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ANALYSIS OF FTR PHASE B
CRITICAL EXPERIMENTS
PART-1

ZPR-III ASSEMBLY 51

January 1970

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DEVELOPMENT REPORT

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PART-1

ZPR-III ASSEMBLY 51

W. R. Young
R. A. Bennett

Reactor and Plant Technology Department
FFTF Division

January 1970

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ANALYSIS OF FTR PHASE B CRITICAL EXPERIMENTS

PART I

ZPR-III ASSEMBLY 51

W. R. Young and R. A. Bennett

ABSTRACT

Critical experiments in support of the design of the Fast Flux Test Reactor have been carried out in ZPR-III Assemblies 48, 48A, 51, and 52a, b, c, d, e, and f, by personnel of Argonne National Laboratory. This report presents the results and analysis of the experiments performed in Assembly 51.

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ANALYSIS OF FTR PHASE B CRITICAL EXPERIMENTS

PART I

ZPR ASSEMBLY 51

W. R. Young and R. A. Bennett

INTRODUCTION

The purposes of the Fast Flux Test Facility (FFTF) Critical Experiments Programs^(1,2) are to provide experimental data for evaluation of calculational methods being used in the design of the Fast Test Reactor (FTR), and to provide experimental verification of the design of the FTR. The program has been divided into phases A, B, and C. Phases A and B are providing verification of calculational models and some basic design data. Phase C will be an engineering mockup study. Phase A consisted of a study of control rod configurations in a plutonium fueled nickel-sodium reflected system.^(3,4,5)

This report presents the results of an analysis of the first series of Phase B experiments. The experiments were performed by Argonne National Laboratory and Battelle-Northwest personnel in ZPR III, Assembly 51.

The goal of this report is to provide experiment-theory correlations indicative of uncertainties in the calculational models used in the FTR design. Where possible, interpretation of disagreements are presented and the impact on FTR design and future understanding of operation are noted.

Several sets of experiments were performed, including critical mass and edge-drawer reactivity worths, spatial distributions of reaction rates, and small sample reactivity worths, distributed sodium voiding reactivity worth, large sample core reactivity worths, fuel compaction and fuel movement reactivity worths, fission rate ratios, spectrum measurements, and small sample Doppler reactivity effects. In view of the diversified nature of this series of experiments, each item

will be discussed successively in this report except for the Doppler experiments. The latter will be discussed in a separate report to follow.

SUMMARY

The conclusions reached in the analysis of each experiment are summarized here for convenience in their application to the assessment of uncertainties in FTR design parameters.

These results are, of course, based upon experiments performed in a system approximately one-third the size of the reference FTR. They must, therefore, be used with appropriate reservation until experimental results from larger systems are available. Such experiments are planned for early FY-70.

MULTIPLICATION AND CRITICAL MASS

ZPR III, Assembly 51, was a completely reflected, slightly irregular, cylindrical reactor with a core volume of 307 liters and a critical mass of $212 \text{ kg} \pm 1.1 \text{ kg}$ of fissile material. The fuel was uranium-plutonium with a ^{238}U to ^{239}Pu atom ratio of 4.08:1. Diluents were primarily sodium, steel, and oxygen. The calculated multiplication constant of the as-built system, obtained from a transport-corrected, diffusion-theory, two-dimensional (X-Y) 8-energy group analysis, was 0.987. The convergence of the calculation was 10^{-5} in k . This -1.3% $\Delta k/k$ under-estimate of the effective multiplication constant of Assembly 51, if extrapolated, would indicate present FTR design calculations to overestimate critical mass by 16 kg by overestimating reactor size of fixed enrichments, or to overestimate critical mass by 6 kg by overestimating enrichments for fixed size. The latter is, of course, the expected realistic condition since reactor geometry will be fixed. Such discrepancies, even though uncorrected, are not expected to be particularly disconcerting for the reasons that (1) the FTR is enrichment-zoned and the zone interface may be adjusted, as

necessary, and (2) core lifetime is relatively short and enrichment can be adjusted for subsequent loading. The error in k is consistent with our⁽⁶⁾ recent experience in similar experiment-theory correlations on blanketed systems where differences in experimental and calculated values ranged from -2.4 to 0.6% $\Delta k/k$. Finally, no attempt has been made here to establish the source of this discrepancy with regard to, say, adjustment of cross sections. Such efforts are to be deferred to a time when statistical consistency is established for a number of reflected plutonium reactors.

REACTION RATE DISTRIBUTION

In general, the shapes of the spatial distribution of ^{239}Pu and ^{238}U fissions and the (n,α) reaction in ^{10}B are calculated well within the reactor core. The agreement between experiment and theory begins to deteriorate near the core-reflector boundaries and, in the reflectors, the agreement is inconsistent amongst the three different sensors. Differences as large as 100% were found. The problem results primarily from the rapid softening of the spectrum in the reflector and near the core-reflector interface.

The peak to volume averaged ^{239}Pu fission rate was measured as 1.40 and the calculated value is 1.43. One can, therefore, expect power distributions to be calculable within the core except near the core reflector interface where calculated reaction rates are not reliable. Peripheral control rod burnup may therefore be difficult to compute accurately.

MATERIAL WORTHS

Experiment-theory correlation on central reactivity worths of small samples of ^{239}Pu and ^{238}U are poor. The c/e values, 1.33 ± 0.015 and 1.53 ± 0.02 , respectively, are consistent with the experience reported by others.⁽⁷⁾ This discrepancy has been explored and may be due to the data cross section set or

to discrepancies in β . The significance of the discrepancy is important in analysis of the Doppler experiment where one is actually concerned with small local fuel changes. One is often interested in small density changes over extended regions, e.g., varying fissile density in tests, and in large volume fuel voiding such as whole fuel elements, or bulk core fuel slumping. In the latter, where ^{239}Pu and ^{238}U were removed simultaneously, c/e values were found to be 1.15 - 1.18. The reduction in error is due to cancellation in over-calculating the ^{239}Pu worth positively and over-calculating the absolute value of the negative ^{238}U worth. For ^{10}B , the c/e values for central reactivity worth of small samples were unity.

Spatial distributions for worths of small samples of ^{10}B are calculated well. The poorest agreement occurs at the core reflector interface. In general, the shapes for ^{239}Pu and ^{238}U are also very good but, of course, the absolute magnitude remains poor.

DISTRIBUTED WORTH OF SODIUM

Voiding of 10% of the sodium from large regions of the core is calculated to within 5 to 10%. However, calculations appear very poor for reactivity effects of voiding 30% of the sodium from smaller central regions where the coefficient has a strong positive component and is primarily spectrum dependent. Reactivity calculations are more positive in the range of 30 to 240%.

Should this disagreement prevail in the larger, more FTR-like experiments where the zone of positive sodium worth is thought to be larger, serious uncertainties in the safety analysis could result.

CENTRAL CONTROL REACTIVITY WORTHS

Reactivity worths of central control rods of B_4C and Ta were found to have c/e values of 0.92 ± 0.02 and 0.94 ± 0.02 , respectively.

The c/e values obtained in analyses of central natural B_4C rods worth measurements performed in ZPR-III, Assembly 48A ranged up to 1.05. One estimates, therefore, that a minimum uncertainty in control rod calculation is $\pm 10\%$.

NEUTRON SPECTRA

The general shapes of the neutron spectra at four assembly locations ranging radially from the core center into the reflector were well represented by calculations. The fine detail arising from scattering resonances of assembly materials were not, and cannot be, reproduced with a 26-group cross section set. Subtle reactivity effects dependent upon these fine spectral details may, with the spectrum properly integrated, prove calculable with the coarse structure.

DESCRIPTION OF ZERO POWER REACTOR III (ZPR-III)

The facilities of the ZPR-III at the National Reactor Testing Stations, Idaho Falls, Idaho are owned by the United States government and operated, under contract with the U.S. Atomic Energy Commission, by Argonne Laboratories. Detailed descriptions of the facility have been published.^(8,9)

Briefly, the ZPR-III can be described as a split table critical facility. Reactor materials typically 1/8 to 1/4-in. thick are loaded as flat plates. The plates are placed in long drawers that fit into horizontal matrix tubes. One matrix cell is made up of a matrix tube, the drawer, and the drawer contents. A matrix cell is, on the average, 2.182 in. high by 2.178 in. wide.

A small air gap always remains between the planes of stainless steel formed from the mating of the drawer front wall in each assembly when the halves of the assembly are brought together. Drawer walls are 0.032 in. thick and are perforated to reduce the minimum amount of stainless steel in ZPR-III assemblies. The minimum separation of fuel at the axial mid-plane is equivalent to a two-drawer wall thickness or 0.064 inches. An uncertainty of ± 0.005 in. in the reference axial drawer position, a ± 0.004 -in. uncertainty in the angular tolerance of each drawer front, and a ± 0.004 -in. uncertainty in the angular tolerance of each mating fuel piece must be added to the minimum separation. These uncertainties apply to fuel piece locations in each half of the assembly. The total gap then will be estimated as 0.077 ± 0.013 .

Figure 1 is a picture of the ZPR-III. Assembly 51 was loaded when the picture was taken. The three dark matrix cells in the right-half are empty safety and control drawer positions. The drives for the safety and control drawers are apparent on the outside of the left half.

FIGURE 1. ZPR-III Loaded with Assembly 51

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DESCRIPTION OF ZPR-III ASSEMBLY 51⁽⁴⁾

Assembly 51 was a 307 $\times$ irregular right cylinder. Figure 2 shows a cross section transverse to its cylindrical axis. Each matrix cell of the figure represents a ZPR-III matrix tube, a drawer, and its contents.

The core height was 86.51 cm and a volume equivalent core would have a radius of 33.647 cm. The axial and radial reflectors were nominally 30.48 cm thick.

The core composition was approximately 36 vol% fuel, 36 vol% sodium, and 25 vol% stainless steel. The radial reflector was composed of 16 vol% sodium, 12 vol% stainless steel, and 61 vol% nickel. The axial reflector had approximately 41 vol% sodium, 17 vol% stainless steel, and 31 vol% nickel.

