.08 15.83 -1.6	14.18 14.24 +0.4	14.37 14.47 +0.7	13.32 13.39 +0.5	8.37 8.69 +3.8		11
		8.97 9.41 +4.9	11.27 11.69 +3.7	14.37 14.46 +0.6	14.19 14.18 -0.1	15.92 15.53 -2.5	14.69 14.44 -1.7	15.92 15.54 -2.4	14.19 14.08 -0.8	14.37 14.35 -0.1	11.27 11.54 +2.4	8.97 9.26 +3.2			12
			8.97 9.22 +2.8	13.33 13.15 -1.4	14.29 14.07 -1.5	13.86 13.73 -0.9	15.30 14.91 -2.6	13.86 13.65 -1.5	14.29 14.06 -1.6	13.33 13.19 -1.1	8.97 9.23 +2.9				13
				8.37 8.60 +2.8	12.22 12.62 +3.3	14.03 14.36 +2.4	12.53 12.43 -0.8	14.03 13.62 -2.9	12.22 12.15 -0.6	8.37 8.48 +1.3					14
						8.17 8.75 +7.1	10.42 10.75 +3.2	8.17 8.19 +0.2							15

a
b
c

Vepco Model
MEASURED
% DIFFERENCE

CORE AVERAGE BURNUP = 13,547 MWD/MTU

	Measured	Vepco Model Predicted	Vepco Model Percent Difference	Vendor Model Percent Difference
Batch 1	14.25	14.20	+0.4	
Batch 2	15.46	15.62	-1.0	Average absolute value 1.3
Batch 3	10.93	10.81	+1.1	

TABLE 4-12

ASSEMBLYWISE ACCUMULATED BURNUP AND BATCH BURNUP SHARING (10^3 MWD/MTU) FOR THE CYCLE 2 OPERATION OF SURRY UNIT 1

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						4.92 5.00 +1.6	6.26 6.16 -1.6	4.92 4.82 -2.0							1
				5.25 5.47 +4.2	7.16 7.28 +1.7	6.99 6.35 -9.2	20.49 20.19 -1.5	6.99 6.22 -11.0	7.16 7.05 -1.5	5.25 5.38 +2.5					2
			5.69 6.02 -10.0	7.62 7.44 -2.4	20.04 20.19 +0.8	21.49 21.57 +0.4	8.64 8.27 -4.3	21.49 21.41 -0.4	20.04 20.03 -0.1	7.62 7.39 -3.0	5.69 6.04 +6.2				3
		5.69 6.02 +5.8	6.48 6.85 +5.7	23.26 23.11 -0.6	22.75 22.47 -1.2	23.16 22.73 -1.9	22.84 22.34 -2.2	23.16 22.92 -1.0	22.75 22.63 -0.5	23.26 23.14 -0.5	6.48 6.79 +4.8	5.69 6.04 +6.2			4
	5.25 5.48 +4.4	7.62 7.42 -2.6	23.26 22.97 -1.3	23.69 23.64 -0.2	7.51 7.54 +0.4	22.11 21.98 -0.6	8.60 8.25 -4.1	22.11 21.95 -0.7	7.51 7.67 +2.1	23.69 23.53 -0.7	23.26 22.98 -1.2	7.62 7.43 -2.5	5.25 5.77 +9.9		5
	7.16 7.22 +0.8	20.04 19.82 -1.1	22.75 22.61 -0.6	7.51 7.56 +0.7	18.31 18.44 +0.7	7.97 8.04 +0.9	19.64 19.42 -1.1	7.97 8.17 +2.5	18.31 18.47 +0.9	7.51 7.62 +1.5	22.75 22.84 +0.4	20.04 19.94 -0.5	7.16 7.61 +6.3		6
4.92 5.16 +4.9	6.99 6.48 -7.3	21.49 21.29 -0.9	23.16 22.47 -3.0	22.11 21.92 -0.9	7.97 8.11 +1.8	24.34 24.29 -0.2	8.79 8.70 -1.0	24.34 24.34 0.0	7.97 8.20 +2.9	22.11 21.96 -0.7	23.16 22.93 -1.0	21.49 21.07 -2.0	6.99 6.60 -5.6	4.92 5.11 +3.9	7
6.26 6.48 +3.5	26.49 20.60 +0.5	8.64 8.49 -1.7	22.84 23.03 +0.8	8.60 8.41 -2.2	19.64 19.86 +1.1	8.79 8.73 -0.7	22.26 22.09 -0.8	8.79 8.59 -2.3	19.64 19.72 +0.4	8.60 8.50 -1.2	22.84 22.99 +0.7	8.64 8.60 -0.5	20.49 20.67 +0.9	6.26 6.38 +1.9	8
4.92 5.16 +4.9	6.99 6.49 -7.2	21.49 21.47 -0.1	23.16 23.04 -0.5	22.11 21.70 -1.9	7.97 8.10 +1.6	24.34 24.26 -0.3	8.79 8.53 -3.0	24.34 23.69 -2.7	7.97 7.95 -0.3	22.11 21.92 -0.9	23.16 22.67 -2.1	21.49 21.54 +0.2	6.99 6.51 -6.9	4.96 5.07 +3.1	9
	7.16 7.25 +1.3	20.04 19.82 -1.1	22.75 22.60 -0.7	7.51 7.64 +1.7	18.31 18.69 +2.1	7.97 8.09 +1.5	19.64 19.39 -1.3	7.97 7.90 -0.9	18.31 18.40 +0.5	7.51 7.49 -0.3	22.75 22.52 -1.0	20.04 19.94 -0.5	7.16 7.38 +3.1		10
	5.25 5.53 +5.3	7.62 7.48 -1.8	23.26 23.08 -0.8	23.69 23.55 -0.6	7.51 7.65 +1.9	22.11 21.85 -1.2	8.60 8.33 -3.1	22.11 21.76 -1.6	7.51 7.61 +1.3	23.69 23.43 -1.1	23.26 22.85 -1.8	7.62 7.44 -2.4	5.25 5.53 +5.3		11
		5.69 6.17 +8.4	6.48 6.95 +7.3	23.26 22.91 -1.5	22.75 22.35 -1.8	23.16 22.85 -1.3	22.84 22.31 -2.3	23.16 22.72 -1.9	22.75 22.25 -2.2	23.26 22.86 -1.72	6.48 6.94 +7.0	5.69 6.07 +6.7			12
			5.69 6.11 +7.4	7.62 7.42 -2.6	20.04 19.97 -0.4	21.49 21.63 +0.7	8.64 8.35 -3.4	21.49 21.44 -0.2	20.04 19.81 -1.1	7.62 7.41 -2.8	5.69 6.11 +7.4				13
				5.25 5.49 +4.6	7.16 7.43 +3.8	6.99 6.53 -6.6	20.49 20.16 -1.6	6.99 6.71 -4.0	7.16 7.46 +4.2	5.25 5.41 +3.1					14
						4.92 5.20 +5.7	6.26 6.57 +5.0	4.92 5.28 +7.3							15

a Vepco Model
b MEASURED
c % DIFFERENCE

CORE AVERAGE BURNUP = 6915 MWD/MTU

	Measured	Vepco Model Predicted	Vepco Model Percent Difference	Vendor Model Percent Difference
Batch 1A	19.63	19.82	-1.0	Average
Batch 2	22.60	22.80	-0.9	absolute
Batch 4A	7.60	7.49	+1.5	value 2.8
Batch 4B	8.40	8.19	+2.6	
Batch 4C	6.25	6.27	-0.3	

ASSEMBLYWISE ACCUMULATED BURNUP AND BATCH BURNUP SHARING (10^3 MWD/MTU) FOR THE CYCLE 1 OPERATION OF SURRY UNIT 2

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						8.96 9.34 +4.2	11.40 11.50 +0.9	8.96 9.11 +1.7							1
				9.82 9.57 -2.6	13.40 13.58 +1.3	15.40 15.34 -0.4	13.75 13.60 -1.1	15.40 15.13 -1.8	13.40 13.31 -0.7	9.82 9.41 -4.2					2
			9.87 10.09 +2.2	14.69 14.63 -0.4	15.73 15.65 -0.5	15.21 15.15 -0.4	16.80 16.56 -1.4	15.21 15.08 -0.9	15.73 15.62 -0.7	14.69 14.68 -0.1	9.87 10.15 +2.8				3
		9.87 10.06 +1.9	12.42 12.41 -0.1	15.83 15.62 -1.3	15.59 15.42 -1.1	17.48 17.35 -0.7	16.10 16.03 -0.4	17.48 17.45 -0.2	15.59 15.61 +0.1	15.53 15.97 +0.9	12.42 12.53 +0.9	9.87 10.13 +2.6			4
	9.82 9.50 -3.3	14.69 14.56 -0.9	15.83 15.69 -0.9	15.58 15.39 -1.2	17.65 17.28 -2.1	16.40 16.20 -1.2	18.02 17.98 -0.2	16.40 16.43 +0.2	17.65 17.56 -0.5	15.58 15.62 +0.3	15.83 15.76 -0.5	14.69 14.79 +0.7	9.82 9.66 -1.6		5
	13.40 13.69 +2.2	15.73 15.71 -0.1	15.59 15.44 -1.0	17.65 17.30 -2.0	16.47 15.98 -3.0	18.19 17.96 -1.3	16.68 16.62 -0.4	18.19 18.09 -0.6	16.47 16.15 -1.9	17.65 17.49 -0.9	15.59 15.49 -0.6	15.73 15.67 -0.5	13.40 13.56 +1.2		6
8.96 9.47 +5.7	15.40 15.57 +1.1	15.21 15.22 +0.1	17.48 17.45 -0.2	16.40 16.26 -0.9	18.19 17.94 -1.4	16.74 16.51 -1.4	18.35 18.21 -0.8	16.74 16.65 -0.5	18.19 18.15 -0.2	16.40 16.27 -0.8	17.48 17.26 -1.3	15.21 15.24 +0.2	15.40 15.22 -1.2	8.96 9.20 +2.7	7
11.40 11.77 +3.3	13.75 13.85 +0.7	16.80 16.91 +0.7	16.10 16.17 +0.4	18.02 18.10 +0.4	16.68 16.57 -0.7	18.35 18.11 -1.3	16.80 16.58 -1.3	18.35 18.22 -0.7	16.68 16.73 +0.3	18.02 17.91 -0.6	16.10 16.07 -0.2	16.80 17.03 +1.4	13.75 13.90 +1.1	11.40 11.55 +1.3	8
8.96 9.49 +5.9	15.40 15.54 +0.9	15.21 15.22 +0.1	17.48 17.52 +0.2	16.40 16.34 -0.4	18.19 18.03 -0.9	16.74 16.47 -1.6	18.35 18.12 -1.3	16.74 16.49 -1.5	18.19 17.86 -1.8	16.40 16.22 -1.1	17.48 17.35 -0.7	15.21 15.34 +0.9	15.40 15.54 +0.9	8.96 9.25 +3.2	9
	13.40 13.48 +0.6	15.73 15.62 -0.7	15.59 15.52 -0.15	17.65 17.48 -1.0	16.47 16.09 -2.3	18.19 17.90 -1.6	16.68 16.47 -1.3	18.19 17.85 -1.9	16.47 15.98 -3.0	17.65 17.31 -1.9	15.59 15.34 -1.6	15.73 15.53 -1.3	13.40 13.53 +1.0		10
	9.82 9.58 -2.4	14.69 14.82 +0.9	15.83 15.91 +0.5	15.58 15.48 -0.6	17.65 17.40 -1.4	16.40 16.22 -1.1	18.02 17.90 -0.7	16.40 16.22 -1.1	17.65 17.40 -1.4	15.58 15.49 -0.6	15.83 15.99 +1.0	14.69 14.88 +1.3	9.82 9.60 -2.2		11
		9.87 10.29 +4.3	12.42 12.65 +1.9	15.83 15.86 +0.2	15.59 15.44 -1.0	17.48 17.24 -1.4	16.10 15.92 -1.1	17.48 17.28 -1.1	15.59 15.43 -1.0	15.83 15.93 +0.6	12.42 12.84 +3.4	9.87 10.34 +4.8			12
			9.87 10.10 +4.3	14.69 14.51 -1.2	15.73 15.52 -1.3	15.21 15.02 -1.3	16.80 16.55 -1.5	15.21 14.98 -1.5	15.73 15.58 -1.0	14.69 14.78 +0.6	9.87 10.37 +5.1				13
				9.82 9.39 -4.4	13.40 13.40 0.0	15.40 15.18 -1.4	13.75 13.43 -2.3	15.40 15.08 -2.1	13.40 13.40 0.0	9.82 9.41 -4.2					14
						8.96 9.09 +1.5	11.40 11.28 -1.1	8.96 9.04 +0.9							15

