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FAST FLUX
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RESEARCH
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DEVELOPMENT
REPORT

**RATIONALE AND PLANS
FOR THE
FTR CRITICAL EXPERIMENTS PROGRAM**

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RATIONALE AND PLANS FOR THE FTR
CRITICAL EXPERIMENTS PROGRAM

by

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June 20, 1967

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RATIONALE AND PLANS FOR THE FTR CRITICAL EXPERIMENTS PROGRAM

R. A. Bennett
P. L. Hofmann

I. Introduction

At the 1966 ANL conference on "Fast Critical Experiments and Their Analysis", a preliminary outline and justification for a series of proposed critical experiments in support of FTR design was presented.⁽¹⁾ Since that time, fairly large design changes in the FTR concept have taken place and more detailed nuclear design information has been generated, so that a revision, and a more detailed justification of the proposed experimental program is in order.

The purpose of this report is to identify the uncertainties in the FTR design which exist due to the lack of critical experiment data, to assess the degree to which additional critical experiments could improve that situation, and to outline briefly the proposed, overall FTR Critical Experiment Program.

Before proceeding into a specific discussion of the proposed FTR program, it is instructive to examine the manner in which critical experiments are used, and to consider briefly the types of programs now under way or planned for the near future at various sites engaged in fast reactor experimental work. This is done in Section II.

In Section III, an assessment of the present status of fast reactor physics calculational methods and accuracy of current nuclear data is presented. Such an assessment is necessary, if one is to evaluate the

nuclear design uncertainties of the FTR and the degree to which they can be removed by critical experiments.

In section IV, the effects of some calculational uncertainties on the FTR nuclear design are enumerated and the possible consequences of these uncertainties are described.

Finally, in section V, a brief outline of the entire critical experiment program is presented.

II. Review of Present Critical Experiment Programs in Support of Pu-Fueled Fast Reactors

At the present time there are four major facilities in the U.S. and Europe that are involved in fast reactor physics critical experiment programs and which can accommodate substantial amounts of Pu fuel.

They are:

ZPR-3	located near Idaho Falls
ZEBRA	located at Winfrith, England
MASURCA	located at Cadarache, France
SNEAK	located near Karlsruhe, Germany

ZPR-3 and ZEBRA have been in operation for considerable time, while MASURCA and SNEAK only recently achieved initial criticality (December 1966). The present and near future experimental programs of these facilities differ in important respects. It is useful to enumerate these differences, so that the planning of future experimental programs in the ZPR-3 facility - in connection with the FTR design - can be carried out on a reasonably firm basis.

Before presenting a detailed discussion of the various programs, some comments concerning the nature of critical experiments in general might be in order. There are basically two types of critical experiments that can be performed:

- (1) Critical experiments can be carried out in direct support of a specific reactor project.
- (2) Critical experiments can be performed for the purpose of generating basic reactor physics information of broad, general interest to reactor technology.

These two points of view have been discussed by various authors. For instance, in the description of the early PWR experiments,⁽²⁾ Krasik and Radkowsky made the following points:

"The aims of the PWR critical experiment program are (1) to provide essential reactor physics information which cannot be obtained by available calculational procedures, (2) to provide general guidance to the analytical nuclear design program, and (3) to substantiate and verify the significant reactor calculations. There are two distinct parts to the critical experiment program which meet substantially different needs.

One part of the program relates specifically to the actual PWR design. The information developed in this part of the program deals primarily with reactor control, flux distributions and the temperature coefficient of reactivity. By its nature, an experiment designed to provide this type of information becomes in many respects a nuclear mockup of the actual reactor design."

"The other part of the experimental reactor physics program - and of equal importance to the first - aims to provide experimental information of considerably broader scope. As remarked above the nuclear mockup experiment provides little or no explanation of basic difficulties which may arise in a design and equally little guidance as to

significant directions in reactor design. The second part of the experimental program, particularly when coupled with a strong theoretical program, supplies this need.

This experimental program, through the study of critical experiments of simple geometry, aims at measurements of such fundamental reactor constants as critical bucklings, migration areas, resonance escape probabilities, thermal utilizations and fast fission factors. The experiments are related to the PWR design in the use of the same reactor core materials. The range of parameter variation is deliberately broadened, however, to demonstrate the trends in the important reactor constants."

