CRITICAL EXPERIMENTS AND COOPERATIVE EVALUATIONS PROGRAM

**SEVENTH QUARTERLY REPORT
APRIL-JUNE 1973**

U.S. ATOMIC ENERGY COMMISSION
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SEVENTH QUARTERLY REPORT

April-June 1973

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TABLE OF CONTENTSPage

	Abstract	
1.	SUMMARY	
	SUBTASK A - Critical Experiment Planning and Analysis	1-1
	SUBTASK B - Cooperative Nuclear Data and Methods Development	1-2
2.	SUBTASK A - CRITICAL EXPERIMENT PLANNING AND ANALYSIS	
	2.1 Scope and Objective	2-1
	2.2 Experimental Planning	2-1
	2.3 ZPPR-3 Phase 3 Doppler Calculations - Pre Analysis	2-9
	2.4 Phase 3 Foil Irradiations - Post Analysis	2-12
	2.5 Computerized Data Evaluation	2-16
	2.6 ZPPR-3 Cross Section Generation	2-24
3.	SUBTASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT	
	3.1 Scope and Objectives	3-1
	3.2 Nuclear Data for Pu-239	3-2
	3.3 Combined Multi-Group Structure	3-8
	3.4 The 3-D Linking Code TRIAL	3-8
	3.5 Unresolved Resonance Treatment in the MINX Code	3-8
	3.6 ACRP Activities	3-10
4.	QUALITY ASSURANCE	
	SUBTASK A - Critical Experiment Planning and Analysis	4-1
	SUBTASK B - Cooperative Nuclear Data and Methods Development	4-1
	REFERENCE	R-1
	APPENDIX A - Prior Reports	A-1
	- List of Contributors	A-2

LIST OF TABLES

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
2.1	ZPPR-3 Phase 3 Computed U-238 Spatial Doppler Reactivity Effect	2-12
2.2	ZPPR-3, Phase 1B Spherical Sodium-Void Calculations	2-26
2.3	Checkout Calculations Performed with Updated Cross Sections Files	2-32
3.1	Adjusted Smooth Fission and Capture Cross Sections for ENDF/B-IV Pu-239	3-4
3.2	Smooth Data Contributions in the Unresolved Resonance Range	3-5
3.3	Remaining Activities for Preparation of Pu-239 Data for ENDF/B-IV	3-6
3.4	Unresolved Resonance Parameters for Pu-239	3-9

LIST OF ILLUSTRATIONS

<u>Figure No.</u>	<u>Title</u>	<u>Page</u>
2.1	Control Position Change from ZPPR-3 to ZPPR-4	2-9
2.2	ZPPR-3 Doppler Measurement Locations	2-11
2.3	ZPPR-3 Phase 3 U-238 Doppler Distribution in Row 137	2-14
2.4	Phase 3 Measured and Calculated U-235 Fission Rate Distribution in Row 137 for the Reference Control Configuration and Two Altered Configurations	2-15
2.5	ZPPR-3 Distorting Influences for Rod #2 U-235 Fission Distribution for Row 137	2-20
2.6	Phase 1B Layout with Numbered Control Positions	2-21
2.7	ZPPR-3 Distorting Influences for Rod #8 U-235 Fission Distribution for Row 137	2-22
2.8	ZPPR-3 Distorting Influence for Rods #2 and #8 U-235 Fission Distribution for Row 137	2-23
2.9	ZPPR Assembly 3 Inner Core Drawer Configuration	2-27
2.10	ZPPR Assembly 3 Outer Core Drawer Configuration	2-28
2.11	ZPPR Assembly 3 One-Dimensional Condensation Model	2-29
2.12	Cross Section Data Processing Block Diagram	2-31
2.13	Comparison of Phase 3 Measured and Calculated (for Current and Updated Cross Section File) U-235 Fission Rate Distribution in Row 137 for the Reference Control Configuration	2-34

SUMMARY

1.1 SUBTASK A - CRITICAL EXPERIMENT PLANNING AND ANALYSIS

In accordance with RRD directions, the scope of subtask A has been extended to include planning, specification and analysis of the ZPPR-4 critical experiment program. Meetings have been held with RRD, WARD, and ANL to establish working procedures and information exchange. The ZPPR-4 program will utilize a simulation of the Demonstration Plant reactor design and will emphasize measurements pertinent to the radial and axial blanket performance. The program will include core performance measurements such as power distribution and control system worth. A transition period between ZPPR-3 and ZPPR-4 is being used to efficiently obtain early measurement of sodium void worth in the Demonstration Plant design.

Analysis of the spatial distribution of the U-238 Doppler reactivity predicts that the value near an inserted control rod is only ~60% of the value near a withdrawn rod. This large gradient in the Doppler reactivity indicates the interpretation of Doppler measurements will be complicated by the need to integrate over the sample location.

Power tilting measurements confirm the large distortion of flux predicted by analysis. A computer code, FONDLE, has been written to manipulate and compare large sets of data. The code has been used for least square fits of data and for comparison of measured and calculated two dimensional power distributions. The code has also been used to demonstrate that the multiplicative distorting influences of a given control rod on the power distribution is relatively independent of the positions of other control rods.

The master cross section file for ZPPR-3 analysis has been revised and updated. Modifications have been made to provide data for better calculation of the fuel spike worth and the sodium void worth. The updated file has little effect on calculated power distributions but does improve the agreement with measured control rod worths and spike worths.

1.2 SUBTASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

Preparation of the nuclear data of Pu-239 for ENDF/B IV continued, with the following activities completed:

- 1) Generation of unresolved resonance parameters,
- 2) Extension of the resolved resonance range to 600 eV using parameters of Ribon,
- 3) Adjustment of the smooth data between 600 eV and 1 MeV for consistency with the final values of σ_f for U-235 and to assure continuity across the upper energy bound of the unresolved resonance range.

The major activities remaining for Pu-239 are 1) the estimation of data uncertainties, 2) a decision on whether to use ENDF/B-III resolved resonance parameters below 300 eV or those of Ribon, and 3) the incorporation of a LASL data evaluation above 1 MeV. The latter effort will provide for a more complete evaluation over that energy range and for better agreement with hard-spectrum benchmark criticals.

The statistical resonance parameters generated for Pu-239 will result in more secondary structure than obtained with the ENDF/B-III parameters. This is particularly true for the total cross section and should permit some degree of self-shielding of the secondary structure with ENDF/B-IV data.

The topical report describing the TRIAL code and the evaluation of the 3-D linking procedure was issued.

Other activities included participation in 1) the specification of the unresolved resonance calculation for the MINX code, 2) establishment of a combined multi-group structure and 3) a meeting of the Advisory Committee on Reactor Physics.

2. SUBTASK A - CRITICAL EXPERIMENT PLANNING AND ANALYSIS

Subtask Leader: S. L. Stewart

2.1 Scope and Objective

Subtask A provides the specification and analysis of the Demonstration Plant Benchmark Critical Experiments in ZPPR Assembly 3. This effort involves scoping analysis to plan the overall experiment program, pre-analysis of experiments for detailed experimental specification and requirements, and detailed post-analysis of experiments to evaluate nuclear data, design, and analysis methods. The overall objective is to resolve major core nuclear design uncertainties for preliminary design of the LMFBR Demonstration Plant.