a — Vepco Model
b — MEASURED
c — % DIFFERENCE

CORE AVERAGE BURNUP = 14,870 MWD/MTU

	Measured	Vepco Model Predicted	Vepco Model Percent Difference	Vendor Model Percent Difference
Batch 1	15.47	15.57	-0.6	
Batch 2	17.02	17.15	-0.8	Average absolute value 1.4
Batch 3	12.04	11.88	+1.4	

TABLE 4-14

ASSEMBLYWISE ACCUMULATED BURNUP AND BATCH BURNUP SHARING (10^3 MWD/MTU) FOR THE CYCLE 2 OPERATION OF SURRY UNIT 2

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						5.74 5.96 +3.8	6.98 7.10 +1.7	5.74 5.93 +3.3							1
				5.98 6.26 +4.7	8.50 8.27 -2.7	10.34 10.37 +0.3	24.71 24.79 +0.3	10.34 10.33 -1.0	8.50 8.62 +1.4	5.98 6.18 +3.3					2
			6.81 7.07 +3.8	10.08 9.96 -1.2	24.04 25.90 -0.5	24.39 24.28 -0.5	26.25 26.04 -0.8	24.39 24.53 +0.6	26.04 26.08 +0.2	10.08 10.06 -0.2	6.81 7.02 +3.1				3
		6.81 7.07 +3.8	9.90 10.04 +1.4	25.13 24.78 -1.4	11.10 10.97 -1.2	25.05 24.85 -0.8	10.91 10.84 -0.6	25.05 24.86 -0.8	11.10 11.10 0.0	25.13 25.30 +0.7	9.90 9.99 +0.9	6.81 7.03 +3.0			4
	5.98 6.21 +3.9	10.08 10.05 -0.3	25.13 25.01 -0.5	11.16 11.13 -0.3	27.20 26.81 -1.4	11.11 10.88 -2.8	27.17 27.05 -0.4	11.11 10.98 -1.2	27.20 27.02 -0.7	11.16 11.10 -0.5	25.13 24.94 -0.8	10.08 10.05 -0.3	5.98 6.34 +6.0		5
	8.50 8.59 +1.1	26.04 25.98 -0.2	11.10 11.06 -0.4	27.20 26.81 -1.4	25.10 24.99 -0.4	26.51 25.98 -2.0	21.82 21.80 -0.1	26.51 26.24 -1.1	25.10 24.53 -2.3	27.20 27.00 -0.7	11.10 10.91 -1.7	26.04 25.83 -0.8	8.50 8.73 +2.7		6
5.74 6.07 +5.8	10.34 10.43 +0.9	24.39 24.20 -0.8	25.05 24.90 -0.6	11.11 10.95 -1.4	26.51 25.98 -2.0	10.94 10.78 -1.5	25.48 25.32 -0.6	10.94 10.68 -2.4	26.51 26.20 -1.2	11.11 10.95 -1.4	25.05 24.83 -0.9	24.39 24.48 +0.4	10.34 10.32 -0.2	5.74 5.95 +3.7	7
6.98 7.28 +4.3	24.71 24.55 -0.7	26.25 26.07 -0.7	10.91 10.85 -0.6	27.17 27.20 +0.1	21.82 22.00 +0.8	25.48 25.55 +0.3	24.08 23.60 -2.0	25.48 25.20 -1.1	21.82 21.93 +0.5	27.17 27.02 -0.6	10.91 10.87 -0.4	26.25 26.19 -0.2	24.71 24.60 -0.5	6.98 7.12 +2.0	8
5.74 6.09 +6.1	10.34 10.51 +1.6	24.39 24.29 -0.4	25.05 24.85 -0.8	11.11 10.88 -2.1	26.51 26.14 -1.4	10.94 10.76 -1.7	25.48 25.04 -1.7	10.94 10.71 -2.1	26.51 26.00 -1.9	11.11 10.96 -1.4	25.05 24.69 -1.4	24.39 24.63 +1.0	10.34 10.57 +2.2	5.74 6.02 +4.9	9
	8.50 8.57 +0.8	26.04 26.03 -0.1	11.10 10.97 -1.2	27.20 26.90 -1.1	25.10 24.62 -1.9	26.51 26.12 -1.5	21.82 21.49 -1.5	26.51 26.00 -1.9	25.10 24.91 -0.8	27.20 26.67 -2.0	11.10 10.86 -2.2	26.04 25.92 -0.5	8.50 8.77 +3.4		10
	5.98 6.25 +4.5	10.08 10.12 +0.4	25.13 25.26 +0.5	11.16 11.04 -1.1	27.20 26.78 -1.5	11.11 10.87 -2.2	27.17 26.87 -1.1	11.11 10.78 -3.0	27.20 26.61 -2.2	11.16 11.01 -1.3	25.13 25.34 +0.8	10.08 10.17 +0.9	5.98 6.35 +6.2		11
		6.81 7.19 +5.6	9.90 10.20 +3.0	25.13 25.19 +0.2	11.10 10.94 -1.4	25.05 24.57 -1.9	10.91 10.71 -1.8	25.05 24.70 -1.4	11.10 10.94 -1.4	25.13 25.23 +0.4	9.90 10.13 +2.3	6.81 7.23 +6.2			12
			6.81 7.14 +4.9	10.08 10.09 +0.1	26.04 25.80 -0.9	24.39 24.48 +0.4	26.25 25.95 -1.1	24.39 24.58 +0.8	26.04 25.85 -0.7	10.08 10.05 -0.3	6.81 7.15 +5.0				13
				5.98 6.23 +4.2	8.50 8.70 +2.4	10.34 10.42 +0.7	24.71 24.93 +0.9	10.34 10.34 0.0	8.50 8.67 +2.0	5.98 6.19 +3.5					14
						5.74 5.93 +3.3	6.98 7.09 +1.6	5.74 5.93 +3.3							15

a — Vepco Model
b — MEASURED
c — % DIFFERENCE

CORE AVERAGE BURNUP = 9,054 MWD/MTU

	Measured	Vepco Model Perdicted	Vepco Model Percent Difference	Vendor Model Percent Difference
Batch 1A	23.60	24.08	-2.0	
Batch 2	25.79	26.00	-0.8	Average absolute value 2.3
Batch 3A	24.14	24.24	-0.4	
Batch 4A	10.80	10.91	-1.0	
Batch 4B	8.01	7.84	+2.2	

Table 4-15

COMPARISON OF PREDICTED AND MEASURED CRITICAL BORON CONCENTRATION
FOR VARIOUS CONTROL ROD CONFIGURATIONS FOR CYCLE 1 OF SURRY UNITS 1 AND 2

Unit	Control Rod Bank Position	Measured Critical Boron Concentration (PPM)	Vepco Model Predicted Critical Boron Concentration (PPM)	Vepco Model Percent Difference	Vendor Model Percent Difference
1	ARO	1196	1168	-2.3	
1	D-Bank In	1077	1050	-2.5	
1	C and D-Banks In	957	942	-1.6	
2	ARO	1182	1168	-1.2	Average absolute value 2.4
2	D-Bank In	1056	1050	-0.6	
2	C and D-Banks In	947	942	-0.5	

Table 4-16

COMPARISON OF PREDICTED AND MEASURED CRITICAL BORON CONCENTRATION
FOR VARIOUS CONTROL ROD CONFIGURATIONS FOR CYCLE 2 OF SURRY UNITS 1 AND 2

Units	Control Rod Bank Position	Measured Critical Boron Concentration (PPM)	Vepco Model Predicted Critical Boron Concentration (PPM)	Vepco Model Percent Difference	Vendor Model Percent Difference
1	ARO	1033	997	-3.5	
1	D-Bank In	917	899	-2.0	
1	C and D-Banks In	800	787	-1.6	Average absolute value 0.7
2	ARO	1408	1401	-0.5	
2	D-Bank In	1325	1312	-1.0	
2	C and D-Banks In	1208	1192	-1.3	

Table 4-17

COMPARISON OF PREDICTED AND MEASURED DIFFERENTIAL BORON WORTH
FOR CYCLE 1 OF SURRY UNITS 1 AND 2

Unit	Predicted Boron Worth (PCM/PPM)	Measured Boron Worth (PCM/PPM)	Vepco Model Percent Difference	Vendor Model Percent Difference
1	11.9	12.1	-1.7	Average absolute
2	11.9	12.2	-2.5	value 5.4

Table 4-18

COMPARISON OF PREDICTED AND MEASURED BORON WORTH
FOR CYCLE 2 OF SURRY UNITS 1 AND 2

Unit	Predicted Boron Worth (PCM/PPM)	Measured Boron Worth (PCM/PPM)	Vepco Model Percent Difference	Vendor Model Percent Difference
1	10.8	10.2	5.9	Average absolute
2	10.3	10.4	-1.0	value 1.0

Table 4-19

COMPARISON OF PREDICTED AND MEASURED INTEGRAL BANK WORTH
FOR CYCLE 1 OF SURRY UNITS 1 AND 2

Unit	Control Rod Bank Position	Measured Integral Bank Worth (PCM)	Vepco Model Predicted Integral Bank Worth (PCM)	Vepco Model Percent Difference	Vendor Model Percent Difference
1	D-Bank In	1480	1379	-6.8	↑ Average absolute value 4.9 ↓
1	C and D-Bank In	1330	1234	-5.1	
2	D-Banks In	1435	1379	-3.9	
2	C and D-Banks In	1309	1234	-5.7	

Table 4-20

COMPARISON OF PREDICTED AND MEASURED INTEGRAL BANK WORTH
FOR CYCLE 2 OF SURRY UNITS 1 AND 2

Unit	Control Rod Bank Position	Measured Integral Bank Worth (PCM)	Vepco Model Predicted Integral Bank Worth (PCM)	Vepco Model Percent Difference	Vendor Model Percent Difference
1	D-Bank In	1051	1079	2.7	↑
1	C and D-Banks In	1331	1202	-9.7	Average absolute value 7.3
2	D-Bank In	880	931	5.8	
2	C and D-Banks In	1244	1249	0.4	↓