MATERIAL DENSITIES

Core

Drawer types designated A and A* make up the bulk of the core drawers. The average isotopic composition of an A and an A* drawer approximated that of the driver zone of a split conical version⁽¹⁰⁾ of the FTR. Figure 3 shows how the A and A* core portion of the drawers were loaded with platelet materials.

All eight safety drawers were spiked with additional plutonium to make their total worth exceed 1.5% $\Delta k/k$. Each safety rod was spiked with fuel removed from the core drawers immediately above and below it. Nonfissile material, removed from safety drawer positions to make room for additional fuel, replaced vacated fuel positions of the donor drawers. This method of spiking minimized perturbations to the critical mass and did not change the material inventory of the region.

FIGURE 2. ZPR-III Assembly 51

FIGURE 3. Core Drawer Loading for ZPR-III Assembly 51

Safety drawer pairs in each half of the assembly were in matrix locations N13, R13, N19, R19. Figure 2 shows the location of the safety drawers, (S), and the fuel depleted drawers immediately above (0) and below, (B) each safety drawer. Table 1 lists the drawer fissile inventory of the critical core and Table 2 itemizes the average isotopic composition of each zone of the assembly.

TABLE 1. Core Loading ZPR-III Assembly 51⁽⁴⁾

	<u>Identity in Figure 2</u>	<u>Number of Drawer Pairs</u>	<u>Fissile Material, kg</u>
Core Loading Fuel Drawers			
A	Even No. Column	55	114.905
A*	Odd No. Column	48	76.606
Control Rod	C	1	2.109
Safety Rod	S	4	12.718
Above Safety Rod	0	4	3.848
Below Safety Rod	B	<u>4</u>	<u>2.564</u>
TOTALS		116	212.75

$k = 1.000566$ (Measured)^(a)

Core Volume = 307.5 l

(a) When the control rod was 4.585 in. withdrawn, the reactor was 8.2 Ih subcritical. Complete insertion of the control rod resulted in an excess reactivity of 58.5 Ih.

Radial Reflector

The radial reflector had an inner nickel-sodium-stainless steel zone resembling the FTR reflector and an outer zone consisting mostly of iron. The iron region, substituted because of the lack of an adequate nickel inventory, made the combined zones nearly equivalent to a single 30 cm thick nickel-sodium-stainless

TABLE 2. Assembly 51 - Average Compositions⁽⁴⁾ and Densities
[atom density ($\times 10^{-24}$ cm³)]

DRAWER TYPE	Pu-239 -241	Pu-240 -242	U-235	U-238	Mo	Na	C	O	Al	Fe	Cr	Ni	Mn	Si
Core Type A	0.001972	0.000170	0.000012	0.005703	0.000224	0.010262	0.003113	0.011479	0.000110	0.01521	0.003783	0.001655	0.000158	0.000185
Core Type A*	0.001498	0.000175	0.000018	0.008506	0.000431	0.008244	0.003113	0.014093	---	0.01600	0.003530	0.001544	0.000147	0.000173
Control Rod	0.001990	0.000162	0.000015	0.007144	0.000197	0.010123	0.002746	0.008241	0.000124	0.016047	0.003992	0.001747	0.000167	0.000196
Safety Rod	0.002990	0.000329	0.000032	0.014635	0.000796	0.004039	0.000916	0.005362	0.000026	0.017198	0.003829	0.001675	0.000160	0.000188
Drawer above) Safety Rod)	0.000902	0.000119	0.000012	0.005705	0.000224	0.010340	0.004150	0.017212	---	0.016122	0.003561	0.001558	0.000149	0.000175
Drawer below) Safety Rod)	0.000597	0.000056	0.000013	0.005829	0.000208	0.010340	0.004150	0.017212	---	0.015803	0.003482	0.001524	0.000145	0.000171
Core Average	0.001727	0.000173	0.000015	0.007049	0.000331	0.009218	0.003113	0.01283	0.000053	0.01562	0.003652	0.001598	0.000152	0.000178
Radial Reflector I	---	---	---	---	---	0.00416	---	---	---	0.007605	0.00187	0.0564	---	0.000090
Radial Reflector II	---	---	---	---	---	---	---	---	---	0.07485	0.00115	0.00049	---	0.000060
Axial Reflector	---	---	---	---	---	0.010321	---	---	---	0.010426	0.002583	0.028926	0.000182	0.000126

steel reflector.⁽¹¹⁾ The iron region was eventually replaced during the transition to the next series of assemblies⁽¹²⁾ at which time adequate nickel was on hand.

The configuration of the radial reflector may be found in Figures 2 and 4. Compositions may be found in Table 2.

Axial Reflectors

The axial reflectors of Assembly 51 had a nominal thickness of 30 cm and a composition simulating that of the expected FTR reflector. The configuration and composition of this region can also be found in Figures 2 and 4 and Table 2.

FIGURE 4. Volume Equivalent Critical Right

EXPERIMENTAL RESULTS AND ANALYSES

Detailed descriptions of experiments and their analyses are presented in this section. In general, each discussion is self-contained except where reference to ANL experiment descriptions are made for convenience.

Both diffusion and transport theory were used in the analyses reported. The specific computer codes used were:

FCC-IV	Reference (13)
DTF IV	Reference (14)
2DF	Reference (15)
2DB	Reference (16)
PERT-IV	Reference (17)

For most calculations of effective multiplication constants, a convergence criterion of 10^{-5} yielding an eigenvalue converged to within $\pm 5 \times 10^{-5} \Delta k/k$ was specified. For sodium void worth, a convergence of 10^{-6} yielding eigenvalues converged to approximately $\pm 5 \times 10^{-6} \Delta k/k$ or the order of ± 2 lh* was specified.

Mesh points were 2 to 4 cm apart in interior core regions, and 4 to 5 cm apart in the reflectors. Near zone boundaries, they were separated by less than 1 cm. Typical mesh structures for a two-dimensional calculation contained between 500 and 900 spatial mesh points.

CRITICAL MASS

Critical mass, even with the best techniques, probably cannot be consistently precalculated closer than several % $\Delta M/M$ to the critical mass because assembly composition, material distribution, and safety rod spiking are generally not known until the assembly is complete. Corrections to predictions must be made after the loading details are known. Assembly 51

was precalculated in September 1967⁽¹⁸⁾ to have a critical mass of 220 kg of *1% $\Delta k/k = 1036 \text{ Ih}^{(4)}$ of fissile material. The assembly went critical in December 1967 with 212 kg fissile mass. The prediction was 4% high in $\Delta M/M$.

The effects of a stepped core edge, safety drawers, air gap, and exact A-A* drawer configuration were omitted from the precalculation. The needed corrections, except for exact concentration, are listed in the next paragraph. The concentration of ^{239}Pu was assumed to be 0.001736 atom/b-cm. Compared with the as-built concentration of 0.001727, this concentration corresponds to a precalculated increase of about 0.3% $\Delta k/k$ or a 1.3 kg decrease in fissile mass.

Cylindricizing Assembly 51

The slightly supercritical system shown in Figure 2 and described in Table 2 is the reference configuration for all Assembly 51 experiments and is formally designated ZPR-III Assembly 51-28.⁽⁴⁾ As shown, it has irregular core and reflector boundaries, spiked safety rods and depleted adjacent core drawers, a two-drawer unit cell of dissimilar composition, a central air gap between halves, and an outer reflector of iron rather than Ni-Na. Corrections to cylindricize and homogenize this system are tabulated as follows:

	<u>K_{eff}</u>	<u>Mass, kg</u>
Assembly condition as in Figures 2 & 4	1.00057	212.75
Corrections:		
Smooth circular boundary, R_{core} = 33.861 cm.	1.00439	215.47
Safety rod & adjacent drawer homogenization	1.00539	
Conversion of Fe reflector to Ni-Na	1.00769	
Air gap	1.00939	
Effect of dissimilar drawers in cell, A-A*	1.01059	
Reduction to unit multiplication	1.0000	202.97

Each corner in the irregular core was either truncated or filled-in to cylindricize the perimeter. Examination of the outline of Assembly 51-28 shows that added core material, on the average is closer to the core center than material that has been removed. Hence, the net effect of cylindricization represents an increase in k and, coincidentally, an increase in mass. Core edge worths were taken from the measured values⁽⁴⁾ and plotted against radius for A and A* drawers (Figure 5.) The materials added or subtracted were given worths corresponding to the radius coincident with the centroid of their volume.

Material subtracted evenly from the edge was assumed to be worth the average of A and A* drawer material at the core radius.

The 2D RZ cylindrical calculations with the diffusion code 2DB gave a critical mass of 228.36 kg fissile. The 2DF RZ S₄ transport calculation for this core gave a calculated eigenvalue of 1.0091. Correcting for core size results in an inferred transport correction to the cylindricized 203 kg core of 1% $\Delta k/k$. The correction is inferred from a recent study showing the transport correction for a 203 kg core to be 0.1% $\Delta k/k$ larger than for a 228 kg core.⁽¹⁹⁾

The calculated eigenvalues were reduced to correspond to the mass of 203 kg using an edge worth of 83 lh/kg. This value is the measured average worth of fissile material removed.⁽⁴⁾ The RZ calculational model is shown in Appendix A. All concentrations in the RZ models were from the average zone compositions.

The method of safety rod spiking, described in Material Densities, did not change the material content of the region of the rods. The calculation of the reactivity effect of the spiking was done with a 2 DF two-dimensional S₄ transport

FIGURE 5. Measured Core Edge Drawer Worths

analysis. The bunching effect is small, 0.10% $\Delta k/k$, and reduces the reactivity of the assembly. The positive effect of lumping in the safety rods is probably offset by the flux peaking in O and B drawers of Figure 2.

The effect of an alternating A, A* distribution, where A drawers are in even numbered column and A* are in the odd numbered ones, was calculated with the code 2DB. The calculations enabled comparison of alternating A-A* columns with a uniform mixture of A and A* in an X-Y geometry. The value of k was increased by 0.12% $\Delta k/k$ in going from the alternate distribution of 51 to a uniform distribution.

With the gap assumed to be 0.077 in., the gap volume would be approximately 0.695 liter. The worth of the air-stainless steel gap in $\Delta k/k$ is deduced from the inferred volume average worth of ^{239}Pu at the midplane to be 0.00365 $\Delta k/k/\text{kg}$, and the mass of fissile material normally in the gap to be 0.47 kg. The result is $\Delta k/k = 0.0017$.