2.2 Experimental Planning

2.2.1 ZPPR Assembly 4

The scope of Subtask A has been extended to include specification and analysis of the experimental program in ZPPR Assembly 4. This program will be a cooperative effort between General Electric Company, Argonne National Laboratory, and Westinghouse Advanced Reactors Division. ZPPR Assembly 4 is intended to provide direct but general experimental support of the Demonstration Reactor Design effort. This means that although the experimental program is directed toward the solution of Demonstration Reactor Design problems, the critical mockup does not need the same degree of similitude as an engineering mockup critical experiment.

The attached outline summarizes the preliminary overall ZPPR-4 critical experiment program plan that has been developed in response to W-ARD's stated design and schedule requirements for the LMFBR Demonstration Plant. The objective of this plan is to structure the ZPPR-4 experiment program to satisfy the highest priority requirements in a timely manner. It should be emphasized that this program outline is preliminary and likely to change as additional input from W-ARD and ANL is received.

The early part of the ZPPR-4 program is focused on developing design information for the radial blanket assembly design. This appears to be the only core component in the Demonstration Plant where the design has not been frozen. In addition to radial blanket information, a series of design confirmation experi-

ments are planned. These experiments will take a lower priority ranking than the radial blanket measurements. Design confirmation experiments are planned in the following areas:

Control System
Ex-Vessel Neutron Detectors
Reactor Performance, e.g., Power Distribution
Flow Orificing
Fuel Management

A four-phase (12-16 month) experimental program is planned for ZPPR-4.

- 1) Beginning-of-Cycle - Initial Core (~4 months)
- 2) End-of-Cycle - Initial Core (~3 months)
- 3) End-of-Cycle - Equilibrium Core (~3 months)
- 4) Middle-of-Cycle - Equilibrium Core (~2 months)

Phase 1, the beginning-of-cycle condition for the initial loading, will be the reference condition against which all other blanket measurements will be compared. This is the only reactor condition which does not have plutonium in the blankets. The plutonium buildup distribution in the blankets will be simulated by Pu spiking the blankets in all experimental configurations except Phase 1. Thus, the blanket conditions will be simulated from the initial loading (Phase 1), to the end of the first operating interval (Phase 2), to the end of an equilibrium operating interval (Phase 3) which is the limiting radial blanket condition. Phase 4 is added to the ZPPR-4 program since the middle of an equilibrium cycle is expected to be the limiting power distribution condition and will provide early verification of estimated power and flow distributions in the reactor core.

Experiments to support licensing efforts are not planned for ZPPR-4, which is too late to effectively support the PSAR effort. Experiments for FSAR support appear to be better performed in the EMC following ZPPR-4.

PRELIMINARY ZPPR-4 OVERALL PROGRAM OUTLINE

PHASE 1 - BEGINNING-OF-CYCLE - INITIAL CORE LOADING

(Approximately 4 months)

1. Load to Critical

Simulate initial core loading and initial control assembly worth and boron concentration.

Blankets will contain depleted U.

2. Power Distribution

The axial and radial fission power distribution will be measured in both the core and blankets by irradiating foils. The gamma heating distribution will also be measured with particular emphasis on the core-blanket interface heating gradient.

3. Control Measurements

Measure control rod worth subcritically for several configurations, including the safety rod worths. Load simulated equilibrium cycle control composition. Measure control rod worth for several configurations. Measure core and blanket power distribution to identify power distribution effects associated with equilibrium cycle control system.

4. Low Level Neutron Monitor

Perform experiments to measure the neutron attenuation between the core and low level neutron detector and between the core and the stored fuel location. (The feasibility of this experiment is not yet established.)

5. Diagnostic Experiments

These experiments are intended to help understand and apply experimental data to Demo design. Experiments may include axial and radial small sample reactivity traverses, high Pu-240 experiment, fuel rearrangement experiment, fuel spiking sensitivity experiment, etc.

PHASE 2 - END-OF-CYCLE - INITIAL CORE LOADING

(Approximately 3 months)

1. Load to Critical

No control rods (CR's) inserted.

Plutonium distributed by fuel and blanket spiking adjustment to simulate core burnup and blanket buildup of Pu at the end of the initial cycle.

2. Power Distribution

The axial and radial fission and gamma power distribution will be measured in both the core and blankets. The power distribution at the core-blanket interface is of primary interest.

3. Control Measurements

These experiments will estimate the change in control worth or calibration during the first operating cycle. Repeat all control rod measurements performed in Phase 1, with both the simulated initial control system and simulated equilibrium control system. The subcritical measurement of the critical control rod configuration(s) of Phase 1 will provide the simulated burnup reactivity effect measurement. Measure single CR influence on blanket power distribution.

4. Diagnostic Experiments

Same as in Phase 1.

PHASE 3 - END-OF-CYCLE - EQUILIBRIUM (OR ASYMPTOTIC) CORE
(Approximately 3 months)

1. Load to Critical

No control rods inserted.

Pu distributed by fuel and blanket spiking adjustment to simulate core burnup and blanket buildup of plutonium.

The blanket Pu distribution resulting from several alternate blanket management schemes may be utilized.

2. Power Distribution

Axial and radial fission and gamma power distribution will be measured in both the core and blankets. The influence of blanket shuffling on blanket power distribution is of particular interest as is the core-blanket power gradient.

3. Control Measurement

The worth of several control rod configurations will be measured subcritically, including safety rods, and configurations planned for Phase 4 experiments. Comparison with measurements in Phase 2 will evaluate the change in control worth due to Pu redistribution. Again, the influence of a single CR on the blanket power distribution will be measured.

4. Diagnostic Experiments

Experiments same as in Phase 1.

PHASE 4 - MIDDLE-OF-CYCLE - EQUILIBRIUM (OR ASYMPTOTIC) CORE
(Approximately 2 months)

1. Load to Critical

MOC control configuration simulated, ~10\$ control inserted.

Plutonium distributed by fuel and blanket spiking adjustment to simulate MOC core and blanket plutonium distribution.

2. Power Distribution

The axial and radial fission and gamma power distribution in the core and blankets will be measured. Particular emphasis is placed on core power distribution and how it is influenced by MOC control rod patterns for both normal and abnormal operating conditions. Measure the influence of control rod patterns on blanket power distribution.

3. Control Measurement

Measure worth of control rod configuration, including safety rods. Repeat control rod measurements performed in Phase 3.

4. Diagnostic Experiments

Same experiments as performed in Phase 1.

2.2.2 ZPPR-3 to ZPPR-4 Transition Experiments

The ZPPR Assembly 3 Phase 3 experimental program includes a large sodium void experiment where sodium is removed from over fifty percent of all core drawers. This sodium must be reloaded back into the core to proceed with ZPPR-4 experiments. With a modes change in core arrangement and control configuration, reloading the sodium will provide a sodium void measurement on a configuration more closely representing the Demonstration Reactor Design. A description of the ZPPR-3 to ZPPR-4 transitional experiments is described below.

Experimental Program

On completion of the measurements currently being performed in the sodium voided environment in ZPPR-3 Phase 3, the Phase 3 experimental program will be interrupted and the following experiments performed.

