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CRITICAL EXPERIMENTS AND COOPERATIVE EVALUATIONS PROGRAM

SECOND QUARTERLY REPORT
JANUARY-MARCH 1972

PREPARED FOR
THE UNITED STATES ATOMIC ENERGY COMMISSION
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SECOND QUARTERLY REPORT
JANUARY-MARCH 1972**

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ABSTRACT

This report is the second in a series of quarterly progress reports published under the Critical Experiments and Cooperative Evaluations Program. Current activities and technical progress for the period January through March, 1972, are reported for two tasks. Task A covers the planning and analysis of critical experiments in ZPPR-3. Task B progress is reported for nuclear data evaluation and testing and methods development in support of Task A.

1. S U M M A R Y

TASK A - CRITICAL EXPERIMENT PLANNING AND ANALYSIS

The specifications for the core loading and experimental plan for ZPPR Assembly 3 have been modified to reflect revisions in the Demonstration Plant design as a result of further analysis, AEC guidelines, and a desire to utilize FTR technology where possible. The program now consists of a short series of experiments, a portion of the original phase 1, using the originally specified core loading, a reduction in core size and execution of the original three phase experimental plan with the smaller core. "Fuel spiking" will be used to maintain criticality where the core size is reduced. Each simulated control rod position will be loaded with B_4C to give a nominal value of 2.2\$ per rod. The total B_4C requirement is 166 kgs.

The critical size of the ZPPR Assembly 3 core (original specification) was predicted to be 1166 matrix positions. Criticality was achieved with a core loading of 1144 matrix positions.

Pre-analysis of control rod measurements indicates substantial flux and power tilting when a peripheral control rod is inserted.

The technique of processing nuclear cross section data for ZPPR-3 analysis is described. A system of material identification in ZPPR-3 cross section files is also described.

TASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

ENDRUN was used to prepare a 59 group nuclear data file from the ENDF/B-III library. Preliminary testing of the file is completed and two benchmarks, ZPR6-7 and ZPPR-2, were calculated using it.

A procedure and two three-dimensional test problems have been specified for evaluating the accuracy of the flux linking technique used in the TRIAL code.

BRD recommendations for a new nuclear data processing code were presented at a meeting in Washington, D. C., on February 24 and 25. The BRD recommendations call for a two-step approach based on the Bondarenko technique but with broader applications than available with current codes.

2. TASK A - CRITICAL EXPERIMENT PLANNING AND ANALYSIS

Task Leader: H. S. Bailey

2.1 Modification of Core Loading and Experimental Plan

The specifications for the core loadings and experimental plan for ZPPR Assembly 3 have been modified to reflect revisions in the Demonstration Plant design as a result of further analysis, AEC guidelines, and a desire to utilize FTR technology where possible. The program now consists of a short series of experiments, a portion of the original Phase 1, using the originally specified core loading, a reduction in core size and execution of the original three phase experimental plan with the smaller core. The large core (original specification) measurements are now labeled "Phase 1A" and subsequent measurements in the smaller core are phases 1B, 2, and 3.

The overall experimental program is outlined in Table 2-1. This modification to the experimental program will provide data of control worth in two different size reactors in addition to providing data more directly applicable to current estimates of demonstration reactor needs. These changes in ZPPR Assembly 3 are expected to extend the experimental program by about 2 months.

The general characteristics of ZPPR Assembly 3 are listed in Table 2-2 for both configurations. The new layout of ZPPR Assembly 3 is shown in Figure 2-1. As indicated, ZPPR-3 still contains 19 simulated control locations in a triangular array with an average pitch of 13 inches. The quantity of B_4C in each control rod position (CRP) will provide a nominal control rod worth of \$2.2. For phases 1B, 2, and 3, ZPPR Assembly 3 Core Zone 1 contains 378 matrix positions (radius 62.0 cm) of which 50 positions simulate control; Core Zone 2 contains 434 matrix positions (radius 90.9 cm) of which 26 simulate control.

Reducing the size of ZPPR-3 will result in a reactivity loss of $\sim 3.3\%$ $\Delta K/K$. This reactivity loss will be compensated by "fuel spiking", the same technique that will be used to compensate for the 12.5\$ and 25\$ worth of control to be loaded into Phases 2 and 3, respectively.

TABLE 2-1

OUTLINE OF ZPPR-3 EXPERIMENTAL PROGRAM

PHASE 1A - END OF CYCLE CONFIGURATION

LARGER CORE LOADING

DRAWER WORTH MEASUREMENTS
REACTION RATE DISTRIBUTIONS
SINGLE CONTROL ROD EXPERIMENTS
ROD INTERACTION MEASUREMENTS
CONTROL SYSTEM WORTH MEASUREMENTS

PHASE 1B*- END OF CYCLE CONFIGURATION

SMALLER CORE LOADING

DRAWER WORTH MEASUREMENTS
REACTION RATE DISTRIBUTIONS
REACTIVITY WORTH TRAVERSES
SINGLE CONTROL ROD EXPERIMENTS
ROD INTERACTION MEASUREMENTS
CONTROL SYSTEM WORTH MEASUREMENTS

PHASE 2* - MID CYCLE CONFIGURATION

~12\$ CONTROL LOADED

LOAD CONTROL AND ENRICHMENT SPIKES
REACTION RATE DISTRIBUTIONS
REACTIVITY WORTH TRAVERSES
SINGLE CONTROL ROD MEASUREMENTS
EFFECT OF CONTROL INSERTION ON POWER
DISTRIBUTION AND CONTROL WORTH
CONTROL SYSTEM WORTH MEASUREMENTS

PHASE 3* - BEGINNING OF CYCLE

~25\$ CONTROL LOADED

LOAD CONTROL AND ENRICHMENT SPIKES
REACTION RATE DISTRIBUTIONS
REACTIVITY WORTH TRAVERSES
DOPPLER MEASUREMENTS
SMALL SAMPLE Na WORTH MAP
LARGE SODIUM VOID WORTH
EFFECT OF SODIUM VOID ON CONTROL WORTH AND DOPPLER
CONTROL SYSTEM WORTH MEASUREMENT

*Experiments performed in small core loading.