Finally, the outer radial reflector of iron was replaced with a Ni-Na reflector of the same nominal composition as the inner radial reflector and $\Delta k/k$ was measured to be 0.0023.⁽¹¹⁾

A 2DB X-Y calculation was made of Assembly 51-28. Here the details of the radial structure were included. The X-Y calculational model is shown in Appendix B.

The effective axial buckling was inferred from the R-Z calculation and a 2 DB 1D cylindrical model possessing the exact composition and radial mesh of the 2D model. The buckling was found to be 6.82 m^{-2} .

Cross Sections

All calculations were based upon a 26-group cross section set. Sets having fewer energy groups were collapsed from the

26 group set by averaging over a calculated spectrum. These 26-group cross sections, a Battelle-Northwest set, are a modified version of the Bondarenko set.^(20,21) They are read from the data tape by the computer code, FCC IV,⁽¹²⁾ which also computes the resonance shielding factor, punches out cards and, if desired, collapses them to a few-group set.

Core cross sections have been weighted for heterogeneities by cell calculations, while reflector cross sections are used unaltered from a homogeneous fundamental mode calculation.

The ZPR-III-51 loadings were made up of flat plates or cans, as in the case of sodium, 1/8 to 1/4 in. in thickness. The plates were so loaded into stainless steel drawers that the long dimensions of the plates and drawers were parallel with the core axis. Heterogeneous effects arising because of the finite size of the plates yield a difference calculated to be 0.74% in k between a homogeneous model and a more realistic model in which the plates are simulated. Table 3 gives data on the sources of the 0.74% difference in k as calculated with a 2DB RZ, 8-group, homogeneous model, and a 2DB RZ, 8-group heterogeneous model. The largest contributors to errors in a homogeneous model appear to result from an excessive capture to fission ratio for fuel materials and excessive leakage, largely because of a harder spectrum in the fuel of the heterogeneous model.

TABLE 3. Heterogeneity Corrections for Core Material for Assembly 51

Reactions	Homogeneous to Heterogeneous: Correction in $\Delta k/k$
Increased Pu and U utilization	+0.0006
Decreased absorptions in SS and sodium	+0.0001
Decrease in α_{40} , α_{40} , α_{28} due to spectrum shift in fuel	+0.0054
Decrease in leakage	<u>+0.0013</u>
Total	+0.0074

Heterogeneity effects were incorporated into the calculational model through flux-volume weighted cross sections. The flat plates of reactor materials were approximated by infinite planes in an S 12 one-dimensional, 26-group DTF-IV⁽¹⁴⁾ transport calculation. Periodic boundary conditions were used at the cell boundary. The calculational model of the A-A* drawer pair is shown in Figure 6. Concentrations in each region of the cell are listed in Table 4. Sodium, stainless steel, and other diluents were lumped together in five regions which correspond to similar regions in the actual loading. In a like manner, fuel was lumped into four regions. The volume average concentrations of the isotopes in Table 4 correspond to the concentrations of Table 2. The exceptions, Mo and Si, were not included in the cell calculation because of lack of data at that time. They were added to the mix later.

The cross sections were averaged over the cell flux and volume using

$$\bar{\phi} \bar{\sigma} V_{\text{cell}} = \int_{\text{cell}} \phi(X) \sigma(X) dv$$

where $\bar{\phi}$ is the average flux in the cell, V_{cell} is the cell volume, and $\bar{\sigma}$ is the weighted cross section. The average transport cross section, $\bar{\Sigma}_{\text{tr}}$ is inversely weighted to make the leakage from the cell $B^2 \bar{\phi} / 3 \bar{\Sigma}_{\text{tr}}$ equal to

$$V_{\text{cell}} B^2 \bar{\phi} / 3 \bar{\Sigma}_{\text{tr}} = B^2 \int_{\text{cell}} \frac{\phi(X)}{3 \bar{\Sigma}_{\text{tr}}(X)} dV.$$

When cell 26-group cross sections were collapsed to fewer groups, they were collapsed as a mix. Inverse linear weighting was used for transport cross sections.

FIGURE 6. Unit Cell for Heterogeneity

TABLE 4. Isotopic Concentrations in Model Unit Cell

<u>Region</u>	<u>Thickness, cm</u>	<u>Mix</u>	<u>Isotope</u>	<u>Density,</u> <u>10^{24} atom/cm³</u>	
1	0.723	1	C	0.004545	Coolant, clad and structure
			Fe	0.02278	
			Cr	0.005339	
			Ni	0.002336	
			Mo	0.0004782	
			Al	0.0008073	
			Na	0.013514	
			O	0.01867	
2	0.9525	2	Pu-239 + 241	0.0037544	ZPPR + U-238
			Pu-240	0.0005562	
			U-235	0.00002929	
			U-238	0.01537	
3	2.5307	1			
4	0.6338	3	Pu-239 + 241	0.012735	Pu + U-238 (metal)
			Pu-240	0.0006769	
			U-235	0.00004281	
			U-238	0.003773	
5	1.447	1			
6	0.953	4	Pu-239 + 241	0.0042037	SEFOR + U-238
			Pu-240	0.0004416	
			U-235	0.00009202	
			U-238	0.04826	
7	2.16	1			
8	0.9615	2			
9	0.7225	1			

Results

The calculational and experimental results are shown in Table 5. There is a small difference in the X-Y analysis and the RZ analysis, but they appear consistent. The differences are indicative of the problems in cylindricization of the core, and the X-Y model is more representative of the calculational accuracy of our codes and data set.

TABLE 5. Critical Calculations, Assembly 51

	<u>Mass, kg</u>	<u>K_{eff}</u>
Measured Assembly 51-28	212 ± 1.1	1.000
Transport Corrected X-Y 2D Diffusion Calculation	212	0.9842
Cylindricized Assembly 51-28, Inferred from measured edgeworths	203 ± 1.1	1.000
Transport Corrected R-Z2D Diffusion Calculation	203	0.9878

REACTION RATE TRAVERSES

Radial and axial reaction rate distributions of ^{239}Pu (n,f), ^{238}U (n,f), and ^{10}B (n, α) were measured with gas filled proportional counters.⁽²²⁾ Measured and calculated results are shown in Figures 7 through 12. Radial traverses were through the P row 1.25 in. from the midplane in Half 1, and axial traverses were along matrix position P-16. The chambers were secured to a stainless steel sled made of a 23-in. length of thin walled tubing that moved through a traverse tube. The traverse tubing was built into the specially loaded drawers illustrated in Figures 13 and 14. These drawers are shown for illustration only and are not drawn to scale. The lengths of the sensitive volumes of the chambers were: ^{239}Pu fission chambers - 1 in.; ^{238}U fission chamber - 1 in.; and $^{10}\text{B-F}_3$ chamber - 2 in. Each traverse was performed twice (fuel normal and fuel rotated). Fuel was rotated to study the effect of neighboring fuel orientation on reaction rates.

FIGURE 7. Radial ^{239}Pu Fission Rate Distribution

FIGURE 8. Axial ^{239}Pu Fission Rate Distribution

FIGURE 9. Radial ^{238}U Fission Rate Distribution

FIGURE 10. Axial ^{238}U Fission Rate Distribution

FIGURE 11. Radial ^{10}B (n, α) Rate Distribution

FIGURE 12. Axial ^{10}B (n, α) Reaction Rate Distribution

0"	2"	17"	21"
		Na_2CO_3	Na
		U_3O_8	Na
		ZPPR FUEL	Ni
		Na_2CO_3	Na
		Na	Na
		Na_2CO_3	Ni
		SS	Na
		U-238	Ni
		Pu METAL	Na
		Na	Ni

A DRAWER

0"	2"	17"	21"
		Na_2CO_3	Na
		U-238	Na
		SEFOR FUEL	Ni
		Na_2CO_3	Na
		Na	Na
		Na_2CO_3	Ni
		U_3O_8	Na
		ZPPR FUEL	Na
		Na	Ni

A* DRAWER

FIGURE 13. Drawer Loadings for Radial Traverses

90° ROTATED CONFIGURATION

NORMAL CONFIGURATION

FIGURE 14. Drawer Loadings for Axial Traverses

Rotation of core drawer plates for radial traverses was done in the Half II, P row. All P row core plates in Half II were rotated 90° about a line parallel to the cylinder axis. These plates were then lying horizontal in the drawer rather than vertical. The rotation of core plates for axial traverses is shown in Figure 15.

Axial reaction rates were calculated in 26 energy groups with DTF-IV using an S4 angular quadrature and slab geometry. Radial leakage was calculated from DB^2 , using a radial buckling that made the assembly critical. Input cross sections were the same as those previously mentioned. The slab had a core region of 43.255-cm half-height, and a 30 cm thick axial reflector. Atom densities for these regions are given in Table 2.

Radial reaction rates for ^{239}Pu , ^{238}U , and ^{10}B , were obtained from an 8-energy group, 2D, X-Y, 2DB, diffusion calculation. Axial leakage rates were approximated using an effective buckling of 0.000682 cm^{-2} , the axial buckling for 1D infinite cylinder diffusion calculation, to give the same eigenvalue as the 2D, RZ, calculation. The X-Y model is shown on the data sheet for this calculation in Appendix B-2. Reactor cross sections are the same as used for the 8-group critical mass calculation. The X-Y model is slightly different than Figure 2, the loading for Assembly 51. In particular, the core loading in matrix positions O-10, Q-10, and P-10 are different. The O-10 and P-10 drawers have been slipped down one-half matrix cell height (2.8 cm) in order to achieve one-fourth core symmetry for the calculation. The calculated activities are along a line 1.383 cm above the center of the P row. Differences between the calculational model and the experimental geometry were investigated and found to cause a maximum calculational underestimation of the plutonium fission rate in P-10 of about 2.2%.

FIGURE 15. Radial ^{239}Pu Worth Distributions

No calculated correction was made for the finite chamber size, but the count rates were plotted for a location corresponding to the center of the sensitive volume of the chamber.

