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AEC RESEARCH AND DEVELOPMENT REPORT

JANUARY 1972

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CRITICAL EXPERIMENTS AND COOPERATIVE EVALUATIONS PROGRAM

FIRST QUARTERLY REPORT
OCTOBER-DECEMBER 1971

PREPARED FOR
THE UNITED STATES ATOMIC ENERGY COMMISSION
UNDER CONTRACT NO. AT(04-3)-189 PROJECT AGREEMENT NO. 59

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CRITICAL EXPERIMENTS AND COOPERATIVE EVALUATIONS PROGRAM
FIRST QUARTERLY REPORT
OCTOBER-DECEMBER 1971

Approved:

P. Greebler
P. Greebler, Manager
Nuclear Engineering

Approved:

J. H. Field
J. H. Field
Program Manager

Prepared for the
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SUNNYVALE, CALIFORNIA 94086

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1 - S U M M A R Y

TASK A - PLANNING AND ANALYSIS OF ZPPR ASSEMBLY 3

The ZPPR Assembly 3 configurations will simulate a control system of 19 control rods, each control rod with a nominal value of -1.8 . The assembly core will be a 2 enrichment zone core with a height of 36 inches. The critical configuration is estimated to contain 568 drawers (radius 76.0 cm) in zone 1 of which 48 simulate control and 552 drawers in zone 2 (radius 106.8 cm) of which 28 drawers simulate control. The experimental program is divided into 3 phases, where phases 1, 2, and 3 represent the end, middle, and beginning of an operating interval, respectively. Phases 1, 2, and 3 are designed to contain about 0, \$10, and \$20 of inserted control, respectively. Reactivity compensation of this inserted control will be accomplished by fuel spiking, i.e., replacing selected U_3O_8 plates with Pu-Al and Fe_2O_3 plates.

The simulated control system will use B_4C as poison material. Each control position will occupy 4 matrix positions in a 2x2 array. The total B_4C required for ZPPR-3 is estimated to be 220 Kg. Analysis of control rod experiments in ZPPR-2 was used as a basis for normalization of the B_4C material requirements.

The experimental program has been specified.

TASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

Testing for ENDF/B, Version III was completed and the results sent to Brookhaven National Laboratory. A multi-group data set, based on ENDF/B-III, was obtained from HEDL.

Conversion of the LDX computer code to the GE-635 computer has been completed and checkout of the ETOX code on the GE-635 is continuing.

ENDF/B-III data is being processed with the ENDRUN code into a 59-group file. This file will be used for ZPPR-III analyses.

Testing with fictitious tedium isotopes shows excellent agreement between the ENDRUN, ETOX and ETOG processing codes.

A Doppler coefficient benchmark problem using SEFOR results has been specified.

Programming of a preliminary, test version of the TRIAL code is completed and checked out. Testing of the technique is being continued and specifications for an output processor are being written.

A purchase order for computer time on the Idaho IBM-360-75 has been signed with Aerojet Nuclear Corp.

2. TASK A - PLANNING AND ANALYSIS OF ZPPR ASSEMBLY 3

Task Leader: H. S. Bailey*

2.1 ZPPR Assembly 3 Configuration

Preliminary planning of ZPPR Assembly 3 (carried out under Task N of PA-10 during the first quarter of FY72) defined an experimental configuration similar to ZPPR Assembly 2 while containing most of the major features of the General Electric Demonstration Reactor. ZPPR-3 was defined as a 2 enrichment zone assembly, with each zone containing ~50% of the core volume. The drawer fuel loading remain the same as ZPPR-2. The major difference between ZPPR-2 and ZPPR-3 was the simulation of 37 control locations in ZPPR-3 and a larger core radius. Each control location would remove fuel from 2 drawers and contain a moderator, BeO. B_4C would be used as control material.

During this quarter, several changes in the General Electric Demonstration Reactor design have occurred. Some of these design changes are incorporated in the ZPPR-3 configuration. In particular, the simulated control system is changed. Assembly 3 will have 19 simulated control locations, with each control position occupying 4 drawer locations in a 2x2 array. The control locations will contain no BeO. The control locations will contain either sodium cans (control out) or a mixture of B_4C plates and sodium cans (control in). The quantity of B_4C in each location is specified to have a nominal value of \$1.8 when compared with the control out condition. The general characteristics of ZPPR Assembly 3 are listed in Table 2-1.