TABLE 2-2

GENERAL CHARACTERISTICS OF ZPPR ASSEMBLY 3

	<u>Configuration A (Phase 1A)</u>	<u>Configuration B (Phases 1B,2,3)</u>
Core Height	36 inches	36 inches
Core Diameter	~82 inches	~72 inches
Number of Simulated Control Locations	19	19
Poison Material	B ₄ C	B ₄ C
Enrichment of Poison Material	Natural to ~90% B-10	
Inserted Control Worth	0,10\$,20\$	0,12.5\$,25\$
Nominal Control Rod Worth	\$1.8 per rod	\$2.2 per rod

ZPPR HALF 1

FIGURE 2-1 - CONTROL CONFIGURATION FOR ZPPR ASSEMBLY 3

Fuel spiking is accomplished by replacement of a 1/4 inch U_3O_8 plate by a 1/8 inch Pu-Al plate and a 1/8 inch Fe_2O_3 plate in selected drawers. The spiking pattern will attempt to simulate changes in the plutonium enrichment during all three phases while keeping the core geometry constant for all three phases.

Approximately 16 and 24 evenly spaced spiked drawers are needed in core zones 1 and 2, respectively, to achieve critical in Phase 1B. Tentative fuel spiking requirements for Phases 1B, 2 and 3 are listed in Table 2-3.

A tentative fuel spiking plan is shown in Figure 2-2 for Phase 1B and Figure 2-3 for Phase 3. The spiked drawers, for each phase, are located in a manner where the plutonium content is equally distributed throughout the core. Because of the two drawer cell loading in core zone 2, consideration is given to the angular distribution of plutonium in Zone 2. A summary of drawer content and plutonium values is listed in Table 2-4 for each octant of core zone 2.

The B_4C requirements to simulate the control system in ZPPR Assembly 3 have been reevaluated and normalized to ZPPR Assembly 2 control rod experiments. The simulated control system is required to have a nominal value of \$2.20 at each of the 19 control locations. The total B_4C requirements are estimated to be 166 Kg natural B_4C .

The calculational technique utilized to evaluate the B_4C requirements is described in Section 2.3 of Reference 1. A curve of calculated central worth vs. B_4C mass is presented in Figure 2-4. The uncorrected and the normalized relative control rod worth vs. radial position is shown in Figure 2-5. The normalized curve includes all correction factors (assumed, in this analysis to be the same for ZPPR-3 and ZPPR-2).

A summary of the B_4C mass required to give \$2.20 at each control rod location is presented in Table 2-5. The mass calculated is based upon a reactor radius of 90.88 cm. Included in the table is the corrected worth of each rod assuming the rod to be placed into the present critical assembly Phase 1A. The central rod worth is ~30% lower (~1.55\$) in the larger core size.

TABLE 2-3

ZPPR ASSEMBLY 3 SPIKING REQUIREMENTS

<u>Phase</u>	<u># Spikes Core Zone 1</u>	<u># Spikes Core Zone 2</u>
1	16	24
2	38	50
3	60	76

ZPPR HALF 1

FIGURE 2-2 - PRELIMINARY SPIKING PLAN OF PHASE 1 FOR ZPPR ASSEMBLY 3

FIGURE 2-3 - PRELIMINARY SPIKING PLAN OF PHASE 3 FOR ZPPR ASSEMBLY 3

TABLE 2-4

Pu-239 VALUES FOR CORE ZONE 2

Octant*	Drawer Type	Normal Core		Phase 1		Phase 3	
		# Drawers	Pu Value	# Drawers	Pu Value	# Drawers	Pu Value
A	Spiked	0		3		10.5	
	Even	25		25		25	
	Odd	<u>26.5</u>		<u>23.5</u>		<u>16</u>	
		51.5	1.25525-3	51.5	1.31502-3	51.5	1.46443-3
B	Spiked	0		3		8.5	
	Odd	26		26		26	
	Even	<u>24.5</u>		<u>21.5</u>		<u>16</u>	
		50.5	1.28010-3	50.5	1.34105-3	50.5	1.45279-3
C	Spiked	0		3		9	
	Odd	24.5		24.5		24.5	
	Even	<u>26</u>		<u>23</u>		<u>18</u>	
		50.5	1.25500-3	50.5	1.31596-3	50.5	1.45459-3
D	Spiked	0		3		10	
	Odd	26.5		26.5		26.5	
	Even	<u>25</u>		<u>22</u>		<u>14</u>	
		51.5	1.27985-3	51.5	1.33962-3	51.5	1.46266-3

*Octants are shown in Figure 2-3.