2.2.2.1 Outer Core and Control Modification

Reload the outer core and control rod positions (CRP-s) to the configuration shown in Figure 2.1. The following loading changes will be required to make this modification:

32 drawers RB → OC
 18 drawers OC → RB
 24 drawers OC → IC
 32 drawers CRP → IC
 32 drawers OC → CRP

In addition, change the inserted control rods from the reference Phase 3 configuration (#2,4,6,8,10,12,14,16, and 18) to a 7 control rod configuration where control rod #1 is inserted and the 13 kg B_4C control rod loading will be inserted into control rod positions, 9,11,13,15,17, and 18. Thus, control rod position 1 contains ~4.58 kgs B_4C and control rod positions 9,11,13,15,17, and 19 each contains 13.02 kgs B_4C . The net effect of changing the outer core and CRP configuration and control insertion pattern is to increase reactivity by an estimated 1.3\$. Thus, about 2\$ of fuel spikes will be removed before the above loading changes are initiated. After all loading changes have been completed,

adjust fuel spikes to achieve a critical condition. Modify the sodium void pattern to the configuration which gives the expected maximum Na void reactivity effect for the above control rod configuration.

2.2.2.2 Sodium Void Measurements

Sodium will be progressively loaded back into the assembly in steps which reverse in principle the sodium voiding measurements already performed in Phase 3, e.g., reload sodium in cylindrical rings starting from the outside of the void pattern in at least 5 steps. The reactivity effect of each step will be measured.

2.2.2.3 Sodium-In Neutronic Measurements

At the completion of Experiment 2.1.2.2, all of the sodium will be loaded back into the assembly. The assembly will be subcritical by the sodium void reactivity effect. The assembly will be brought to a critical condition by loading fuel spikes and the following experiments performed.

a. Power Distribution Measurements

The radial power will be measured in the upper 180° sector of one-half of the assembly. Irradiation of enriched uranium foils will be the primary measurements with sufficient plutonium foils to allow normalization to a plutonium based power distribution. No U-238 foils will be irradiated.

b. Single Control Rod Measurements

All control rod worths will be measured subcritically with the seven control pattern (#1,9,11,13,15,17, and 19) fully inserted in the modified outer core arrangement. The worth of the following fully inserted single control rods will be measured.

- (1) Control rod 2
- (2) Control rod 8
- (3) Control rod 10

2.2.2.4 Reload to Phase 3

Reload the reactor back to the original Phase 3 arrangement in the following steps measuring the reactivity effect of each step.

- a. Reload to Phase 3 control pattern, i.e., #2,4,6,8,10,12,14,16, and 18 inserted. The reactivity change of this step is estimated to be -0.3\$.

CONTROL POSITION CHANGE FROM ZPPR-3 TO ZPPR-4

ZPPR HALF I

(SHADED AREA REPRESENTS MODIFIED OUTER ZONE)

FIGURE 2.1

- b. Reload control pitch back to original Phase 3 control pitch. The reactivity change of this step is estimated to be -3.0\$.
- c. Re-adjust outer core boundaries back to original Phase 3 arrangement. The reactivity effect of this step is estimated to be +2.0\$. At this point the reactor should be ~\$1.3 subcritical. Add fuel spikes until criticality is again achieved.
- d. Perform Phase Na-in U-238 Doppler measurements.
- e. Measure Phase 3 Na-in radial small sample reactivity traverses.

2.3 ZPPR-3 Phase 3 Doppler Calculations - Pre Analysis

A set of calculations have been performed to determine the spatial distribution of the U-238 Doppler reactivity effect with the ZPPR-3 Phase 3 reference control rod configuration, i.e., #2,4,6,8,10,12,14,16 and 18 fully inserted. These calculations were performed in two dimensional (x,y) geometry and 6-energy groups on the ARC system in Idaho. The spatial distribution of the Doppler effect was evaluated using "exact perturbation theory". The real flux distribution was computed for the reference Phase 3 conditions where all cross sections are defined at 300°k. The adjoint flux distribution was computed for the perturbed condition where the U-238 cross sections are defined at 700°k. The cross sections were generated from ENDF/B-III data and have been previously described.⁽¹⁾

The four characteristic positions where Doppler measurements will be performed in Phase 3 are identified in Figure 2.2 along with the reference Phase 3 control pattern. Actually only two positions will be used, 135-32 and 136-31, with the control rods repositioned to provide the environment of positions 135-42 and 136-43. The computed U-238 Doppler reactivity effect at these four positions is listed in Table 2.1. The values shown in Table 2.1 are computed for the U-238 in the ZPPR fuel drawer and represent an axial average of the Doppler effect.

Examination of Table 2.1 indicates the U-238 Doppler effect is substantially reduced near an inserted B₄C control rod as expected. The computed value in drawer 136-31, next to control rod #2, is only about 60 percent of the value

ZPPR-3 PHASE 3
DOPPLER MEASUREMENT LOCATIONS

- Position of Doppler Measurements.
- ▨ Inserted Control Rod

TABLE 2.1

ZPPR-3 PHASE 3 COMPUTED U-238 SPATIAL DOPPLER REACTIVITY EFFECT
 (300°k-700°k)

<u>Position, Drawer No.</u>	<u>300°k-700°k Reactivity, %Δk/kg U²³⁸</u>
136-31*	-0.77×10^{-3}
135-32	-1.17×10^{-3}
135-42	-1.12×10^{-3}
136-43**	-1.26×10^{-3}

* Next to an inserted control rod.

** Next to a withdrawn control position.

in drawer 136-43, next to withdrawn control position #5. In fact, the computed value in drawer 136-31 is only 65 percent of the value computed for the adjacent drawer 135-32. This indicates a severe gradient in the local Doppler effect in the vicinity of inserted B_4C control rod. To illustrate this gradient, the relative local U-238 Doppler effect in row 137 is plotted in Figure 2.3. For reference purposes the computed U-235 fission distribution is also plotted in Figure 2.3.

Inspection of Figure 2.3 shows the severity of the U-238 Doppler gradient between an inserted and an adjacent withdrawn control rod. The Doppler gradient observed in Figure 2.3 is more severe than indicated from Table 2.1 since the values in Table 2.1 identify the average value in a ZPPR drawer. Figure 2.3 also contains the radial variation in Doppler effect. Further inspection also indicates that a substantial gradient exists, even across one ZPPR drawer. For reference purposes the width of a ZPPR drawer is indicated in Figure 2.3.

These observations lead to some serious questions about the ability to properly analyze and interpret the proposed small sample Doppler measurements in ZPPR-3, Phase 3.

2.4 Phase 3 Foil Irradiations - Post Analysis

The irradiated foil data from the power tilting experiments in Phase 3 have been received. The experimental values were compared with the predicted values which appeared in the previous quarterly progress report.⁽²⁾ This comparison is shown in Figure 2.4 where the experimental data for the two tilted experiments is plotted on the same figure which previously appeared.

As observed in Figure 2.4 the power distortion caused by inserting 1 rod and withdrawing another control rod (rod swap) can be quite large. This large distortion is predicted analytically and verified by the irradiated foil data. The agreement between calculations and measured data in these experiments is remarkable. This is particularly surprising since this experiment was intended to create flux distortions and tilting. It should be remembered that this pre-analysis was performed using diffusion theory with local heterogeneity corrections.

ZPPR-3 PHASE 3 U-238 DOPPLER DISTRIBUTION
IN ROW 137

FIGURE 2.3

FIGURE 2.4 PHASE 3 MEASURED AND CALCULATED U-235 FISSION RATE DISTRIBUTION
IN ROW 137 FOR THE REFERENCE CONTROL CONFIGURATION AND TWO ALTERED CONFIGURATIONS

It appears that although control rod can cause large power distortions, these distortions can be accurately predicted with relatively unsophisticated analytical techniques.

2.5 Computerized Data Evaluation

There exists a need to be able to manipulate large amounts of computed and experimental values to properly evaluate and analyze various analytical techniques for predicting power distribution. These comparisons between experimental and calculated results are difficult to evaluate due to the bulk of data. With the large volume of experimental data generated in the ZPPR-3 program, the proper utilization of the experimental data required the development of an automatic technique for data manipulation and comparison.