A similar viewpoint was expressed by Snyder in his paper describing the early critical experiment programs at KAPL.⁽³⁾ Regarding the PPA, he stated:

"Its primary purpose was to obtain information used in the early stages of design work. However, improvements in technique have made it possible to use the PPA for accurate check of the nuclear performance of a final reactor design. An additional function of the apparatus has been to determine parameters and properties of materials useful for future power reactors in further development programs."

More recent viewpoints tend to favor the "generalized" critical experiment approach. For instance, in a recent ANL talk, Loewenstein stated:⁽⁴⁾

"The amount of flexibility available to perform detailed engineering mockup experiments seems to be somewhat less as one contemplates the current projected plutonium fueled designs of interest. This is not considered very limiting since analytical and nuclear data developments are constantly improving.

The nature of some problems facing the smaller earlier design reactors are also somewhat different from those facing the current and future designers. It is judged that this will ultimately be reflected in a major reconsideration about the desirability to perhaps even dispense with the detailed engineering mockup in favor of a series of related critical assemblies and/or zoned systems. These may, in fact, provide better information for confident reactor design. Useful data for a reactor design can be generated as a function of system configuration, gross composition, and isotopic composition, within a parametric but realistic framework. The role of the critical experiment in the reactor itself must also be considered. The potential of the dry and wet critical experiments cannot be neglected with at least the first loading of the reactor. Significant long term reactivity data can be generated by the reactor itself through periodic zero power tests. These can be correlated with small sample replacement experiments in critical assemblies to isolate nuclear from mechanical burnup effects. The latter pertain to fuel growth and/or redistribution."

Kouts and Palmedo⁽⁵⁾ also argue for a more systematic approach to critical experiments. In discussing the characteristics of a "Good Set of Experiments", they have identified five features of importance:

"1. To assure adequacy, they should extend somewhat beyond the expected range of design interest. Specific designs can then be analyzed by interpolation between understood experiment, rather than by extrapolation into an untested region.

2. There should be measurements of as many as possible of the quantities directly related to the criteria that the analysis must satisfy. The calculation of important quantities is thus tested firsthand, and therefore more reliably.

3. The experiments should be geometrically simple, or at worst their geometric complexity should be of the same kind as that expected to exist in the reactors. This avoids the need to incorporate in the analysis of experiments unnecessary geometric complexity that itself introduces errors. Otherwise, the features of the analysis most in doubt could be those that are required to deal with the geometric problems, and the parts of the analysis needed for reactor design would remain untested.

4. The set of experiments should be large enough to permit use of statistical methods in analyzing trends. Usually, at least three experiments at different values of x are needed to establish the shape of a function $f(x)$ in a certain range. Most often, the number is larger.

5. The experiments should be useable as a diagnostic set to permit testing individual aspects of the theory. This means that the experiments should be planned so as to test each aspect separately. This will be accomplished if it is seen how the experimental results depend on variation of one parameter at a time.

The above features are not uniquely requirements of experiments in support of the fast breeder program. They are fundamental features of good experimental science, placed in the specific frame of reference discussed here."

The present (1966-1967) ZPR-III program is primarily concerned with more general experiments for the purpose of supplying basic data which might be useful in the design of future fast nuclear reactors. The European programs, on the other hand, generally concentrate on supplying nuclear design information for specific reactor concepts and designs.

In the case of the FTR, the complexity of the machine is of such a nature, that detailed critical experiments are absolutely essential. In the following paragraphs a brief description of the present ZPR-III programs and of the European programs is given to provide some perspective for judging the proposed FTR experiments.

A. The ZPR-III Program

Following the SEFOR experiments in ZPR-III (Assembly 47), a lengthy series of experiments on Pu assemblies with degraded neutron spectra has been under way.⁽⁶⁾ The purpose of this program is to supply integral neutronics data to serve as the basis for theory-experiment correlations.

To facilitate the analysis of the proposed critical configurations, a strong effort is being made to retain a "clean" geometry as much as possible. Six different assemblies will be considered (P1-P6) incorporating a systematic variation in U/Pu ratio and graphite and sodium concentrations.

The experiments are expected to provide information on enrichment requirements, Doppler coefficients and sodium coefficients, and neutron spectra. It is hoped that these integral data will be of use to nuclear designers of future fast breeder reactors.