FIGURE 4.1

SURRY UNIT 1 - CYCLE 1

CRITICAL BORON CONCENTRATION
vs.
BURNUP

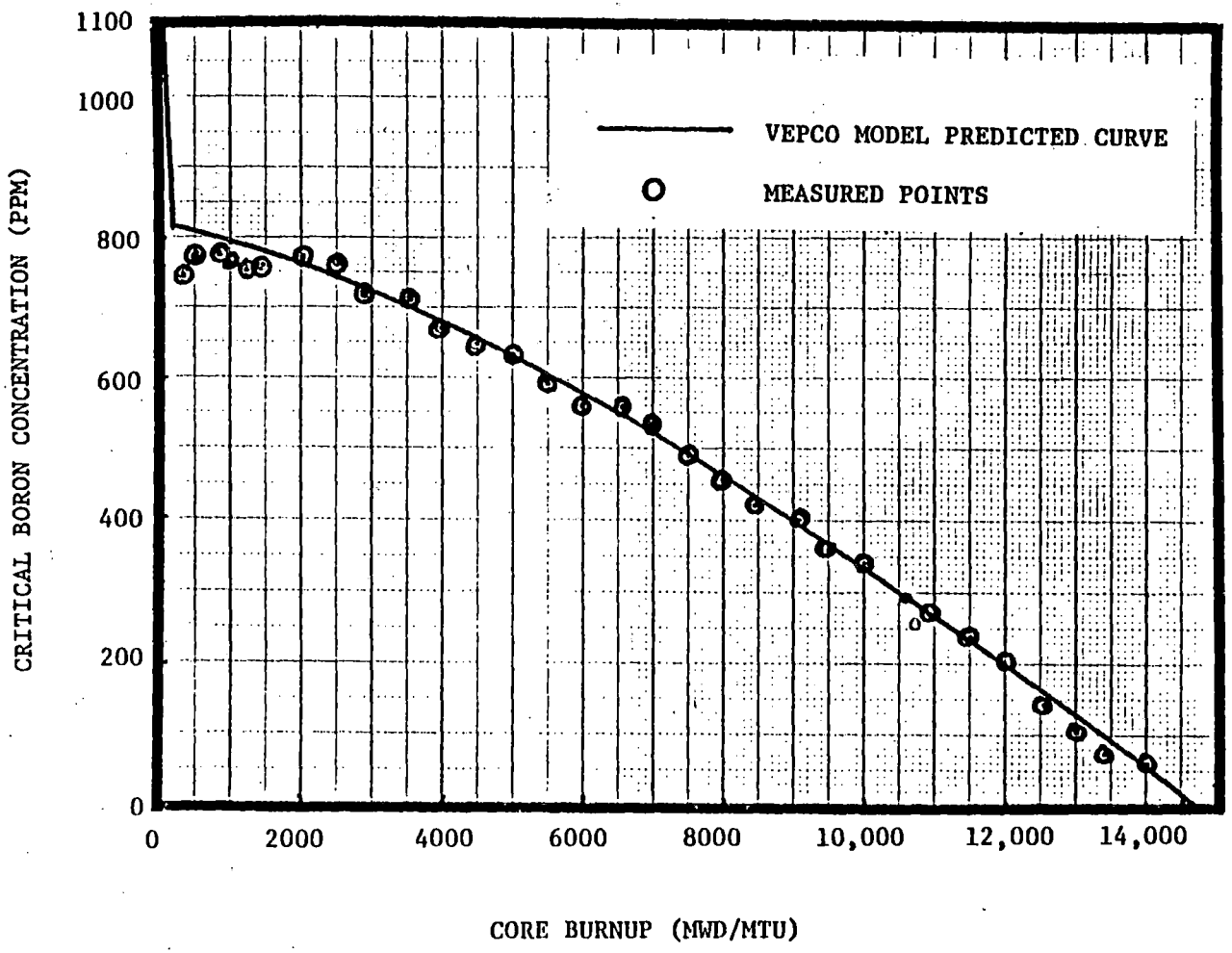


FIGURE 4.2

SURRY UNIT 1 - CYCLE 2

CRITICAL BORON CONCENTRATION
vs.
BURNUP

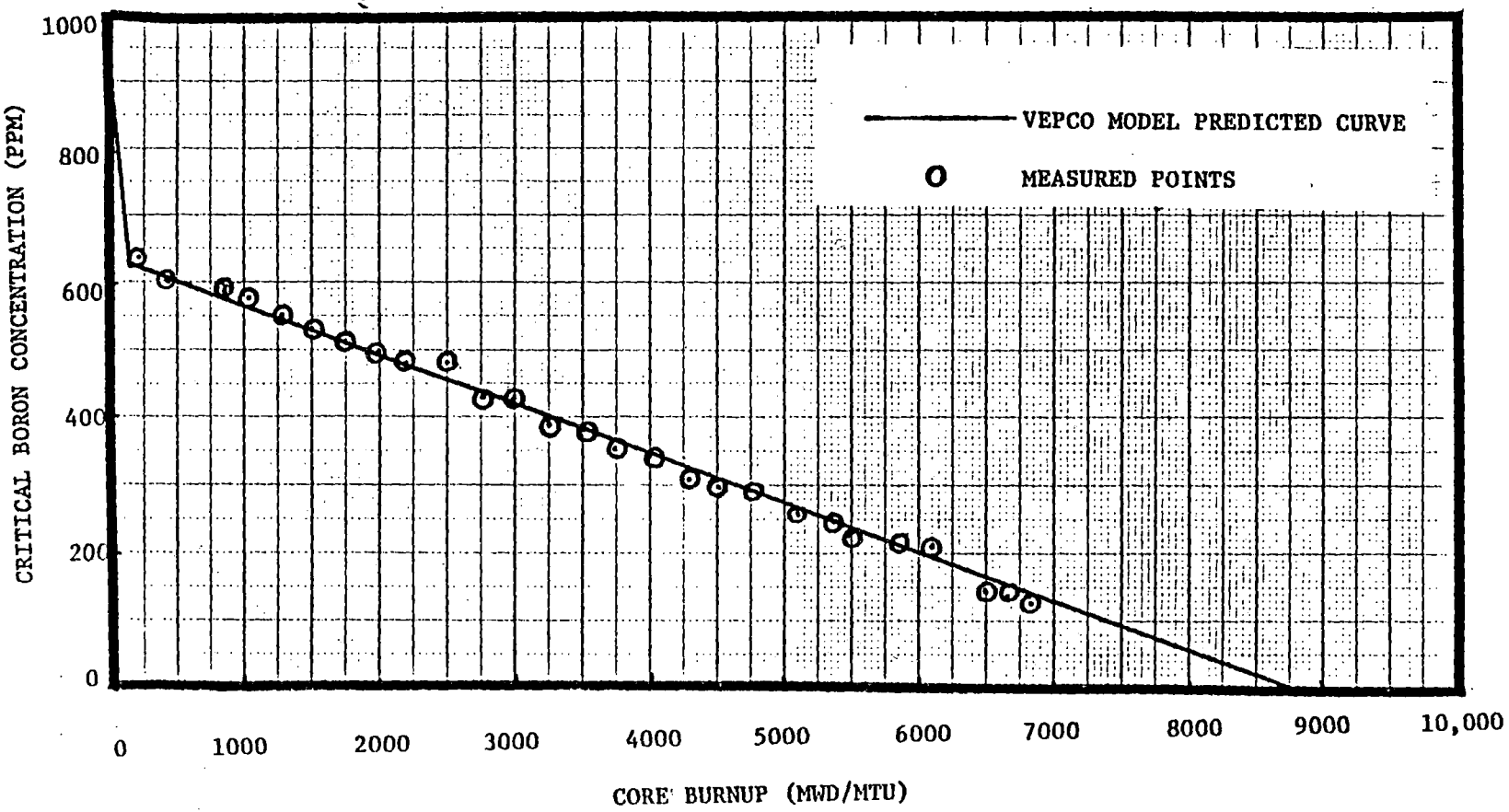


FIGURE 4.3

SURRY UNIT 2 - CYCLE 1

CRITICAL BORON CONCENTRATION
vs.
BURNUP

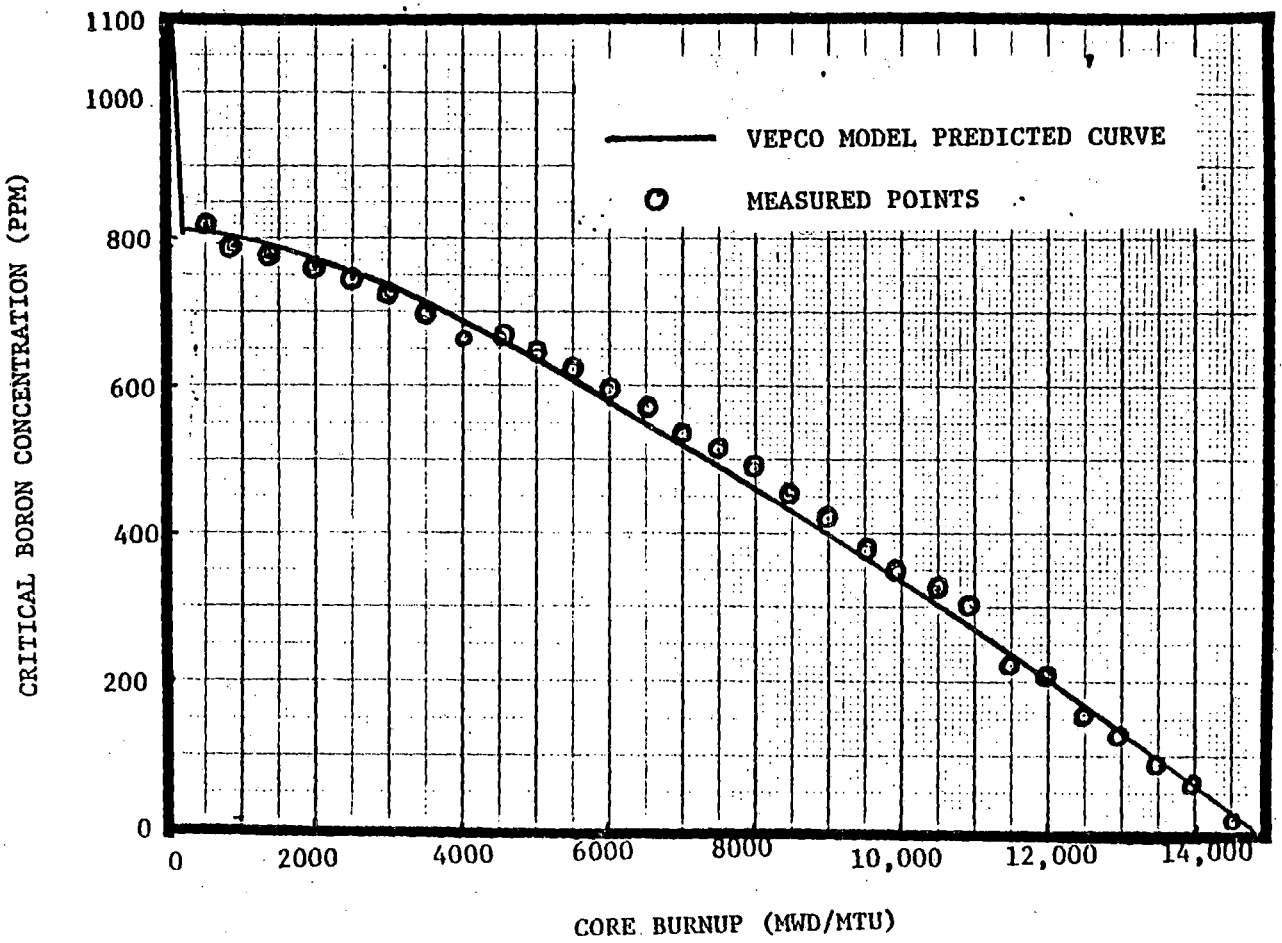


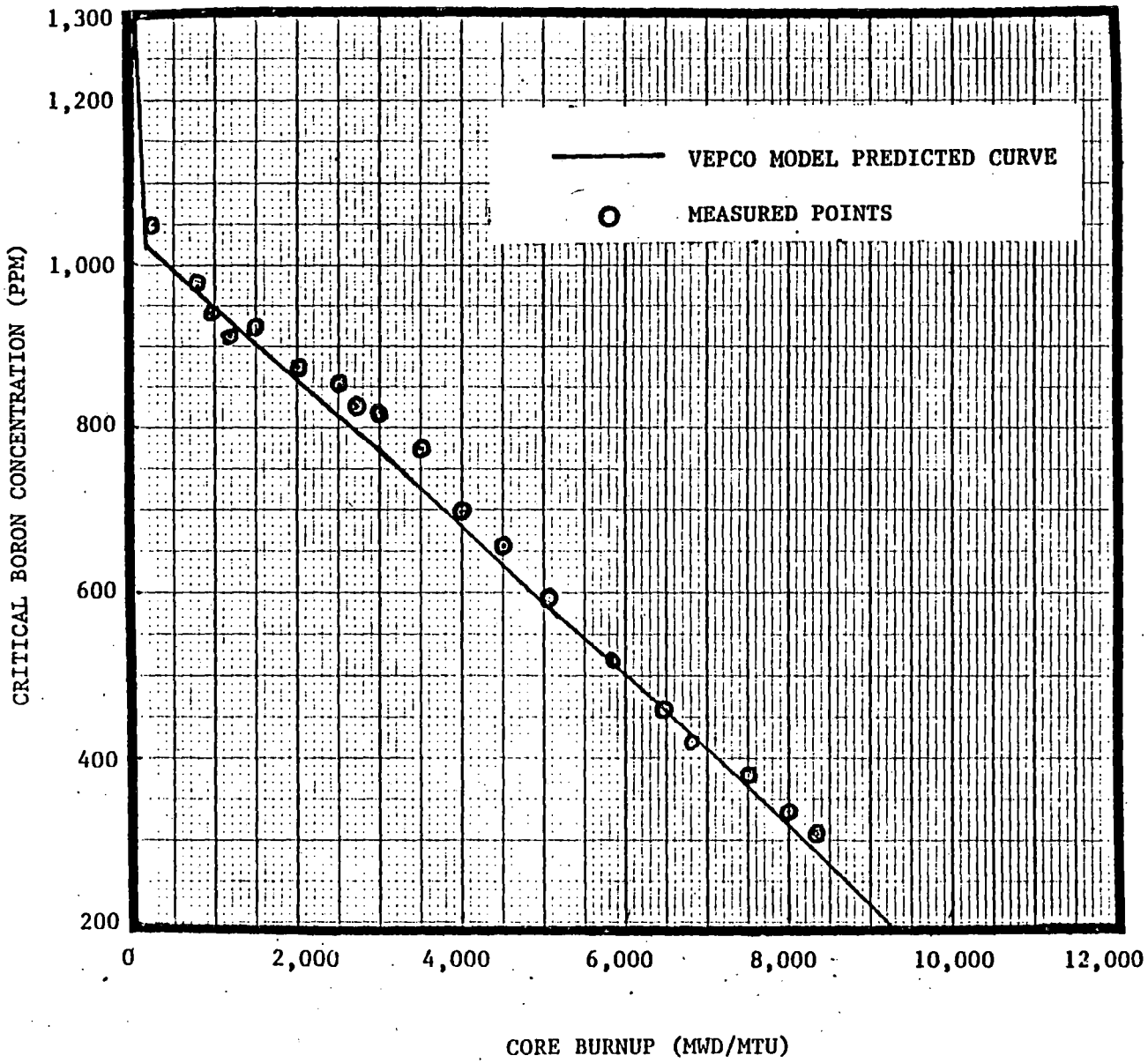
FIGURE 4.4

SURRY UNIT 2 - CYCLE 2

CRITICAL BORON CONCENTRATION

vs.

BURNUP



SECTION 5 - SUMMARY AND CONCLUSION

The PDQ07 discrete model, which contains the NULIF, HAFIT, SHUFFLE, PAPDQ, and PDQ07 computer codes, is operational at Vepco for the purpose of performing detailed reactor physics analyses and supporting the evaluation of core performance. The accuracy of the PDQ07 discrete model has been established through extensive comparisons of calculations with measurements from the Surry reactors. The results of these comparisons are:

1) Assembly average power distributions are predicted typically within a standard deviation of 2%, with a maximum standard deviation of 4.3% for low power flux maps (where uncertainties in the data are greater due to the low neutron flux level and the drift in the power level) and a maximum standard deviation of 3.6% for flux maps at power levels greater than 10%.
2) Peak rod $F^N_{\Delta H}$ values are predicted typically within 2.5% with a maximum deviation of 4.3% between predicted and measured.
3) Assembly average burnups are predicted typically within 2.5% and batch average burnups within 1.5%.
4) Critical soluble boron concentrations are predicted typically within 30 ppm and boron worth within 3%.
5) Control rod bank integral worths are predicted typically within 6% with a maximum deviation of 9.7%.

In addition, the accuracy of the Vepco model was verified by comparison of the accuracy of Vepco results (i.e, comparison of predicted values with

measurement data) with the accuracy of similar results obtained from a vendor model. These comparisons indicated that the standard deviation and/or percentage difference between the Vepco PDQ07 discrete model calculations and reactor measurement data were within acceptable industry standards.

Verification, as well as improvements to the PDQ07 discrete model, will continue to be made as more experience is obtained through the continued application of the model to the Surry and North Anna reactors.

SECTION 6 - REFERENCES

1. W. A. Wittkopf, et. al., NULIF - "Neutrum Spectrum Generator, Few Group Constant Calculator, and Fuel Depletion Code", BAW-10115, June 1976.

2. H. H. Hassan, et. al., "Babcock and Wilcox Version of PDQ07 - User's Manual", BAW-10117P, December 1975.

3. H. H. Hassan, et. al., "Shuffle - Program to Perform Fuel Shuffle in Nuclear Reactor Core", BAW-422, Rev. 1, July 1975.

4. H. H. Hassan, W. A. Wittkopf, et. al., "HAFIT", BAW-425, July 1973.

5. H. H. Hassan, et. al., "Zeus/General Service Programs", BAW-423, Rev. 1, July 1975.

6. Private correspondence from the Babcock and Wilcox Company to the Virginia Electric and Power Company dated February 3, 1971, and October 6, 1971.

7. Final Safety Analysis Report - Surry Power Station Units 1 and 2, Virginia Electric and Power Company.

8. R. D. Klatt, W. D. Leggett III, and L. D. Eisenhart, "FOLLOW - a Code Providing a Standard Reactivity Follow Procedure by Calculating Effective Critical Boron Concentrations as a Function of Burnup", WCAP-7482, February 1970.

9. W. D. Leggett III, and L. D. Eisenhart, "The INCORE Code", WCAP-7149, December 1967.

10. W. D. Leggett III, "TOTE - a Code for Totaling Local Burnup, Isotopics, and Uranium Values", WCAP-7309, March 1969.

APPENDIX A

