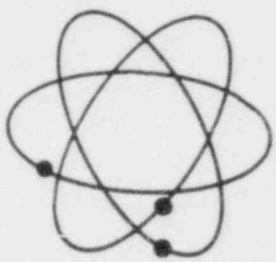
VEP-FRD-19A
JULY, 1981

Vepco

THE PDQ 07 DISCRETE MODEL



FUEL RESOURCES DEPARTMENT
VIRGINIA ELECTRIC AND POWER COMPANY

VEP-FRD-19A

THE PDQ07 DISCRETE MODEL

by

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Nuclear Fuel Engineering Group
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Virginia Electric and Power Company
Richmond, Virginia

July, 1981

Recommended for Approval:

Martin L. Bowling

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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, RAPDQ, 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

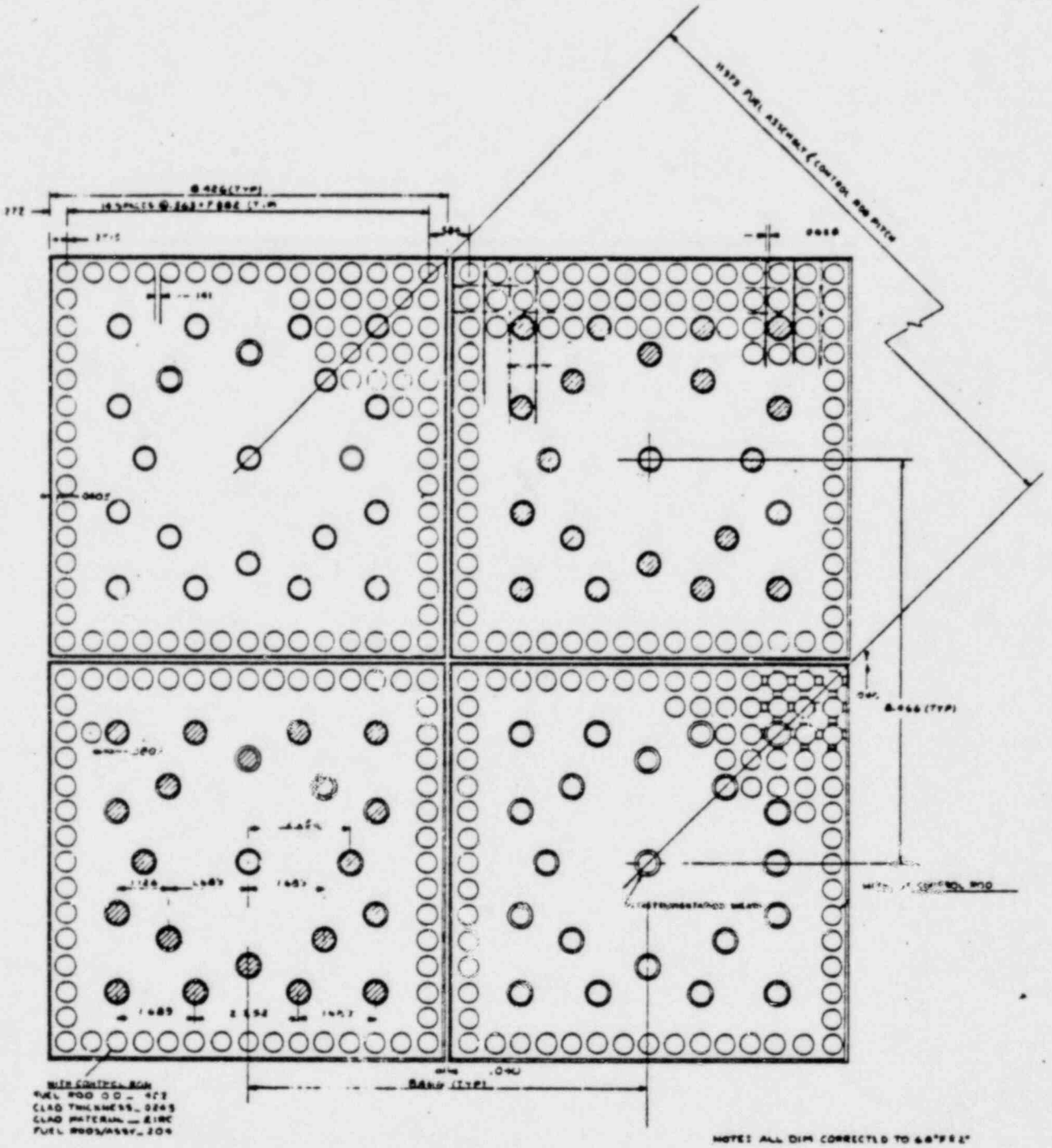
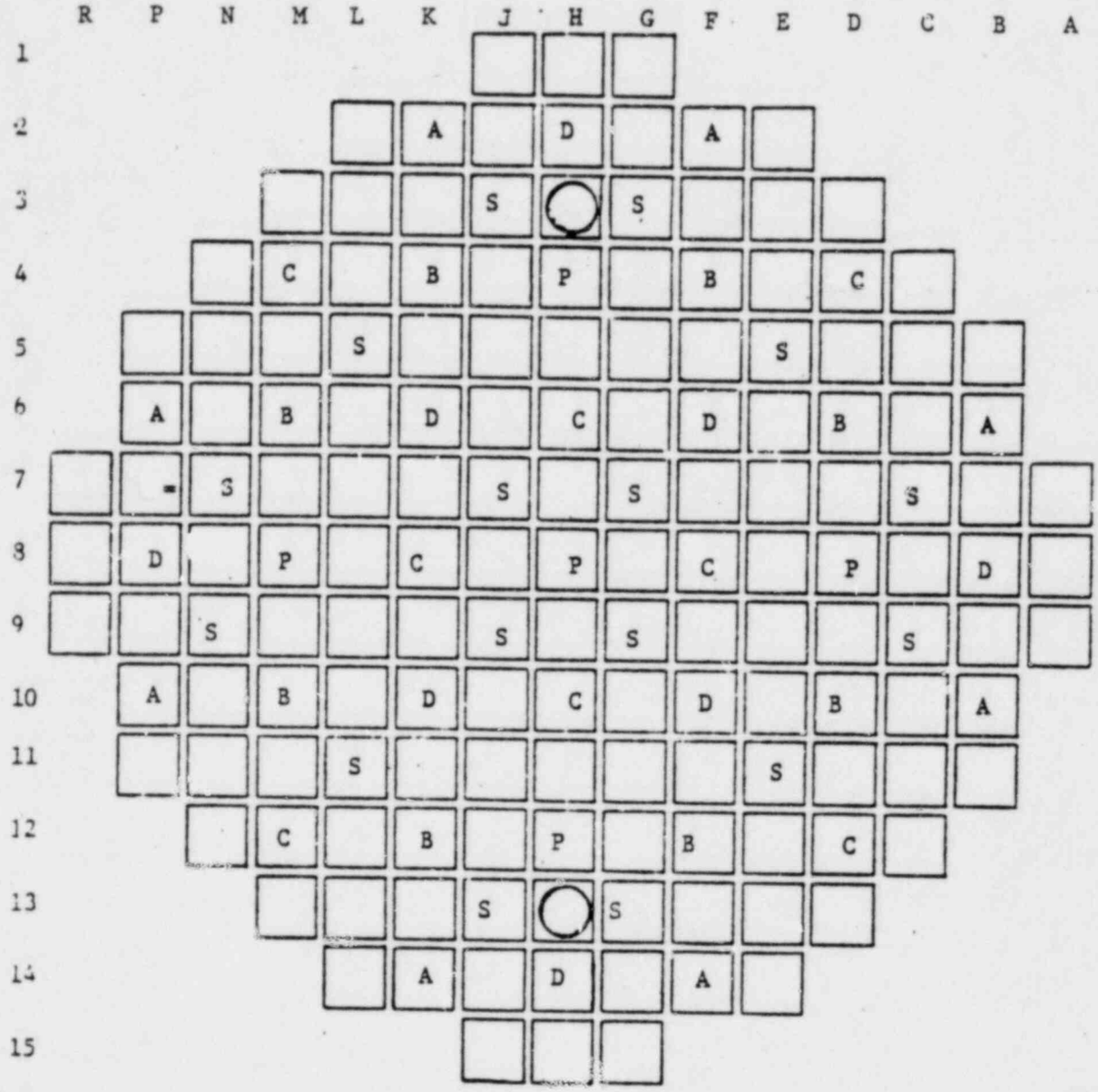


FIGURE 2-2

CONTROL ROD BANK LOCATIONS



CONTROL ROD ASSEMBLY BANKS

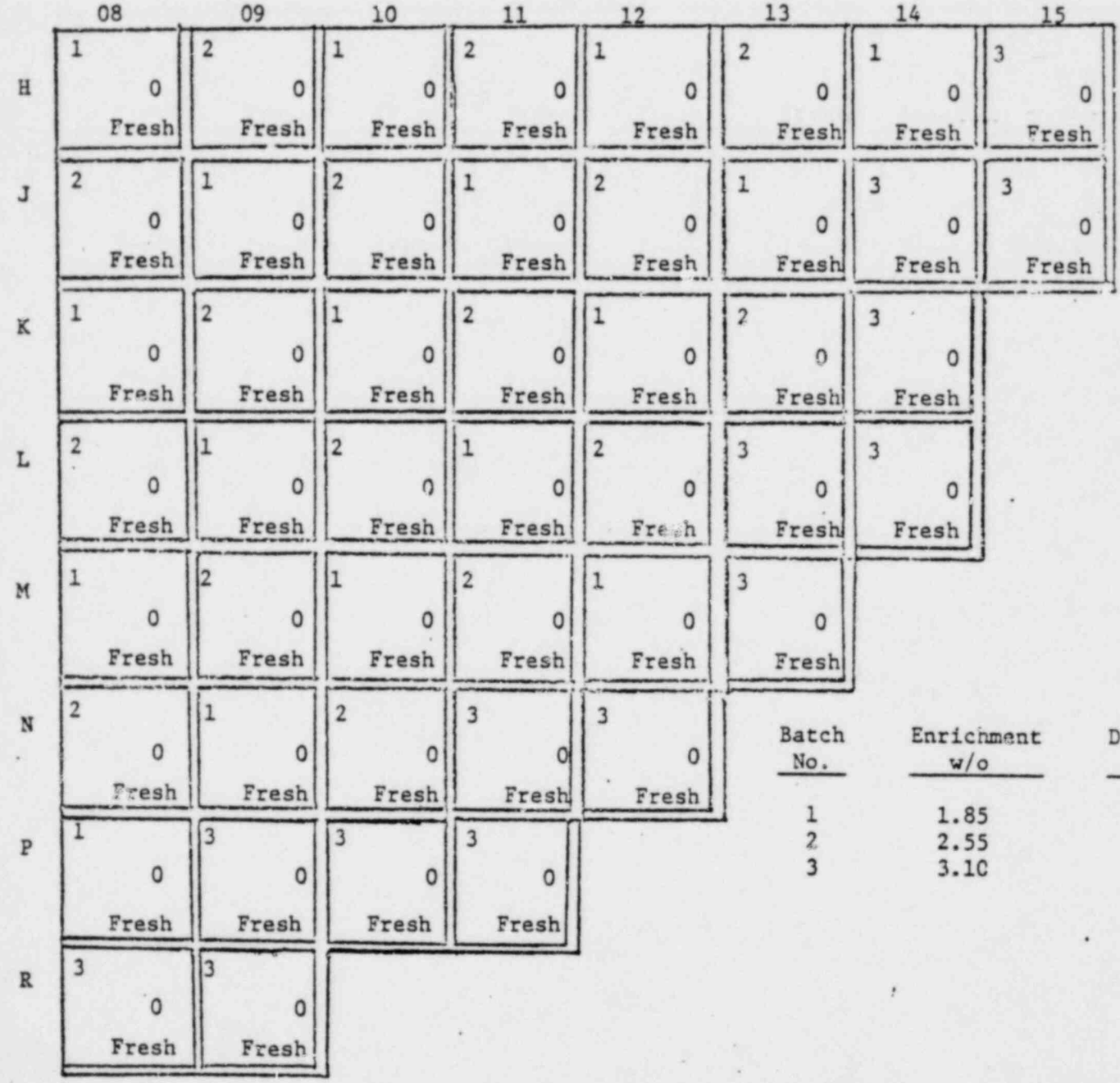
Function	Number of Assemblies
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown (S)	16
Part Length (P)	5
	53