B. Fast Reactor Critical Experiment Programs in Europe

The recent IMFBR physics team visit to Europe⁽⁷⁾ provided an opportunity to review, in fair detail, the fast critical experimental programs in France, Germany and England. Detailed discussions were held concerning the present programs, as well as the programs planned for the near future. It is clear that the European programs (unlike the ANL program) are strongly project oriented, though some time has been reserved for more general research. The following summaries are taken from reference (7).

1. The French Program, MASURCA

MASURCA is a very elaborate fast critical facility. It can accommodate about 2 tons of Pu. At present, only 175 kg of Pu are available (155 kg is 8% Pu-240, 20 kg is 4% Pu-240). By the end of the year, an additional 750 kg of U-235 will be available.

MASURCA went critical for the first time 12/15/66. It had a core mockup similar to ZEBRA-6. This assembly 1A has a lattice of three

graphite rods per fuel rod. It has no sodium. MASURCA 1 was supposed to contain only Pu. However, since there was a fairly large discrepancy in critical mass, U-235 had to be added and the assembly became 1A.

The discrepancy in critical mass was about 10%.

Plans for MASURCA are as follows: Assembly 1A experiments will be completed, and the reactor will then be switched to 1B which is analogous to 1A, except that uranium replaces plutonium. The 1B composition will then also be examined in an exponential experiment in connection with HARMONIE, in a pulsed experiment in the German SUAK facility, and finally, as part of a fast-thermal experiment (ERMINE) in the French MINERVE reactor. This is a rather interesting sequence of experiments on the same reactor composition. It is hoped that the MINERVE experiments will reproduce the spectra found in MASURCA so that reactivity coefficients could be measured there, and MASURCA time saved. This experimental series is expected to be completed in 1967. Years 1968 and 1969 are expected to be devoted to PHENIX mockups. The PHENIX experiments will be done with the available 175 kg of Pu, and, hence, they will of necessity have to be zoned experiments to simulate the actual 800 kg loading.

2. The German Program, SNEAK

The first SNEAK core is U-235 loaded to simulate the ZPR-III assembly 41. The purpose of this experiment is to obtain a checkpoint. Early in 1967, the Germans expect to go to a Pu core to simulate, first a steam reactor, and then a Na reactor.

SNEAK-1 is not completely identical to ZPR-41, but it comes quite close. It has a 400 lit core with about 500 kg of U-235. The reflecting blanket is depleted uranium. The critical inventory based on ZPR-41 was expected to be 502 kg, and SNEAK went critical with 504 kg. All measurements have been made and agree very well with expectations.

Future SNEAK programs are as follows:

- SNEAK 3A Mockup of medium size steam reactor
 600 lit core, to study heterogeneities.
- SNEAK 3B Mockup of medium size steam reactor
 using a two-zone Pu-U core, since
 only 175 kg of Pu will be available.
- SNEAK 2 Mockup of large (~ 1400 lit) Na cooled
 prototype reactor.
- SNEAK 4 Simulation of Na prototype.
- SNEAK 5 Simulation of steam prototype.

Detailed preparations for the SNEAK-3 experiments are now under way. The height of the 600 lit core will be 81 cm, diameter is 97 cm, the fissile fuel is in oxide form.

In 3A, heterogeneity effects will be studied by bunching platelets. Many preparatory calculations have been done for this experiment.

After 3A, the assembly will be converted to 3B by introducing a central (~ 250 lit) PuO_2/UO_2 zone. Fission products will be mocked up to study their effects, and Pu sector experiments will be performed to measure Doppler coefficients.

A hot Doppler experiment might possibly be done in KRITO. This is a Dutch facility similar to STARK.

3. The British Program, ZEBRA

ZEBRA is the major facility of the British fast reactor physics program. ZEBRA went critical in 1962.

The present critical configuration is ZEBRA-7, which is in direct support of PFR. This program was started in May of 1966 and is expected to last until July 1967. Most of this 14 months period will be taken up directly with PFR experiments, but some more general experiments will also be done during this time. At a later date, it is quite possible that there may be another series of PFR mockup experiments, particularly if major PFR design changes occur. Mockup programs could be started on 2 to 3 months notice.