The Assembly 3 layout with simulated control locations is shown in Figure 2-1. The calculated critical size for ZPPR Assembly 3 is the following: Core zone 1 contains 568 drawers (radius 76.0 cm) of which 48 are control drawers, Core Zone 2 contains 552 drawers (radius 106.73 cm) of which 28 are control drawers.

The Assembly 3 experimental program is separated into 3 phases; Phase 1, 2, and 3 represent the control conditions at the end, middle and beginning of a fuel cycle, respectively. A brief description of the 3 phases of ZPPR-3 is given below:

*Contributors are listed in Appendix A.

TABLE 2-1

GENERAL CHARACTERISTICS OF ZPPR ASSEMBLY 3

Core Height	36 inches
Core Diameter	~82 inches
Number of simulated control locations	19
Poison material	B ₄ C
Enrichment of poison material	Natural to ~90% B-10
Inserted Control Worth	0, 10\$, 20\$
Nominal control rod worth	\$1.8 per rod

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ZPPR HALF 1

FIGURE 2-1 - CONTROL CONFIGURATION FOR ZPPR ASSEMBLY 3

- Phase 1: Simulate conditions at the end of an operating cycle - all control rods are withdrawn.
- Phase 2: Simulate the conditions near the middle of an operating cycle - approximately one half of the burnup control is inserted, i.e., ~\$10, with several insertion patterns.
- Phase 3: Simulate conditions at the beginning of an operating cycle - all burnup control is inserted, i.e., ~\$20. The insertion pattern will simulate a realistic beginning-of-cycle control pattern.

The critical size of all three phases will be determined by the critical loading of phase 1. When control is inserted for phase 2 and phase 3, ~10\$ and 20\$, respectively, criticality is maintained by "fuel spiking". Fuel spiking is accomplished by replacing selected 1/4 inch U_3O_8 plates with 1/8 inch Pu-Al and 1/8 inch F_2O_3 plates, thus changing the plutonium enrichment. This keeps the core geometry constant during all three phases, with the plutonium enrichment changing as in an operating power reactor. A tentative fuel spiking plan is shown in Figure 2-2 for phase 3.

2.2 B_4C Requirements

The B_4C requirements to simulate control system in ZPPR Assembly 3 have been evaluated and normalized to ZPPR Assembly 2 control rod experiments. The simulated control system is required to have a nominal value of 1.8\$ at each of the 19 control locations. The B_4C requirements are summarized below:

Phase 2 normal configuration	6 rods @ 1.8\$/rod = 68.5 Kg
Phase 3 normal configuration	12 rods @ 1.8\$/rod = 131.5 Kg
Total system	19 rods @ 1.8\$/rod = 220 Kg

ZPPR HALF I

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

2.3 Analytical Method

A diffusion-perturbation theory technique was used to evaluate the B_4C requirements. All analyses were performed with one-dimensional diffusion theory (cylindrical geometry) in 29 energy groups. Region and energy dependent axial buckling values were based on leakage data from a 6 group 2-dimensional (R-Z) diffusion theory calculation.

The technique consists of evaluating the central worth of control rods containing various amounts of B_4C . This provides a curve of calculated central worth vs. B_4C mass (Figure 2-3). The relative worth as a function of radius is evaluated by perturbation calculations of the "control in" and "control out" composition of the control location. The uncorrected relative control worth of Assembly 3 as a function of radius is shown in Figure 2-4.

Analysis of control rod experiments in ZPPR Assembly 2 by the same technique allows normalization of ZPPR-3 B_4C requirements to ZPPR-2 control rod experiments. Normalization is possible at 2 positions, $r=0$ and $r=86.2$ cm which corresponds to ZPPR-2 control rod positions A and C, respectively. Normalization is accomplished by dividing the uncorrected relative worth, by the ZPPR-2 C/E (calculated to experimental) value at these 2 radii. The normalized curve of relative worth vs. radius is also shown in Figure 2-4. The normalized curve also includes all correction factors. For this analysis, it is assumed that correction factors, e.g., transport, heterogeneity, will be the same for both ZPPR-3 and ZPPR-2 control rods.

The worth of a control rod at position r is determined by multiplying the central calculated value by the relative "corrected" value at position r from Figure 2-4. Conversely, to define the B_4C mass that provides a given control worth at position r , the desired worth (1.8%) is divided by the relative "corrected" value at position r . This provides the computed worth of that control rod, if it were located at the center. The appropriate B_4C mass is then obtained from Figure 2-3.