2.5.1 FONDLE, A Computer Code for Manipulating Data

The computer code FONDLE was designed and programmed to compare ZPPR-3 measured and calculated results and to help interpret these comparisons. FONDLE can manipulate large amounts of data such that comparisons can be made between different sets of experimental data or between experimental and calculated results. Hopefully, FONDLE will be able to help in uncovering subtle points of interest to the reactor designer which might remain undetected due to the large amounts of data necessary to process.

FONDLE is capable of doing the following types of manipulations:

- (a) The code can accept input information (i.e., fluxes) from a number of different computational codes, such as SN2D and ARC fluxes from ANL- Idaho.
- (b) Calculate reaction rate distributions using input fluxes and cross sections.
- (c) Calculate a least-square normalization fit using selected data points from the first array, and have all of the second array multiplied by the normalization value. A least square fitting of data is advantageous in that it provides the reactor designer with quantitative bias factors and uncertainties which are statistically meaningful. In FONDLE, normalization of the calculated array to selected data points in the experimental array is done by least square fitting (LSF) techniques.

- (d) Normalize selected data points (i.e., zones) from an input array with a constant value. This is useful if each set of fluxes is to be normalized differently by zones.
- (e) Unfold arrays about symmetry boundaries. This is useful when comparing computed data from calculations which have different symmetry assumptions, such as 1/4 core versus 1/2 core representations.
- (f) Select a small portion of a larger array, to be used by any of the other functions of FONDLE. This is helpful when only a small part of the larger array is required.
- (g) Provide pointwise ratios of two different data sets. By properly selecting the data arrays one is able to identify the change between the two data sets. This is advantageous in that it is possible to identify the distorting influence of control rods on power distribution. The distorting influence can be defined as the perturbed distribution (control rod inserted) divided by the unperturbed distribution (control rod withdrawn).
- (h) Provide a pointwise product of two or more different data arrays.
- (i) Interpolate data points to new points. This is needed because the irradiated foil position may be different from calculated activity point. This requires the computed values to be interpolated to provide a value at the same spatial position as the foil.

FONDLE will also plot any of the above results (a through i) in two types of pictorial representations, either a two-dimensional contour plot, and/or a three-dimensional perspective view. Finally FONDLE can manipulate the two-dimensional data arrays such that one or several of the above functions can be applied in any sequence that the user deems necessary.

2.5.2 Influence of Control Rods on Power Distributions in ZPPR-3

The computer code FONDLE has been used to determine the distorting influence of control rods on the reactor power distribution. The distorting influence of a control rod will be defined, in this case, as the perturbed U-235 fission distribution (control rod inserted) divided by the unperturbed U-235 fission distribution (control rod withdrawn). An example of this is presented in Figure 2.5,

where the distorting influence of control rod #2 on Phase 1B is shown through the center of the assembly. For reference purposes, the ZPPR-3 core layout is shown in Figure 2.6 with the control rod positions numbered 1 through 19. As observed in Figure 2.5, the distorting influence ranges from $\sim .56$ within control rod #2 to ~ 1.07 on the far side of the reactor core. In Figure 2.7 the distorting influences are plotted for control rod #8.

The range of influences run from $\sim .34$ in control rod #8 to ~ 1.10 at the other side of the core. As one can see control rod #8 creates more power tilting than rod #2, along with more power depression around the inserted rod.

By comparing the distorting influences of control rod #8 on Phase 1B, where there are no other control rods inserted, and Phase 2, where control rods #2, 4, 6, 10, 14, and 18 are inserted, one observes significant results. The distorting influence of a control rod on the power distribution is independent of presence of other control rods. This conclusion is demonstrated in Figure 2.7. The solid dots represent the pointwise influence function computed for Phase 1B, where no other control rod are inserted, the circled dots represent the pointwise influence function computed for Phase 2, where control rods 2, 4, 6, 10, 14, and 18 were also inserted. As observed in Figure 2.7 the two sets of influence function values are almost identical. Even in Phase 2 where rods #2 and #8 are inserted next to each other, the largest influence error between the rods is approximately 3 percent.

The significance of this conclusion is evident since one may then synthesize a power distribution by combining the influence functions of the inserted control rods with an undisturbed power distribution. To test the validity of combining influence functions, the product of the distorting influences of rods #2 and #8 are shown in Figure 2.8 along with the computed distorting influence of the combination of control rods #2 and #8. As observed the range is from $\sim .31$ in control rod #8 to approximately 1.16 on the far side of the reactor. The difference between the computed influences and that of product of the influences show remarkable agreement around the two inserted control rods. And again, the largest error is to be found on the far side of the reactor with an error of ~ 1.5 percent.

FIGURE 2.5
ZPPR-3 DISTORTING INFLUENCES FOR ROD #2
U-235 FISSION DISTRIBUTION FOR ROW 137

Q

FIGURE 2.6

ZPPR HALF I

PHASE 1B LAYOUT WITH NUMBERED CONTROL POSITIONS

FIGURE 2.7
ZPPR-3 DISTORTING INFLUENCES FOR ROD #8
U-235 FISSION DISTRIBUTION FOR ROW 137

FIGURE 2.8
ZPPR-3 DISTORTING INFLUENCE FOR RODS # 2 AND #8
U-235 FISSION DISTRIBUTION FOR ROW 137

The information presented in Figure 2.8, tends to confirm the possibility that power distributions may be predicted by a simple synthesis technique involving the distorting influence functions of control rods and an undistorted power distribution.

2.6 ZPPR-3 Cross Section Generation

2.6.1 Cross Section Data Processing

The master cross section library for analysis of ZPPR Assembly 3 has been updated and modified. Several areas in the analysis of ZPPR-3 experimental data indicated the necessity for updated cross sections.

Comparison of ZPPR-3 calculations and experimental data indicates that the reactivity effect of the fuel spikes was overestimated⁽²⁾. The master cross section library contained only the Pu-239 and Pu-240 cross sections to describe the fuel spike. The overestimate of the fuel spike worth was probably caused by neglecting the changes in heterogeneity corrections of adjacent materials due to the presence of the fuel spike. Cross sections describing the adjacent materials were added to the master library.

The poor agreement with experiment⁽³⁾ for the Phase 1B partial sodium voiding calculations showed the requirement for regenerating flux weighted sodium-void cross sections. The sodium-void cross sections for the partial sodium voiding calculations were flux weighted (corrected because of the fine flux distribution in the drawer caused by the heterogeneous distribution of the materials) with the 29-energy group non-voided flux weighting factors. In updating the master cross section library, one-dimensional, 59-energy group, S16, transport calculations were utilized to compute the flux weighting factors applied to the sodium voided cross sections.

Updated sodium-void cross sections will also be required in the analysis of the U-238 Doppler experiments where the assembly is partially sodium-voided.

Updating of the master cross section library began with the 59-energy group cross sections generated by TDOWN⁽⁴⁾, which are described in the Fourth Quarterly Report⁽⁵⁾. These cross sections were corrected because of the fine flux distribution in the drawer caused by the heterogeneous distribution of the materials.

To represent a homogenized drawer accurately, the group cross sections of each material were flux weighted to provide the same reaction rates as the heterogeneous description of the drawer. One-dimensional S16 transport calculations in 59-energy groups were utilized to compute these fluxes.