◯ = SOURCE ASSEMBLY LOCATIONS

Figure 2-3

SURRY UNITS 1 AND 2 -- CYCLE 1

FUEL LOADING



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 J13	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
yy
zz

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

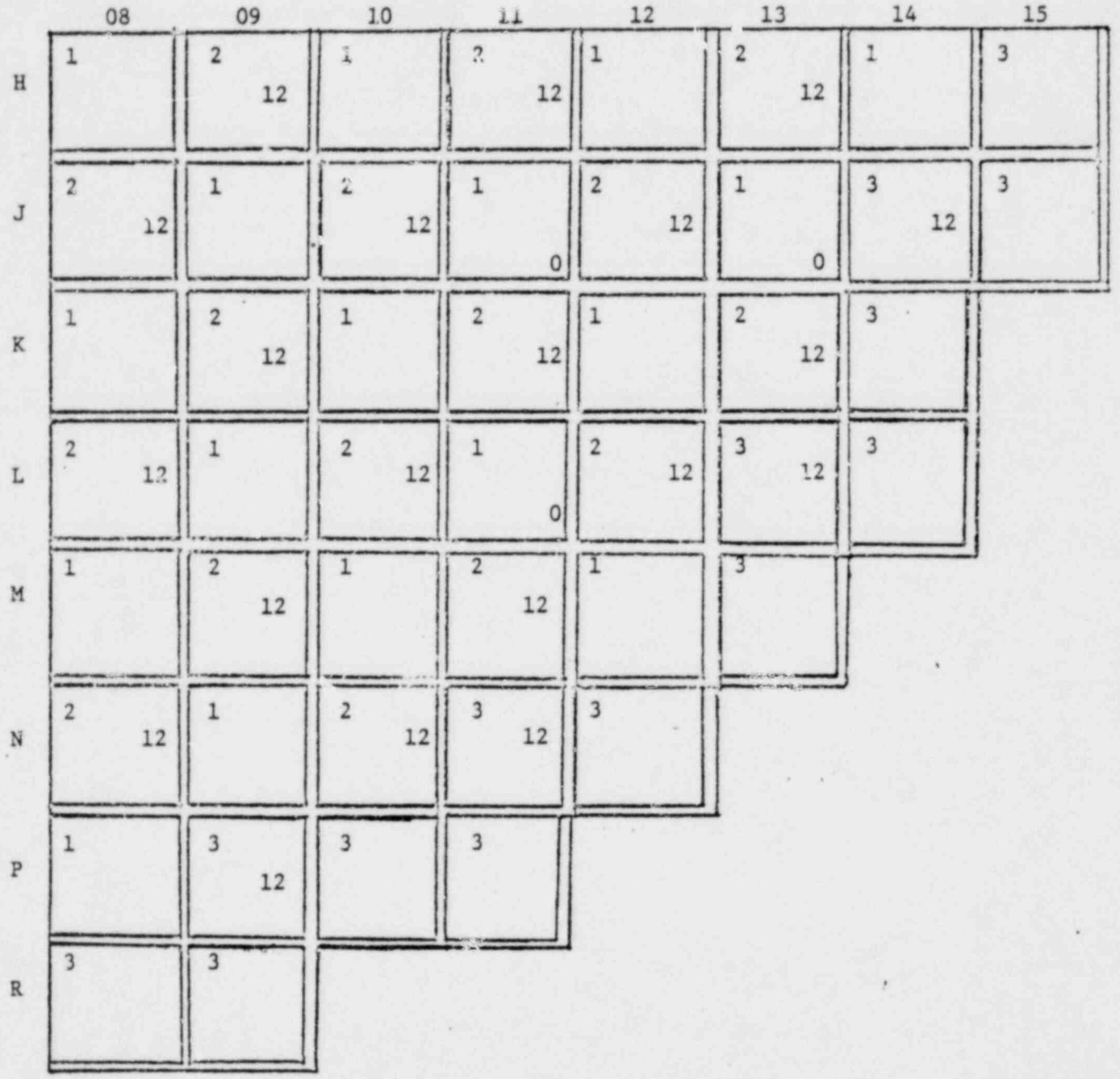


Figure 2-7

SURRY UNIT 1 -- CYCLE 2

BURNABLE POISON ROD LOADING

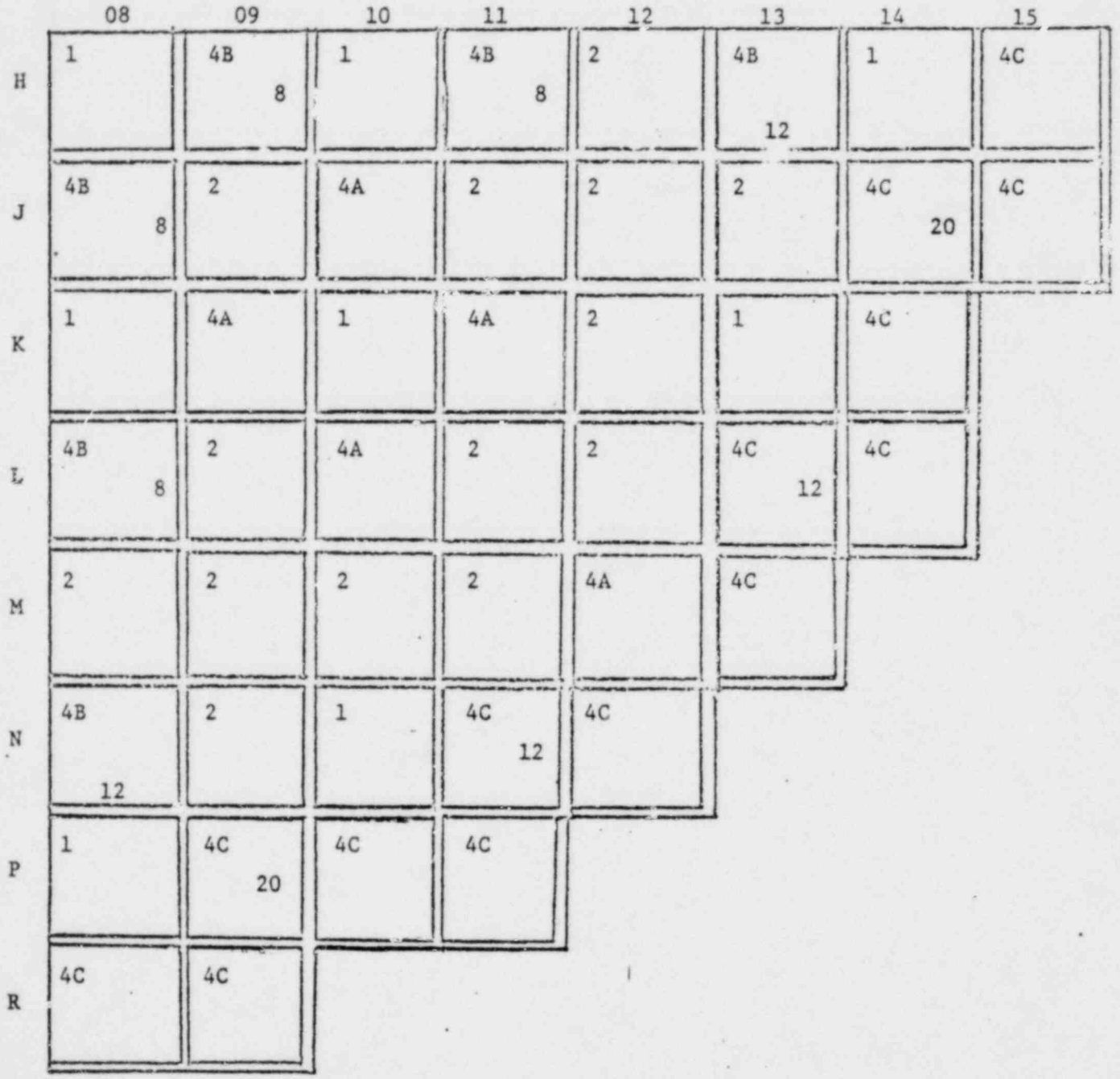
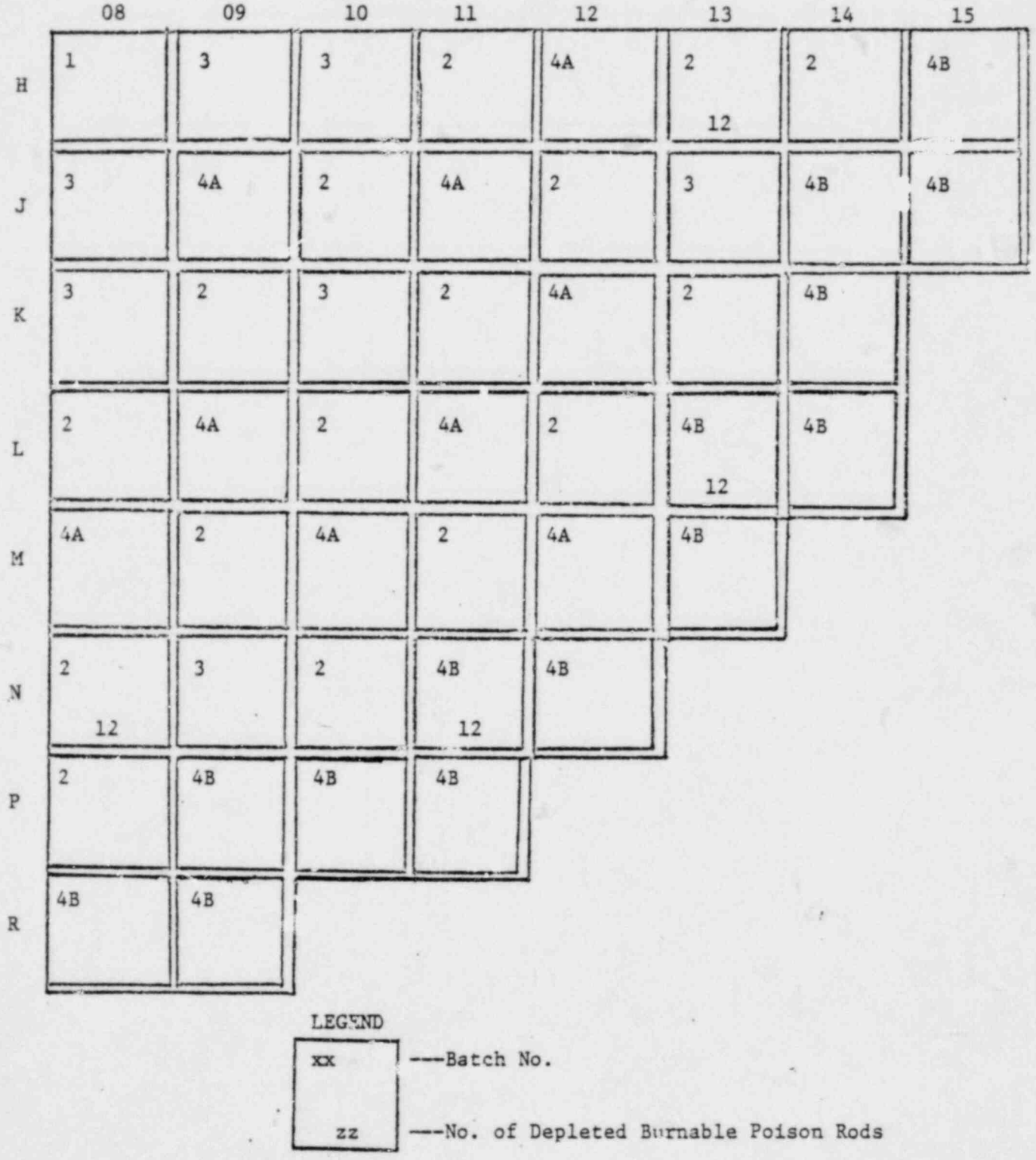