The B_4C mass determined by this technique, to give 1.8% at each control location, is presented in Table 2-2. A summary of ZPPR-2 control analysis is presented in Table 2-3, showing the data used for normalization.

FIGURE 2-4 - RELATIVE ZPPR-3 CONTROL ROD WORTH VS. RADIAL POSITION

2-8

Relative Control Rod Worth

1.0

.8

.6

.4

.2

0

Radius (cm)

0

10

20

30

40

50

60

70

80

90

100

Computed ZPPR-3

ZPPR-3 normalized
to ZPPR-2 experiments

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TABLE 2-2

ZPPR ASSEMBLY 3 B₄C REQUIREMENTS

<u>Radius (cm)</u>	<u># Elements</u>	<u>Normalized & Corrected Rod Worth</u>	<u>B₄C Mass/ Element, Kg</u>	<u>Total B₄C Mass, Kg</u>
0	1	.923	5.13	5.13
38.7	2	.833	6.04	12.08
41.1	4	.821	6.19	24.76
73.0	4	.562	12.44	49.76
75.2	2	.532	13.70	27.40
77.3	2	.504	15.40	30.80
84.5	<u>4</u>	.397	17.36**	<u>69.44</u>
	19			219.37

**17.36 Kg - \$1.5/element @ 84.5 cm

TABLE 2-3

SUMMARY OF CONTROL MOCKUP EXPERIMENTS IN ZPPR-2

	<u>Experiment, \$</u>	<u>Calculated, \$</u>	<u>C/E</u>
Worth 4 drawer sodium hole relative to core (A)	-0.82	-0.84 ⁽¹⁾	1.024
Worth central control (A) relative to sodium	-2.51	-2.72 ⁽¹⁾	1.083 ⁽²⁾
Worth central control (A) relative to core	-3.33	-3.56 ⁽¹⁾	1.069
Worth Rod (B) relative to core	-2.92	-3.19 ⁽³⁾	1.092
Worth Rod (C) relative to core	-1.478	-1.580 ⁽³⁾	1.069
Worth Rod (C) relative to Na	- .896	-1.032 ⁽³⁾	1.152 ⁽²⁾
Worth Na hole (C) relative to core	-0.58	-0.55 ⁽³⁾	0.948

- 1) 1-D diffusion in 29 energy groups
- 2) values used for normalization
- 3) computed by perturbation technique

2.4 Experimental Program

An outline of the ZPPR-3 experimental program is listed below with the ANL estimate of required time. This program was developed as a cooperative effort with the ZPPR staff.

ZPPR Assembly 3 Work Plan Outline

I. Phase I, End of Cycle Start ~1/18/72

A. Fuel Inspection and Approach to Critical

A yearly inspection of all Pu-U-Mo fuel plates is a ZPPR operational requirement. This inspection will be done during the conversion of Assembly 2 to the Assembly 3 configuration. Figure 2-5 shows the estimated critical loading with the 19 control rod locations, numbered 1 through 19 for reference purposes. In the approach to critical the Zone 1- Zone 2 boundary will be fixed as indicated in Figure 2-5. Criticality will be achieved by adjusting the size of Zone 2.

B. Initial Adjustments and Measurements

The initial critical configuration will be adjusted, if necessary, and ZPPR operational measurements performed.

C. Power Distribution Measurements

A detailed power distribution will be obtained to compare with comparable measurements in Phase 2 and 3. Power distribution is obtained from the fission reactions of U-235, U-238, and Pu-239 foils. The foil locations are indicated in Figure 2-6.

D. Small Sample Reactivity Traverses

A radial and an axial small sample reactivity traverse will be measured for each of the following materials: Pu-239, Pu-240, U-235, B-10, Ta, Na, BeO, Pu-241, Fe, Ni, and Cr.

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ZPPR HALF I

FIGURE 2-5 - ASSEMBLY 3 WITH NUMBERED CONTROL ROD POSITIONS

FIGURE 2-6 - FOIL LOCATIONS FOR POWER DISTRIBUTION MEASUREMENTS

A - Pu-239, U-238, U-235 foils @ 3, 9, 15, 21, and 24"
and U-235 @ 6, 12, 18, 27 and 31"

B - Pu-239, U-238, U-235 foils @ 3, 9, 15" **ZPPR HALF I**

C - U-235 foils @ 3"

E. Single Control Rod Experiments

These experiments will provide data to verify the design of control rods for Phases 2 and 3 and provide early data analysis of control rod effects.