Description of the INCORE, TOTE, and FOLLOW Computer Codes

A-1 INCORE COMPUTER CODE DESCRIPTION

INCORE is a data analysis computer code used to process information obtained from the movable incore instrumentation system, and is therefore, the primary computer program for core follow analysis. Input to the INCORE program consists of:

1) A description of reactor conditions when the measurements were made (such as power level, control rod positions, etc.)
2) Incore detector readings including which flux thimbles were used and neutron cross sections of the sensor
3) Analytical information calculated by the PDQ07 discrete model
4) Options specifying which thimbles will be employed in local power predictions and what type of calculations are to be performed

INCORE corrects raw pointwise flux measurements for leakage current, changes in power level between measurements, and relative detector sensitivites to determine the pointwise reaction rate in the flux thimbles. The measured reaction rates determined are then compared with expected values.

INCORE computes the relative local power produced by each fuel assembly and in the peak fuel rod for each assembly. Local relative power is computed as:

$$P_{mi} = \{ \sum_{j=1}^{n} W_j \, (R_{mj} \times \frac{P_{ci}}{R_{cj}}) \}$$

where P_{mi} is normalized so that $\sum_{i=1}^{157} P_{mi}/157=1$

and P_{mi} = Measured power for the i^{th} location (which corresponds to an assembly for $1 < i < 157$ or a fuel rod for $i > 157$)

R_{mj} = Measured reaction rate for the j^{th} thimble

W_j = Weighting factor for the j^{th} thimble (W_j is based on the distance from the i^{th} location to the j^{th} thimble)

P_{ci} = Power calculated for the i^{th} location by the PDQ07 discrete model

R_{cj} = Reaction rate calculated for the j^{th} thimble by the PDQ07 discrete model

n = Number of thimbles used for measuring power in the i^{th} location

Different ratios of power to reaction rate (P_{ci}/R_{cj}) obtained from the PDQ07 discrete model are used depending on the control rod configurations at each elevation.

INCORE calculates 1) the relative power for each assembly and quadrant based on the assembly average local powers and 2) the twenty largest values of $F^N_{\Delta H}$ in descending order with an identifying number so that hot spot (peak rod) locations in the core can be determined.

A-2 TOTE COMPUTER CODE DESCRIPTION

The TOTE computer code is an isotopic and burnup follow program which calculates material concentrations for the fuel and accumulated burnup based on measured power distributions (obtained from the INCORE code) and tables of material concentrations as a function of burnup. The INCORE code outputs burnup rate information for every fuel batch and/or assembly, (the total burnup rate for the fuel region is given as well as the value for each of four axial segments of approximately equal length). The burnup rate is given as the megawatt-hours generated in a given fuel quantity per 1,000 megawatt-hours generated in the core.

Input to the TOTE code consists of:

1) the core energy (megawatt-hours) associated with each INCORE burnup rate deck
2) a description of each fuel region (including metric tons of uranium, corresponding INCORE source number, previous burnup, isotopic depletion type, etc.)
3) tables of the change in up to ten material concentrations with burnup
4) the burnup rate decks from INCORE

Core average, fuel batch and/or assembly, and material concentrations are outputed by TOTE. Interpolation for material concentrations as a function of burnup is quadratic (generally using the two preceding and one succeeding table entry). A description of each fuel region (item (2) above) is output for subsequent TOTE runs.