^{239}Pu Fission

The shape of the radial fission rate distribution of ^{239}Pu is calculated well in the core but begins to fail near the core reflector boundary. The difficulties in reaction rate calculations at the interface arise from the resonance self-shielding factors of the activation cross sections and the spectral discontinuity that results from collapsing cross sections for adjacent regions over greatly different spectra. Figures 7 and 8, the ^{239}Pu reaction rate distributions, show a reference calculated curve where the resonance self-shielding factors for the ^{239}Pu fission cross sections were calculated for a core composition. Another curve labeled "reflector cross sections" was calculated with infinite dilution, reflector-averaged cross sections. Standard deviations from statistical fluctuations of measured count rates are between ± 0.3 and $\pm 0.5\%$ for the radial, and less than $\pm 0.3\%$ for the axial traverses.

Calculated fission rate distributions are low in both the axial and the radial reflectors, even when ^{239}Pu shielding factors calculated for an infinite dilution are used. A comparison of Figures 7 and 8 shows that the discrepancy, $\sim 10\%$, is about the same in the axial as in the radial reflector. The axial reflector has about 41 vol% sodium as compared with 17 vol% in the radial reflectors.

A 2DB, R-Z, 8-group calculation and a 26-group, 1D slab, DTF-IV calculation were made for comparison with the measured ^{239}Pu (n,f) axial distributions. These results are plotted in Figure 8.

R-Z calculations of the axial ^{239}Pu (n,f) distributions are in better agreement with measurement than the 1D slab calculation. Slab calculations depend upon DB^2 representation of the radial leakage in which the B^2 is an average buckling that gives the eigenvalue of the R-Z calculation. This selection of B^2 is not a strong function of the volumetric leakage in the reflector because of the relatively low importance of this region. The 2DB calculations show that the average geometric buckling of the core-reflector will be smaller than the core buckling and larger than the reflector buckling. A large DB^2 in the reflector will depress the flux there and could cause the low calculated ^{239}Pu (n,f) rate. Even though the 26-group calculation should be better, the radial leakage approximation dominates the calculational results so that the slab calculation does not give good agreement.

^{238}U Fission

The general shape of the measured radial distribution of ^{238}U fission rate, shown in Figure 9, is predicted fairly well by calculation. The depletion of the fissile content of the "B" drawers, centered near a radius of 17 cm above and below the B-row, clearly decreases the very fast flux in that region and, hence, reduces the fast fissions in ^{238}U .

For the axial ^{238}U fission rate distribution, given in Figure 10, the in-core agreement is good except at the mid-plane. The depression in the fission rate at 0 cm is due to the gap between the halves and the stainless steel plane formed from drawer fronts on both sides of the gap. The gap discontinuity was not included in the calculational model.

Standard deviations from statistical fluctuations of measured count rates are between ± 0.3 and $\pm 1.7\%$ for all measured points of Figures 9 and 10.

$^{10}\text{B} (n, \alpha)$

Calculated and measured $^{10}\text{B} (n, \alpha)$ reaction rate distributions are plotted in Figures 11 and 12. The statistical counting uncertainties are a maximum of $\pm 0.6\%$ for the radial traverse, and $\pm 0.3\%$ for the axial traverse. The size of the plotted points are approximately the size of these uncertainties.

There is only one calculational curve (Figure 12) for the axial reaction rate distribution because this calculation was done in 26 groups. Since ^{10}B is not a resonance absorber, core and reflector averaged cross sections for this case are identical.

Major deviations of calculated and measured ^{10}B reaction rates appear at the core-reflector interface and at the assembly boundaries. Deviations at the assembly boundary may stem from some neutrons reflected back into the reflector by the assembly structure, and from room return neutrons. Neither of these were included in the calculational model. Although ^{10}B is not a resonance absorber, the resonance region may cause difficulty in ^{10}B reaction rate calculations at discontinuities.

Resonance cross section shelf-shielding factors used to calculate the activation flux are calculated from a homogeneous model. This model is not necessarily valid at the core-reflector discontinuity and, hence, the flux is in question at this region.

Effects of Plate Rotation

Activity traverses were taken for platelet geometries, both normal and adjacent to the traversing path rotated 90° . The alternate geometries should give an indication of the effect of finite size plates on local reaction rates. An inspection of Figures 7 through 12 shows the effect of adjacent fuel orientation on the shape of activation distributions could be as large as one percent.

A comparison can be made of the effect of fuel rotation on the peak-to-average ^{239}Pu fission rate distribution. The peak-to-average axial fission rate distributions, as inferred from experiment, are 1.265 ± 0.003 for fuel normal, compared to 1.263 ± 0.003 for fuel rotated. The same comparisons for the radial ^{239}Pu fission distribution give 1.399 ± 0.003 for fuel normal, and 1.395 ± 0.003 for fuel rotated.

Comparisons of Reaction Rate Distributions

Virtually all ^{238}U fissions are with neutrons at energies above 1.0 MeV. Fission rate distributions of ^{238}U are not significantly affected by resonance flux, by room return neutrons, or by thermal neutrons. It is clear that deviations of calculated distributions from measured distributions for this isotope are much smaller than for the ^{239}Pu or ^{10}B reaction rate distributions. Furthermore, it is probable that the absence of these three complicating factors is responsible for the better agreement of the ^{238}U calculated and measured distributions.

The ^{239}Pu (n,f) and ^{10}B (n, α) rates appear to be undercalculated at the core-reflector interface and near the outer reflector boundary. The ^{239}Pu (n,f) rates alone are low inside the reflector at the thermal reaction peak. The problems at the core-reflector and near the assembly boundary were discussed in a preceding section. A disagreement in relative reaction rates of ^{239}Pu (n,f) and ^{10}B (n, α) in the reflector was observed in ZPR-III, Assembly 48 A. Here the ^{239}Pu (n,f) rate was calculated correctly but the ^{10}B (n, α) rate was over-calculated at the reflector thermal peak.⁽⁵⁾ There are obvious inconsistencies in the ^{10}B and ^{239}Pu (n, α) and n,f) cross sections and in the relative calculated spectra of ZPR-III, Assemblies 51 and 48 A. The exact nature

of these inconsistencies is not known, but it is highly unlikely that both the ^{239}Pu fission cross sections and the ^{10}B capture cross sections are correct.

A review of the cross section sets, including resonance parameters, is in progress. Definitive experiments to check individual cross sections and resonance parameters are needed. An experiment in which the ^{239}Pu and ^{238}U data sets can be checked independently would be especially useful.

MATERIAL WORTH TRAVERSES

Radial and axial small sample reactivity worth distributions were measured for ^{239}Pu , ^{238}U , and ^{10}B in Assembly 51.⁽²³⁾ The samples were an annulus of 9.315 g of ^{239}Pu , a cylinder of 85.58 g of ^{238}U with 0.18 g ^{235}U , and an annulus of boron with 0.38 g of ^{10}B and 0.329 g of ^{11}B . Perturbation calculations of the boron worth were in close agreement with the measurements. The calculated reactivities of small ^{239}Pu and ^{238}U samples at the core center were 35% high for ^{239}Pu and 53% high for ^{238}U when compared with the measured values.

The distributions of small ^{239}Pu , ^{238}U , and ^{10}B sample reactivity worths, as measured and calculated for Assembly 51, are plotted in Figures 15 through 20. The axial worth distribution is along drawer P16 and the radial worth distribution is 1-1/4 in. from the reactor midplane for each of the three isotopes.

Calculations of ^{239}Pu and ^{238}U worths were made in 24-energy groups with the PERT IV^(16,17) perturbation code. Fluxes and adjoint fluxes were from 2DB. The calculational model was the same as for the RZ critical mass (see Appendix B-3). Earlier calculations on material worths in Assembly 51 were made in 8 energy groups. Although the

FIGURE 16. Axial ^{239}Pu Worth Distributions

FIGURE 17. Radial ^{238}U Worth Distributions

FIGURE 18. Axial ^{238}U Worth Distributions

FIGURE 19. Radial ^{10}B Worth Distributions

FIGURE 20. Axial ^{10}B Worth Distributions

8-group and 24-group calculations were on slightly different models, they were similar enough to conclude that results of the 8-group and a 24-group perturbation calculation do not differ significantly except for ^{238}U which is quite sensitive to group structure. The 24-group calculation of ^{238}U central worth was 53% more negative than the measurement, compared with the 8-group calculation of ^{238}U central worth calculated to be 30% more negative than the measurements.

Other investigators⁽⁷⁾ have explored possible contributions to the discrepancy between measured and calculated fuel worths. They pointed out that much better agreement exists between calculated and measured critical masses, and between calculated and measured fission ratios than in small sample fuel worths and concluded that the comparisons of calculated and measured small sample worths were of limited value at this time.

In all cases, an overcalculation of the absolute worth at the core-reflector interface just inside the reflector is experienced. The overcalculation is in addition to that at the core center for ^{239}Pu and ^{238}U and an exception to the correctly calculated value of the ^{10}B worth at the core center. Both ^{239}Pu (n,f) and ^{10}B (n, α) calculated and measured reaction rates show a maximum immediately outside the core. It is probable that the measured reaction rates of these two isotopes are correct and that the worth measurements are inconsistent, either because of perturbations caused by the injection of the experiment or because the experiment is insensitive to the small maxima.

^{239}Pu worth is overcalculated in most of the ZPR-III and ZEBRA plutonium assemblies measured to date.⁽⁷⁾ Figure 21 is a plot of the measured plutonium worth per fuel atom per kg as a function of critical mass for six of these assemblies.

FIGURE 21. Measured ^{239}Pu Central Worths for Various Plutonium Assemblies

The calculated-to-experiment value, c/e, is also included for each assembly. Assembly 49 had the same compositions as 48 except that the core sodium had been removed. Assembly 50 compositions were the same as 49 except that carbon was placed in the vacated sodium locations. The c/e for ^{239}Pu central worth in Assembly 49 is 6 to 10% lower than for the other assemblies.

Assuming the measurements to be correct, either the plutonium reaction rates, the plutonium fission neutron source distributions, or both, are not consistent in the three calculations. Carbon and sodium cross sections are not suspect because the anomalous behavior of the c/e occurs between Assemblies 48 and 49 when the carbon content does not change, and between Assemblies 49 and 50 when sodium content does not change.