1. Control Rod Worths at 3 Radii

The B_4C required to provide a control rod value of $-\$1.8 \pm .05$ at control rod position (CRP) 8 will be measured. The worth of estimated $\$1.8$ control rods at CRP's 1 and 2 will be measured. The worth of the simulated control rods 2 and 8 will also be measured at CRP 1.

2. Plate/Pin Comparison

This experiment will examine the differences in worth and power distribution of a natural B_4C control rod mockup in plate and pin composition at CRP 1.

3. Perturbation of Power Distribution due to Control Rods

This experiment will measure the change in power distribution due to the experiments of E.1. Foil measurements are made for each rod insertion at the positions indicated in Figure 2-7.

4. Sodium Worth in CRP 1

F. Rod Interaction Experiment

The interaction effects of control rods located in CRP's 2, 8, and 9 will be investigated.

G. Control Rod Group Experiments

This is the last experiment in Phase 1. The worth of the safety system, Phase 3 control configuration, and Phase 2 control configurations are measured subcritically. The final measurement will have control rods 2, 4, 6, 10, 14, and 18 inserted. This is the control configuration for the Phase 2 approach to critical.

II. Phase II, Middle-of-Cycle Condition

A. Approach to Critical

Control rods 2, 4, 6, 10, 14, and 18 are fully inserted. Criticality is achieved by adding the Pu/Al fuel spikes at preselected positions.

ZPPR HALF I

FIGURE 2-7 - FOIL LOCATIONS FOR EXPERIMENT E.3.

B. Power Distribution

Measurements are similar to those of Phase I.C.

C. Small Sample Reactivity Measurements

Same as Phase I.D.

D. Single Control Rod Experiments

The worth of control rods in position 1, 3, and 8 are measured individually.

E. Flux Peaking

This examines extremes in control rod configuration. Control rods will be loaded into positions 2, 3, 4, 5, 6, and 7 and then into positions 9, 11, 13, 15, 17, and 19. The central rod worth (CRP 1) will be measured and flux distribution measured for each configuration.

F. Partial Rod Insertion Effects

Two configurations with partially inserted control rods will be examined in this experiment. Three-dimensional fission distributions will be determined for both configurations.

G. Control Rod Group Experiments

This is the final experiment of Phase II. The subcritical k_{eff} of the system will be measured with the simulated safety system inserted, all shim rods inserted, and the Phase III control configuration inserted.

III. PHASE III, Beginning-of-Cycle Condition

A. Approach to Critical

With 12 control rods inserted, additional Pu/A₂ spikes will be added to selected positions to achieve criticality.

B. Power Distributions

Same as Phase I.C.

C. Small Sample Reactivity Traverses

Same as Phase I.D.

D. U-238 Doppler Effect

The U-238 Doppler effect will be measured near the core midplane at three different radii and distances from control positions.

E. Small Sample Na Worth Distributions

These measurements are in addition to III.C and will provide a small sample sodium worth map for the core.

F. Large Na Void Experiments

Axial and radial voiding patterns will be investigated. Axial voiding will be measured for a 5x5 drawer area at the core center in 6 inch steps from the midplane i.e., $\pm 6"$, $\pm 12"$, $\pm 18"$. Radial voiding will be accomplished in steps over a height of $\pm 12"$ from the core midplane until the core Zone 1 boundary is reached.

G. Sodium Void Neutronics

These experiments are to determine the effect of sodium voiding on previously measured data. Small sample Na worth measurements, worth of control rods, U-238 Doppler measurements, and small sample reactivity measurements already measured in Phase III will be repeated in the sodium void conditions.

This completes the ZPPR Assembly 3 experimental program. The total program is estimated by ANL to take ~50 weeks.

3. TASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

Task Leader: B. A. Hutchins*

3.1 Nuclear Data

Data tapes containing ENDF/B-III data were received from Brookhaven National Laboratory, and Phase 1 data testing was performed on Fe and Pu-239. Both materials were found to conform to the ENDF/B procedures limits,⁽¹⁾ although it would be beneficial for the Fe (n, γ) data to be given in more detail while the secondary energy distributions for (n,2n) and the inelastic continuum have unnecessary detail. A review of the total cross section value at the minimum of the 24 keV "window" was also recommended. No changes were recommended for the Pu-239 data. Results of this phase 1 testing were sent to Brookhaven National Laboratory for evaluation of the status of these materials at the November meeting of CSEWG.