TABLE 2-5

ZPPR ASSEMBLY 3 B₄C REQUIREMENTS

<u>Radius (cm)</u>	<u># Elements</u>	<u>Normalized and Corrected Rod Worth</u>	<u>B₄C Mass/ Element, KC</u>	<u>Total B₄C Mass, KG</u>	<u>Rod Worth in Critical Assembl r = 107.9 cm</u>
0	1	.923	4.47	4.47	1.55
33.15	2	.834	5.28	10.56	1.59
33.33	4	.833	5.29	21.16	1.57
57.52	4	.614	8.60	34.40	1.48
57.84	2	.612	8.65	17.30	1.41
66.29	2	.492	12.77	25.54	1.65
66.66	<u>4</u>	.485	13.14	<u>52.56</u>	1.34
	19			165.99	

2.2 Phase 1A Work Plan

I. Gross Power Distribution

This experiment retains the intent of the power distribution measurements of Section I.C. of the General Work Plan while providing additional data for analytical checks of the axial buckling. Additional axial distribution measurements have been added and the number of Pu-239 foils has been reduced to conserve them for the future phases.

The distribution of the reactors U-235 (n,f), U-238 (n,f), and Pu-239 (n,f) will be measured. The reaction U-238 (n, γ) will also be obtained from the U-238 foil data, though this is not a primary objective of the experiment. The foil packets and positions listed below replace those described in the General Work Plan.

A statistical error (count rate only) of + 2% for Pu-239 (n,f) and U-235 (n,f), and + 5% for U-238 (n,f) in the cor region is desired.

A. Axial Buckling Test in Vicinity of the CRP(s)

In matrix positions 134-22, 137-22, 137-33, and 137-36, U-235 foils are irradiated at 3-inch intervals between 3 and 33 inches. At 18 inches (core/axial blanket interface) foils are irradiated across the interface. In the same matrix positions, Pu-239 foils are irradiated at 3, 12, and 18 inches and U-238 foils are irradiated at 6-inch intervals from 3 to 33 inches.

B. Gross Power Distribution Scan

In all of the foil locations labeled B and C in Fig. 2 of the General Work Plan, excluding those utilized in A above, U-235 and U-238 foils will be irradiated at 3 inches.

II. Small Sample Reactivity Traverses

This experiment is to be used for analytical normalization and is a minimum version of the experiments listed in Section I.D. of the General Work Plan. The number of samples has been reduced as well as the spatial points and the axial traverse deleted.

Small sample reactivity worth measurements of Pu-239, B-10, Ta, U-238, U-235, Cf, and Na will be measured at the centers of matrix positions 137-33, 34, 35, and 36 relative to a point external to the assembly. These traverses are done in the 137 row in the slot. The shape of the worth curves will be determined, where possible, by kinetic data obtained while the sample is in motion. The desired precision of the static measurements, excluding Na, is $\pm 0.5\%$ and for Na ± 5 to 10% .

III. Single Control Rod Experiments

The single control rod experiments of the General Work Plan, Section I.E.1, will be done with the exception that the B_4C quantities projected for use in the rest of the ZPPR-3 program should be used. The estimated B_4C quantity for each control rod position is listed below:

<u>CRP #</u>	<u>Mass B_4C per CRP, kg</u>	<u>Total Mass, kg</u>
1	4.5	4.5
2,3,4,5,6,7	5.3	31.8
9,11,13,15,17,19	8.6	51.6
8,10,12,14,16,18	13.0	<u>78.0</u>
		165.9

These B_4C quantities are estimated to provide a single control rod worth of 2.2\$ in the later phases of ZPPR-3. Each group of six control rods should have identical B_4C loadings. The priority ranking is 2,3,4,5,6, and 7 having first priority and 8,10,12,14,16, and 18 having second priority in case each group cannot have identical loadings. Use of these B_4C quantities in control rod experiments of Phase 1A will provide a more direct comparison with later experiments and the total B_4C requirements are substantially reduced. All measurements are subcritical and no foil irradiations will be done in conjunction with this experiment.

IV. Control Rod Group Experiments

Control rods should be simulated with the B_4C quantities listed in Section III. These experiments represent an ordered insertion of the rods in the selected groups to measure the reactivity worth and the interaction effects of the rods. These experiments will provide a means of examining the available methods of subcritical measurements at these extreme critical levels as well as providing data from symmetric geometries for analytical comparison.

Using the rod numbers as indicated in Fig. 1 of the General Work Plan, the following sequence of measurements are made. In each case the rods contain a B_4C loading estimated to be worth -2.2\$ in the Phase 1B configuration.

Step ^a	CRP(s)		
	Added	Removed	In Assembly
1	8,14	- -	8,14
2	11,17	8,14	11,17
3	10,16	11,17	10,16
4	9,15	10,16	9,15
5	2,5	9,15	2,5
6	4,6	5	2,4,6
7	10,14,18	- -	2,4,6,10,14,18
8	8,12,16	- -	2,4,6,8,10,12,14,16,18
9	3,5,7,9,11,13, 15,17,19	- -	2,3,4,5,6,7,8,9,10,11,12, 13,14,15,16,17,18,19
10	- -	2,3,4,5,6,7	8,9,10,11,12,13,14,15,16, 17,18,19
11	- -	8,10,12,14,16,18	9,11,13,15,17,19

^aIn steps 1 through 4 all of the rods have identical loadings as per rod 8 (Section III). In steps 5 through 11 the rods are loaded as indicated in text.

Recent analysis indicates substantial flux tilting from an inserted off-center control rod. Steps 1 and 5, when compared with single rod measurements of rods 2 and 8 in Section III, should provide some indication of the influence of flux tilting on control rod worth.