^{238}U worth is overcalculated in absolute worth in all assemblies also. The worth of ^{238}U per atom of fuel per kg as a function of light atom concentration is shown plotted in Figure 22. The c/e for each isotope except 48A, whose worth has been included to illustrate a point to be explained later, is also plotted. The worth function is linear except for the nickel reflected assemblies ZPR-III, 51, and 56E. In connection with the calculated ^{238}U worth, it is notable that the c/e is nearly constant for ZPR-III Assemblies 48, 49, and 50, but diverges drastically for the nickel reflected assemblies ZPR-III 51 and 56E.

The constancy of the fractional calculated error in ^{238}U worth in Assemblies 48, 49, and 50, implies the inconsistency in ^{239}Pu worth to be in the plutonium cross sections or in the fission source distribution. The divergent behavior of the c/e ratio of ^{238}U for nickel reflected assemblies implies miscalculation in the denominator of the perturbation equation

FIGURE 22. Measured ^{238}U Central Worths for Various Plutonium Assemblies

unless the spectrum at the core center has been affected by the nickel reflector. However, a change in the denominator is not consistent with the plutonium worth that does not show the larger calculated error for these cases. Other limitations of the perturbation approximation will now be considered.

Small sample fissile material worths are considerably more sensitive to the calculated real and adjoint spectrum than a material without significant secondary neutron emission. The first order calculated perturbation worth of a fissioning nuclide contains a term proportional to

$$\int dV \int dE \phi^\dagger(E, V) \chi(E, V) \int dE' \phi(E', V) \delta[\nu \Sigma_f(E', V)] \quad (17)$$

where $\chi(E, V)$ is the spectrum of reactor fission neutrons. The volume integral is over the whole reactor. For a small central sample, the expression reduces to

$$\left[\int dE \phi^\dagger(E) \chi(E) \int dE' \phi(E') \delta[\nu \Sigma_f(E')] \right]$$

where the parameters are evaluated at the core center. This term is directly proportional to the adjoint weighted fission source. For this term to be the main source of error in fuel worths, it necessarily would be about 16% too large for ^{239}Pu or 7% too small for ^{238}U . While a 7% error in the adjoint weighted spectrum of ^{238}U fission neutrons or a 16% over-calculation of the ^{239}Pu adjoint weighted fission spectrum is possible, the simultaneous existence of these two conditions is improbable.

Consider next the normalization of the perturbation worths. If the adjoint weighted fission source averaged over the entire reactor is small, then the normalization term for the perturbation equation

$$\frac{1}{K} \int dV \int dE \chi(E, V) \phi^\dagger(E, V) \int dE' \phi(E', V) \nu \Sigma_f(E, V) \quad (17)$$

may be large and the sum of all terms will be correspondingly greater in absolute magnitude.

The absolute fission ratios for ^{238}U and ^{239}Pu as calculated by 2DB for a central core region indicate (see following discussion of Central Fission Ratios) the integral fission rates to be sufficiently well represented that only a small fraction of the approximately 30% discrepancy between measured and calculated ^{239}Pu worth is likely to arise from these reactions because of the strong dependence of the ^{239}Pu worths on the fission rates.

It has been pointed out⁽⁷⁾ that the inverse kinetics technique of control rod calibration used for the ZPR-III is very sensitive to the value of β , the delayed neutron fraction assumed in analysis. An error in the calibration may account for some of the discrepancy between calculated and measured reactivities. However, the excellent agreement for ^{10}B worths, $c/e = 1$, and the better agreement when fertile and fissile material are simultaneously removed, c/e of 1.15 ± 0.02 to 1.18 ± 0.02 , indicate inconsistencies in spectra used in the perturbation approximation, inconsistencies in the data set, or both.

DISTRIBUTED SODIUM WORTH

Partial sodium voiding experiments were conducted over zones of various sizes and shapes to obtain integral data dependent upon the spatial variation of the sodium worth function ranging from positive at the core center to negative out towards the core boundary.

Although the system contained both elemental sodium and sodium carbonate, only the elemental sodium was removed (see Figure 3) to eliminate the necessity for making corrections for carbon and oxygen removal. This approach does, however, introduce a heterogeneity problem. As shown in earlier

experiments^(24,25,26) and their analyses, the worth of sodium removed is sensitive to the composition of neighboring core components. The care needed in the selection of methods for cross section averaging is discussed in the following, and experimental data and calculations are compared for these experiments in Table 6.

TABLE 6. Sodium Worths

Experiment	Axial Location(a) of Perturbation	Number of Drawers Perturbed		Material Removed or Added, g		Reactivity Change, Δk		c/e
		Type A	Type A*	Sodium	Stainless Steel	Experiment	Calculation	
I	0 to 7 in.	6	12	1038.	-0.84	+9.1 ^(b)	+12.0	
II	0 to 7 in.	6	12	2474.16	+82.24	-18.6 ^(b)	-4.0	
III	7 to 17 in.	6	12	1436.16	+83.26	-24.9 ^(b)	-17.0	
IV	0 to 7 in.	22	20	2768.0	-1.84	+10.4 ^(b)	+24.0	
1/4 Core	0 to 17 in.	24		2474.42	+82.32	-51.9 ^(c)	-48.6	0.936 ^(c)
1/2 Core	0 to 17 in.	42		4329.78	+144.06	-88.8 ^(c)	-86.0	0.968 ^(c)
Full Core	0 to 17 in.	112		11541.32	+364.16	-201.6 ^(c)	-207 (PERT) -192 (2DB)	1.026 ^(c) 0.952 ^(c)

(a) Both halves
(b) $\pm 0.02 \Delta k$
(c) $\pm 2\%$

For measurement of axial dependence, sodium was removed from nine drawers to form a volume centered about the core midplane. The midplane is defined to be perpendicular to the core axis. In Experiment I, the volume extended from 0 to 7 in. on both sides of the midplane. In Experiment II, the volume extended the full core length. In Experiment III, the volume extended from 7 to 17 in. on both sides of the midplane. These volumes are illustrated in Figure 23.

CORE REGION OF SODIUM VOIDING

IV

X-Y PLAN VIEW OF CORE,
ZPR-III ASSEMBLY 51, SODIUM WORTH
EXPERIMENTS I, II, III

IV
+10.4

III
-24.8

II
-18.6

EXPERIMENT I
+9.1

X-Z VIEWS OF ASSEMBLY 51 SODIUM VOIDING EXPERIMENT

FIGURE 23 Sodium Voiding Experiments

Two measurements for investigating the radial dependence of sodium removal were performed. In each, from 0 to 7 in. of sodium was removed on both sides of the core midplane. The removal of sodium in Experiment I from the 9 drawers formed by a central 3×3 array served as a base point. In Experiment IV, sodium was removed from the 21 drawers from a central 5×5 array with the four corner drawers excluded.

Distributed partial sodium void experiments were done in one quadrant, in two adjacent quadrants, and over the whole core. In each experiment, as shown in Figure 3, only the interior column of elemental sodium was removed over the entire core length of each A-A* drawer pair. Experimental and calculated values are included in Table 6. The 2DB and PERT calculations are compared with the whole core partial voiding measurement, but only PERT calculations for the $1/4$ and $1/2$ core voiding experiments were performed. PERT calculations were made by integrating the small sample sodium worth over the voided volume.

In calculating sodium void reactivity changes, care must be taken to include the location effect of the voided volume in the drawer. This effect is particularly important for small volumes and when the sodium is close to resonance absorbers such as ^{238}U .⁽²⁵⁾ Cell calculations to yield flux-volume weighted cross sections for sodium void experiments were made on a cell in which the location of the sodium in the drawer was carefully duplicated. Sodium removal was simulated by removing sodium from that portion of the cell and leaving the appropriate amount of stainless steel in the volume. The model for these cell calculations was nearly the same as the actual drawer loading except that the stainless steel of the cans, drawers, and matrix tubes was smeared into each region

in proportion to its proximity to that region. The model for the partial sodium void over 1/4, 1/2, and the full core was the same as the model for the R-Z critical mass calculation. In the case of the perturbation calculations, fluxes for input to PERT-IV were calculated from the same unperturbed model.

LARGE SAMPLE MATERIAL WORTHS

Since the FTR will always be loaded with substantial in-core heterogeneities, it is of interest to know how well major changes in local core composition can be calculated. To this end, the reactivity change effected by removing and replacing a central core matrix tube with sodium filled drawers has been measured. A separate experiment was done to determine the worth of the fuel of the central column. The worths of poison columns of B_4C and of tantalum were measured to support the FTR control system design and the evaluation of calculational models.

The calculational models for these analysis are given in Appendix E-8, E-9, and E-10.

Large sample material worths have an uncertainty of 2% of the measured reactivity change. The uncertainty arises from source strength uncertainties and statistical variations in flux monitor count rates.

Sodium Substituted for Central Driver Drawer⁽²⁴⁾

Plutonium, uranium, stainless steel plates, and all other materials were removed from the central matrix tube, P16, and replaced with sodium filled cans. The sodium section extending through the axial reflectors was 58 in. long excluding drawer ends and spring gaps. Table 7 summarizes the added sodium and stainless steel loadings.

TABLE 7. Composition of Pl6 Matrix Position
Sodium Substituted

	Half One		Half Two		Total
	<u>Reflector</u>	<u>Core</u>	<u>Core</u>	<u>Reflector</u>	
Na, g	580.2	830.0	829.9	579.9	2820.0
SS, g	863.2	1255.2	1255.7	856.4	4230.5

The measured reactivity change was -763 ± 16 lh, whereas the calculated change was -891 lh, giving a c/e ratio of 1.17 ± 0.02 .

Calculations were carried out with the two-dimensional diffusion code 2DB. The central drawer was cylindricized to conserve volume and materials, and the problem was analyzed in R-Z geometry with 8 energy groups.

Worth of Uranium and Plutonium in Central Fuel Drawer⁽²⁷⁾

The reactivity worth of the uranium and plutonium in the Pl6 matrix position was measured also. The ZPPR fuel, Pu metal, U_3O_8 , and depleted uranium plates were removed. Empty stainless steel cans and rearrangement of the normally present stainless steel plates took up the vacated fuel positions. The material and associated reactivity changes, both experimental and calculated, are shown in Table 8.