The ETOX-LDX^(2,3) codes were obtained from HEDL and converted to the GE-635 computer. The LDX code has executed the checkout problem successfully and has been used to compute a ENDF/B benchmark problem. (See below.)

Considerable difficulty was encountered in converting the ETOX code due to the use of non-standard FORTRAN, principally in the indexing. Initially, a new G.E. compiler was tried because of its extended FORTRAN capability. However, this compiler is not yet ready for general use and this attempt had to be abandoned. The ENDF/B mode required by ETOX posed another problem in getting the code checked out. The latest versions of the RIGEL code⁽⁴⁾ was obtained from Brookhaven National Laboratory and has been used to convert ENDF/B-III to the mode used by ETOX. Checkout of ETOX has begun.

In addition to converting ETOX-LDX to the GE-635 computer, an option was added to ETOX which puts the output data file into the GMUG format.⁽⁵⁾ This will allow running of the TDOWN⁽⁵⁾ code with an ETOX output file. This routine is being checked out.

A set of multi-group cross sections generated from ENDF/B-III and processed with the ETOX code were obtained from R. Kiddman, of HEDL. This

*Contributors are listed in Appendix A.

data is not in the standard 0.5 lethargy-width group structure and is only for one temperature, but it was used with the LDX code to compute the fast reactor benchmark assembly ZPR6-7. The value of k_{eff} obtained from LDX is larger by 0.001 than that computed by C. Cowan, at Brookhaven National Laboratory, also using the LDX code and a data set obtained from HEDL. The reason for this difference has not been determined.

Since the ETOX code is not yet operational on the G.E. computer, processing of the ENDF/B-III data with the ENDRUN code⁽⁶⁾ has begun. An additional consideration in the decision to process with the ENDRUN code was the possibility that the broad-group structure required with ETOX (since it does not generate elastic transfer matrices) may not be accurate enough for computations in ZPPR-III. This concern comes about due to the large zones which represent control rod channels and are principally composed of sodium.

A group structure, composed of 59-groups, was specified, with special consideration given to the large resonances of sodium, iron and oxygen. This group structure is given in Table 3-1. The group widths are chosen so that collapsing to the 26-group set, with constant $\Delta u = 0.5$, can be accomplished, with one or more additional groups below group 26. This permits a direct comparison of benchmark results and group-wise cross sections with those computed by other CSEWG organizations. Processing of a number of ENDF/B-III materials into this 59-group set has been completed.

When ETOX becomes operational, it will be used to generate a 26-group data set for the analysis of ZPPR-III. Both the 26- and 59-group sets will be used in the analysis of ZPPR-III and for benchmark problems. Comparison of the results will indicate how important finer structure, and therefore elastic transfer matrices, are for fast reactor analysis.

Special data sets, compiled in the ENDF/B format and labeled tedium isotopes, were prepared by the Codes and Formats Subcommittee of CSEWG for the comparison of the capabilities of the various ENDF/B data processing codes. These twelve isotopes provide data for individual testing of smooth cross section averaging, resolved and unresolved resonance calculations, inelastic cross section averaging and the processing of the File 4 angular distribution data. Processing of the first eight and the twelfth tedium

TABLE 3-1

GROUP STRUCTURE OF 59-GROUP SET

<u>Groups</u>	<u>Δu</u>	<u>Energy Range</u>
1	-.5	10.0 MeV - 16.5 MeV
2-7	.5	.498 MeV - 10.0 MeV
8-25	.25	5.5 keV - .498 MeV
26-40	.1	1.23 keV - 5.5 keV
41-46	.25	275. eV - 1.23 keV
47-57	.5	1.125 eV - 275. eV
58	.6	.617 eV - 1.125 eV
59	-	Thermal

isotopes was carried out with the ENDRUN code. Results were compared with those from the ETOX and ETOG codes run at BNL and with other processing codes by the Codes and Formats Subcommittee. Results of the GE and BNL computations are included in Table 3-2. The values reported are for a 68-group structure of constant 0.25 lethargy width.