Figure 2-8

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

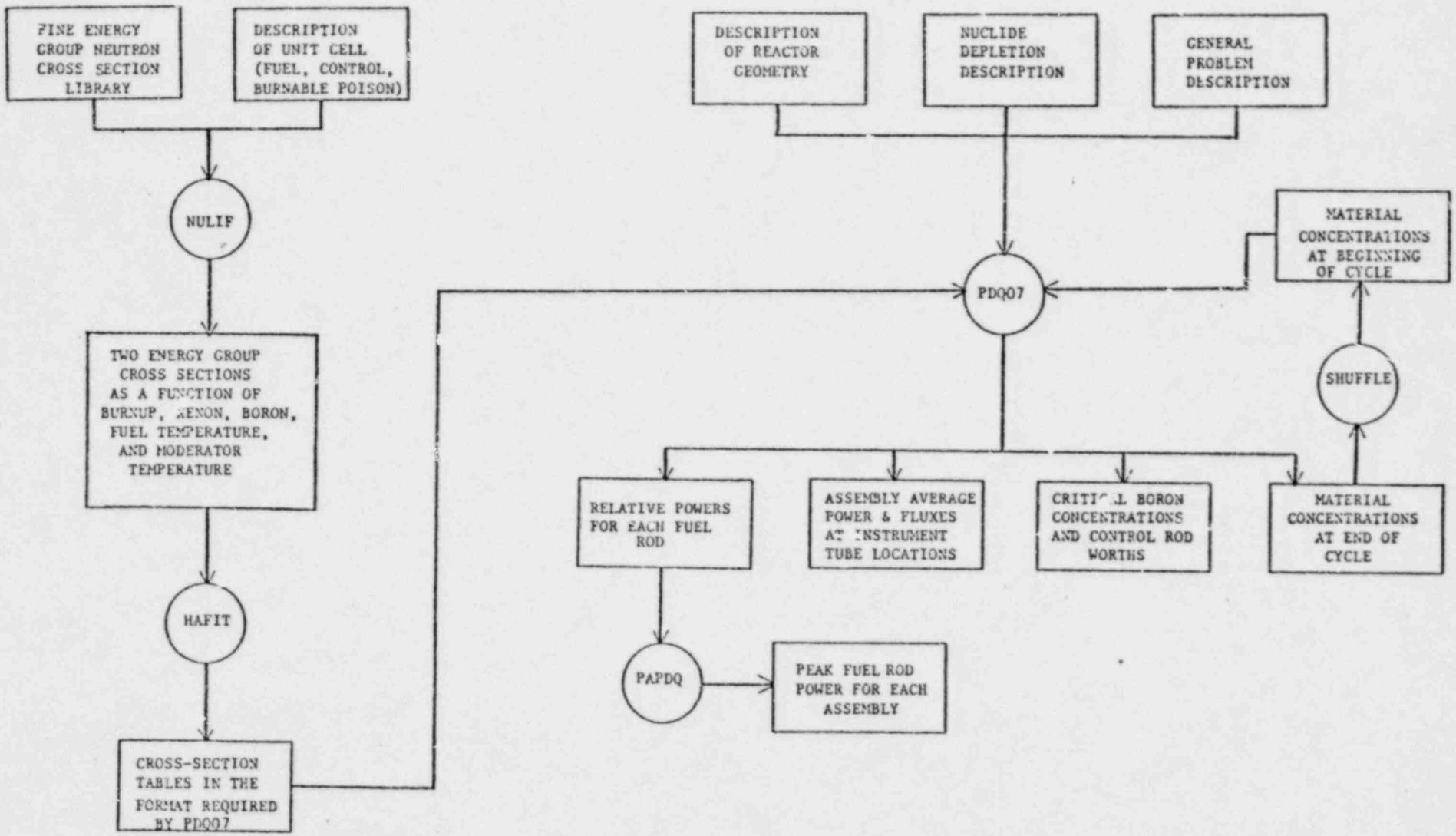
iterative diffusion theory calculation of the neutron flux as a function of radial position. From the neutron flux and cross sections the core power distribution is determined, and subsequently, the fuel and moderator temperature distributions are calculated. Thermal feedback effects are included in 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 HAFNT) 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

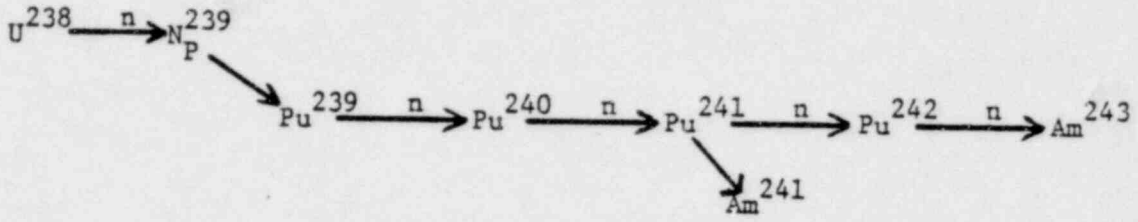
2) Heated perimeter per unit area of flow

Table 3-3

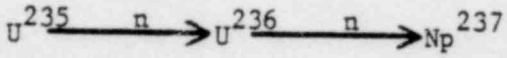
Depletion Equations used in PDQ07

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:

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

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	

Average absolute value 2.0

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	Average absolute value 2.8
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	

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	8[illegible]0	1.6	
23	99	ARO	88[illegible]0	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	Average absolute value 3.6
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	

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	Average absolute value 1.4
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	

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	

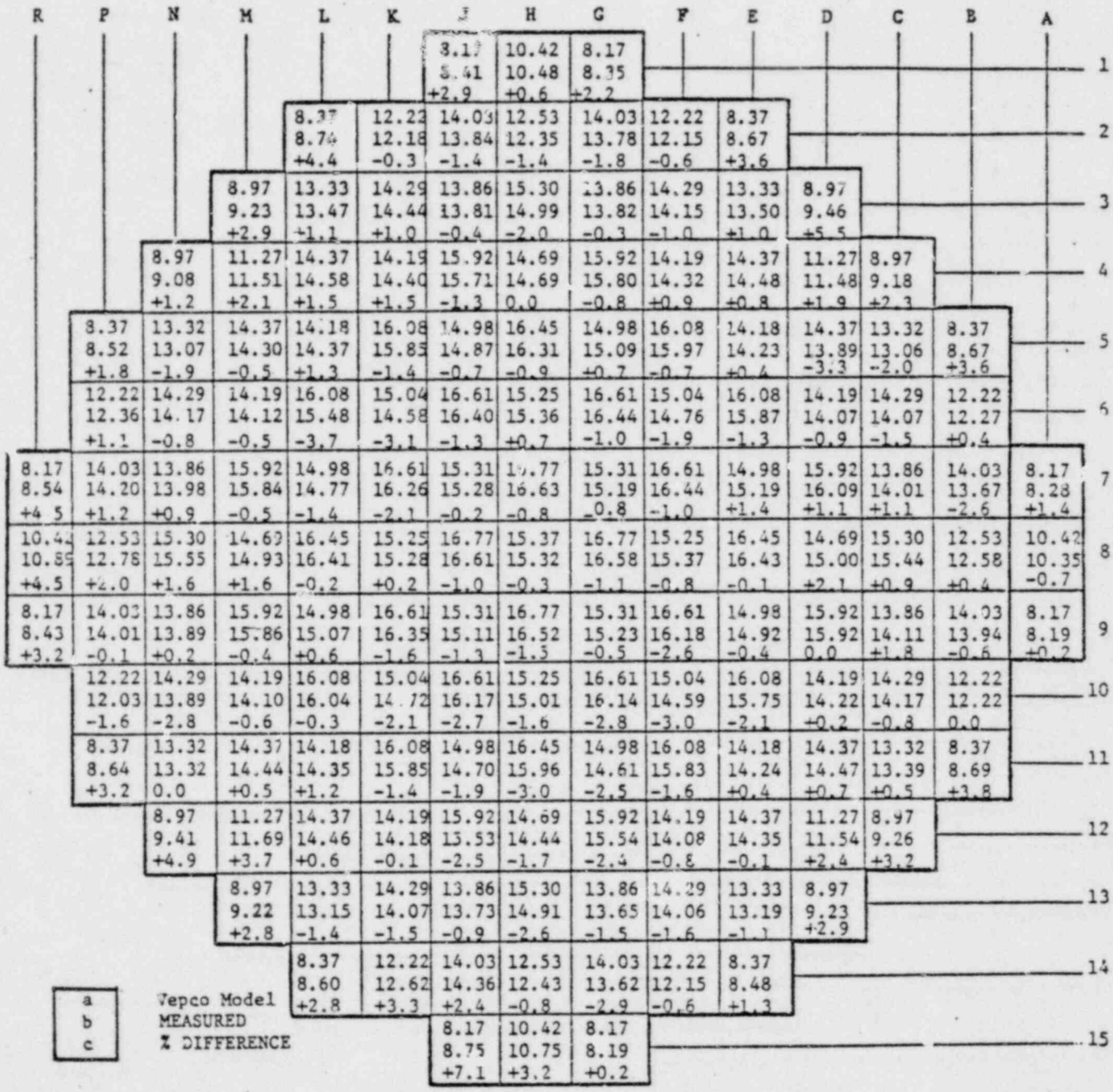
Table 4-10

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	
12	97	ARO	2790	1.331	1.332	+0.1	Average absolute value 1.2
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



1						8.17 8.41 +2.9	10.42 10.48 +0.6	8.17 8.35 +2.2						
2				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				
3			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			
4		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		
5	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	
6	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	
7	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
8	10.44 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
9	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
10	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	
11	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	
12		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		
13			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			
14				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				
15						8.17 8.75 +7.1	10.42 10.75 +3.2	8.17 8.19 +0.2						

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	Average absolute value 1.3
Batch 2	15.46	15.62	-1.0	
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