The 1.15 c/e for the reactivity change on bulk fuel removal from Pl6 in Assembly 51, as shown by the values of Table 8, can be compared with small sample central worths calculated from perturbation theory yielding a c/e of 1.35 and 1.53 for ^{239}Pu and ^{238}U , respectively.

Control Poisons in Central Column

Measurements were made in the central position, Pl6, of the absolute and relative worths of natural boron carbide and tantalum rods.⁽²⁷⁾ Samples of each material having 1/2-in. by

TABLE 8. Uranium and Plutonium Worth Experiment

	<u>Reference Case Mate- rial to be Removed</u>	<u>Perturbed Case Material Added</u>
$^{239}\text{Pu} + ^{241}\text{Pu}$, g	2076.34	0
$^{240}\text{Pu} + ^{242}\text{Pu}$, g	179.56	0
^{235}U , g	12.83	0
^{238}U , g	5978.0	0
Mo, g	94.48	0
O, g	150.86	
Stainless Steel ^(a)	627.75	650.24
Reactivity ^(b) Measured	0	-752
Change Calculated	0	-863
c/e		1.15 ± 0.02

(a) *Stainless Steel Analysis: 73.4% Fe, 17% Cr, 8.4% Ni, 0.75% Mn, and 0.45% Si by weight.*

(b) $\pm 2\%$

2-in. areal profiles and 36-in. lengths were centered on the axis of the 34-in. core. The samples were placed in typical 2-in. by 2-in. core drawers whose contents of approximately 33 vol% sodium, 37 vol% stainless steel, and 18 vol% poison which correspond to 9432 g of Ta or 1262.2 g of B_4C when loaded in the ZPR-III. The worths were measured relative to a 1-in. square axial void centered 1/2 in. above the central axis. Figure 24 illustrates the drawer loadings for the poison experiments. The figure shows how the stainless steel on both sides of the poison was redistributed to maintain spacing of the drawer contents.

FIGURE 24. Drawer Loading for B_4C and Ta Poison Studies

The reactivity worth of the B_4C and Ta was computed with an 8-group, RZ, 2DB calculation and has been compared with the measurements in Table 9.

TABLE 9. Natural B_4C and Ta Worth in P16

Excess Reactivity, Ih		Reactivity Change Relative to Void, Ih		c/e
<u>Test Region</u>	<u>Core</u>	<u>Exp. (a)</u>	<u>Calc.</u>	
Reference	-740	0	-	
B_4C	-1646	-905	-832	0.917 ± 0.02
Ta	-1313	-573	-538	0.935 ± 0.02
B_4C/Ta		1.58	1.55	0.980 ± 0.02

(a) $\pm 2\%$

These results may be compared to experiments performed in ZPR-III Assembly 48, FTR Phase A, experiments, where similar natural B_4C rods were found to have worths of approximately 725 Ih.⁽³⁾ This finding appears consistent, in view of the different critical masses amounting to about 260 kg for Assembly 48 and 212 kg for Assembly 51. Also in Assembly 48 experiments, a similar B_4C rod enriched to approximately 90% boron-10 was found to have a worth of 1700 Ih.

These experimental results show that central tantalum rods having volumes equal to those of B_4C rods have about 2/3 the reactivity worth of B_4C rods. This ratio might be expected to increase for peripheral rods, as illustrated in BNWL-428, "FTR Control Rod Calculations."

In previous work, primarily to compare transport and diffusion results, B_4C worths were found to agree, but tantalum worths were considerably overcalculated.⁽²⁸⁾ The large overcalculation of tantalum worth, 33%, is presumably from neglecting spatial self-shielding of the tantalum rod.

AXIAL MATERIAL WORTHS IN P16

Axial movement of materials would result in reactivity changes in an operating FTR. Two experiments were executed to help assess the reactivity associated with axial movement of material. A slug of core material moved along the central column and measured for reactivity could correspond to axial movement of an entire assembly. The other experiment, fuel compaction, should give some measure of fuel slumping effects.

Axial Worth of Core Material

Axial worth of core material was measured in the central matrix position (P16).⁽²²⁾ Each half of the core portion of the P16 drawer was loaded with a 7-in. segment of core material and two 5-in. segments of canned sodium. Reactivity measurements were taken with the core material segments in each half occupying the following positions as measured from the core midplane; 0-7 in., 5-12 in., and 10-17 in.

The composition of the core material segments is nominally the same as the composition of an A-type drawer. Each sodium segment consisted of sodium filled stainless steel cans. A stainless steel can with nominal dimensions of $1/4 \times 2 \times 5$ in. weighed 45.74 g and was filled with 29.92 g of sodium. The diffusion code 2DB in an R-Z mode and 8 energy groups were used for the analysis of these measurements. The central matrix position (P16) was cylindricized to conserve volume and material. Experimental and calculated reactivity changes due to displacement of core material away from the core midplane are given in Table 10.

The agreement for the 5-12-in. position reflects the ability to calculate the general shape of the fuel worth function within the core. The degeneration of the agreement is

TABLE 10. Axial Worth of Core Material, Pl6 Matrix Position

Position of Core-Material Segments from Core Midplane, in.	Reactivity Change, Ih		c/e
	<u>Experimental</u>	<u>Calculated</u>	
0-7	0.	-	-
5-12	-92.02	-92	1.00 ± 0.02
10-17	-206.78	-235	1.14 ± 0.02

due to the influence of the reflector and accompanying problems of resonance shielding and spectral mismatch in this transition region. Calculational models are shown in Appendices B-11, B-12, and B-13.

Fuel Compaction

The effect of fuel slumping in the FTR was simulated by a fuel compaction experiment.⁽²⁷⁾ A 5-in. section of plutonium metal, depleted uranium, and ZPPR fuel was moved from the 12 to 17-in. axial position in Drawer 1, Pl6, to the 0 to 5-in. position. Vacated positions were filled with canned sodium. Figure 25 shows the rearranged A-drawer. The heavily outlined pieces are those moved for the reactivity measurement. The total worth of this redistribution was measured to be 125 Ih and calculated to be 146 ± 3 Ih, giving a c/e of 1.17 ± 0.02 . A summary of the material movement is given in Table 11.

The c/e for the fuel compaction experiment is consistent with previous experience in calculating fuel worth in the central column of Assembly 51. The calculational model is shown in Appendix B-14.

FIGURE 25. P16 Drawer for Fuel Compaction Experiment

TABLE 11. Material Redistribution in Fuel
Compaction Experiment

	Material Moved from 12-17-in. Axial Position to 0-5-in., Position, g	Material Moved from 0-5-in. Axial Position to 12-17-in., Position, g
^{239}Pu	301.495	-
^{240}Pu	26.117	-
^{241}Pu	3.0882	-
^{242}Pu	0.21685	-
^{238}U	751.69	-
^{235}U	1.61	-
Na		59.84
SS	93.766	91.48
Al	1.93	-

CENTRAL FISSION RATIOS

Absolute central fission ratios were measured with spherical, back-to-back gas-flow fission chambers.⁽²²⁾ Calculated values are compared with the measurements in Table 12. The calculations were done with the fundamental mode diffusion code, FCC, in 26-energy groups, and with 2DB in 8-energy groups. The 2DB calculation was at the center of a cylindrical model of Assembly 51.

NEUTRON SPECTRA⁽²⁵⁾

Neutron energy distributions were inferred from proton recoil spectra as measured with hydrogen bearing gas proportional chambers.^(29,30) The chamber was filled mainly with elemental hydrogen gas for spectrum measurements below neutron energies of 150 keV. Spectrum measurements above neutron energies of 100 keV to 1.5 MeV were taken with methane filled chambers because of the greater stopping power over the

elemental hydrogen gas. The spectra were measured at four in-assembly locations; P-16, N-16, L-16, and J-10, at axial positions adjacent to the midplane.

TABLE 12. ZPR-III Assembly 51-Central Fission Ratios

<u>Fission Ratio</u>	<u>Experiment</u>	Calculated, <u>FCC-IV</u>	<u>c/e</u>	Calculated, <u>2DB</u>	<u>c/e</u>
U-238/U-235	0.0309	0.033	1.060	0.0304	0.983
U-234/U-235	0.214	0.214	1.000	0.216	1.009
U-233/U-235	1.483	1.449	0.978	1.47	0.991
U-236/U-235	0.0685	0.0687	1.003	0.069	1.007
Pu-239/U-235	1.003	1.012	1.009	0.989	0.986
Pu-240/U-235	0.240	0.240	1.000	0.242	1.008

The volume of the Assembly 51 core was reduced to about 70% of its reference value of 307 liters. A fissile mass of 150 kg remained in the core after the reduction. The reduced multiplication constant, to $k_{\text{eff}} = 0.9$, and the reduced spontaneous fission source were necessary to lower the neutron count rate below the saturation level of the system. The reduction does diminish the value of the measurements to understanding the physics of Assembly 51. Calculations showed a decrease in the core size from 307 ℓ to 200 ℓ to be accompanied by a relative increase of 5 to 10% of the flux above 1 MeV. The difference in spectra between the small core and the full sized Assembly 51 complicates the relation between measured spectra and measured reaction rates made in the 307 ℓ core.

The experiment-derived spectra were given for 150 energy intervals. These data are plotted on Figures 26 through 29, along with a calculated spectrum. Powell pointed out⁽³⁰⁾ that

FIGURE 26. Neutron Spectrum, Center of Assembly 51

FIGURE 27. Neutron Spectrum, Midway from Core Center to Core Edge, Assembly 51

FIGURE 28. Neutron Spectrum, Core-Reflector Interface to Core Edge, Assembly 51

FIGURE 29. Neutron Spectrum, Middle of Radial Reflector to Core Edge, Assembly 51

the prominent features are the oxygen, nickel, iron, sodium, and chromium scattering resonances. The oxygen resonance in particular is prominent in the core and the nickel resonance is prominent in the reflector.