The following trends can be seen from Table 3-2:

- a) There is excellent agreement for smooth data.
- b) For resolved resonances, the cross sections from ETOX and ENDRUN agree to within 1%. ETOX shows some differences (approaching 10%) at the resonance peaks (Tedium 3) and in the energy distribution about the peak (Tedium 4).
- c) All three codes agree to within 0.5% for cross sections from unresolved resonances, except where the unresolved range ends within the group, a 10-20% variation occurs (not included in Table 3-2).
- d) ENDRUN and ETOX compute the same resonance self-shielding factors for resolved resonances (Tedium 3 and Tedium 6).
- e) The 10% difference in unresolved resonance self-shielding factors (Tedium 7) for ENDRUN and ETOX is due to differences in the approach to integration. ENDRUN uses a numerical integration while ETOX uses the J function.
- f) The inelastic results are in excellent agreement.

Additional tedium isotopes need to be defined in order to continue these comparisons. Of interest are fission cross section contributions, particularly from unresolved resonances, and overlapping ranges of unresolved resonances.

A Doppler coefficient benchmark problem using SEFOR results has been specified. A preliminary version was reviewed by members of the Data Testing Subcommittee of CSEWG, and the final draft is ready for distribution. The benchmark problem is based upon the steady state, isothermal value of 0.0081 for $T \frac{dk}{dT}$ measured in SEFOR, Core 1.⁽⁷⁾ This value has been substantiated by transient measurements which reduce the standard deviation from ± 0.0014 to ± 0.0010 . The statement of the benchmark problem includes a two-dimensional model, an axial one-dimensional model, and a spherical one-dimensional model, all for homogeneous computations. Corrections for

TABLE 3-2

TEDIUM ISOTOPE RESULTS FROM ENDRUN, ETOX AND ETOG

All cross sections are for infinite dilution. Resonance self-shielding factors are indicated by an asterisk(*) .

TEDIUM ISOTOPE DESCRIPTION	GROUPS	REACTION	ENDRUN	ETOX	ETOG
1. Smooth data $\sigma_s = 1.0$ for all energies	all	σ_s	1.00	1.00	1.00
2. Smooth data $\sigma_c = 1/v$ (1.0@.0253 eV)	35	σ_c	.00376	.00376	.00376
	50	σ_c	.02449	.02449	.02449
	65	σ_c	.15971	.15971	.15971
3. Resolved data one s-wave resonance in group 53	53	σ_c	38.951	39.016	42.235
	53	σ_s	13.967	13.989	15.109
	53	$\sigma_c f^*$ @ $\sigma_o = 100$	.2039	.2037	
		$\sigma_s f^*$ @ $\sigma_o = 100$	.2598	.2595	
4. Resolved data one s-wave resonance at boundary groups 53-54	53	σ_c	24.283	24.3701	20.2116
	54	σ_c	26.0713	26.0708	33.9862
	53	σ_s	9.581	9.611	8.409
	54	σ_s	9.174	9.173	11.584
5. Resolved data many s-wave resonances in group 53	53	σ_c	413.752	414.473	444.900
	53	σ_s	138.905	139.148	149.383
6. Resolved data many p-wave resonances in group 45	45	σ_c	182.557	183.083	186.393
	45	σ_s	62.004	62.181	62.128
	45	$\sigma_c f^*$ @ $\sigma_o = 100$	.3652	.3645	
	45	$\sigma_s f^*$ @ $\sigma_o = 100$	.3797	.3790	

TABLE 3-2 (continued)

<u>TEDIUM ISOTOPE DESCRIPTION</u>	<u>GROUPS</u>	<u>REACTION</u>	<u>ENDRUN</u>	<u>ETOX</u>	<u>ETOG</u>
7. Unresolved data for one s-wave resonance range	39	σ_c	4.861	4.863	4.841
	39	σ_s	17.778	17.784	17.756
	39	$\sigma_c f^*$ @ $\sigma_o=100$	.4508	.4899	
	39	$\sigma_s f^*$ @ $\sigma_o=100$	.3475	.3716	
8. Unresolved data one p-wave resonance range	38	σ_c	.0321	.0321	.0321
	38	σ_c	3.42×10^{-4}	3.42×10^{-4}	-
12. Inelastic data	8	σ_{in}	1.00	1.00	1.00
	9	σ_{in}	1.00	1.00	1.00

heterogeneous effects have been precalculated and are provided, together with a subassembly description, with the benchmark problem.

3.2 Methods Development

Programming of a preliminary test version of the TRIAL code is completed and checked out. This preliminary version contains all of the computations required to obtain three-dimensional fluxes from two-dimensional flux sets. However, no attempt has been made to link TRIAL to other existing codes or to provide sophisticated output until the technique used has been tested and substantiated as being of sufficient accuracy to continue development.