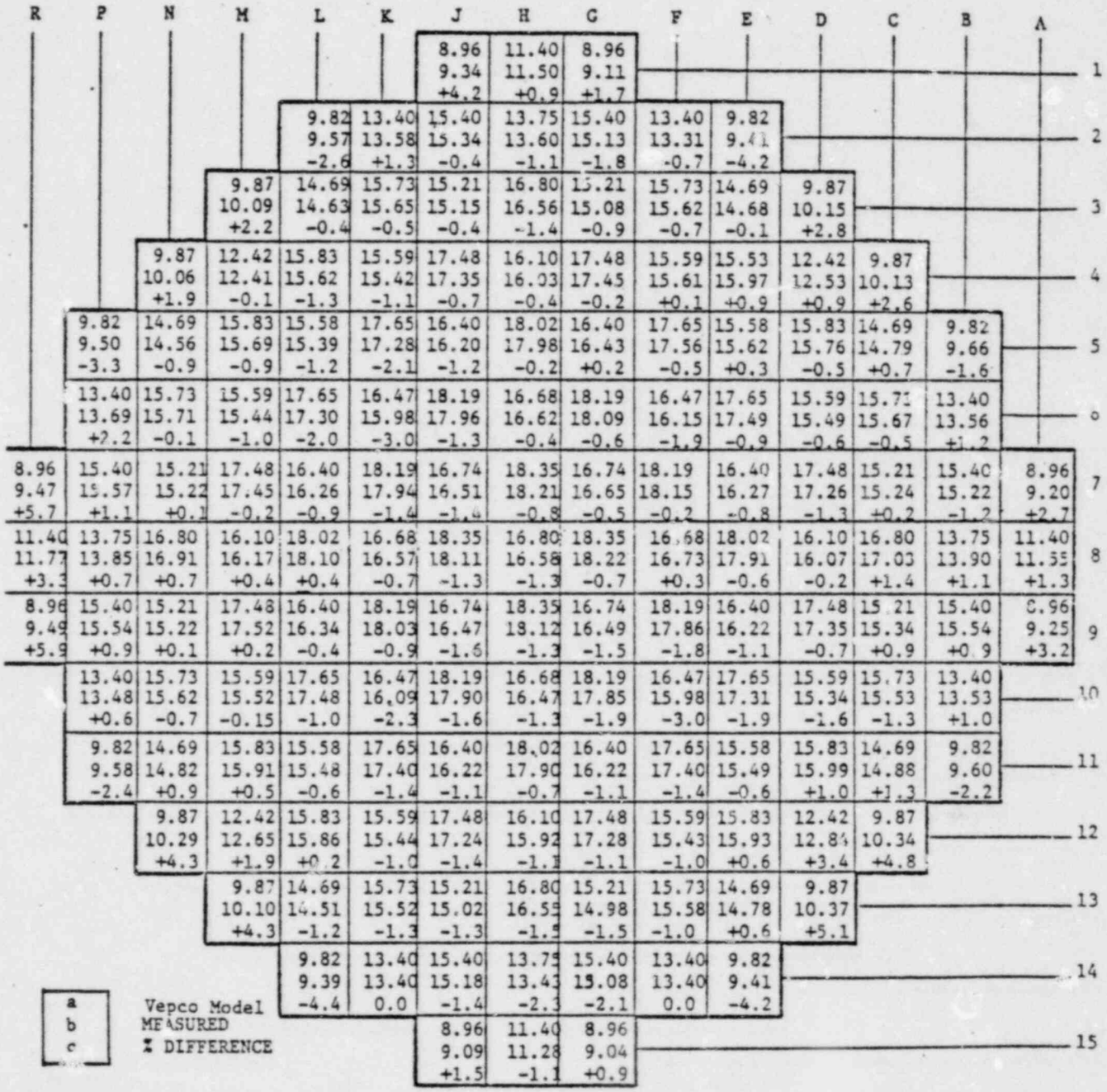
R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
						4.92 5.00 +1.6	6.26 6.16 -1.6	4.92 4.82 -2.0							1
				5.25 5.47 +4.2	7.16 7.28 +1.7	6.99 6.35 -9.2	20.49 20.19 -1.5	6.99 6.22 -11.0	7.16 7.05 -1.5	5.25 5.38 +2.5					2
			5.69 6.02 -10.0	7.62 7.44 -2.4	20.04 20.19 +0.8	21.49 21.57 +0.4	8.64 8.27 -4.3	21.49 21.41 -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.[illegible] -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.[illegible] -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.[illegible]	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
a b c	Vepco Model MEASURED % DIFFERENCE			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

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	

TABLE 4-13

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



1						8.96 9.34 +4.2	11.40 11.50 +0.9	8.96 9.11 +1.7						
2				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				
3			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			
4		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		
5		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
6		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	15.71 15.67 -0.5	13.40 13.56 +1.2
7	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
8	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
9	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
10		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	
11		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	
12			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		
13				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			
14					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				
15							8.96 9.09 +1.5	11.40 11.28 -1.1	8.96 9.04 +0.9						

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	10.94 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 5.93 +3.7	7
6.98 7.26 +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 10.96 -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 25.70 -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	Average absolute value 2.4
2	ARO	1182	1168	-1.2	
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	
2	ARO	1408	1401	-0.5	Average absolute value 0.7
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	↑
1	C and D-Bank In	1330	1234	-5.1	Average absolute value 4.9
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