Spectrum calculations were made with the basic 26-group FCC-IV prepared cross section sets used for all calculations on the standard composition of Assembly 51. The calculations were for a cylindrical core with the size of the cylindricized small core in one dimension, and the axial leakage computed as 2DB separately for the core and reflector. 2DB and the spectra were computed with the height search option of the SN transport code DTF-V. The angular quadrature was S-4. Each spectrum was taken from a radius corresponding to the distance from the center of P16 to the center of the drawer of the chamber location. No allowance for chamber length or local perturbation was made.

The calculation at the center of the core yields a harder spectrum than the measurement. None of the scattering resonances, except a trace of the sodium resonance, are apparent in the P16 central calculation. The absence of the oxygen resonance at 440 keV, shifted in the DTF-V spectrum measurement,⁽³⁰⁾ is especially notable.

As in the P16 calculation and in the other two locations, the sodium resonance is the single surviving feature of the fine spectrum in the N16 location halfway between the core center and the core edge. The calculation fits the general shape of the measured spectrum best in this of all locations.

The calculated spectrum at the core edge, L16, gives a good fit of the overall measured shape but the nickel resonance that begins to deepen the flux depletion at 16 keV is, of course, not present. This location is most difficult to handle

analytically because of the ambivalence of the calculational model for determining the self-shielding of resonance cross sections at an interface.

The softer spectrum in the reflector is apparent in Figure 27. The results at J16 are also shown. The huge nickel scattering resonance, appearing as a depletion in the measured spectrum at 16 keV is fully developed.

ANL has calculated the spectrum with a composite of the ELMOE library of angular scattering data and S2 SNARG, the ANL discrete ordinates transport code in 371 equal lethargy groups.⁽³¹⁾ The oxygen and nickel, and some of the other resonances, were conspicuous and apparently well represented in the calculations.

The measured results have been smoothed by reducing the results to a 15% Gaussian resolution. Even so, these results may show more amplitude fluctuations than in the physical spectrum because of statistical counting variations, counter limitations, incomplete gamma discrimination, and instabilities in the differential unfolding process. Double scatter in the chamber will accentuate peaks and neighboring valleys. Wall effects produce asymmetric skew that are difficult to assess.

From a consideration of the experimental data and the ANL calculations, the coarse 26-group structure of the cross section set is obviously adequate only for calculation of the general spectrum shape. Using the 26 group set may have a small effect on gross reactivity calculations such as critical mass, but may obscure more subtle reactivity coefficients.

For example, the 440 keV scattering resonance of oxygen has a significant effect on the neutron spectrum at its peak. The reduction at this point may have a significant effect on the calculated sodium void coefficient for small central volumes.

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APPENDIX A
CROSS SECTION ENERGY GROUP STRUCTURE

APPENDIX A

CROSS SECTION ENERGY GROUP STRUCTURE

Energy Structure for 26-Group Cross Section Set

<u>Group</u>	<u>Fission Source</u>	<u>Energy Limit</u>	<u>Velocity</u>	<u>Delta U</u>
1	2.0000-02	6.5000+00	3.5311+09	4.8000-01
2	9.8000-02	4.0000+00	2.7700+09	4.8000-01
3	1.9000-01	2.5000+00	2.1899+09	4.8000-01
4	2.6800-01	1.4000+00	1.6388+09	5.7000-01
5	1.9600-01	8.0000-01	1.2388+09	5.7000-01
6	1.3500-01	4.0000-01	8.7595+08	6.9000-01
7	5.8000-02	2.0000-01	6.1939+08	6.9000-01
8	2.2000-02	1.0000-01	4.3798+08	6.9000-01
9	9.0000-03	4.6500-02	2.9866+08	7.7000-01
10	3.0000-03	2.1500-02	2.0308+08	7.7000-01
11	1.0000-03	1.0000-02	1.3850+08	7.7000-01
12	0.0000	4.6500-03	9.4444+07	7.7000-01
13	0.0000	2.1500-03	6.4220+07	7.7000-01
14	0.0000	1.0000-03	4.3798+07	7.7000-01
15	0.0000	4.6500-03	2.9866+07	7.7000-01
16	0.0000	2.1500-04	2.0308+07	7.7000-01
17	0.0000	1.0000-04	1.3850+07	7.7900-01
18	0.0000	4.6500-03	9.4444+06	7.7000-01
19	0.0000	2.1500-05	6.4220+06	7.7000-01
20	0.0000	1.0000-05	4.3798+06	7.7000-01
21	0.0000	4.6500-03	2.9866+06	7.7000-01
22	0.0000	2.1500-06	2.0308+06	7.7000-01
23	0.0000	1.0000-06	1.3850+06	7.7000-01
24	0.0000	4.6500-03	9.4444+05	7.7000-01
25	0.0000	2.1500-07	6.4220+05	7.7000-01
26	0.0000	2.5200-08	2.1986+05	7.7000-01
SUM	1.00000+00			

Energy Structure for 8-Group Cross Section Set

<u>New Group</u>	<u>Old Groups Per New Group</u>	<u>Fission Source For New Groups</u>	<u>Lower Energy Limit, MeV</u>
1	3	3.08000-01	2.5
2	3	5.99000-01	0.4
3	3	8.90000-02	0.0465
4	3	4.00000-03	0.00465
5	2	0.00000	0.001
6	2	0.00000	0.000215
7	4	0.00000	0.00001
8	6	0.00000	0.0000000252

APPENDIX B
CALCULATIONAL MODELS

APPENDIX B

CALCULATIONAL MODELS

EXPLANATION OF CALCULATIONAL MODEL DATA SHEETS

Cross Section Sets

- 522226 Flux-volume weighted A-A* macroscopic core cross sections collapsed over central core zone fluxes to 8-energy groups
- 522326 Flux-volume weighted A-A* macroscopic core cross sections collapsed over out core zone fluxes to 8-energy groups
- 511921 Flux-volume weighted safety rod macroscopic cross sections collapsed over central core zone fluxes to 8-energy groups
- FCC 74 8-group macroscopic radial reflector cross sections collapsed from central reflector fluxes
- FCC 110 8-group macroscopic axial reflector cross sections collapsed from axial reflector fluxes
- FCC 125 8-group macroscopic test zone cross sections collapsed from test zone fluxes
- 522427 Flux-volume weighted heavy safety rod macroscopic cross sections collapsed over safety rod, slab model, fluxes to 8-energy groups
- 522730 Flux-volume weighted tantalum rod macroscopic cross sections collapsed over tantalum rod, slab model, fluxes.

CRITICAL, MASS, XY CORE

B-2
CALCULATION # 2DB123

← EXPERIMENTAL ASSEMBLY # 51-28

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____CRITICAL MASS 212.08 kg

ABBREVIATIONS:

☒ _____ IN HALF #1, _____ IN HALF #2☐ = SAFETY DRAWER☐ = CONTROL DRAWER☒ = ENRICHED DRAWER OF CELLOTHER: MEASURED MASS 212.75
K_{eff} 1.00056

ABBREVIATIONS:

☐ = SAFETY DRAWER☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.:

CORRECTED MASS 212.08 kg

← CALCULATIONAL MODEL

CODE 2DB GEOMETRY XYDIFFUSION _____, if S_N, N = _____SYMMETRY 1/4SOURCE OF ATOM DENSITIES ANLX-SECT. { GROUPS 8
HOMO./HETERO. Ref1/CoreCORE HT. 86.511 cmEFFECTIVE CORE RADIUS 33.647 cmREFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cmBUCKLING 6.82 M⁻²CALCULATED K_{eff} = 0.9738CALCULATED MASS = 210.75CORE GAP -0.0017TEMPERATURE 300 °K 307 °K -0.0002

SAFETY ROD POSITION _____

DIFFUSION + TRANSPORT S 4 0.01

HOMOG. → HETEROGENEOUS _____

OTHER: +0.00131-0.00092+0.00189CORRECTED DIFFUSION K_{eff} 0.97418CORRECTED TRANSPORT K_{eff} 0.98418

CRITICAL MASS, CYLINDRICIZED CORE

B-3

CALCULATION # 2DB51

EXPERIMENTAL ASSEMBLY # 51 CYLINDRICIZED

EFFECTIVE CORE RADIUS 32.91 cmCORE HEIGHT 86.51 cmREFLECTOR THICKNESS { RADIAL 30 cm
AXIAL 30 cmCRITICAL MASS 202.87 kg

ABBREVIATIONS:

☒ IN HALF #1, _____ IN HALF #2☐ = SAFETY DRAWER☐ = CONTROL DRAWER☒ = ENRICHED DRAWER OF CELL

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER☐ = CONTROL DRAWER

OTHER:

CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZDIFFUSION _____, if S_N , $N =$ _____SYMMETRY 1/2

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. Ref1/CoreCORE HT. 86.51 cmEFFECTIVE CORE RADIUS 34.9061 cmREFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cmBUCKLING 0CALCULATED K_{eff} = 1.000CALCULATED MASS = 228.57

CORE GAP _____

TEMPERATURE -0.00017

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S_4 0.01000

HOMOG. → HETEROGENEOUS _____

OTHER: -25.7 kg -0.0205

MASS CORRECT.:

CORRECTED MASS 202.87 kgCORRECTED DIFFUSION K_{eff} 0.9778CORRECTED TRANSPORT K_{eff} 0.9888

SODIUM VOID CALCULATIONAL MODEL
SODIUM OUT 5 x 5, 0-7 in.