Testing has been completed to the point of verifying the coding. This was done with models of the SEFOR core and a Demonstration Plant control cell. In this testing phase, one-dimensional results were used to determine two-dimensional fluxes for pre-computed two-dimensional problems. If the one-dimensional fluxes are taken directly from the two-dimensional results, this procedure should reproduce the two-dimensional results, thus testing the code programming. If the one-dimensional fluxes are from independent problems, it tests the sensitivity of the results to the validity of the initial solutions used. The first step has been successfully completed and the second is in progress.

A complete test of the technique requires a three-dimensional problem of the type TRIAL is typically meant to solve. Since the greatest need is for rapid computation of k_{eff} and power distributions in a reactor with partial control inserted, the test model has been chosen to consist of a core, radial and axial blankets and six partially-inserted control rods. Presently, one-dimensional solutions are being used to arrive at the critical dimensions and poison concentrations for the control rods.

While TRIAL currently outputs only fluxes, the information needed from three-dimensional computations includes the following:

1. Fluxes normalized to power by mesh-point, region, group and group-integrated.

2. Peak fluxes and mesh points at which they occur.
3. Fission power densities by mesh-point and line and integrated over lines and regions.
4. Peak power densities and mesh points at which they occur.
5. Reaction rate distributions along given lines.
6. Fuel cycle information, such as region dependent reaction rate cross sections.
7. Neutron balances by region, including directional leakages. (Needed to obtain bucklings for equivalent calculations in fewer dimensions.)
8. Energy deposition by mesh point and region.
9. General perturbation capability by mesh point and region.

Programming specifications are being written for this processor. However, it is being specified in a way in which it can be easily applied to any set of three-dimensional fluxes, rather than just to TRIAL output.

A purchase order has been signed with Aerojet Nuclear Corp. of Idaho Falls for computation time on their IBM 360-75 computer. It is planned that the ZPPR-3 computations on this computer will mostly consist of ARC runs and will be submitted through the ANL remote terminal in Idaho. A 29-group GE nuclear data file for ZPPR-3 has been sent to ANL for conversion to the ARC format.

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APPENDIX A - LIST OF CONTRIBUTORS

PROGRAM MANAGEMENT

P. Greebler

TASK A - PLANNING AND ANALYSIS OF ZPPR ASSEMBLY 3

Task Leader: H. S. Bailey

Contributors:

S. Crick
A. Hartman
S. Stewart

TASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

Task Leader: B. A. Hutchins

Contributors:

M. D. Kelley
J. E. Turner

DISTRIBUTION

Director, Contracts Division U. S. Atomic Energy Commission San Francisco Operations Office 2111 Bancroft Way Berkeley, California 94704	1	Division of Reactor Development and Technology, Headquarters U. S. Atomic Energy Commission Washington, D. C. 20545 Attn: Assistant Director, Project Management	2
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Division of Reactor Development and Technology, Headquarters U. S. Atomic Energy Commission Washington, D. C. 20545 Attn: Assistant Director, Plant Engineering	1	Assistant Director for Pacific Northwest Programs, RDT U. S. Atomic Energy Commission P. O. Box 550 Richland, Washington 99352	1
Division of Reactor Development and Technology, Headquarters U. S. Atomic Energy Commission Washington, D. C. 20545 Attn: Assistant Director, Program Analysis	1	RDT Site Office U. S. Atomic Energy Commission Argonne National Laboratory Building 2, Argonne, Illinois 60439	1