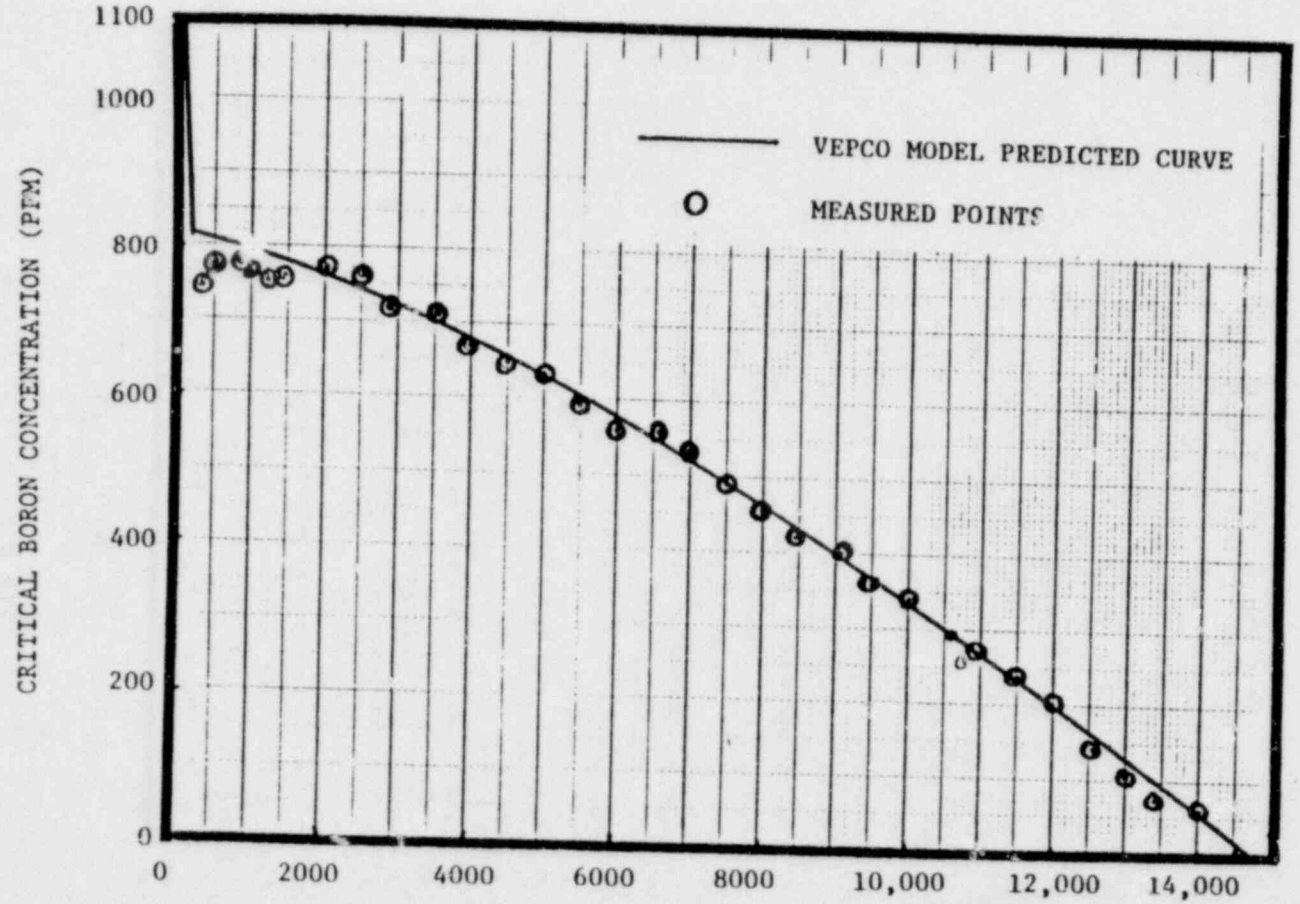


FIGURE 4.2

SURRY UNIT 1 - CYCLE 2

CRITICAL BORON CONCENTRATION
vs.
BURNUP

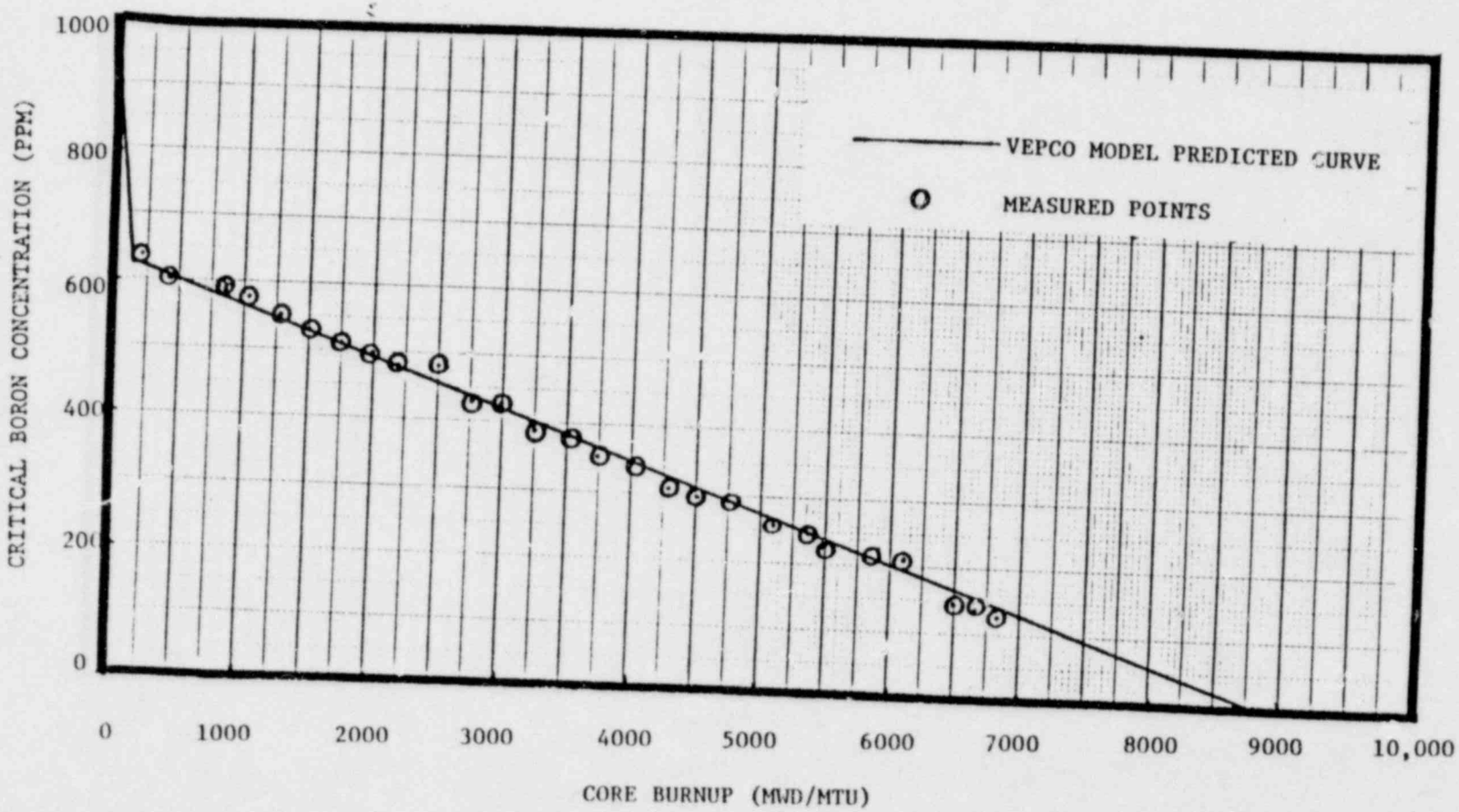


FIGURE 4.3

SURRY UNIT 2 - CYCLE 1

CRITICAL BORON CONCENTRATION
vs.
BURNUP

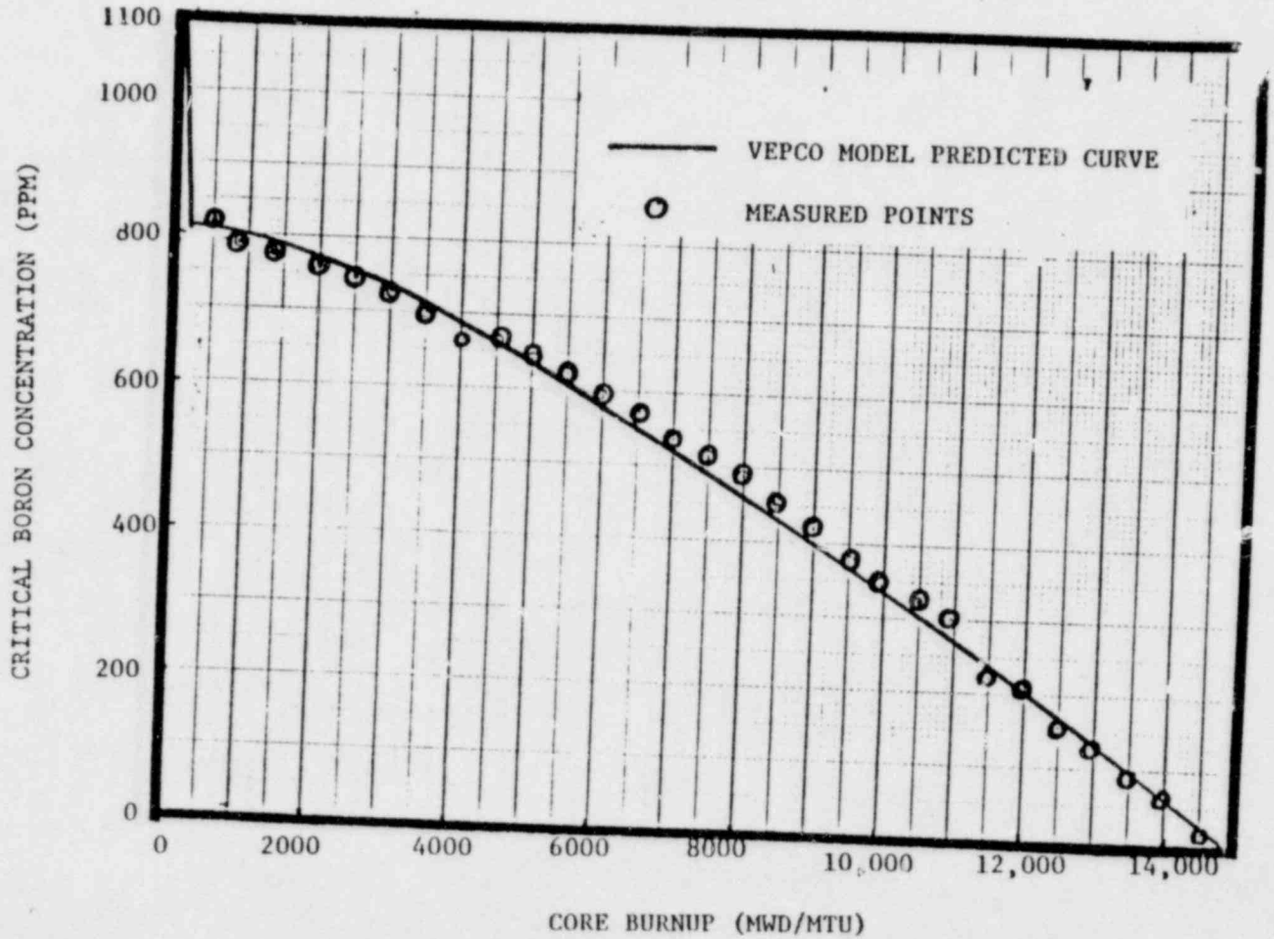


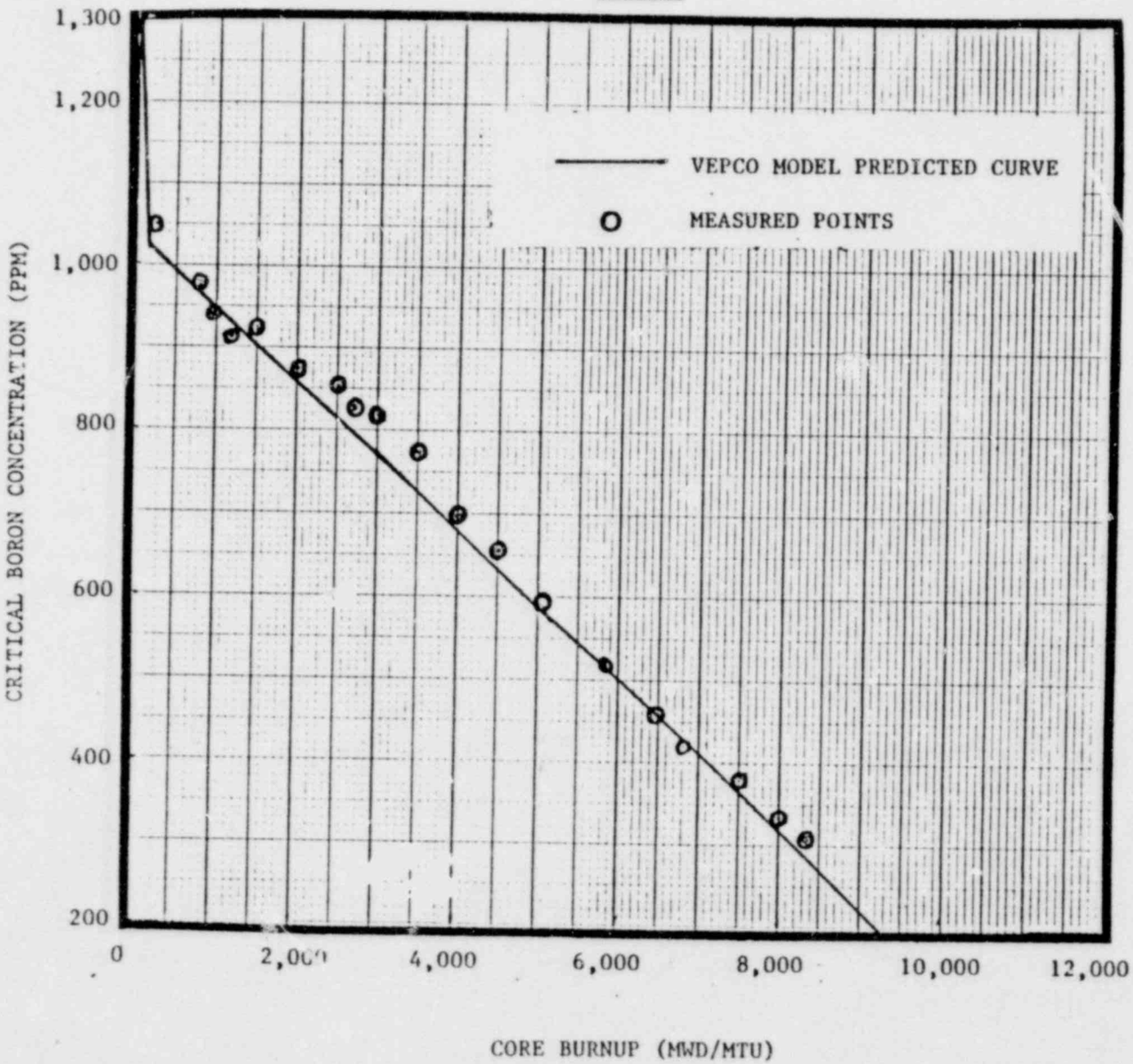
FIGURE 4.4

SURRY UNIT 2 - CYCLE 2

CRITICAL BORON CONCENTRATION

vs.

BURNUP



SECTION 5 - SUMMARY AND CONCLUSION

The PDQC7 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., "ULIF - "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:

$$\begin{aligned}\text{corrected or nominal boron concentration} = {} & \text{measured boron concentration} \\ & + \text{inverse boron worth} \times \Big[\text{off-nominal reactivity correction due to rod group 1 position} \\ & + \text{off-nominal reactivity correction due to rod group 2 position} \\ & + \text{off-nominal reactivity correction due to moderator temperature} \\ & + \text{off-nominal reactivity correction due to power} \\ & + \text{off-nominal reactivity correction due to xenon and samarium behavior}\Big]\end{aligned}$$