2.3 Pre-Analysis of Phase 1

Pre-analysis of Phase I of ZPPR-3 has been performed using the G.E. modified version of ENDF/B-II, described later in this report. Calculations consist of one-dimensional (cylindrical geometry) diffusion theory in 29 energy groups, and two-dimensional (R-Z and X-Y) diffusion calculations in 6 energy groups. These calculations are normalized to the computed k_{eff} of ZPPR-2 (1.004). The model used in these calculations is given in Figures 2-6 and 2-7, the energy group structures are given in Table 2-6. The material concentrations were supplied by W. P. Keeney⁽²⁾ in the form of drawer loading specifications. The predicted critical size for ZPPR-3 Phase I is as follows:

	Radius (cm)	No. of Drawers	Volume (l)
Inner Core	76.0	568*	1662.16
Control Zone	95.67	900	2633.88
Outer Core	108.87	598*	1748.68
Radial Blanket	145.37	912	5336.20
Radial Reflector	153.97	253	1738.64
Inner Axial Blanket	76.0	568	1659.26
Outer Axial Blanket	108.87	598	1745.63
Spring Gap	145.37	-	94.27
Axial Reflector	145.37	-	2026.21
Core Half Height	45.80		
Axial Blanket	91.52		
Spring Gap	92.23		
Axial Reflector	107.49		

*This includes control rod locations.

2-17

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FIGURE 2-6 - ZPPR ASSEMBLY 3 R-Z MODEL

ZPPR HALF I

FIGURE 2-7

TABLE 2-6

GROUP STRUCTURES

Group	<u>29 Group Structure</u>		<u>6 Group Structure</u>	
	Delta Lethargy	Lower Energy	Delta Lethargy	Lower Energy
1	.5	6.065 MeV	2.0	1.353 MeV
2	.5	3.679	2.0	.1832
3	.5	2.231	2.0	24.79 keV
4	.5	1.353	2.0	3.355
5	.5	.8208	2.0	454.0 eV
6	.5	.4979	7.5	.2511
7	.5	..3020		
8	.5	.1832		
9	.5	.1111		
10	.5	67.38 keV		
11	.5	40.87		
12	.5	24.79		
13	.5	15.03		
14	.5	9.119		
15	.5	5.531		
16	.5	3.355		
17	.5	2.035		
18	.5	1.234		
19	.5	748.5 eV		
20	.5	454.0		
21	.5	275.4		
22	.5	167.0		
23	.5	101.3		
24	.5	61.44		
25	.5	37.27		
26	.5	22.60		
27	1.5	5.043		
28	1.5	1.125		
29	1.5	.2511		

The fissile material in the ZPPR-3 Phase I at startup will be:

Isotope	Mass (kg)		Radial Blanket	Axial Blanket	Total
	Inner core	Outer Core			
Pu-239	509.3	839.44			1348.74
Pu-241	9.37	15.39			24.76
U-235 (fuel)	3.17	5.21			8.38
U-235 (U_3O_8)	4.11	2.25	51.08	20.55	77.99
Total	525.95	862.29	51.08	20.55	1459.87

Since these calculations were made, ZPPR Assembly 3 has gone critical with 1144 drawers or 107.86 cm to the outer core edge.

Using this size, a 6 group, two-dimensional R-Z calculation gave a k of 1.0008.

Two-dimensional X-Y calculations were performed using Figure 2-7 as a model. Using a 6 group cross section set the power distribution and the worth of two control rod locations, positions 1 and 8, were calculated using the ARC system in Idaho. Figure 2-8 shows the power distribution for the Phase I condition, where all control rod locations are filled with sodium and steel. Figure 2-9 shows the depression on the power distribution when the center control rod is inserted. The worth of this rod was computed to be \$1.66, or about 8% lower than what was previously estimated, \$1.80.⁽¹⁾ Figure 2-10 shows the effect of inserting a control rod on the outer edge, the worth of this rod was computed to be \$1.94, or about 8% higher than what was previously estimated. The distortion caused by the outer control rod can be more easily seen in Figure 2-11. A factor of two difference in power from core edge to core edge exists when the outer control rod is inserted. With both the center and outer edge control rods inserted, the power tilting is still a factor of two edge-to-edge difference, as can be seen in Figure 2-12.

The differences between previously computed control worths⁽¹⁾ and current calculations indicate the sensitivity of control worth as a function of core size and rod position. Relatively small changes in core size can

FIGURE 2-8

POWER DISTRIBUTION IN ROW 137

FIGURE 2-9

FIGURE 2-10

FIGURE 2-11

POWER DISTRIBUTION IN ROW 137

FIGURE 2-12

POWER DISTRIBUTION IN ROW 137

make substantial differences in control worth. For example, previous control worth estimates assume a critical size of ~ 106 cm, whereas current computations assume ~ 109 cm. This small increase in core radius reduces the central control worth $\sim 8\%$, because the control is in a lower relative flux environment, and increases the outer control worth. The worth of the outer control increases even though the reactor average flux level is lower (for the same reactor power). The outer control location actually experiences an increase in flux (and importance) as the core radial dimension increases. This increase can be illustrated by examining the relative radial position of the control rod as $R_{\text{control}}/R_{\text{core}}$. Since R_{control} remains constant and R_{core} increases, the relative radial position of the control rod changes, tending to increase outer control worth.

2.4 Cross Section Data Processing

The cross sections used in the pre-analysis of Phase I of ZPPR Assembly 3 were based upon a modified version of ENDF/B-II⁽³⁾ data generated in early 1971. These modifications had been restricted to the important fuel isotopes and were determined by minimizing the sum of the squares of the differences between the calculated and measured eigenvalues for eleven benchmark critical and experimental assemblies (including SEFOR). The adjusted eigenvalues, determined in this manner, agreed to within 0.5% of the measured values for all eleven assemblies.