B-4

CALCULATION # 2DB56

EXPERIMENTAL ASSEMBLY #

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ _____ IN HALF #1, _____ IN HALF #2

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

☒ = ENRICHED DRAWER OF CELL

OTHER: _____

CALCULATIONAL MODEL

CODE 2DB GEOMETRY _____

DIFFUSION _____, if S_N , $N =$ _____

SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51 cm

EFFECTIVE CORE RADIUS 34.9 cm

REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm

BUCKLING 0

CALCULATED K_{eff} = 1.00078

CALCULATED MASS = 229 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION + TRANSPORT S _____

HOMOG. + HETEROGENEOUS _____

OTHER: K, BARE CASE; Na IN 1.00055
 $\Delta K/K$ +0.00023

ABBREVIATIONS:

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

OTHER:

MASS CORRECT :

CORRECTED MASS 229 kg

CORRECTED DIFFUSION K_{eff} _____

CORRECTED TRANSPORT K_{eff} _____

← EXPERIMENTAL ASSEMBLY #

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ IN HALF #1, _____ IN HALF #2

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

☒ = ENRICHED DRAWER OF CELL

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

OTHER:

NO-CORE, SODIUM OUT

MASS CORRECT.:

CORRECTED MASS 229 kg

← CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZ

DIFFUSION _____, if S_N , $N =$ _____

SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51 cm

EFFECTIVE CORE RADIUS 34.9 cm

REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm

BUCKLING 0

CALCULATED K_{eff} = 0.99911

CALCULATED MASS = 229 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S _____

HOMOG. → HETEROGENEOUS _____

OTHER: K BASE CASE; Na IN 0.99899
 $\Delta K/K$ +0.00012

CORRECTED DIFFUSION K_{eff} _____

CORRECTED TRANSPORT K_{eff} _____

CALCULATION # 2DB54

← EXPERIMENTAL ASSEMBLY

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ _____ IN HALF #1, _____ IN HALF #2☐ = SAFETY DRAWER☐ = CONTROL DRAWER☒ = ENRICHED DRAWER OF CELL

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.:

CORRECTED MASS 229 kg

← CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZDIFFUSION _____, if S_N , $N =$ _____SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORECORE HT. 86.51 cmEFFECTIVE CORE RADIUS 34.9 cmREFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cmBUCKLING 0CALCULATED K_{eff} = 0.99827CALCULATED MASS = 229 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S _____

HOMOG. → HETEROGENEOUS _____

OTHER: K , BASE CASE; NA IN 0.99830
 $\Delta K/K$ -0.00003CORRECTED DIFFUSION K_{eff} _____CORRECTED TRANSPORT K_{eff} _____

← EXPERIMENTAL ASSEMBLY # _____

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ _____ IN HALF #1, _____ IN HALF #2

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

☒ = ENRICHED DRAWER OF CELL

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.:

CORRECTED MASS 229 kg

← CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZ

DIFFUSION _____, if S_N , $N =$ _____

SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51 cm

EFFECTIVE CORE RADIUS 34.9 cm

REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm

BUCKLING 0

CALCULATED K_{eff} = 0.99912

CALCULATED MASS = 229 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S _____

HOMOG. → HETEROGENEOUS _____

OTHER: K , BASE CASE; NA IN 0.99928
 $\Delta K/K$ 0.00016

CORRECTED DIFFUSION K_{eff} _____

CORRECTED TRANSPORT K_{eff} _____

← EXPERIMENTAL ASSEMBLY #

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ _____ IN HALF #1, _____ IN HALF #2

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

☐ = ENRICHED DRAWER OF CELL

OTHER: _____

← CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZ

DIFFUSION _____, if S_N , $N =$ _____

SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51 cm

EFFECTIVE CORE RADIUS 33.7 cm

REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm

BUCKLING 0 B_4C Ta

CALCULATED $K_{eff} = 0.96852$ 0.97128

CALCULATED MASS = 210 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S _____

HOMOG. → HETEROGENEOUS _____

OTHER: Base Case K_{eff} 0.97636 0.97636
 $\Delta K/K$ 0.00803 0.00520

ABBREVIATIONS:

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.:

CORRECTED MASS 210 kg

CORRECTED DIFFUSION K_{eff} _____

CORRECTED TRANSPORT K_{eff} _____

EXPERIMENTAL ASSEMBLY

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ _____ IN HALF #1, _____ IN HALF #2☐ = SAFETY DRAWER☐ = CONTROL DRAWER☐ = ENRICHED DRAWER OF CELL

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.:

CORRECTED MASS 227 kg

CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZDIFFUSION _____, if S_N , $N =$ _____SYMMETRY 1/2 Core

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORECORE HT. 86.51 cmEFFECTIVE CORE RADIUS 34.9 cmREFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cmBUCKLING 0CALCULATED K_{eff} = 0.99218CALCULATED MASS = 227 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION $\rightarrow$ TRANSPORT S _____HOMOG. $\rightarrow$ HETEROGENEOUS _____OTHER: K , BASE CORE, P16 IN 1.00078
 $\Delta K/K$ -0.0085CORRECTED DIFFUSION K_{eff} _____CORRECTED TRANSPORT K_{eff} _____

← EXPERIMENTAL ASSEMBLY

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ _____ IN HALF #1, _____ IN HALF #2☐ = SAFETY DRAWER☐ = CONTROL DRAWER☒ = ENRICHED DRAWER OF CELL

OTHER: _____

← CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZ

DIFFUSION _____, if S_N , $N =$ _____

SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51 cm

EFFECTIVE CORE RADIUS _____

REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm

BUCKLING 0

CALCULATED K_{eff} = 0.99245

CALCULATED MASS = _____

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S _____

HOMOG. → HETEROGENEOUS _____

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.:

CORRECTED MASS _____

CORRECTED DIFFUSION K_{eff} _____CORRECTED TRANSPORT K_{eff} _____

FUEL MOVEMENT 0-7 in.

CALCULATION # 2DB75

B-11

EXPERIMENTAL ASSEMBLY #

EFFECTIVE CORE RADIUS

CORE HEIGHT

REFLECTOR THICKNESS { RADIAL
AXIAL

CRITICAL MASS

ABBREVIATIONS:

IN HALF #1, IN HALF #2

S = SAFETY DRAWER

C = CONTROL DRAWER

E = ENRICHED DRAWER OF CELL

OTHER:

ABBREVIATIONS:

S = SAFETY DRAWER

C = CONTROL DRAWER

OTHER:

MASS CORRECT.:
CORRECTED MASS 238 kg

CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZ

DIFFUSION, if S_N , $N =$

SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51

EFFECTIVE CORE RADIUS 35.6 cm

REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm

BUCKLING 0

CALCULATED K_{eff} = 1.000

CALCULATED MASS = 238 kg

CORE GAP

TEMPERATURE

SAFETY ROD POSITION

DIFFUSION + TRANSPORT S

HOMOG. + HETEROGENEOUS

OTHER:

CORRECTED DIFFUSION K_{eff}

CORRECTED TRANSPORT K_{eff}

FUEL MOVEMENT 5-12 in.

CALCULATION # 2DB76

B-12

EXPERIMENTAL ASSEMBLY #

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☐ IN HALF #1, _____ IN HALF #2

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

☐ = ENRICHED DRAWER OF CELL

OTHER: _____

CALCULATIONAL MODEL

CODE 2DB GEOMETRY RZ

DIFFUSION _____, if S_N , $N =$ _____

SYMMETRY 1/2 CORE

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51 cm

EFFECTIVE CORE RADIUS 35.63 cm

REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm

BUCKLING 0

CALCULATED K_{eff} = 0.99911

CALCULATED MASS = 238 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S _____

HOMOG. → HETEROGENEOUS _____

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.:
CORRECTED MASS 238 kg

CORRECTED DIFFUSION K_{eff} _____
CORRECTED TRANSPORT K_{eff} _____

FUEL MOVEMENT 10-17 in.

B-13
CALCULATION # 2DB77

← EXPERIMENTAL ASSEMBLY #

EFFECTIVE CORE RADIUS _____

CORE HEIGHT _____

REFLECTOR THICKNESS { RADIAL _____
AXIAL _____

CRITICAL MASS _____

ABBREVIATIONS:

☒ IN HALF #1, _____ IN HALF #2

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

☒ = ENRICHED DRAWER OF CELL

OTHER: _____

← CALCULATIONAL MODEL

CODE _____ GEOMETRY _____

DIFFUSION _____, if S_N , $N =$ _____

SYMMETRY _____

SOURCE OF ATOM DENSITIES _____

X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE

CORE HT. 86.51 cm

EFFECTIVE CORE RADIUS 35.63 cm

REFLECTOR THICKNESS RADIAL 30 cm

AXIAL 30 cm

BUCKLING 0

CALCULATED K_{eff} = 0.99773

CALCULATED MASS = 238 kg

CORE GAP _____

TEMPERATURE _____

SAFETY ROD POSITION _____

DIFFUSION → TRANSPORT S _____

HOMOG. → HETEROGENEOUS _____

OTHER: _____

ABBREVIATIONS:

☐ = SAFETY DRAWER

☐ = CONTROL DRAWER

OTHER:

MASS CORRECT.: _____

CORRECTED MASS 238 kg

CORRECTED DIFFUSION K_{eff} _____

CORRECTED TRANSPORT K_{eff} _____

FUEL COMPACTION

B-14
CALCULATION # 2DB71-
72

EXPERIMENTAL ASSEMBLY #
EFFECTIVE CORE RADIUS 34.9 cm
CORE HEIGHT 86.51 cm
REFLECTOR THICKNESS { RADIAL 30 cm
AXIAL 30 cm
CRITICAL MASS 229 kg
ABBREVIATIONS:
[] IN HALF #1, [] IN HALF #2
[S] = SAFETY DRAWER
[C] = CONTROL DRAWER
[E] = ENRICHED DRAWER OF CELL
OTHER: K = 1.000204

CALCULATIONAL MODEL
CODE 2DB GEOMETRY RZ
DIFFUSION, if S_N , N =
SYMMETRY FULL CORE
SOURCE OF ATOM DENSITIES
X-SECT. { GROUPS 8
HOMO./HETERO. REFL/CORE
CORE HT. 86.51 cm
EFFECTIVE CORE RADIUS 34.9 cm
REFLECTOR THICKNESS RADIAL 30 cm
AXIAL 30 cm
BUCKLING 0 1.001622
CALCULATED K_{eff} =
CALCULATED MASS =
CORE GAP
TEMPERATURE
SAFETY ROD POSITION
DIFFUSION + TRANSPORT S
HOMOG. + HETEROGENEOUS
OTHER: $\Delta K/K$ +0.001418

ABBREVIATIONS:

[S] = SAFETY DRAWER

[C] = CONTROL DRAWER

OTHER:

FUEL OUT ZONE

FUEL COMPACTED ZONE

MASS CORRECT.:

CORRECTED MASS

CORRECTED DIFFUSION K_{eff}

CORRECTED TRANSPORT K_{eff}

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