A-3 FOLLOW PROGRAM DESCRIPTION

The FOLLOW computer code processes reactor operation data and calculates nominal boron concentrations. The FOLLOW code is designed to describe the nearly linear relationship between available core reactivity and cycle burnup at nominal* conditions. It is most convenient to use boron as a measure of core reactivity with off-nominal corrections being made for power, xenon and samarium, temperature, and control rods in terms of their boron worth. These off-nominal corrections are made using the following equation:

corrected or nominal boron concentration = measured boron concentration + inverse boron x [off-nominal reactivity correction due to rod group 1 position + off-nominal reactivity correction due to rod group 2 position + off-nominal reactivity correction due to moderator temperature + off-nominal reactivity correction due to power + off-nominal reactivity correction due to xenon and samarium behavior]

The boron concentration in Surry Units 1 and 2 is typically measured one to three times per day. After proper normalization, this data is plotted against cycle burnup and forms the "boron depletion curve". Since this curve is well behaved and nearly linear from the beginning to the end of the cycle, it can provide the following information:

1) Extrapolation to end-of-cycle life for verifying refueling dates.
2) Rate of loss of reactivity with burnup for confirmation of design predictions.

3) Best-estimate of beginning cycle, hot-full-power criticality under equilibrium conditions.

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* Nominal conditions are associated with hot full power, equilibrium xenon conditions, and all control rods withdrawn from the core.

APPENDIX B

REPRESENTATIVE INCORE OUTPUT USING THE VEPCO PDQ07 DISCRETE MODEL

FIGURE B-1

INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT BEGINNING OF LIFE CONDITION

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
	PREDICTED					0.66	0.87	0.66				PREDICTED			
	MEASURED					0.72	0.95	0.72				MEASURED			1
	PCT DIFFERENCE					9.2	9.5	9.2				PCT DIFFERENCE			
				0.62	0.98	1.10	1.00	1.10	0.98	0.62					
				0.62	1.02	1.12	0.98	1.12	1.02	0.62					2
				-1.1	4.4	1.7	-1.7	1.7	4.4	-1.1					
			0.63	0.94	1.05	1.05	1.14	1.05	1.05	0.94	0.63				
			0.65	0.95	1.03	1.04	1.09	1.04	1.03	0.95	0.65				3
			4.1	0.7	-1.8	-0.7	-4.7	-0.7	-1.8	0.7	4.1				
		0.63	0.77	0.99	1.02	1.15	1.08	1.15	1.02	0.99	0.77	0.63			
		0.65	0.79	0.99	1.02	1.18	1.07	1.18	1.02	0.99	0.79	0.65			4
		3.9	2.3	-0.3	0.1	2.5	-1.5	2.5	0.1	-0.3	2.3	3.9			
	0.62	0.94	0.99	1.00	1.14	1.10	1.20	1.10	1.14	1.00	0.99	0.94	0.62		
	0.61	0.96	1.02	1.02	1.12	1.09	1.17	1.09	1.12	1.02	1.02	0.96	0.61		5
	-1.5	1.4	3.3	2.1	-2.3	-0.7	-2.2	-0.7	-2.3	2.1	3.3	1.4	-1.5		
	0.98	1.05	1.02	1.14	1.10	1.21	1.13	1.21	1.10	1.14	1.02	1.05	0.98		
	0.97	1.05	1.04	1.13	1.06	1.18	1.11	1.18	1.06	1.13	1.04	1.05	0.97		6
	-0.3	-0.2	1.9	-1.5	-3.6	-2.8	-2.2	-2.8	-3.6	-1.5	1.9	-0.2	-0.3		
0.66	1.10	1.05	1.15	1.10	1.21	1.14	1.24	1.14	1.21	1.10	1.15	1.05	1.10	0.66	
0.69	1.11	1.08	1.12	1.08	1.17	1.13	1.23	1.13	1.17	1.08	1.12	1.08	1.11	0.69	7
4.5	0.8	3.6	-2.5	-1.7	-3.9	-1.1	-1.1	-1.1	-3.9	-1.7	-2.5	3.6	0.8	4.5	
0.87	1.00	1.14	1.08	1.20	1.13	1.24	1.16	1.24	1.13	1.20	1.08	1.14	1.00	0.87	
0.90	1.02	1.15	1.07	1.16	1.11	1.22	1.15	1.22	1.11	1.16	1.07	1.15	1.02	0.90	8
3.4	2.7	0.7	-0.9	-3.3	-2.3	-1.6	-0.4	-1.6	-2.3	-3.3	-0.9	0.7	2.7	3.4	
0.66	1.10	1.05	1.15	1.10	1.21	1.14	1.24	1.14	1.21	1.10	1.15	1.05	1.10	0.66	
0.69	1.11	1.08	1.12	1.08	1.17	1.13	1.23	1.13	1.17	1.08	1.12	1.08	1.11	0.69	9
4.5	0.8	3.6	-2.5	-1.7	-3.9	-1.1	-1.1	-1.1	-3.9	-1.7	-2.5	3.6	0.8	4.5	
	0.98	1.05	1.02	1.14	1.10	1.21	1.13	1.21	1.10	1.14	1.02	1.05	0.98		
	0.97	1.05	1.04	1.13	1.06	1.18	1.11	1.18	1.06	1.13	1.04	1.05	0.97		10
	-0.3	-0.2	1.9	-1.5	-3.6	-2.8	-2.2	-2.8	-3.6	-1.5	1.9	-0.2	-0.3		
	0.62	0.94	0.99	1.00	1.14	1.10	1.20	1.10	1.14	1.00	0.99	0.94	0.62		
	0.61	0.96	1.02	1.02	1.12	1.09	1.17	1.09	1.12	1.02	1.02	0.96	0.61		11
	-1.5	1.4	3.3	2.1	-2.3	-0.7	-2.2	-0.7	-2.3	2.1	3.3	1.4	-1.5		
		0.63	0.77	0.99	1.02	1.15	1.08	1.15	1.02	0.99	0.77	0.63			
		0.65	0.79	0.99	1.02	1.18	1.07	1.18	1.02	0.99	0.79	0.65			12
		3.9	2.3	-0.3	0.1	2.5	-1.5	2.5	0.1	-0.3	2.3	3.9			
			0.63	0.94	1.05	1.05	1.14	1.05	1.05	0.94	0.63				
			0.65	0.95	1.03	1.04	1.09	1.04	1.03	0.95	0.65				13
			4.1	0.7	-1.8	-0.7	-4.7	-0.7	-1.8	0.7	4.1				
				0.62	0.98	1.10	1.00	1.10	0.98	0.62					
				0.62	1.02	1.12	0.98	1.12	1.02	0.62					14
				-1.1	4.4	1.7	-1.7	1.7	4.4	-1.1					
	STANDARD					0.66	0.87	0.66				AVERAGE			
	DEVIATION					0.72	0.95	0.72				PCT DIFFERENCE			15
	=0.027					9.2	9.5	9.2				= 2.3			

Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	1	1	0	0	ARO

FIGURE B-2

INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT BEGINNING OF LIFE CONDITION

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
	PREDICTED					0.48	0.54	0.48				PREDICTED			
	MEASURED					0.50	0.56	0.50				MEASURED			1
	PCT DIFFERENCE					2.8	5.2	2.8				PCT DIFFERENCE			
				0.75	1.04	0.90	0.40	0.90	1.04	0.75					
				0.75	1.03	0.88	0.39	0.88	1.03	0.75					2
				-0.0	-1.4	-2.4	-2.1	-2.4	-1.4	-0.0					
			0.81	1.16	1.18	1.04	1.02	1.04	1.18	1.16	0.81				
			0.83	1.16	1.14	1.00	0.96	1.00	1.14	1.16	0.83				3
			2.7	0.1	-3.8	-3.8	-5.6	-3.8	-3.8	0.1	2.7				
		0.81	0.99	1.19	1.15	1.24	1.15	1.24	1.15	1.19	0.99	0.81			
		0.82	1.02	1.20	1.15	1.20	1.11	1.20	1.15	1.20	1.02	0.82			4
		1.6	3.3	1.4	-0.1	-2.9	-4.0	-2.9	-0.1	1.4	3.3	1.6			
	0.75	1.16	1.19	1.07	1.07	1.11	1.26	1.11	1.07	1.07	1.19	1.16	0.75		
	0.77	1.16	1.19	1.09	1.09	1.10	1.23	1.10	1.09	1.09	1.19	1.16	0.77		5
	2.5	0.7	0.8	2.0	1.6	-0.6	-2.9	-0.6	1.6	2.0	0.8	0.7	2.5		
	1.04	1.18	1.15	1.07	0.51	1.09	1.16	1.09	0.51	1.07	1.15	1.18	1.04		
	1.07	1.16	1.15	1.08	0.53	1.10	1.16	1.10	0.53	1.08	1.15	1.16	1.07		6
	2.2	-1.6	0.4	1.1	3.6	0.5	-0.6	0.5	3.6	1.1	0.4	-1.6	2.2		
0.48	0.90	1.04	1.24	1.11	1.09	1.14	1.30	1.14	1.09	1.11	1.24	1.04	0.90	0.48	
0.51	0.92	1.07	1.25	1.12	1.09	1.14	1.31	1.14	1.09	1.12	1.25	1.07	0.92	0.51	7
6.8	2.7	2.5	0.5	1.1	-0.5	0.4	0.2	0.4	-0.5	1.1	0.5	2.5	2.7	6.8	
0.54	0.40	1.02	1.15	1.26	1.16	1.30	1.25	1.30	1.16	1.26	1.15	1.02	0.40	0.54	
0.57	0.41	1.03	1.16	1.24	1.15	1.30	1.26	1.30	1.15	1.24	1.16	1.03	0.41	0.57	8
5.6	3.3	1.4	0.8	-1.6	-0.6	-0.2	0.3	-0.2	-0.6	-1.6	0.8	1.4	3.3	5.6	
0.48	0.90	1.04	1.24	1.11	1.09	1.14	1.30	1.14	1.09	1.11	1.24	1.04	0.90	0.48	
0.51	0.92	1.07	1.25	1.12	1.09	1.14	1.31	1.14	1.09	1.12	1.25	1.07	0.92	0.51	9
6.8	2.7	2.5	0.5	1.1	-0.5	0.4	0.2	0.4	-0.5	1.1	0.5	2.5	2.7	6.8	
	1.04	1.18	1.15	1.07	0.51	1.09	1.16	1.09	0.51	1.07	1.15	1.18	1.04		
	1.07	1.16	1.15	1.08	0.53	1.10	1.16	1.10	0.53	1.08	1.15	1.16	1.07		10
	2.2	-1.6	0.4	1.1	3.6	0.5	-0.6	0.5	3.6	1.1	0.4	-1.6	2.2		
	0.75	1.16	1.19	1.07	1.07	1.11	1.26	1.11	1.07	1.07	1.19	1.16	0.75		
	0.77	1.16	1.19	1.09	1.09	1.10	1.23	1.10	1.09	1.09	1.19	1.16	0.77		11
	2.5	0.7	0.8	2.0	1.6	-0.6	-2.9	-0.6	1.6	2.0	0.8	0.7	2.5		
		0.81	0.99	1.19	1.15	1.24	1.15	1.24	1.15	1.19	0.99	0.81			
		0.82	1.02	1.20	1.15	1.20	1.11	1.20	1.15	1.20	1.02	0.82			12
		1.6	3.3	1.4	-0.1	-2.9	-4.0	-2.9	-0.1	1.4	3.3	1.6			
			0.81	1.16	1.18	1.04	1.02	1.04	1.18	1.16	0.81				
			0.83	1.16	1.14	1.00	0.96	1.00	1.14	1.16	0.83				13
			2.7	0.1	-3.8	-3.8	-5.6	-3.8	-3.8	0.1	2.7				
				0.75	1.04	0.90	0.40	0.90	1.04	0.75					
				0.75	1.03	0.88	0.39	0.88	1.03	0.75					14
				-0.0	-1.4	-2.4	-2.1	-2.4	-1.4	-0.0					
	STANDARD					0.48	0.54	0.48				AVERAGE			
	DEVIATION					0.50	0.56	0.50				PCT DIFFERENCE			15
	=0.021					2.8	5.2	2.8				= 1.9			

Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	1	2	0	0	D-In

FIGURE B-3

INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT BEGINNING OF LIFE CONDITIONS

K	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
	PREDICTED					0.61	0.79	0.61				PREDICTED			
	MEASURED					0.61	0.80	0.61				MEASURED			1
	PCT DIFFERENCE					0.9	0.4	0.9				PCT DIFFERENCE			
				0.60	0.90	1.02	0.95	1.02	0.90	0.60					
				0.61	0.93	1.02	0.92	1.02	0.93	0.61					2
				3.1	3.8	-0.2	-3.4	-0.2	3.8	3.1					
			0.63	0.91	1.01	1.03	1.12	1.03	1.01	0.91	0.63				
			0.64	0.93	1.03	1.02	1.07	1.02	1.03	0.93	0.64				3
			1.7	1.4	1.5	-1.1	-4.7	-1.1	1.5	1.4	1.7				
		0.63	0.80	1.00	1.05	1.17	1.11	1.17	1.05	1.00	0.80	0.63			
		0.63	0.80	1.00	1.04	1.15	1.09	1.15	1.04	1.00	0.80	0.63			4
		0.0	0.3	-0.3	-0.3	-1.1	-1.7	-1.1	-0.3	-0.3	0.3	0.0			
	0.60	0.91	1.00	1.04	1.18	1.15	1.23	1.15	1.18	1.04	1.00	0.91	0.60		
	0.60	0.90	1.00	1.04	1.18	1.17	1.23	1.17	1.18	1.04	1.00	0.90	0.60		5
	-0.1	-0.9	-0.4	-0.4	0.1	2.2	0.1	2.2	0.1	-0.4	-0.4	-0.9	-0.1		
	0.90	1.01	1.05	1.18	1.16	1.26	1.19	1.26	1.16	1.18	1.05	1.01	0.90		
	0.90	1.02	1.04	1.18	1.17	1.26	1.19	1.26	1.17	1.18	1.04	1.02	0.90		6
	-0.1	0.1	-0.4	-0.4	1.1	-0.2	-0.4	-0.2	1.1	-0.4	-0.4	0.1	-0.1		
0.61	1.02	1.03	1.17	1.15	1.26	1.21	1.29	1.21	1.26	1.15	1.17	1.03	1.02	0.61	
0.62	1.01	1.03	1.16	1.15	1.27	1.21	1.29	1.21	1.27	1.15	1.16	1.03	1.01	0.62	7
1.4	-1.3	-0.3	-0.5	0.5	0.7	-0.0	-0.5	-0.0	0.7	0.5	-0.5	-0.3	-1.3	1.4	
0.79	0.95	1.12	1.11	1.23	1.19	1.29	1.22	1.29	1.19	1.23	1.11	1.12	0.95	0.79	
0.79	0.94	1.10	1.10	1.25	1.19	1.29	1.22	1.29	1.19	1.25	1.10	1.10	0.94	0.79	8
-0.1	-1.0	-2.2	-0.8	1.0	-0.0	0.0	-0.1	0.0	-0.0	1.0	-0.8	-2.2	-1.0	-0.1	
0.61	1.02	1.03	1.17	1.15	1.26	1.21	1.29	1.21	1.26	1.15	1.17	1.03	1.02	0.61	
0.62	1.01	1.03	1.16	1.15	1.27	1.21	1.29	1.21	1.27	1.15	1.16	1.03	1.01	0.62	9
1.4	-1.3	-0.3	-0.5	0.5	0.7	-0.0	-0.5	-0.0	0.7	0.5	-0.5	-0.3	-1.3	1.4	
	0.90	1.01	1.05	1.18	1.16	1.26	1.19	1.26	1.16	1.18	1.05	1.01	0.90		
	0.90	1.02	1.04	1.18	1.17	1.26	1.19	1.26	1.17	1.18	1.04	1.02	0.90		10
	-0.1	0.1	-0.4	-0.4	1.1	-0.2	-0.4	-0.2	1.1	-0.4	-0.4	0.1	-0.1		
	0.60	0.91	1.00	1.04	1.18	1.15	1.23	1.15	1.18	1.04	1.00	0.91	0.60		
	0.60	0.90	1.00	1.04	1.18	1.17	1.23	1.17	1.18	1.04	1.00	0.90	0.60		11
	-0.1	-0.9	-0.4	-0.4	0.1	2.2	0.1	2.2	0.1	-0.4	-0.4	-0.9	-0.1		
		0.63	0.80	1.00	1.05	1.17	1.11	1.17	1.05	1.00	0.80	0.63			
		0.63	0.80	1.00	1.04	1.15	1.09	1.15	1.04	1.00	0.80	0.63			12
		0.0	0.3	-0.3	-0.3	-1.1	-1.7	-1.1	-0.3	-0.3	0.3	0.0			
			0.63	0.91	1.01	1.03	1.12	1.03	1.01	0.91	0.63				
			0.64	0.93	1.03	1.02	1.07	1.02	1.03	0.93	0.64				13
			1.7	1.4	1.5	-1.1	-4.7	-1.1	1.5	1.4	1.7				
				0.60	0.90	1.02	0.95	1.02	0.90	0.60					
				0.61	0.93	1.02	0.92	1.02	0.93	0.61					14
				3.1	3.8	-0.2	-3.4	-0.2	3.8	3.1					
	STANDARD					0.61	0.79	0.61				AVERAGE			
	DEVIATION					0.61	0.80	0.61				PCT DIFFERENCE			15
	=0.013					0.9	0.4	0.9				= 0.9			

Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	1	31	90	1,300	ARO

FIGURE B-4

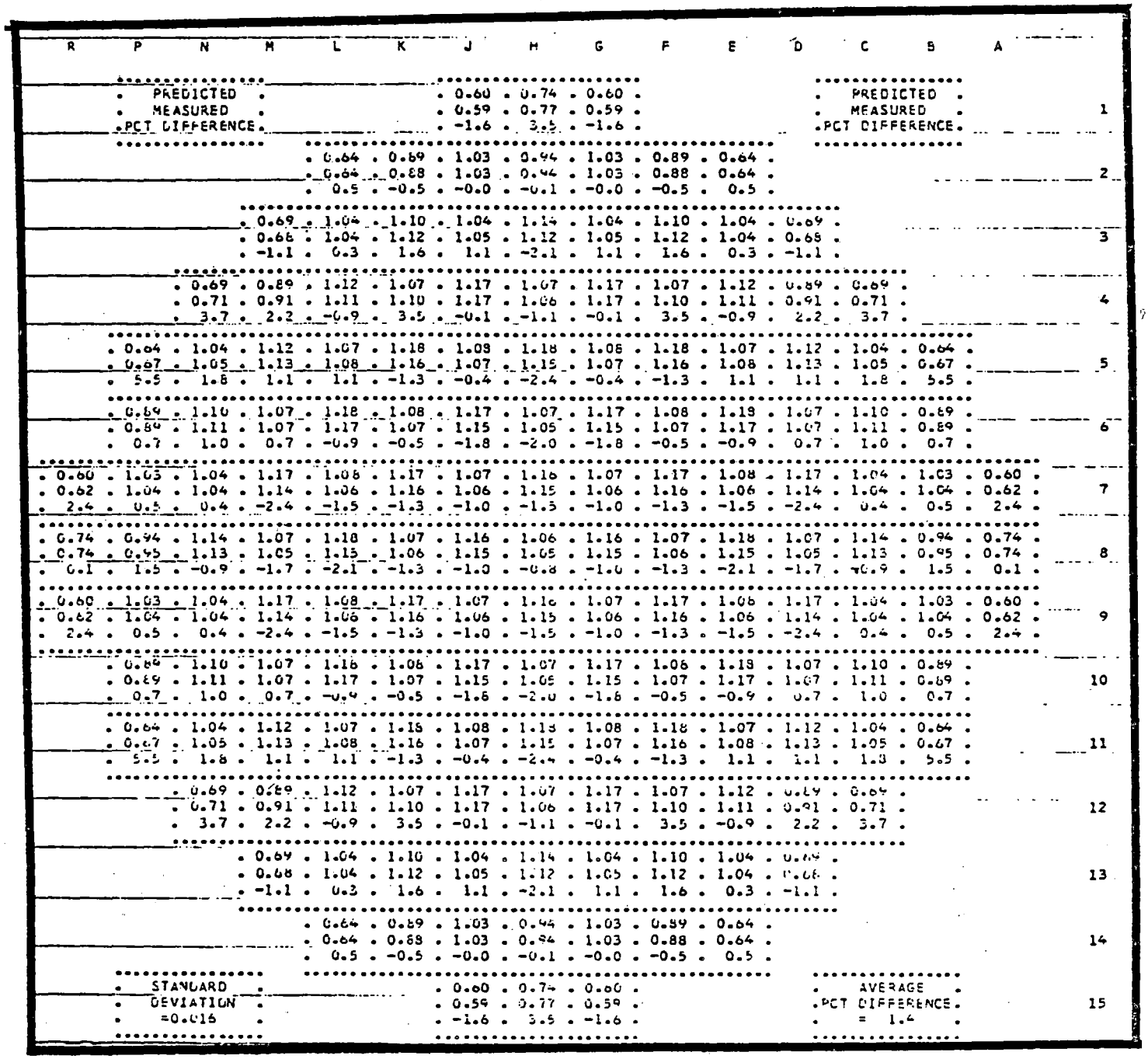
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT MIDDLE OF LIFE CONDITIONS