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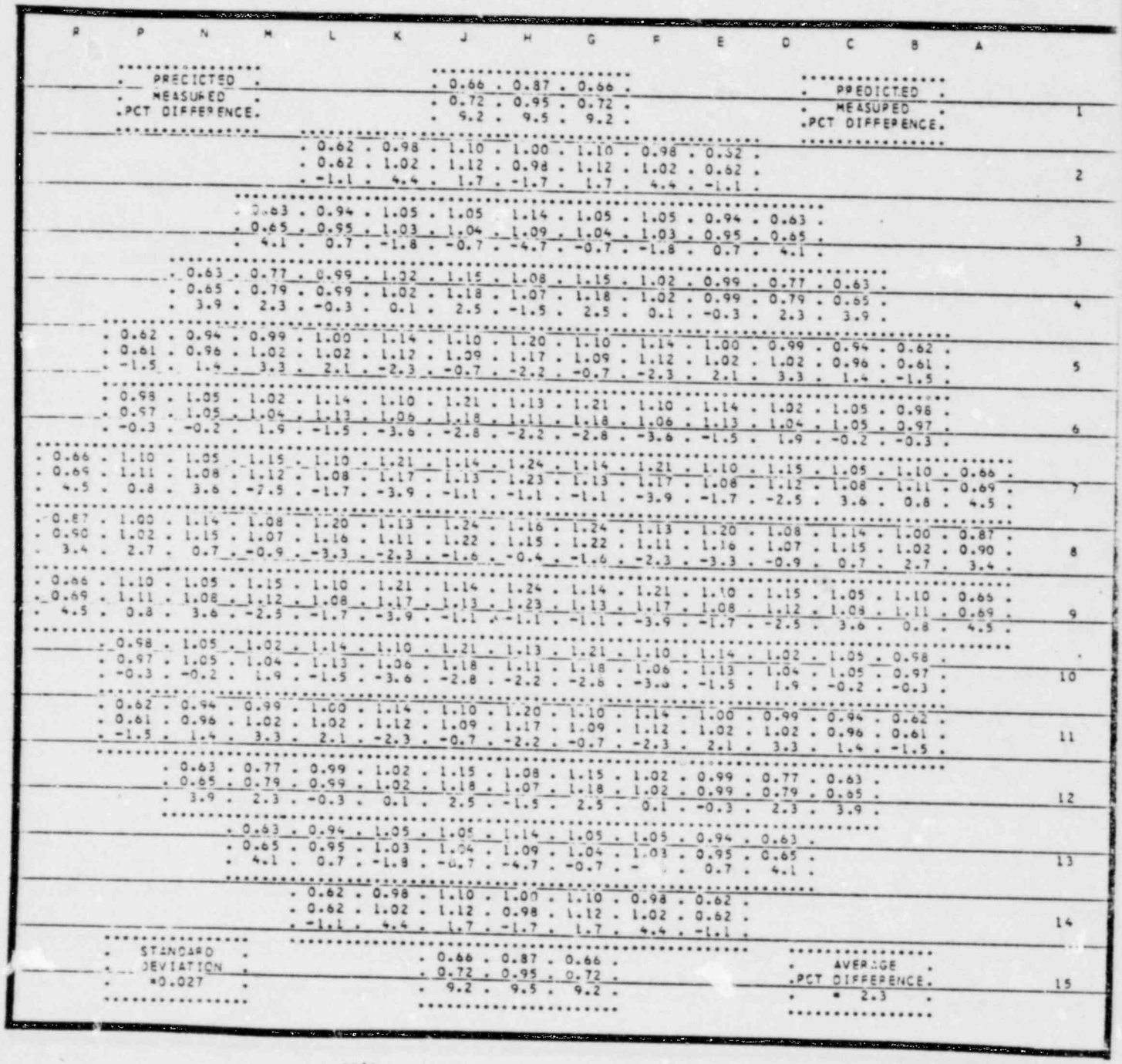
* Nominal conditions are associated with hot full power, equilibrium xenon conditions, and all control rods withdrawn from the core.

APPENDIX B

REPRESENTATIVE INCORE OUTPUT USING THE VEPCO PDQ07 DISCRETE MODEL

FIGURE B-1

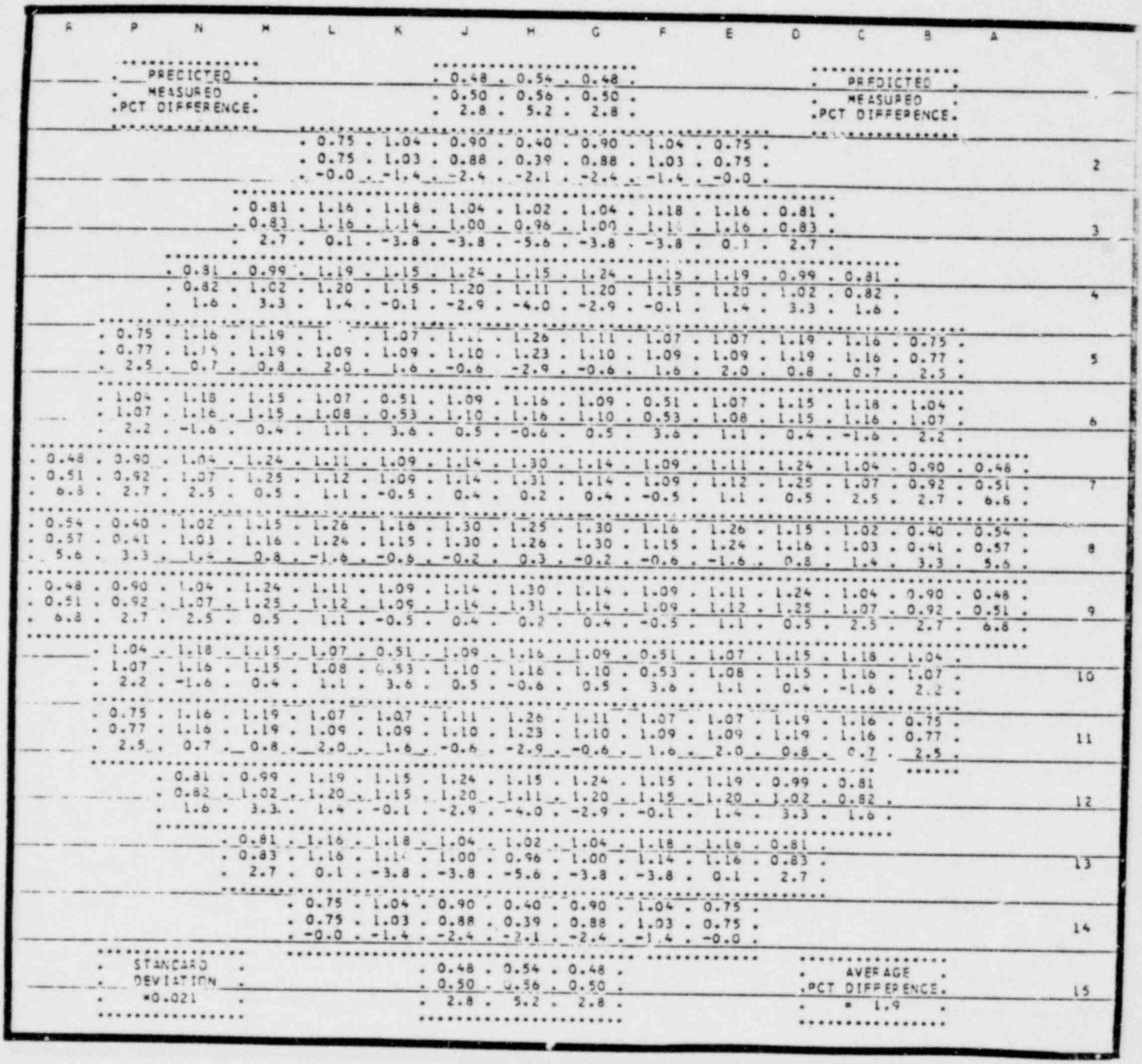
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT BEGINNING OF LIFE CONDITION



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

FIGURE B-2

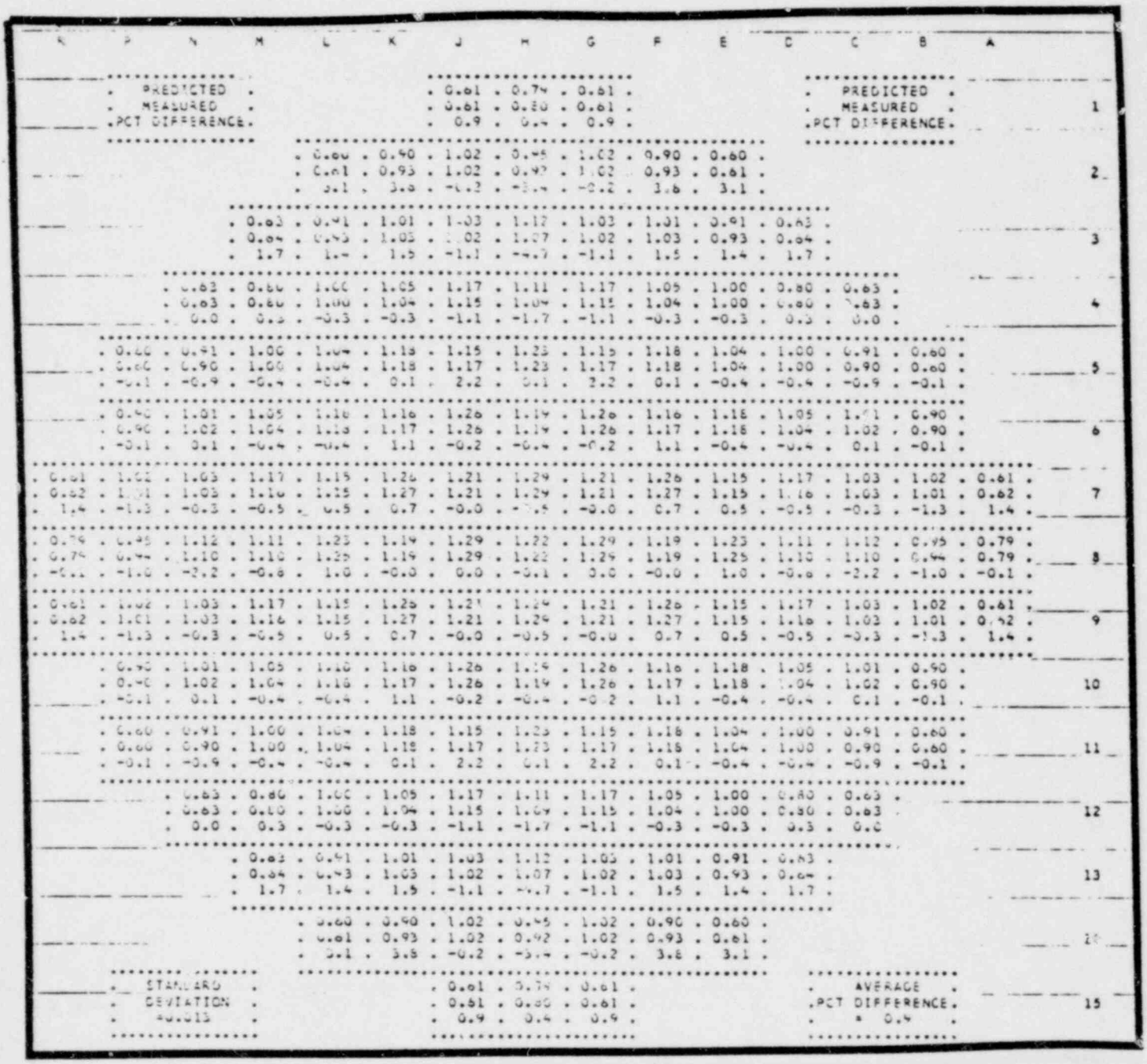
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT BEGINNING OF LIFE CONDITION