Multigroup cross sections were obtained by processing through the ENDRUN⁽⁴⁾ code. This generates infinitely-dilute cross sections plus appropriate resonance self-shielding factors from pointwise data and resonance parameters given in the ENDF/B format.⁽⁵⁾ This file level is called a GMUG library, and consists of the following data for a single ENDF/B material:

1. Infinitely-dilute cross sections for elastic scattering, inelastic scatter, elastic removal, fission, capture, $(n,2n)$, and total reactions for each group of a specified multigroup structure (maximum of 100 groups).
2. Group average values of ξ , $\bar{\mu}$ elastic, $\bar{\nu}$ (average neutrons emitted per fission), and χ (fission neutron fraction).

3. Downscattering matrices including elastic scattering, inelastic scattering, and (n,2n) reactions. (Maximum of 70 source and 49 downscattering groups).
4. Resonance self-shielding factors for up to 3 material temperatures and 5 values of σ_0 (parameter specifying total cross section per atom of resonance material). These self-shielding factors are used together with the infinitely-dilute cross sections for elastic scattering, fission, capture, and total reactions to obtain effective group constants, as well as to compute the contribution this material makes to σ_0 , when computing self-shielding for other materials (maximum of 100 groups with self-shielding).

TDOWN⁽⁶⁾ then generates a spatial and composition dependent cross section set from the GMUG file. The self-shielding of the cross sections is accomplished by computing σ_0 and using interpolation routines to determine the self-shielding factors for the σ_0 and input temperature. The elastic removal cross section is corrected for the energy distribution of the neutron flux seen by the material.

The ZPPR-2 drawer configurations are shown in Figure 2-13. In order to use the Bell rational approximation available in TDOWN, each drawer configuration had to be homogenized into two regions, as shown in Figures 2-14 through 2-16. The concentration values given in these figures is the homogenized value over the entire drawer. TDOWN calculates the new concentrations for the region widths shown.

A drawer loading to represent the sodium hole was also computed in TDOWN.

The cross sections from TDOWN were further corrected because of the fine flux distribution in the drawer caused by the heterogeneous distribution of materials. In order to represent a homogenized drawer accurately, the group cross sections of each material from each plate need to be flux weighted to provide the same reaction rates as the heterogeneous description of the drawer. A one-dimensional S_{16} transport calculation, or equivalent, is required for the desired accuracy in computing these groupwise fluxes.

FIGURE 2-13
ZPPR ASSEMBLY 2 DRAWER CONFIGURATIONS

CORE ZONE 1

Region 1 $r_1 = .279405 \text{ cm}$		Region 2 $r_2 = 2.540015 \text{ cm}$	
Atoms/cm ³ x 10 ⁻²⁴		Atoms/cm ³ x 10 ⁻²⁴	
Pu-239	8.433-4	Na	8.7964-3
Pu-240	1.117-4	Ox	1.3116-2
Pu-241	1.540-5	Fe	1.2576-2
Pu-242	1.800-6	Cr	2.7018-3
U-235	5.153-6	Ni	1.2210-3
U-238	2.384-3	Mn	2.095-4
Mo	2.306-4	Si	1.3070-4
		Cu	1.900-5
		C	2.950-5
		U-235	7.150-6
		U-238	3.166-3

Dimensions from Table I, ZPR-TM-24, June 1970.

Compositions from Table X, ZPR-TM-47, January 1971.

FIGURE 2-14

CORE ZONE 2
(Drawers A & B)

Region 1 $r_1 = .279405$ cm		Region 2 $r_2 = 2.2447$ cm	
Atoms/cm ³ x 10 ⁻²⁴		Atoms/cm ³ x 10 ⁻²⁴	
Pu-239	1.2741-3	U-235	3.720-6
Pu-240	1.687-4	U-238	1.5896-3
Pu-241	2.320-5	Na	8.5643-3
Pu-242	2.800-6	Ox	1.17613-2
U-235	7.7855-6	Fe	1.38521-2
U-238	3.6019-3	Cr	2.5229-3
Mo	3.4100-4	Ni	1.1601-3
		Mn	2.020-4
		Si	1.1300-4
		Cu	2.030-5
		C	2.330-5

FIGURE 2-15

RADIAL BLANKET

Region 1 $r_1 = 0.7950$		Region 2 $r_2 = 1.7405$	
Atoms/cm ³ x 10 ⁻²⁴		Atoms/cm ³ x 10 ⁻²⁴	
U-235	1.2750-5	U-235	1.2750-5
U-238	5.5427-3	U-238	5.5427-3
Ox	9.2370-3	Ox	1.0895-2
		Fe	6.9233-3
		Cr	1.9915-3
		Ni	8.9780-4
		Mn	1.5740-4
		C	1.0128-3
		Si	8.890-5
		Na	6.3981-3

FIGURE 2-16

The flux weighting factors used on Phase I cross sections were obtained from ANL-Idaho⁽⁷⁾. These values were confirmed with an S_{16} computation for Zone 1 of ZPPR-2.

A similar method of flux weighting is used to generate cross sections for use in two-dimensional (R,Z) calculations, where the control rod locations are homogenized in with the fuel. The computation is performed by defining a cell or "macro-cell" which consists of a control rod surrounded by fuel. Material cross sections are corrected (flux weighted) to provide the same reaction rates when the control rod is homogenized with the fuel.

The same technique will be used to process ENDF/B-III data. It is planned to use ENDF/B-III data for most analysis of experimental data.