To account for the changes in heterogeneity corrections of adjacent materials due to the presence of a fuel spike, a two drawer "cell" model was described for the inner (Figure 2.9) and outer (Figure 2.10) core. These models were used to compute the fine flux distributions described above. The two drawer "cell" provides a more accurate positioning of the spike within the drawer. Previous one drawer descriptions placed the effective center of the Pu-Al plate at the drawer center. The two drawer model places the Pu-Al plate at the drawer edge and provides a uniform distribution of the spike influence on adjacent materials.

To compute the buckling values for each cell calculation described above, the Phase 1B core configuration was described in spherical geometry. The sphere was defined to conserve the total volume of the inner and outer core zones, radial blanket, and shield. An identical calculation was performed for the sodium-voided condition. The sodium was voided from the inner core zone and to a radius of two inches into the outer core zone. The buckling values for each cell were combined with the group dependent diffusion coefficient and input as a removal cross section to each composition defined in the cell calculations. The constant removal term imposes the reactor spectrum upon the cell calculation.

Upon completion of the flux weighting of the 59-energy group cross sections, the spherical calculations were repeated utilizing the flux weighted cross sections. The results are presented in Table 2.2. Examination of these results indicates that the flux weighting increases the calculated sodium void worth by approximately 5 percent.

The flux weighted 59-energy group cross sections were condensed to 6-energy groups using SN1D⁽⁶⁾. The one-dimensional, cylindrical geometry, condensation model is shown in Figure 2.11. The radii for zone 3 (Figure 2.11) were determined by computing the average radius for control rod positions #2, 3, 4, 5, 6, and 7 and utilizing that computed radius as the center of the volume associated with the 24 matrix positions occupied by control rod positions 2, 3, 4, 5, 6, and 7.

TABLE 2.2

ZPPR-3, Phase 1B, Spherical Sodium-Void Calculations

	<u>Resonance Shielded Cross Sections</u>	<u>Flux Weighted Cross Sections</u>
Eigenvalue, Sodium-In	1.00447	1.01453
Eigenvalue, Sodium-Out	1.01791	1.02861
Delta (Out-In)	0.01344	0.01408
Void Worth (\$)*	4.27	4.47

$$*\beta = .003147$$

FIGURE 2.9

ZPPR ASSEMBLY 3 INNER CORE DRAWER CONFIGURATION

MATRIX
U ₃ O ₈
SODIUM
Fe ₂ O ₃
Pu-U-Mo
Fe ₂ O ₃
SODIUM
U ₃ O ₈ *
U ₃ O ₈ *
MATRIX
U ₃ O ₈
SODIUM
Fe ₂ O ₃
Pu-U-Mo
Fe ₂ O ₃
SODIUM
U ₃ O ₈
MATRIX

0.0
0.22225
0.85725
2.08915
2.48285
3.04165
3.43535
4.66725
5.02285
5.26415
5.74675
6.38175
7.61365
8.00735
8.56615
8.95985
10.19175
10.82675
11.04900

PLATE MATERIAL CONCENTRATIONS (ATOMS/CM³X10⁻²⁴)

MATERIAL	U ₃ O ₈	Pu-U-Mo	Fe ₂ O ₃	SODIUM 1/2 INCH	MATRIX	Pu-Al
U-235	3.442-5	6.200-5				
U-238	1.565-2	2.797-2				
O	4.169-2		3.761-2			
Mo		2.524-3	2.132-5	1.433-5	1.620-4	
Pu-241		1.670-4				7.400-5
Pu-240		1.382-3				1.356-3
Pu-239		9.846-3				2.739-2
C			2.000-5	2.000-5	2.650-4	
Al			1.497-5	4.072-6	3.000-6	2.850-3
Fe			4.191-2	7.222-3	5.932-2	
Cr			4.724-3	2.077-3	1.690-2	
Ni			2.878-3	9.562-4	7.401-3	
Mn			3.626-4	1.714-4	1.493-3	
Si			2.326-4	1.060-4	8.910-4	
Cu			3.600-5	2.088-5	2.270-4	
Na				2.340-2		
Pu-242		1.000-6				1.000-6

* U₃O₈ plate replaced with Fe₂O₃ and Pu Al plates for spiked configuration.

ZPPR ASSEMBLY 3 OUTER CORE DRAWER CONFIGURATION

FIGURE 2.10

0.0	MATRIX
0.22225	Fe ₂ O ₃
0.57785	Pu-U-Mo
1.13665	Fe ₂ O ₃
1.53035	SODIUM
2.76225	U ₃ O ₈
3.39725	SODIUM
3.99415	Fe ₂ O ₃
4.38785	Pu-U-Mo
4.94665	Fe ₂ O ₃
5.30225	MATRIX
5.74675	SODIUM
6.97865	Fe ₂ O ₃
7.37235	Pu-U-Mo
7.93115	Fe ₂ O ₃
8.23485	SODIUM
9.55675	U ₃ O ₈ *
9.95045	U ₃ O ₈ *
10.19175	SODIUM
10.78865	MATRIX
11.04900	

PLATE MATERIAL CONCENTRATIONS (ATOMS/CM³X10⁻²⁴)

MATERIAL	U ₃ O ₈	Pu-U-Mo	Fe ₂ O ₃	SODIUM 1/2 INCH	MATRIX	Pu-Al	SODIUM 1/4 INCH
U-235	3.442-5	6.200-5					
U-238	1.565-2	2.797-2					
O	4.169-2		3.761-2				
Mo		2.524-3	2.132-5	1.433-5	1.620-4		1.670-5
Pu-241		1.670-4				7.400-5	
Pu-240		1.382-3				1.356-3	
Pu-239		9.846-3				2.739-2	
C			2.000-5	2.000-5	2.650-4		2.326-5
Al			1.497-5	4.072-6	3.000-6	2.850-3	8.362-6
Fe			4.191-2	7.222-3	5.932-2		9.977-3
Cr			4.724-3	2.077-3	1.690-2		2.876-3
Ni			2.878-3	9.562-4	7.401-3		1.348-3
Mn			3.626-4	1.714-4	1.493-3		2.328-4
Si			2.326-4	1.060-4	8.910-4		1.401-4
Cu			3.600-5	2.088-5	2.270-4		2.702-5
Na				2.340-2			2.164-2
Pu-242		1.000-6				1.000-6	

* U₃O₈ plate replaced with Fe₂O₃ and Pu-Al plates for spiked configuration.

FIGURE 2.11

ZPPR ASSEMBLY 3 ONE-DIMENSIONAL CONDENSATION MODEL

Matrix
Locations

	4	105.6	24	252.4	48	386	834	328
(1)	CRP #1							
(2)	INNER CORE							
(3)	CRP #2,3,4,5,6,7							
(4)	INNER CORE							
(5)	CRP #8,9,10,11,12,13,14,15,16,17,18,19							
(6)	OUTER CORE							
(7)	RADIAL BLANKET							
(8)	RADIAL SHIELD							
0.0	6.37823	33.38029	36.85559	62.65616	66.43776	91.32234	129.69937	141.97829

() Zone Designation.

Two separate condensations were performed. The first condensation utilized the heterogeneous Phase 1B core configuration. To analyze the effect of plutonium enrichment and the presence of boron upon cross section condensation, the second condensation utilized the heterogeneous Phase 3 core configuration and reference control configuration (rods # 2, 4, 6, 8, 10, 12, 14, 16, and 18 inserted). For this condensation the sodium filled control rod positions were homogenized with the simulated control rods in zones 3 and 5 (i.e., sodium filled control rod positions 3, 5, and 7 were homogenized with simulated control rods 2, 4, and 6 in zone 3).