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

FIGURE B-3

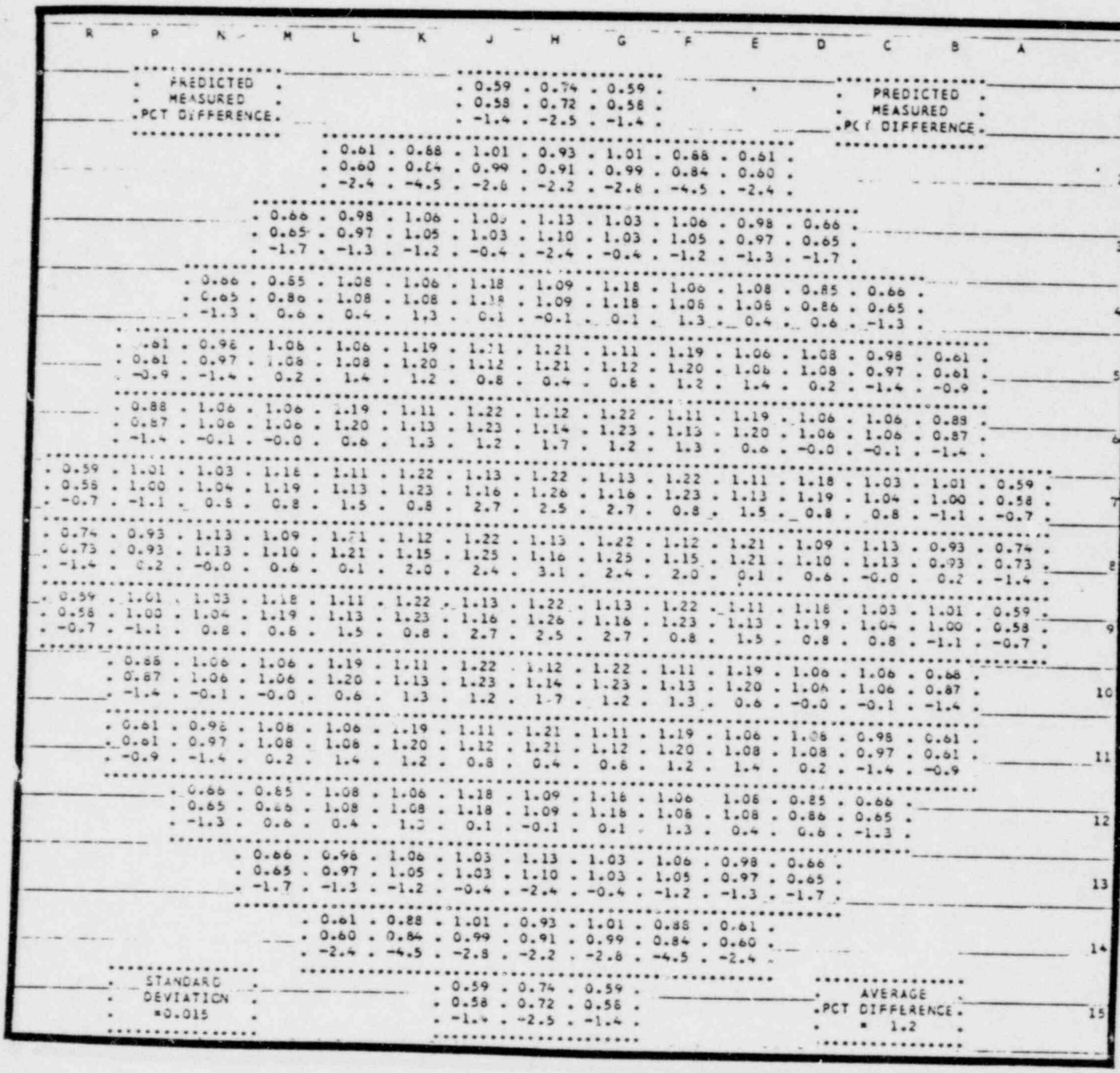
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT BEGINNING OF LIFE CONDITIONS



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

FIGURE B-4

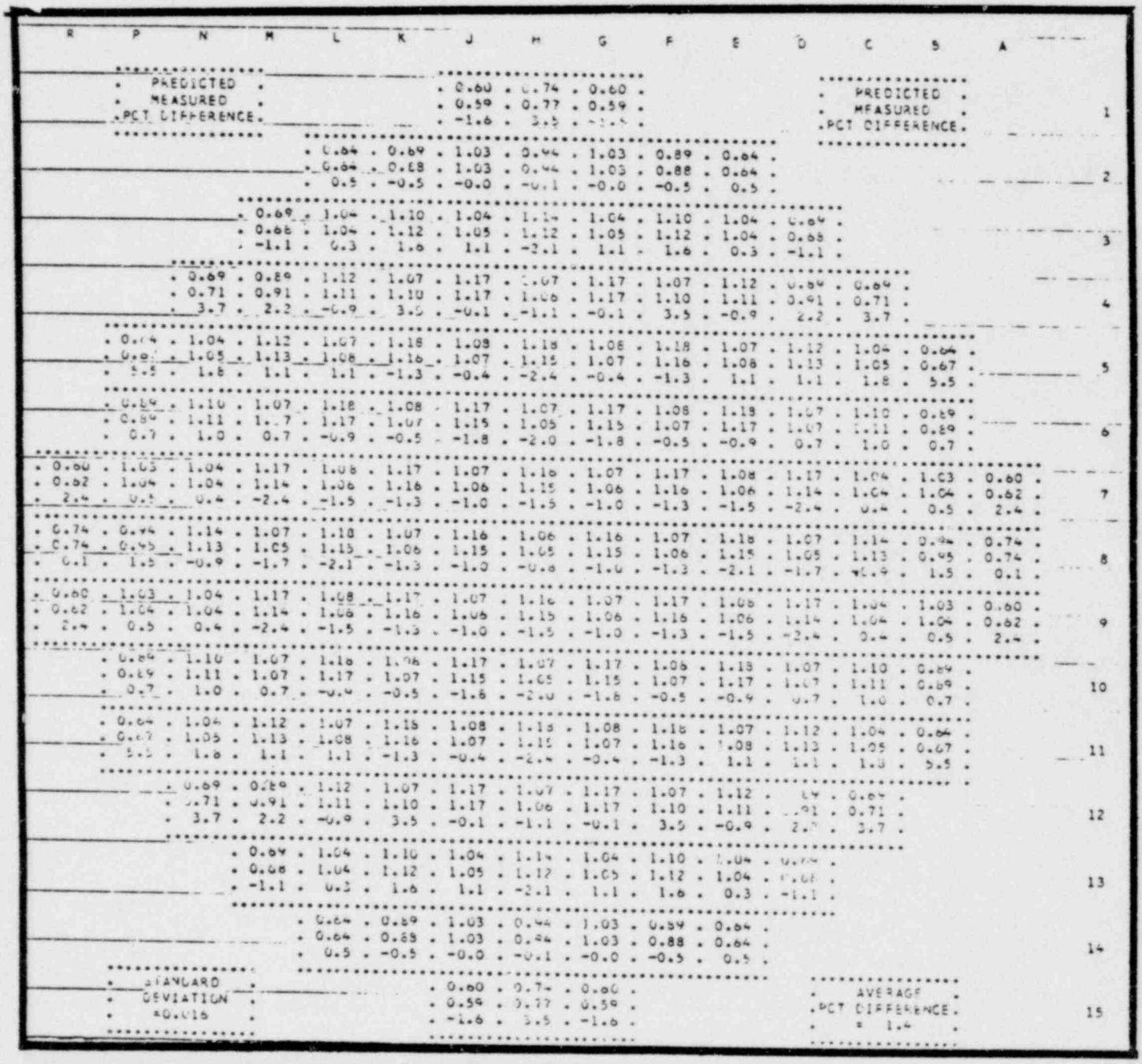
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT MIDDLE OF LIFE CONDITIONS



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

FIGURE B-5

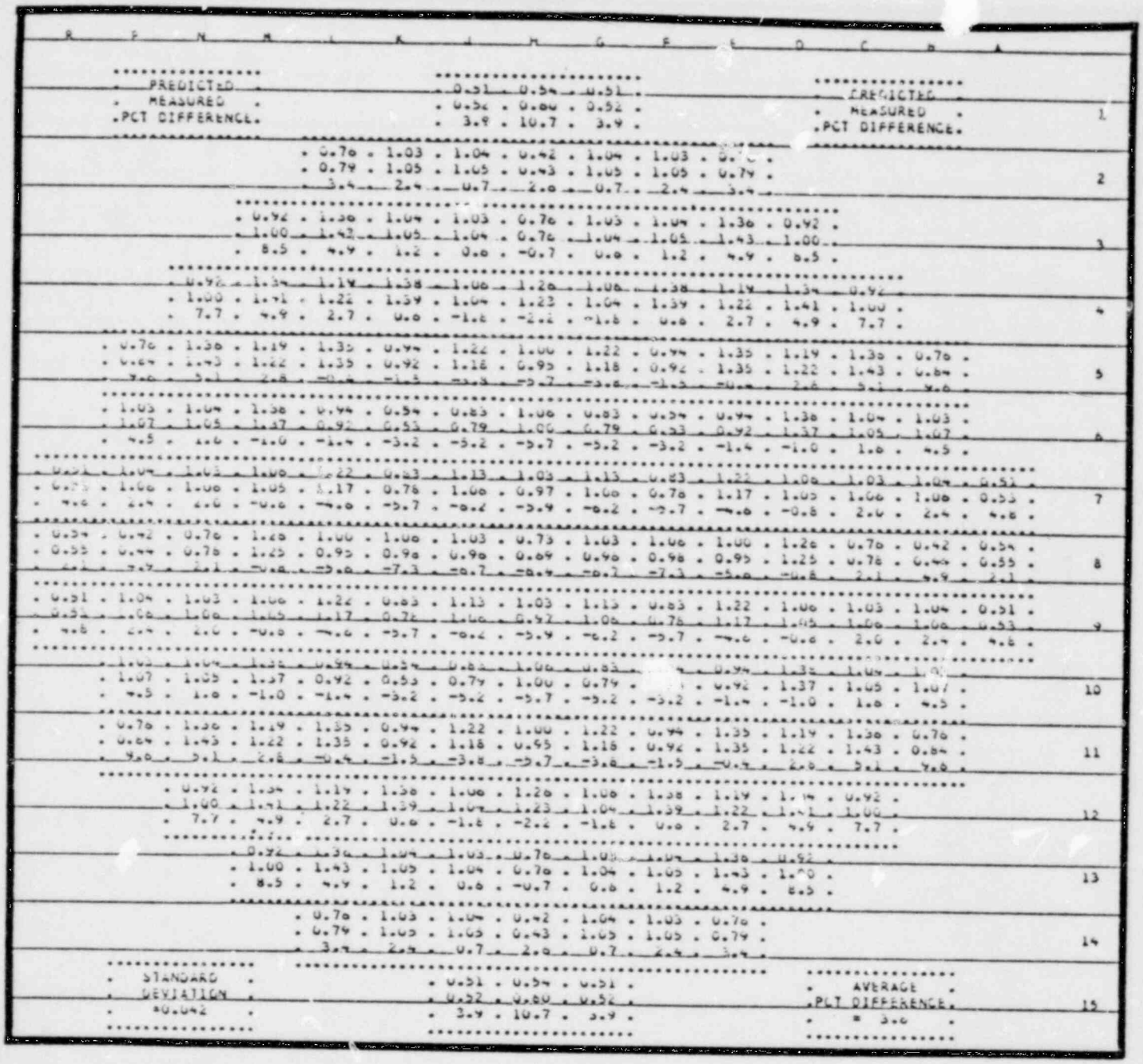
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR INITIAL CORE AT END OF LIFE CONDITIONS