A System of Material Identification in ZPPR-3 Cross Section Files

A system of identification to describe, in SNLIB format,⁽⁸⁾ the various materials available in analysis work has been defined.

This method was generated to establish a master cross section library for the study of ZPPR Assembly 3. All other cross section files will be generated, either through transferring or condensing cross sections, from the master file. Ideally, a master file should make provisions to contain, in a comprehensive form, all the cross sections required for analysis. This system of material identification is an attempt to satisfy this criteria.

In using the SNLIB number, which is a four-digit integer, the following procedure is used to calculate the proper integer:

1. The atomic number is used to calculate the first and second digits for the non-fissionable elements. For fissionable materials, add to 50 the isotope number and use the last two digits. For example, Uranium-28 would be 238 plus 50 or 288. Therefore, the first two digits used would be 88. The exceptions to this rule are listed in Table 2-7.

TABLE 2-7

EXCEPTIONS TO THE RULE

	<u>First Two Digits</u>
B-10	95
B-11	96

2. The third digit will denote zone of influence number. This is a number assigned to one region or a group of regions that have similar flux spectra. An example where this would be useful would be in the radial and axial reflectors. Although the material concentrations are different, the same cross sections would apply. Figure 2-17 contains the zone of influence numbers for ZPPR Assembly 3.
3. The fourth "digit" will denote the type of shielding the cross sections contain. The numbering scheme for the fourth "digit" was selected such that regions with non-fissionable materials would have numbers between 0 and 9. For regions with fissionable material the fourth "digit" will range from 0 to 19. Care must be taken in numbering the zones of influence to insure that no overlapping of numbers occurs.

Definitions of the fourth "digit" will be further defined as the analysis proceeds, but in Table 2-8 a sample listing is shown of what the definitions might look like.

Using Table 2-8 and Figure 2-17, some examples of a four-digit material number using this system is given below.

1162 denotes	Sodium cross sections in core zone 2, that have been corrected for plate heterogeneity.
1182 denotes	Sodium cross sections in core zone 1, that have been corrected for plate heterogeneity.
1187 denotes	Sodium cross sections in the control channel of core zone 1, that have been corrected for plate heterogeneity.
8500 denotes	Uranium-235 infinitely dilute.
8590 denotes	Uranium-235 in core zone 1, that have been corrected for plate heterogeneity in the Pu-U-Mo plate.
8591 denotes	Uranium-235 in core zone 1, that have been corrected for plate heterogeneity in the U_3O_8 plate.
8596 denotes	Uranium-235 in the core zone 1 that have been corrected for plate heterogeneity in the Pu-U-Mo plate - with sodium-filled control holes in zone 1.

TABLE 2-8

FOURTH DIGIT DEFINITIONS

<u>Digit</u>	<u>Denotes Cross Sections</u>
0	Infinitely dilute
1	Resonance self-shielding factors included
2	Flux weighting of type 1 between plates
3	Flux weighting of type 2 between cells
4	Resonance self-shielding - Na out
5	Flux weighting of type 4 between plates - Na out
6	Flux weighting of type 5 between cells - Na out
7	Type 2 cross sections in control channel
8	Type 2 cross sections in control channel - Na out
9	Not used
10	Heterogeneous correction of cross sections in fuel plate
11	Heterogeneous correction of cross sections outside the fuel plate
12	Cross sections at 700°K
13	Cross sections at 700°K - Na out
14	Cross sections at 1400°K
15	Cross sections at 1400°K - Na out
16	Heterogeneous correction of cross sections in fuel plate (w/Na holes)
17	Heterogeneous correction of cross sections outside fuel plate (w/Na holes)
18	Not used
19	Not used

FIGURE 2-17 ZPPR ASSEMBLY 3 R-Z MODEL

○ = Zone of Influence Number

3. TASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

Task Leader: B. A. Hutchins

3.1 Nuclear Data

A 59 group generalized cross section file (GMUG)⁽⁶⁾ containing the necessary materials for benchmark calculations and ZPPR-3 analysis has been prepared from the ENDF/B-III libraries. Table 3-1 lists the isotopes on the GMUG file. Three isotopes; Pu-240, U-238 and Pu-239 have temperature dependent shielding factors. Pu-240 and U-238 each have data on the GMUG file in two separate materials because the file has limitations of three temperatures per material.

The ENDRUN⁽⁴⁾ code was used to process all materials except the Be(n,2n) scattering matrix and the iron data. Be(n,2n) on the GMUG file is based on Perkins analysis.⁽⁹⁾ The iron data was not processed by ENDRUN because of the multi-level resolved resonances of ENDF/B-III iron. Generalized twenty-nine group ENDF/B-III iron cross section data were obtained from R. Kidman of WADCO and these data were expanded to 59 groups and placed on the GMUG file.

Preliminary benchmark calculations were carried out using ENDF/B-II iron. The eigenvalues for benchmarks ZPR6-7⁽¹⁰⁾ and ZPPR-2⁽⁷⁾ were .985 and .993, respectively, using the version two iron. The two benchmarks were recalculated with version three iron data and the results are given in Tables 3-2 and 3-3. The eigenvalues, activation ratios and reactivity worths are compared with those values obtained by using ENDF/B-II modified by Pitterle.⁽¹¹⁾ The preliminary values in Tables 3-2 and 3-3 show that there is close agreement between ENDF/B-II modified and ENDF/B-III results.