RDT Senior Site Representative U. S. Atomic Energy Commission P. O. Box 2325 San Diego, California 92112	1	Director Chemical Engineering Division Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois 60439	1
Division of Naval Reactors Chief, Nuclear Materials Branch U. S. Atomic Energy Commission Washington, D. C. 20545	1	FFTF Fuels Department Pacific Northwest Laboratory P. O. Box 999 Richland, Washington 99352	2
Director Division of Reactor Licensing U. S. Atomic Energy Commission Washington, D. C. 20545	3	RDT Site Office U.S. Atomic Energy Commission Atomics International P. O. Box 1446 Canoga Park, California 91304	1
Director Metallurgy Division Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois 60439	2	RDT Site Office U.S. Atomic Energy Commission Argonne National Laboratory P. O. Box 2108 Idaho Falls, Idaho 83401	1
Manager, FFTF Project Pacific Northwest Laboratory P. O. Box 999 Richland, Washington 99352	6	Division Leader (CMB) Chemistry and Metallurgy Division Los Alamos Scientific Laboratory P. O. Box 1663 Los Alamos, New Mexico 87544	1
Manager Chemistry and Metallurgy Division Pacific Northwest Laboratory P. O. Box 999 Richland, Washington 99352	1	Director Metallurgy and Materials Science Division Brookhaven National Laboratory Upton, New York 11973	1
RDT Site Office U. S. Atomic Energy Commission Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee 37830	1	Division Chief, M & S Division NASA - Lewis Research Center 21000 Brookpark Road Cleveland, Ohio 44135	1
Director Division of Reactor Standards U. S. Atomic Energy Commission Washington, D. C. 20545	2	General Manager Westinghouse Electric Corporation Bettis Atomic Power Laboratory P. O. Box 79 West Mifflin, Pennsylvania 15122	1
Division of Technical Information Extension U. S. Atomic Energy Commission P. O. Box 62 Oak Ridge, Tennessee 37831	135	Associate Manager Materials Engineering Department Battelle Memorial Institute Columbus Laboratories 505 King Avenue Columbus, Ohio 43201	1
Director LMFBR Program Office Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois 60439	2		

Manager, Nuclear Laboratories Combustion Engineering Inc. Nuclear Division Prospect Hill Road Windsor, Connecticut 06095	1	Manager, Research United Nuclear Corporation Research and Engineering Center Grasslands Road Elmsford, New York 10523	1
Manager, Plutonium Chemistry and Ceramics Fuels Development Nuclear Materials and Equipment Corporation Plutonium Laboratory Leechburg, Pennsylvania 15656	1	Director Nuclear Safety Program Oak Ridge National Laboratory P. O. Box Y Oak Ridge, Tennessee 37830	3
Head, Fuels and Materials Atomic Power Development Associates 1911 First Street Detroit, Michigan 48226	1	K-2 Group Leader Reactor Division Los Alamos Scientific Laboratory P. O. Box 1663 Los Alamos, New Mexico 87544	1
Director Metals and Ceramics Division Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee 37830	2	Westinghouse Electric Corporation Technical Director Advanced Reactors Division P. O. Box 158 Madison, Pennsylvania 15663	2
Director, Liquid Metal Engineering Center Atomics International P. O. Box 1449 Canoga Park, California 91304	1	Manager Liquid Metal Information Center Atomics International P. O. Box 1449 Canoga Park, California 91304	1
Manager—Advanced Development Activity General Electric Company Knolls Atomic Power Laboratory P. O. Box 1072 Schenectady, New York 12301	1	Irradiation Coordinator EBR-II Project Argonne National Laboratory P. O. Box 2528 Idaho Falls, Idaho 83401	1
Director, LMFBR Technology Program Atomics International P. O. Box 309 Canoga Park, California 91304	1	Experiment Manager EBR-II Project Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois 60439	1
Director Nuclear Development Center The Babcock and Wilcox Company Atomic Energy Division Lynchburg, Virginia 24501	1	Director Reactor Safety & Analysis Division Argonne National Laboratory 9700 South Cass Avenue Argonne, Illinois, 60439	9
Laboratory Assistant Director Gulf General Atomic Incorporated P. O. Box 608 San Diego, California 92112	1	Technical Director, FFTF Project Hanford Engineering Development Laboratory P. O. Box 1970 Richland, Washington 99352	1

U.S. Atomic Energy Commission
Region II - Division of Compliance
Suite 818
230 Peach Tree Street, N. W.
Atlanta, Georgia 30303

Dr. Knox M. Broom, Jr.
Nuclear Specialist
Middle South Services, Inc.
P.O. Box 61000
New Orleans, La.

Division of Reactor Development
and Technology Headquarters
U.S. Atomic Energy Commission
Washington, D. C. 20545
Attn: Chief Of Reactor Physic Branch

1 Assistant Director of Pacific Northwest Programs, RDT 1
U.S. Atomic Energy Commission
P.O. Box 550
Richland, Washington 99352

1 Technical Director, FFTF Project 1
Hanford Engineering Development Laboratory
P.O. Box 1970
Richland, Washington 99352

Division of Reactor Development
and Technology Headquarters
U.S. Atomic Energy Commission
Washington, D. C. 20545
Attn: Chief Of Core Design

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