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
	PREDICTED					0.59	0.74	0.59				PREDICTED			
	MEASURED					0.58	0.72	0.58				MEASURED			
	PCT DIFFERENCE					-1.4	-2.5	-1.4				PCT DIFFERENCE			1
				0.61	0.88	1.01	0.93	1.01	0.88	0.61					
				0.60	0.84	0.99	0.91	0.99	0.84	0.60					2
				-2.4	-4.5	-2.8	-2.2	-2.8	-4.5	-2.4					
			0.66	0.98	1.06	1.03	1.13	1.03	1.06	0.98	0.66				
			0.65	0.97	1.05	1.03	1.10	1.03	1.05	0.97	0.65				3
			-1.7	-1.3	-1.2	-0.4	-2.4	-0.4	-1.2	-1.3	-1.7				
		0.66	0.85	1.08	1.06	1.18	1.09	1.18	1.06	1.08	0.85	0.66			
		0.65	0.86	1.08	1.08	1.18	1.09	1.18	1.08	1.08	0.86	0.65			4
		-1.3	0.6	0.4	1.3	0.1	-0.1	0.1	1.3	0.4	0.6	-1.3			
	0.61	0.98	1.06	1.06	1.19	1.11	1.21	1.11	1.19	1.06	1.08	0.98	0.61		
	0.61	0.97	1.08	1.08	1.20	1.12	1.21	1.12	1.20	1.08	1.08	0.97	0.61		5
	-0.9	-1.4	0.2	1.4	1.2	0.8	0.4	0.8	1.2	1.4	0.2	-1.4	-0.9		
	0.88	1.06	1.06	1.19	1.11	1.22	1.12	1.22	1.11	1.19	1.06	1.06	0.88		
	0.87	1.06	1.06	1.20	1.13	1.23	1.14	1.23	1.13	1.20	1.06	1.06	0.87		6
	-1.4	-0.1	-0.0	0.6	1.3	1.2	1.7	1.2	1.3	0.6	-0.0	-0.1	-1.4		
0.59	1.01	1.03	1.18	1.11	1.22	1.13	1.22	1.13	1.22	1.11	1.18	1.03	1.01	0.59	
0.58	1.00	1.04	1.19	1.13	1.23	1.16	1.26	1.16	1.23	1.13	1.19	1.04	1.00	0.58	7
-0.7	-1.1	0.8	0.8	1.5	0.8	2.7	2.5	2.7	0.8	1.5	0.8	0.8	-1.1	-0.7	
0.74	0.93	1.13	1.09	1.21	1.12	1.22	1.13	1.22	1.12	1.21	1.09	1.13	0.93	0.74	
0.73	0.93	1.13	1.10	1.21	1.15	1.25	1.16	1.25	1.15	1.21	1.10	1.13	0.93	0.73	8
-1.4	0.2	-0.0	0.6	0.1	2.0	2.4	3.1	2.4	2.0	0.1	0.6	-0.0	0.2	-1.4	
0.59	1.01	1.03	1.18	1.11	1.22	1.13	1.22	1.13	1.22	1.11	1.18	1.03	1.01	0.59	
0.58	1.00	1.04	1.19	1.13	1.23	1.16	1.26	1.16	1.23	1.13	1.19	1.04	1.00	0.58	9
-0.7	-1.1	0.8	0.8	1.5	0.8	2.7	2.5	2.7	0.8	1.5	0.8	0.8	-1.1	-0.7	
	0.88	1.06	1.06	1.19	1.11	1.22	1.12	1.22	1.11	1.19	1.06	1.06	0.88		
	0.87	1.06	1.06	1.20	1.13	1.23	1.14	1.23	1.13	1.20	1.06	1.06	0.87		10
	-1.4	-0.1	-0.0	0.6	1.3	1.2	1.7	1.2	1.3	0.6	-0.0	-0.1	-1.4		
	0.61	0.98	1.08	1.06	1.19	1.11	1.21	1.11	1.19	1.06	1.08	0.98	0.61		
	0.61	0.97	1.08	1.08	1.20	1.12	1.21	1.12	1.20	1.08	1.08	0.97	0.61		11
	-0.9	-1.4	0.2	1.4	1.2	0.8	0.4	0.8	1.2	1.4	0.2	-1.4	-0.9		
		0.66	0.85	1.08	1.06	1.18	1.09	1.18	1.06	1.08	0.85	0.66			
		0.65	0.86	1.08	1.08	1.18	1.09	1.18	1.08	1.08	0.86	0.65			12
		-1.3	0.6	0.4	1.3	0.1	-0.1	0.1	1.3	0.4	0.6	-1.3			
			0.66	0.98	1.06	1.03	1.13	1.03	1.06	0.98	0.66				
			0.65	0.97	1.05	1.03	1.10	1.03	1.05	0.97	0.65				13
			-1.7	-1.3	-1.2	-0.4	-2.4	-0.4	-1.2	-1.3	-1.7				
				0.61	0.88	1.01	0.93	1.01	0.88	0.61					
				0.60	0.84	0.99	0.91	0.99	0.84	0.60					14
				-2.4	-4.5	-2.8	-2.2	-2.8	-4.5	-2.4					
	STANDARD					0.59	0.74	0.59				AVERAGE			
	DEVIATION					0.58	0.72	0.58				PCT DIFFERENCE			15
	=0.015					-1.4	-2.5	-1.4				= 1.2			

Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	1	48	91	6,780	ARO

FIGURE B-5

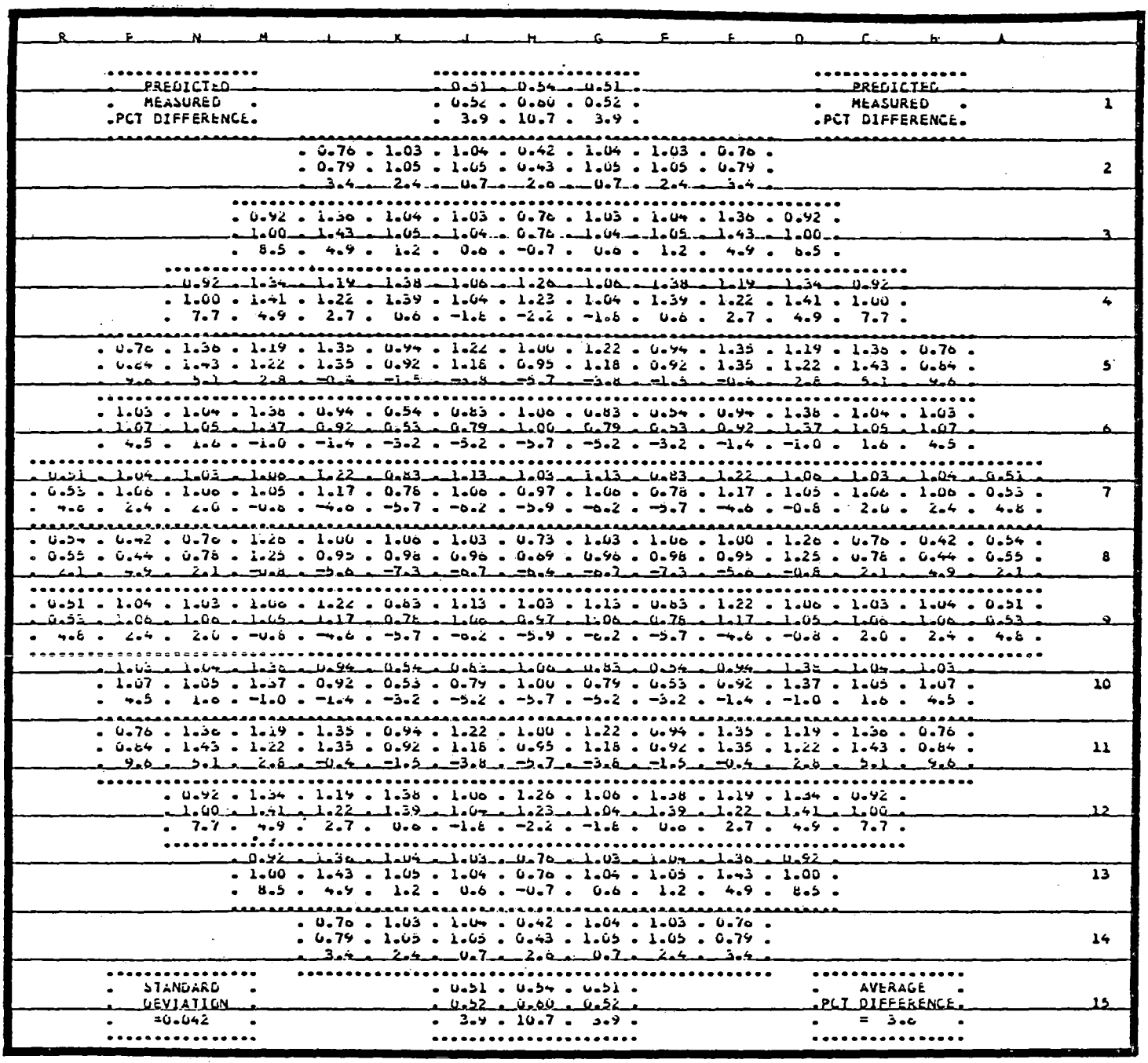
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT END OF LIFE CONDITIONS



Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	1	64	98	13,650	ARO

FIGURE B-6

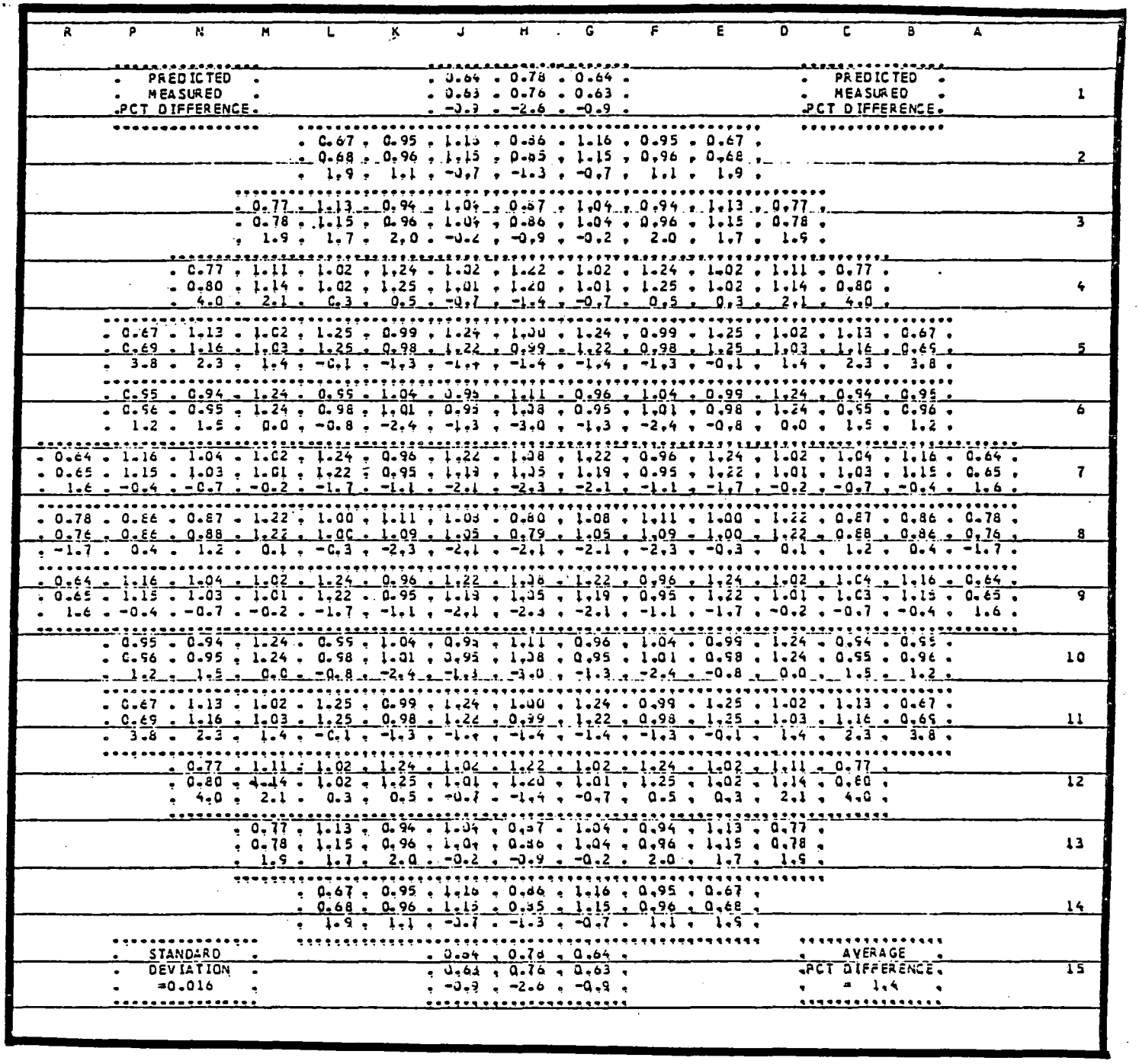
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR RELOAD CORE AT BEGINNING OF LIFE CONDITION



Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	2	2	2	0	D-In

FIGURE B-7

INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR RELOAD CORE AT BEGINNING OF LIFE CONDITIONS



Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	2	12	97	2,790	ARO

FIGURE B-8

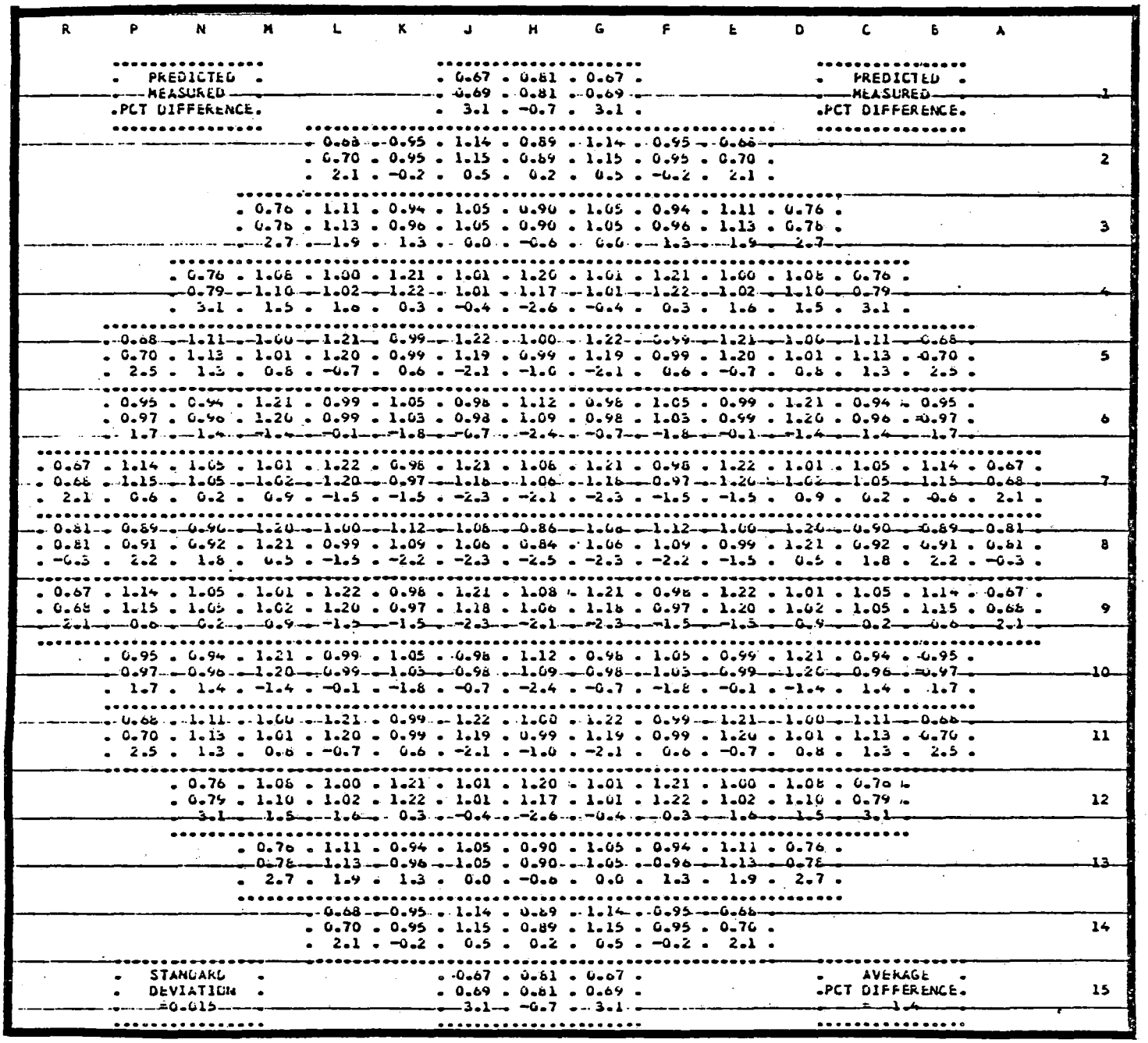
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR RELOAD CORE AT MIDDLE OF LIFE CONDITIONS

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
	PREDICTED					0.65	0.79	0.65				PREDICTED			
	MEASURED					0.65	0.76	0.65				MEASURED			1
	PCT DIFFERENCE					-0.7	-3.7	-0.7				PCT DIFFERENCE			
				0.68	0.95	1.15	0.87	1.15	0.95	0.68					
				0.69	0.96	1.14	0.86	1.14	0.96	0.69					2
				2.0	0.9	-0.9	-1.9	-0.9	0.9	2.0					
			0.77	1.13	0.94	1.04	0.88	1.04	0.94	1.13	0.77				
			0.78	1.14	0.96	1.03	0.86	1.03	0.96	1.14	0.78				3
			1.1	1.1	1.6	-0.7	-1.7	-0.7	1.6	1.1	1.1				
		0.77	1.10	1.01	1.23	1.01	1.21	1.01	1.23	1.01	1.10	0.77			
		0.80	1.12	1.01	1.22	1.00	1.19	1.00	1.22	1.01	1.12	0.80			4
		3.6	1.4	-0.1	-1.0	-1.0	-1.9	-1.0	-1.0	-0.1	1.4	3.6			
	0.68	1.13	1.01	1.24	0.98	1.23	1.00	1.23	0.98	1.24	1.01	1.13	0.68		
	0.70	1.15	1.02	1.23	0.98	1.21	0.99	1.21	0.98	1.23	1.02	1.15	0.70		5
	3.8	1.8	0.7	-0.7	-0.9	-1.3	-0.6	-1.3	-0.9	-0.7	0.7	1.8	3.8		
	0.95	0.94	1.23	0.98	1.04	0.97	1.11	0.97	1.04	0.98	1.23	0.94	0.95		
	0.97	0.95	1.23	0.99	1.02	0.96	1.08	0.96	1.02	0.99	1.23	0.95	0.97		6
	1.4	1.4	-0.3	0.2	-1.5	-0.7	-2.8	-0.7	-1.5	0.2	-0.3	1.4	1.4		
0.65	1.15	1.04	1.01	1.23	0.97	1.21	1.07	1.21	0.97	1.23	1.01	1.04	1.15	0.65	
0.67	1.15	1.03	1.01	1.21	0.96	1.19	1.05	1.19	0.96	1.21	1.01	1.03	1.15	0.67	7
2.0	-0.0	-1.4	-0.2	-1.3	-0.4	-1.3	-2.1	-1.8	-0.4	-1.3	-0.2	-1.4	-0.0	2.0	
0.75	0.87	0.88	1.21	1.00	1.11	1.07	0.82	1.07	1.11	1.00	1.21	0.88	0.87	0.79	
0.78	0.88	0.89	1.21	1.00	1.09	1.05	0.81	1.05	1.09	1.00	1.21	0.89	0.88	0.78	8
-1.9	0.4	0.9	-0.1	0.2	-2.0	-1.7	-1.9	-1.7	-2.0	0.2	-0.1	0.9	0.4	-1.9	
0.65	1.15	1.04	1.01	1.23	0.97	1.21	1.07	1.21	0.97	1.23	1.01	1.04	1.15	0.65	
0.67	1.15	1.03	1.01	1.21	0.96	1.19	1.05	1.19	0.96	1.21	1.01	1.03	1.15	0.67	9
2.0	-0.0	-1.4	-0.2	-1.3	-0.4	-1.3	-2.1	-1.8	-0.4	-1.3	-0.2	-1.4	-0.0	2.0	
	0.95	0.94	1.23	0.98	1.04	0.97	1.11	0.97	1.04	0.98	1.23	0.94	0.95		
	0.97	0.95	1.23	0.99	1.02	0.96	1.08	0.96	1.02	0.99	1.23	0.95	0.97		10
	1.4	1.4	-0.3	0.2	-1.5	-0.7	-2.8	-0.7	-1.5	0.2	-0.3	1.4	1.4		
	0.68	1.13	1.01	1.24	0.98	1.23	1.00	1.23	0.98	1.24	1.01	1.13	0.68		
	0.70	1.15	1.02	1.23	0.98	1.21	0.99	1.21	0.98	1.23	1.02	1.15	0.70		11
	3.8	1.8	0.7	-0.7	-0.9	-1.3	-0.6	-1.3	-0.9	-0.7	0.7	1.8	3.8		
		0.77	1.10	1.01	1.23	1.01	1.21	1.01	1.23	1.01	1.10	0.77			
		0.80	1.12	1.01	1.22	1.00	1.19	1.00	1.22	1.01	1.12	0.80			12
		3.6	1.4	-0.1	-1.0	-1.0	-1.9	-1.0	-1.0	-0.1	1.4	3.6			
			0.77	1.13	0.94	1.04	0.88	1.04	0.94	1.13	0.77				
			0.78	1.14	0.96	1.03	0.86	1.03	0.96	1.14	0.78				13
			1.1	1.1	1.6	-0.7	-1.7	-0.7	1.6	1.1	1.1				
				0.68	0.95	1.15	0.87	1.15	0.95	0.68					
				0.69	0.96	1.14	0.86	1.14	0.96	0.69					14
				2.0	0.9	-0.9	-1.9	-0.9	0.9	2.0					
	STANDARD					0.65	0.79	0.65				AVERAGE			
	DEVIATION					0.65	0.76	0.65				PCT DIFFERENCE			15
	=0.014					-0.7	-3.7	-0.7				= 1.3			

Unit	Cycle	M/D Map	Power(%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	2	16	100	4,520	ARO

FIGURE B-9

INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR RELOAD CORE AT END OF LIFE CONDITIONS



Unit	Cycle	M/D Map	Power (%)	Core Burnup (MWD/MTU)	Control Rod Configuration
2	2	23	99	8,850	ARO

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

R. H. LEASBURG
VICE PRESIDENT
NUCLEAR OPERATIONS

August 10, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. D. G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
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NPF-4
NPF-7

Gentlemen:

VEPCO NUCLEAR DESIGN
TOPICAL REPORTS

In response to the requests in the letters from Mr. Robert L. Tedesco dated May 18, 1981, May 20, 1981 and May 13, 1981 accepting the Vepco topical reports VEP-FRD-19, VEP-FRD-20, and VEP-FRD-24, respectively, for reference in licensing actions by Vepco, we have issued revised versions of these reports which include the NRC evaluation letter and its attachment between the title page and the abstract. Five (5) copies of each of these revised versions of the reports are enclosed for your use.

If you have any questions, please contact us.

Very truly yours,

R. H. Leasburg
Vice President
Nuclear Operations

Enclosure

cc: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing

Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

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