3.2 Methods Development

Two linking approaches for obtaining three-dimensional fluxes from two-dimensional flux sets using the TRIAL code have been specified. For an

TABLE 3-1

LISTING OF MATERIALS IN THE GMUG3 FILE

Isotope	ENDF/B-III No.	Temperatures*(°K)	σ_0 Range**
Manganese 55	1019	300	ID
Copper	1087	300	ID
Pu-240	1105	300 900 2100 1500 3000 3900	$10-10^4$
Pu-241	1106	300	ID
Molybdenum	1111	300	ID
Chromium	1121	300	$1-10^3$
Nickel	1123	300	$1-10^3$
Oxygen-16	1134	300	$10-10^4$
Aluminum-27	1135	300	ID
Carbon-12	1140	300	ID
Silicon	1151	300	ID
Beryllium-9	1154	300	ID
Boron-10	1155	300	ID
Sodium-23	1156	300	$1-10^3$
U-235	1157	300	$10-10^4$
U-238	1158	300 900 2100 1500 3000 3900	$1-10^3$
Pu-239	1159	300 900	$10-10^4$
B-11	1160	300	ID
Pu-242	1161	300	ID
Iron	1180	300	$1-10^3$

*Temperatures are specified for isotopes even though they are not temperature dependent.

**ID represents infinitely dilute only - no shielding factors are included on the file for this material.

TABLE 3-2

RESULTS FOR BENCHMARK ZPR6-7

	59 group ENDF/B-III	29 group ENDF/B-II Modified	Measured
k_{eff}	.9902	.9903	1.00
<u>Activation Ratios</u> @ core center			
$\sigma_f(\text{U-235})/\sigma_f(\text{Pu-239})$	1.105	1.092	1.0474
$\sigma_f(\text{Pu-240})/\sigma_f(\text{Pu-239})$	.1919	.1936	.1850
$\sigma_c(\text{U-238})/\sigma_f(\text{Pu-239})$	.158	.156	-
$\sigma_f(\text{Pu-241})/\sigma_f(\text{Pu-239})$	.1318	.1372	-
<u>Reactivity Worths</u> ($10^{-5} \Delta k/k/\text{mole}$) @ core center			
U-235	35.4	34.6	-
U-238 (depleted)	-2.49	-2.48	-2.11
Ru-239	42.95	42.80	42.39
Pu-240	4.383	4.275	-
Na	-.189	-.165	-.159
Cr	-.341	-.327	-
Fe	-.265	-.303	-.264
Ni	-.469	-.499	-.416

TABLE 3-3

RESULTS FOR BENCHMARK ZPPR-2

	59 group ENDF/B-III	29 group ENDF/B-II Modified	Measured
k_{eff}	1.000	.9992	1.00
<u>Activation Ratios</u> @ core center			
$\sigma_f(\text{U-238})/\sigma_f(\text{U-235})$	.02124	.02165	.0201
$\sigma_f(\text{Pu-239})/\sigma_f(\text{U-235})$	.9073	.9179	.9372
$\sigma_f(\text{Pu-240})/\sigma_f(\text{U-235})$	.1811	.1784	.1704
$\sigma_c(\text{U-238})/\sigma_f(\text{U-235})$	.1426	.1429	-
<u>Reactivity Worths*</u> ($10^{-5} \Delta k/k/\text{mole}$) @ core center			
U-235	18.46	18.22	
Pu-239	22.36	22.47	
Pu-241	32.31	32.41	
Na	.0062	.0151	
Cr	-.0694	-.0608	
Fe	-.0426	-.0702	
Ni	-.0572	-.0879	
Al	-.0270	-.0299	
C	-.033	-.030	
Mo	-.8526	-1.228	

*Measured values of reactivity worths are not given because the normalization of the calculated values required for comparison is not known. A core height of 91.6 cm was used in determining worths.

(x,y,z) model, of a cylindrical reactor, they are the following:

Approach 1: To obtain three-dimensional fluxes by running only two-dimensional problems.

1. Compute (x,y) fluxes at chosen z levels.
2. Compute one (r,z) solution, using the nearest cylindrical approximation possible to the (x,y,z) model.
3. With TRIAL, link the (x,y) flux sets using the (r,z) flux set to get 3-dimensional fluxes.

Approach 2: To obtain changes in three-dimensional fluxes which had been computed in three dimensions due to a major perturbation in the (x,y,z) model (such as movement of control rods).

1. Perform one (x,y,z) calculation for the median configuration.
2. Obtain (x,y) flux sets from the (x,y,z) results at chosen z levels.
3. Run one (r,z) solution using the nearest cylindrical approximation to the median (x,y,z) configuration.
4. Test the linking approximation by running TRIAL to link the (x,y) flux sets from step 2 using the (r,z) set from step 3. Try various (x,y) sets and axial linking functions to give best results.
5. Perform one (r,z) solution using the nearest cylindrical approximation to the perturbed (x,y,z) configuration.
6. With TRIAL, link the (x,y) flux sets from step 2 using the (r,z) set from step 5 to obtain three-dimensional fluxes in the perturbed configuration. Apply results from step 4 in choosing the (x,y) sets and the linking functions.
7. Repeat steps 5 and 6 for each perturbation about the median configuration.

Approach 1 has the advantage of not requiring a pre-calculation in three dimensions. However, the results will be dependent upon the choice of axial bucklings and the treatment of non-multiplying solutions in the (x,y) problems. Approach 2 has potential for rapidly computing fluxes as

control rod positions are modified, but will require an initial three-dimensional solution. In both approaches the results will have some sensitivity to the number and axial positions of the (x,y) solutions and linking functions used. This evaluation program includes determination of this sensitivity as well as sensitivity to axial bucklings used in the (x,y) solutions of the first approach.