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

FIGURE B-6

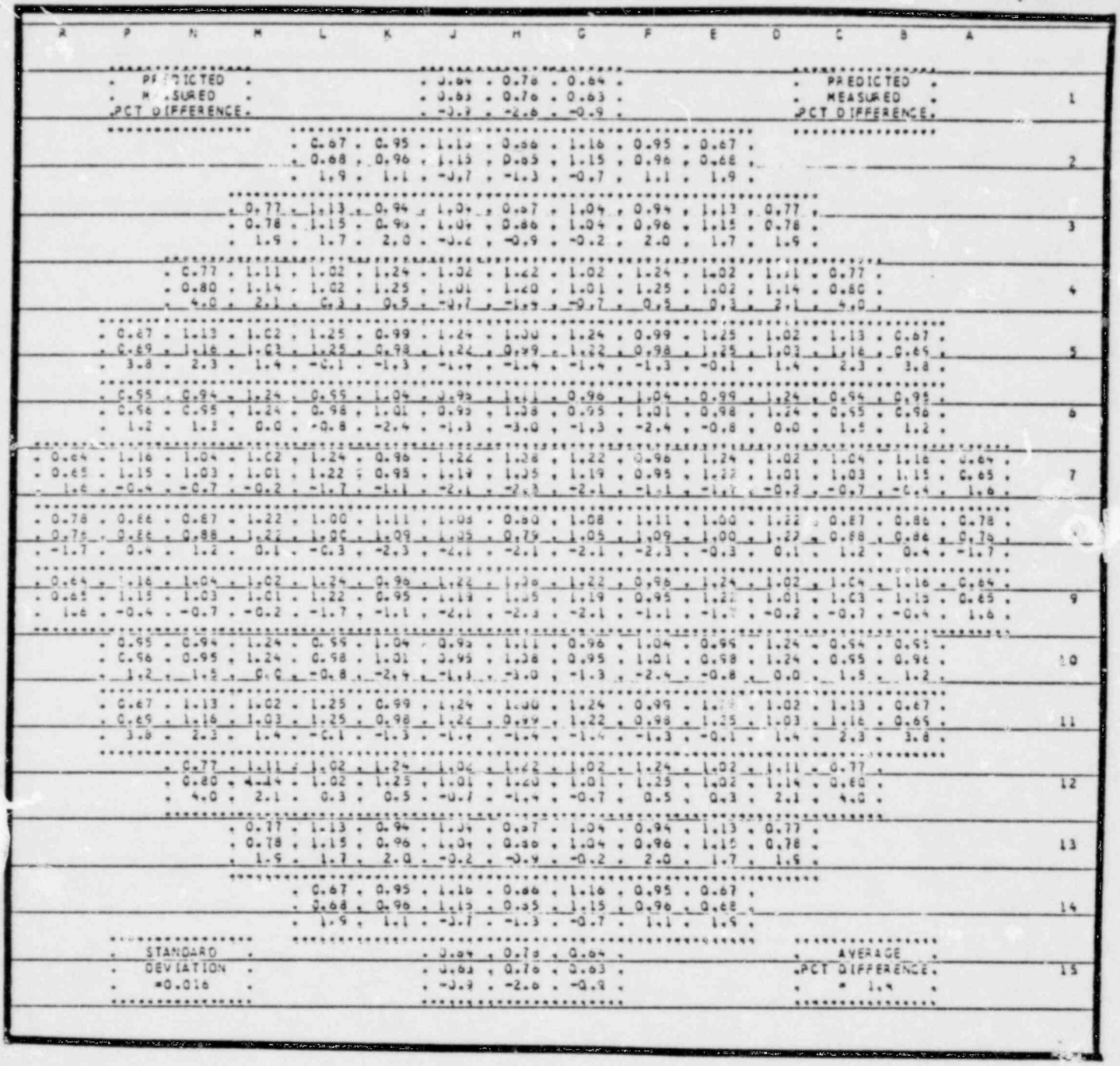
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR RELOAD CORE AT BEGINNING OF LIFE CONDITION



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

FIGURE B-7

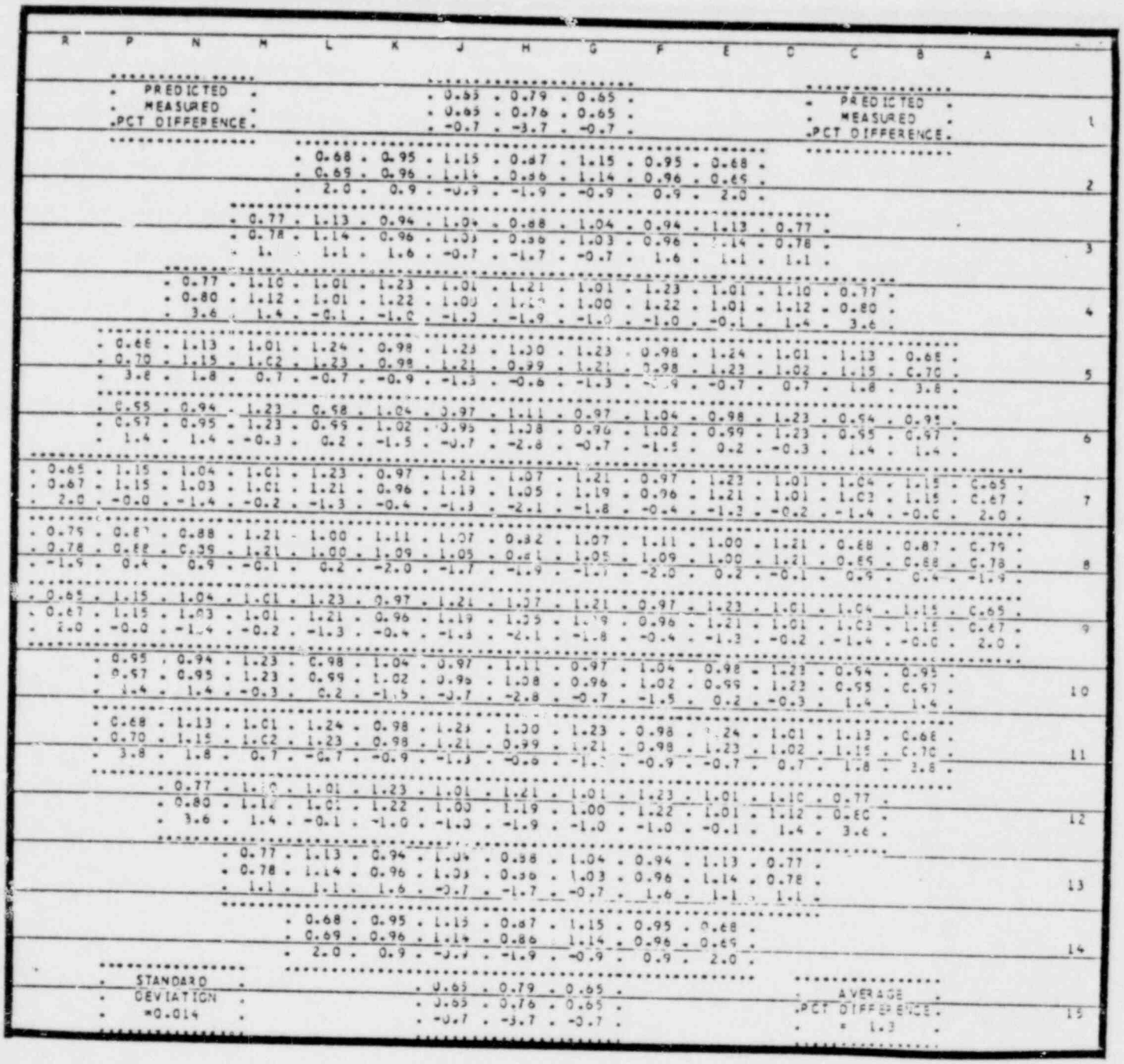
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR RELOAD CORE AT BEGINNING OF LIFE CONDITIONS



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

FIGURE B-8

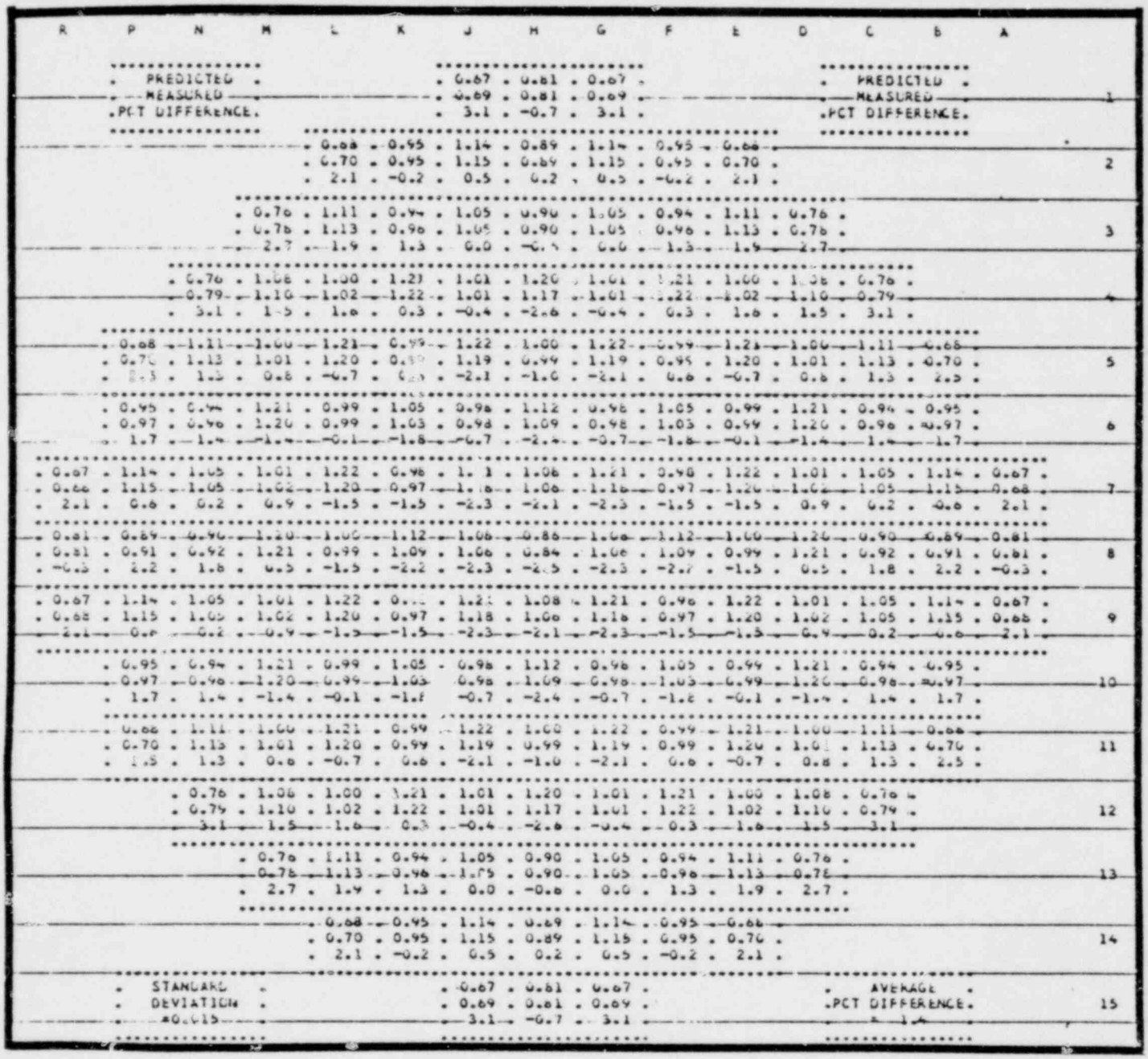
INCORE CALCULATED ASSEMBLYWISE AVERAGE POWER DISTRIBUTION
FOR RELOAD CORE AT MIDDLE OF LIFE CONDITIONS



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