The process of generating the 29- and 6-energy group cross section libraries is illustrated by the simplified block diagram shown in Figure 2.12.

Upon completion of checkout of the non-sodium-voided cross section file the sodium-voided cross sections will be condensed and checked out. The 59-energy group master cross section library will then contain all of the cross sections required for the analysis of ZPPR Assembly 3.

2.6.2 Cross Section Data Testing

To test the updated 6-energy group ENDF/B-III cross section files (generated as described in Section 2.6.1 of this report), several two-dimensional (x,y) diffusion theory calculations were repeated. The computed results are compared with identical calculations utilizing the current cross section file and experimental control rod worths and summarized in Table 2.3.

Comparison of the calculated eigenvalues, power distributions and U-235 fission rate distributions for case 1 (Table 2.3) and case 5 indicates that the use of rod-in fuel cross sections to describe the fuel reaction rates when control rods are withdrawn has a negligible influence on the six group calculation. The reverse (rod-out fuel cross sections to describe fuel reaction rates when a control rod is inserted) comparison, with similar results, was reported in the Fourth Quarterly Report⁽⁵⁾. A number of other comparisons are possible where cross sections collapsed in one spectral environment are used to compute values in a different environment. The results of these comparisons indicate that the computed results are relatively insensitive to the flux spectrum used for cross section condensation i.e., Phase 1B or Phase 3 neutron flux spectrum. This

FIGURE 2.12

CROSS SECTION DATA PROCESSING BLOCK DIAGRAM

TABLE 2.3

CHECKOUT CALCULATIONS PERFORMED WITH UPDATED
GROSS SECTIONS FILES

Case	Phase	Description	Measured ^(B) Worth (\$)	Eigenvalue Current File	Corrected ^(C) Worth (\$) ^(A)	C/E	Eigenvalue Updated File	Corrected ^(C) Worth (\$) ^(A)	C/E
1	1B	No Control	-	0.99717	-	-	0.99293	-	-
2	1B	Control Rod #8 Inserted ^(D)	1.97	0.99104	1.79	0.91	0.98654	1.87	0.95
3	1B	Control Rods #2,4,6,8,10,12, 14,16, and 18 Inserted ^(E)	21.45	0.92826	20.35	0.95	0.92467	20.10	0.94
4	3	Control Rods #2,4,6,8,10,12, 14,16, and 18 Inserted	-	1.00539	-	-	0.99718	-	-
5	1B	No Control ^(F)	-	-	-	-	0.99291	-	-

(A) $\beta = 0.003147$ (B) Average of fission counter and BF_3 proportional counters outside radial and axial reflectors.

(C) Corrected for transport and number of energy groups.

(D) Control rod cross sections are from the Phase 3 condensation.

(E) Control rod and fuel cross sections are from the Phase 3 condensation.

(F) Fuel cross sections are from the Phase 3 condensation.

conclusion is limited to the spectral changes which occur during a fuel cycle in a Demonstration sized reactor i.e., spectral changes due to changes in plutonium and compensating control compositions.

One observes in Table 2.3, small changes in calculated control worth with the two cross section files. The computed U-235 fission distribution also agrees well. Figure 2.13 shows the U-235 fission distribution with the reference Phase 3 control insertion pattern. Values are computed with the updated cross section file and the current cross section file and compared against the experimental data. As observed in Figure 2.13, both computed distributions agree well with experimental values.

Perhaps the biggest difference observed with the two files is the change in computed k_{eff} between Phase 1B and Phase 3. The updated cross section file computes only a $+0.42\% \Delta k/k$ difference in absolute k_{eff} ($0.9929 \rightarrow 0.9972$) between Phase 1B and 3 while the current file yields a difference of $0.82\% \Delta k/k$, ($0.9972 \rightarrow 1.0054$). Since both conditions are known to be exactly critical, the differences noted above reflect an error in either the calculation of control worth or fuel spikes. As indicated above the updated file significantly reduces that error.

One of the current observations of these comparative calculations is the surprising sensitivity of the computed reaction rate distributions. The geometric assumptions involved computational modeling. This apparent sensitivity is currently being investigated.

FIGURE 2.13
COMPARISON OF PHASE 3 MEASURED AND CALCULATED (FOR CURRENT AND UPDATED
CROSS SECTION FILE) U-235 FISSION RATE DISTRIBUTION IN ROW 137 FOR THE
REFERENCE CONTROL CONFIGURATION.

3. SUBTASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

Subtask Leader: B. A. Hutchins

3.1 Scope and Objectives

Subtask B directly supports Subtask A by providing evaluated nuclear data and computational methods which are used in the pre- and post-analyses of the ZPPR-3 experiments. To provide this service, Subtask B work scope includes participation in 1) the cooperative activities of the Cross Section Evaluation Working Group; 2) nuclear data processing; 3) computer code evaluation, acquisition and development; and 4) support of the efforts to coordinate code development.

Subtask B objectives include the evaluation of nuclear data for one material for Version IV of ENDF/B and the participation in other activities of the Cross Section Evaluation Working Group in determining Benchmark problems and testing data, with emphasis on data for LMFBR shielding. Participation in the code coordination activities is continued and new computer code requirements for LMFBR design will be recommended, as well as the specifications reviewed. The 3-D linking code, TRIAL, is to be sent to the Argonne Code Center and a report describing the code and an evaluation of the method will be issued.

3.2 Nuclear Data for Pu-239

Preparation of the ENDF/B-IV nuclear data for Pu-239 is continuing, with the goal of having all data, with the exception of fission product and delayed neutron yields, to the National Neutron Cross Section Center by August 1, 1973. This schedule would allow for the preliminary distribution of the Version IV data by NNCSC in September. During the present reporting period, the following data preparation has been completed:

- 1) Statistical resonance parameters have been generated for the unresolved resonance range (described in more detail below).
- 2) The resolved resonance range will be extended up to 600 eV, by use of the parameters generated by Ribon⁽⁸⁾. However, it has not yet been decided if Ribon's parameters will be used between 1 and 300 eV or if the ENDF/B-III parameters⁽⁹⁾ will be used together with Ribon's spin assignments. R. Gwin, of ORNL, is making a comparison between the results from these two sets and recent ORNL measurements of the fission and capture cross sections.
- 3) Adjustments were made to the smooth (File 3) data between 600 eV and 1 MeV for purposes of a) providing consistency with measured Pu-239 to U-235 fission ratios using the final U-235 fission cross sections provided by H. Alter, and b) assuming continuity across the top of the unresolved resonance energy range. Adjustments for consistency with U-235 were only necessary between 40 and 90 keV, with the values at other energies remaining as reported in the previous quarterly report⁽²⁾. Table 3-1 lists the adjusted fission and capture cross sections. Smooth contributions to the cross sections in the unresolved resonance range were added only above 8.03 keV and are shown in Table 3-2. Aside from inelastic scattering, which is all pointwise data, very little smooth data were needed to assure a continuous transition at the upper end of the unresolved range.

The latest values from the NBS measurements of the total cross section were received from R. A. Schrack. They are only slightly

different from the CSISRS values, used to evaluate σ_t , and do not result in any changes in the cross sections for Version IV.