Using the benchmark assembly ZPR6-7⁽¹⁰⁾ as a guide, a simplified reactor was specified to test the TRIAL approaches. A description of this reactor is given in Figure 3-1.* Two three-dimensional solutions of this reactor will be carried out; the first having the boron control rods at point A, shown on the R-Z plane. In the second problem the rods will be withdrawn to position B. Results from these two problems will allow evaluation of both approaches, with the second problem being treated as a perturbation of the first. These two problems are being prepared to be calculated by the CITATION⁽¹²⁾ code at ORNL. CITATION will use 4-group diffusion theory for the analysis.

The composition descriptions and geometric dimensions for the TRIAL problem are given in Table 3-4.

BRD recommendations for the computational capability of a new nuclear data processing code were presented at a meeting in Washington, D. C. on February 24 and 25. The primary BRD need for such a code is to unify the processing of data over a broad range of LMFBR design computations, such that consistent (datawise) computations can be performed with a number of different calculational techniques and approximations (such as diffusion theory vs. (S_n, P_0) transport theory). Thus, the first version of this code, at least, should unify existing processing techniques and not rely upon the development of improved algorithms.

For the data processed by this new code to be of any use in BRD design computations, it must include information which will allow rapid adjustment of cross sections for different degrees of resonance self-shielding. This, then, requires that a two-step processing technique,

*The x-y planes do not have curved boundaries as shown in the figure. Boundaries are blocked at x-y intervals closest to the curve.

FIGURE 3-1
DESCRIPTION OF TRIAL TEST PROBLEM

Dimensions are in centimeters

TABLE 3-4

PRESCRIPTION FOR ANALYSIS OF TRIAL TEST PROBLEM

Regional Compositions

Material	Core	Control Rod	Sodium Hole	Blanket
Pu-239	.00093*	-	-	-
U-238	.0058	-	-	.040
Iron	.015	.004	.013	.005
Sodium	.009	-	.017	-
Oxygen	.014	.015	-	-
Boron-10	-	.013	-	-

*Estimate based on 1-D diffusion solution

Geometry and Mesh

Core radius (approx. mean) 75 cm - 16 mesh intervals

1/2 core height 75 cm - 10 mesh intervals

radial blanket thickness (approx. mean) 40 cm - 4 mesh intervals

axial blanket thickness 30 cm - 4 mesh intervals.

such as used in the ETOX-1DX,⁽¹³⁾ ENDRUN-TDOWN^(4,6) or SUPERTOG-XSDRN⁽¹⁴⁾ code systems be specified. The Step-1 code generates, from ENDF/B, infinitely-dilute group cross sections, transfer matrices and information needed to self-shield the resonances. The step-2 code uses the data from the first step to self-shield the cross sections and matrices for specific reactor models. This adjustment in step=2 must be very rapid since it may be made many times in a single problem.

Consideration of different techniques for applying self-shielding adjustments in step-2 indicated that the direct use of resonance parameters, such as in the codes SUPERTOG-XSDRN⁽¹⁴⁾ and SDX,⁽¹⁵⁾ is significantly slower than the Bondarenko approach of using pre-computed self-shielding factors. Since the Bondarenko approach has been shown to give excellent agreement both with more precise methods and measurements for important LMFBR design parameters⁽¹⁶⁾, the incentive, at this time, for using anything other than this approach is small. Therefore, BRD recommended that a two-step approach, using the Bondarenko technique, be specified for the new processing code.

Detailed BRD recommendations for both the step-1 and step-2 codes are shown in Table 3.5.

General requirements which are also important to BRD are the following:

1. Conformance to the programming standards⁽¹⁷⁾ and exclusive use of the standard interfaces⁽¹⁸⁾, specified by the Committee on Computer Code Coordination. Additional standard interfaces must be specified.
2. Establishment of procedures which assure that (a) the step-1 code will remain current with the ENDF/B Procedures Manual, and (b) modifications to either code can be rapidly made all installations using them.

TABLE 3-5

RECOMMENDATIONS FOR NEW PROCESSING CODES

Basic Approach: 2-Step, Bondarenko-Type, with Expanded Capabilities

Step 1: Basically ETOX or ENDRUN Approach.

Requirements include:

- P_0 Elastic Scattering Matrices
- Fission Neutron Spectrum by Isotope
- Rapid Unresolved Resonance Averaging
- Saving of Pointwise Cross Sections from Resolved Resonances
- σ_0 Adjustment for Overlapping Ranges
- Rhomberg Integration of Fine Data Including Self-Shielding Factors
- Optional Averaging of Gamma Production, Gamma and Heating Rate Data
- Partial Computation and Results Possible
- Self-Shielding of Inelastic Scattering

Step 2: Basically LDX and TDOWN Adjustments Applied.

Requirements Include:

- P_0 Elastic Scattering Matrices
- Fission Neutron Spectrum Generation from Isotopes
- Broad User Options for Adjustments Applied
- Multi-region Cell Heterogeneity Corrections
 - Cylindrical or Slab
 - Resonance Heterogeneity
 - Multi-group Flux Variations
 - Application of Cell Factors to Group Constants
- (S_n, P_ℓ) 1-D flux Solution Capability
- Group Collapsing
- Optional Processing of Gamma Production, Gamma and Heating Rate Data
- Self-Shielding of Inelastic Scattering
- Group-Collapsing of Self-Shielding Factors and Cross Sections

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APPENDIX A - LIST OF CONTRIBUTORS

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