Data preparation activities for Pu-239 that must still be completed are listed in Table 3-3. While smooth cross sections for fission and capture had previously been evaluated to 8 MeV, L. Stewart has offered to provide a reevaluation of all data above 1 MeV based on the recent LASL evaluation⁽¹⁰⁾ but modified to be consistent with the U-235 fission cross sections used in ENDF/B-IV. This will result in a more complete reevaluation above 1 MeV, including scattering and threshold reactions, and in better agreement with the hard-spectrum benchmark experiments. There are no plans to modify the Pu-239 data in the thermal range (below 1 eV) even though agreement with thermal benchmark problems has not been satisfactory with ENDF/B-III. The one exception is a possible change in $\bar{v}_{th}$.

The statistical resonance parameters for Pu-239 were generated with the UR computer code at NNCSC, based on evaluated, high-resolution fission and capture cross sections from ORNL. The original plan for generating these parameters was to vary the s-wave strength function and the (0,1)-state average fission widths to achieve predetermined energy-dependent values for the fission, capture, and total cross sections, as was done for ENDF/B-III. While the fission and capture cross sections show much secondary structure, measurements of the total cross section are very sparse (essentially limited to those of Uttley) and not of high enough resolution to show similar secondary structure. This is a source of concern since the secondary structure will have some degree of self shielding that will not be included in the processing of ENDF/B data unless that structure is accurately included in the total cross section.

The procedure actually used to generate the unresolved parameters was to vary the same parameters, but only as needed to obtain the desired fission and capture cross sections, letting the total take whatever values that resulted. This procedure provided more structure in the total cross section than is possible to deduce from the Uttley data, but the averages are in excellent agreement with Uttley up to 3 keV and run about 0.5 barns low above that energy. This comparison is shown in Figure 3-1. The average parameters from this analysis will be used for ENDF/B-IV and are listed in Table 3-4.

Table 3.1
Adjusted Smooth Fission and Capture Cross Sections
for ENDF/B-IV Pu-239

<u>Energy (keV)</u>	$\sigma_f(b)$		$\sigma_c(b)$	
	<u>Previous</u>	<u>Adjusted</u>	<u>Previous</u>	<u>Adjusted</u>
40	1.604	1.604	.825	.825
43	1.603	1.60	.833	.834
45	1.604	1.59	.865	.860
47	1.605	1.57	.900	.866
48	1.606	1.56	.899	.870
50	1.609	1.57	.944	.880
53	1.615	1.625	.857	.860
55	1.618	1.64	.848	.860
57	1.619	1.630	.862	.867
57.3	1.619	1.628	.866	.870
58	1.619	1.623	.866	.870
60	1.618	1.618	.880	.880
63	1.614	1.614	.894	.894
65	1.610	1.610	.904	.904
67	1.605	1.610	.915	.915
70	1.598	1.618	.920	.930
75	1.585	1.67	.869	.910
76.4	1.581	1.62	.898	.910
80	1.568	1.573	.924	.926
85	1.556	1.556	.963	.963

Table 3.2

Smooth Data Contributions in the Unresolved Resonance Range

E (keV)	Resonance Contribution				Smooth Contribution					Total Cross Sections				
	σ_f	σ_c	σ_{es}	σ_t	σ_f	σ_c	σ_{es}	σ_{in}	σ_t	σ_f	σ_c	σ_{es}	σ_{in}	σ_t
0.6	5.35	8.65	14.62	28.62	0	0	0	0	0	5.35	8.65	14.62	0	28.62
8.03	2.15	1.29	11.72	15.16	0	0	0	0	0	2.15	1.29	11.72	0	15.16
8.5	2.16	1.11	11.52	14.79	0	0	0	.112	.112	2.16	1.11	11.52	.112	14.802
9.0	1.94	1.18	11.51	14.63	0	0	0	.19	.19	1.94	1.18	11.51	.19	14.82
9.5	1.90	1.14	11.45	14.49	0	0	0	.243	.243	1.90	1.14	11.45	.243	14.733
10.0	1.86	1.10	11.40	14.36	0	0	0	.2714	.2714	1.86	1.10	11.40	.2714	14.6314
12.5	1.79	.98	11.27	14.04	0	0	0	.2892	.2892	1.79	.98	11.27	.2896	14.3292
15.0	1.73	.865	11.15	13.745	0	0	0	.296	.296	1.73	.865	11.15	.296	14.047
17.5	1.70	.789	11.07	13.559	0	0	0	.3016	.3016	1.70	.789	11.07	.3016	13.8606
20.0	1.67	.714	11.00	13.384	0	0	0	.307	.307	1.67	.714	11.00	.307	13.691
22.5	1.65	.650	10.87	13.17	0	0	0	.3097	.3097	1.65	.650	10.87	.3097	13.4797
25.0	1.64	.585	10.75	12.975	+.0025	+.0025	.024	.313	.342	1.6425	.5875	10.774	.313	13.317
25.0	0	0	0	0	1.6425	.5875	10.774	.313	13.317	1.6425	.5875	10.774	.313	13.317

GEAP-13771-7

Table 3.3

Remaining Activities for Preparation of Pu-239
Data for ENDF/B-IV

<u>ACTIVITY</u>	<u>RESPONSIBLE INDIVIDUAL OR ORGANIZATION</u>	<u>TARGET DATE</u>
1. BCD Information and Data Uncertainties	B. A. Hutchins, G.E.	8/1/73
2. $\bar{\nu}$ (E)	B. A. Hutchins, G.E.	8/1/73
Delayed Neutron Yields & Spectra	H. Hummel, ANL	9/21/73
3. F. P. Yield Data	R. Schenter, HEDL	9/21/73
4. Resolved Resonance Parameters	R. Gwin, ORNL	7/1/73
5. Smooth Data to Supplement Resolved Resonances	R. Gwin, ORNL	8/1/73
6. Smooth Data for Continuity Across Boundaries	B. A. Hutchins, G.E.	8/1/73
7. Adjustment of Smooth Elastic Scattering <1MeV	NNCSC	8/15/73
8. Provide Evaluation above 1MeV	L. Stewart, LASL	8/1/73
9. Photon Production Cross Sections	R. LaBauve, LASL	7/15/73
0. Generation of New Material	NNCSC	9/1/73

FIGURE 3.1

COMPARISON OF σ_t FOR Pu-239 AS
MEASURED BY UTTLLEY AND COMPUTED
FROM THE UNRESOLVED RESONANCE PARAMETERS
FOR ENDF/B-IV

Table 3-4 also gives the average pointwise fission, capture and total cross sections that are computed with these parameters. Since these fission and capture values were used to generate the parameters, they also correspond to the cross sections that were evaluated from the ORNL differential data. For this evaluation, R. Gwin provided higher-resolution data in the energy range between 600 eV and 10 keV. These data were renormalized to produce the average cross sections that were previously evaluated for this range for the wider energy bins⁽²⁾. The 42 pointwise values in Table 3-4 were then chosen between 600 eV and 25 keV to provide the experimentally indicated structure in these two cross sections.

While this parameter analysis has resulted in achieving some structure in the total cross section for ENDF/B-IV, the actual structure could be even more marked. This points to a strong need for measurements over this range with energy resolutions at least as fine as 25 eV up to 1 keV and 250 eV above 1 keV.

The Pu-239 evaluation was described at the May 23-24 meeting of the Cross Section Evaluation Working Group.

3.3 Combined Multi-Group Structure

A request was made to LASL to modify the proposed 228-group structure to include 0.1 lethargy-width groups over the 3 keV sodium resonance. This modification was made and will allow for more accurate calculations of the spatial self-shielding of this resonance.

3.4 The 3-D Linking Code TRIAL

The topical report describing the TRIAL code and the evaluation of the 3-D linking procedure was issued.⁽¹²⁾

3.5 Unresolved Resonance Treatment in the MINX Code

A meeting was held with C. Weisbin, R. LaBauve and P. Soran, of LASL; R. Cullen, of LLL; and O. Ozer, of BNL, to arrive at a plan of action for incorporating unresolved resonance calculations into the MINX code by September, 1973. It was concluded that adaptation to MINX of the ETOX routines for unresolved resonance

Table 3.4

Unresolved Resonance Parameters for Pu-239

Energy (keV)	$r_n^o(0,0)$	$r_n^o(0,1)$	$r_f(0,1)$	σ_f	σ_c	σ_s
.600	9.492-4	3.368-4	6.503-3	5.35	8.65	14.62
.640	7.405-4	2.627-4	7.853-3	4.30	6.80	13.08
.725	5.798-4	2.057-4	2.154-2	4.05	4.62	11.90
.775	9.324-4	3.309-4	4.120-2	6.92	5.72	13.52
.825	6.528-4	2.316-4	8.858-2	5.86	3.51	11.77
.875	6.665-4	2.365-4	1.750-2	3.97	4.75	12.33
.925	8.739-4	3.101-4	4.515-2	6.10	4.80	13.04
.975	1.584-3	5.621-4	6.021-2	10.50	6.65	17.05
1.040	9.207-4	3.267-4	5.343-2	6.25	4.48	13.13
1.16	9.366-4	3.323-4	4.353-2	5.70	4.42	13.25
1.25	8.252-4	2.928-4	1.987-2	4.10	4.39	12.95
1.35	8.446-4	2.997-4	3.444-1	7.40	2.15	11.72
1.55	6.374-4	2.262-4	3.737-3	2.25	3.70	12.17
1.70	9.465-4	3.358-4	1.123-2	3.50	4.23	13.53
1.80	9.768-4	3.466-4	2.418-2	4.10	3.80	13.39
1.85	1.069-3	3.795-4	5.823-2	5.35	3.40	13.40
1.95	7.044-4	2.500-4	1.285-3	2.05	3.54	12.43
2.1	8.451-4	2.999-4	4.786-3	2.55	3.64	12.99
2.2	9.003-4	3.194-4	6.832-3	2.75	3.60	13.17
2.4	1.049-3	3.721-4	1.729-2	3.50	3.44	13.60
2.45	1.187-3	4.211-4	3.017-2	4.30	3.40	14.03
2.6	9.829-4	3.488-4	5.370-3	2.65	3.45	13.46
2.75	8.792-4	3.185-4	1.188-1	4.45	2.00	12.08
2.8	1.102-3	3.909-4	3.550-2	3.95	2.90	13.42
2.9	4.900-4	7.258-4	3.719-2	3.65	3.90	17.28
3.25	7.900-4	2.803-4	5.863-2	3.22	1.95	11.90
3.75	9.646-4	3.423-4	2.038-2	2.78	2.39	12.77
4.25	8.531-4	3.027-4	1.237-2	2.22	2.16	12.35
4.75	9.037-4	3.207-4	2.430-2	2.50	1.93	12.31
5.25	9.455-4	3.355-4	1.055-2	2.15	2.02	12.54
5.75	9.036-4	3.206-4	2.173-2	2.28	1.73	12.19
6.25	8.938-4	3.171-4	2.065-2	2.18	1.64	12.10
6.75	8.053-4	2.857-4	1.261-2	1.85	1.55	11.85
7.25	7.965-4	2.826-4	2.516-2	2.01	1.37	11.67
8.25	8.842-4	3.137-4	3.685-2	2.20	1.27	11.74
8.75	7.550-4	2.679-4	5.435-2	2.12	1.06	11.30
9.25	8.404-4	2.982-4	1.234-2	1.75	1.30	11.72
9.75	7.344-4	2.606-4	5.800-2	2.05	0.97	11.19
10.00	7.789-4	2.764-4	2.813-2	1.86	1.10	11.40
15.00	7.844-4	2.783-4	2.840-2	1.73	0.865	11.15
20.00	5.086-4	3.129-4	4.942-2	1.67	0.714	11.00
25.00	2.829-4	3.124-4	8.080-2	1.64	0.585	10.75

calculations was the only approach that could be completed on schedule. The lack of a same-sequence overlap correction in ETOX is of concern, but the effects of this omission could not be detected in comparisons of benchmark calculations. It was agreed that a more state-of-the-art treatment, such as that in the SDX code, is required for the subsequent version of the MINX code.

3.6 ACRP Activities

A General Electric representative attended the May 16-18 meeting of the Advisory Committee on Reactor Physics. Reports were made at this meeting on the planning and analysis of the ZPPR-3 experiments, on the activities of the ANS Standards Committee 19, Physics of Reactor Design, and on LMFBF computer usage at General Electric.

4. QUALITY ASSURANCE

SUBTASK A - CRITICAL EXPERIMENT PLANNING AND ANALYSIS

Quality Assurance efforts are primarily concentrated on obtaining the best cross section data available for analysis and comparing analytical results with Argonne National Laboratory (ZPPR) analytical efforts. Cross sections used in analysis will be based on ENDF/B-III data, which is the current industry standard. Analytical results are compared with similar efforts by the ZPPR staff. A system of identifying cross section data used in ZPPR Assembly 3 analysis is in effect to reduce the possibility of inadvertently using incorrect data. A minimum of two levels of General Electric-Breeder Reactor Department management review all formal publications, consist with General Electric-Breeder Reactor Department procedures.

SUBTASK B - COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT

The cooperative nature of the Subtask B activities provides for a test of quality within themselves. Nuclear data evaluation and testing is performed as part of the Cross Section Evaluation Working Group Program. This working group has established procedures for reviewing and comparing results obtained from all the participating organizations, in order to determine the quality and adequacy of the data and data processing methods used by these organizations. In a like manner, neutronics methods development is a cooperative program, requiring reviews of specifications and comparison of results. Consequently, reporting of progress generally includes a comparison of results with those reported by other organizations. Computer codes developed under Subtask B conform to the standard interfaces and programming practices specified by the Committee on Code Coordination.

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12. C. L. Cowan, P. Greebler, B. A. Hutchins, M. D. Kelley, R. Protsik, "A Description and Evaluation of the Flux Linking Approach for Computing Three-Dimensional Fluxes - The Computer Code TRIAL", General Electric Company, GEAP-13931, March 1973.

APPENDIX A. PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

- ° Monthly Progress Letters, Nos. 1 through 15 from October 1971 through May 1973.
- ° The quarterly reports under the title "Critical Experiments and Cooperative Evaluations Program" follow:

GEAP-13771-1
First Quarterly Report
October-December 1971

GEAP-13771-2
Second Quarterly Report
January-March 1972

GEAP-13771-3
Third Quarterly Report
April-June 1972

GEAP-13771-4
Fourth Quarterly Report
July-September 1972

GEAP-13771-5
Fifth Quarterly Report
October-December, 1972

GEAP-13771-6
Sixth Quarterly Report
January-March, 1973

- ° In addition, the following topical reports have been issued:

GEAP-13850
Specification of ZPPR Assembly 3,
Phase 1 Experimental Program;
H. S. Bailey and S. L. Stewart
(October 1972)

GEAP-13931
A Description and Evaluation of the Flux Linking Approach
for Computing Three-Dimensional Fluxes - The Computer
Code TRIAL (March, 1973)

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