The kind of experiments already done, or planned for the present PFR mockup series, is as follows:

- (1) Critical Mass Data: Fuel enrichment; effect of pin spacers (W, Mo, Nb) on reactivity; fuel rearrangement effects on reactivity.
- (2) Reaction Rate Data: Power distribution, particularly near boundaries; effect of moderating reflectors; breeding gain measurements; determination of optimum breeding blanket dimensions.
- (3) Control Rod Design: Worth of rods, using actual rod pin design; rod interaction effects; spatial heat rate distributions within the control rod assembly; effect of rods on flux distribution.
- (4) Gamma Flux Distributions: Gamma flux distribution in core, blanket and shield regions, this is done with gamma scans.
- (5) Shield Design: Determination of starting spectra for LIDO experiments; flux spectrum measurements overlap the LIDO experiments.

(6) Instrument Sensitivity Data: Neutron and gamma intensities; mockup goes out to instrument positions.

(7) Doppler Coefficients: Most of this information goes back to the ZEBRA 5 experiments.

(8) Sodium Coefficients: Effect of total and local sodium voiding. This is done in sector experiments.

(9) Miscellaneous: Possible measurements of power coefficients; spontaneous fission source; mockup of fission products to determine their effect on reactivity (no plans at present to determine their effect on Doppler and sodium coefficients); effect of Pu-240 on Na coefficient (this was done with 5% and 10% Pu-240 material, nothing startling was discovered).

(10) General Physics Experiments: Central fission ratios and scans; activation ratios; effect of heterogeneities; Rossi α measurements; Li^6 spectrometry.

To accomplish these extensive PFR critical experiment goals, various versions of assembly 7 are being looked at. The main practical difficulty is the lack of adequate Pu and Na, so that zoned experiments must be done. Partial Al for Na substitution is used, also aluminum oxide is being used. The following assemblies have been identified.

(1) ZEBRA-7A: In this assembly the inner PFR zone is divided into two zones. The central zone represents a good simulation of PFR containing about 150 kg PuO_2/UO_2 fuel (1/4 core volume) and Na. The zone is large enough to study central spectra and axial reaction rates.

(2) ZEBRA-7B: In this assembly the inner zone is more homogenized. This assembly is best for radial reaction rate studies and control rod worth studies. Ta and B_4C rod worths were measured, singly, in pairs, and in diametrically opposite compositions. Some shadowing was observed. The early three region PFR blanket design was represented in a 120° sector. At 100w enough intensity is obtained to do measurements in the outer shield mockup.

(3) ZEBRA-7C: In this assembly the core moderating reflector has been eliminated and is replaced by blanket breeder material.

(4) ZEBRA-7D: The outer graphite breeder blanket is eliminated to improve instrument performance.

(5) ZEBRA-7E: This assembly will deal with variations in the inner shield zone. It is not yet in progress.

Along the line of more general physics experiments the main objectives are improvement of nuclear data from integral experiments (e.g. α Pu-239) and identification of assemblies one should build to highlight uncertainties in nuclear data.

This brief review of the European programs gives an indication of their strong project orientation. This is particularly true for the British experiments in support of PFR. It is expected that the critical experiments in support of FTR design would follow a similar pattern.

III. An Assessment of the Uncertainties of Calculated Nuclear Parameters for Small Fast Reactors

Assessments of the uncertainties of calculated nuclear parameters of fast reactors have been attempted by various authors. The purpose

of this section is to call attention to some of these attempts, to point out some of their more interesting findings - particularly as they apply to small FTR-type reactors - and to recommend a set of expected uncertainties for the more important neutronic reactor characteristics so that a basis for FTR experimental planning can be established.

The task of estimating uncertainties is quite difficult, since uncertainties of predicted nuclear characteristics generally stem from a variety of interrelated sources.

Three such "error sources" can be identified quite readily. They are:

1. Errors due to uncertainties in nuclear data.
2. Errors due to inadequate calculational models and methods.
3. Uncertainties due to unforeseen changes in reactor design.

The nature of the first two items is fairly self-evident. The third item may require some explanation. Core design generally requires a series of iterations among nuclear, thermal-hydraulic, and mechanical design approaches. A first iteration rapidly establishes the gross core size and overall core volume fractions for a given power level. For given constituents and core size a first estimate of fuel enrichment is then made. The fuel enrichment, or the resulting fertile-to-fissile fuel ratio, now becomes a very important determinant for other nuclear parameters, such as sodium and Doppler coefficients, flux spectra and spatial distributions, control rod worths, etc. For instance, for a given core, the Doppler coefficient is almost directly proportional to the fertile-to-fissile ratio. If the initial enrichment estimate of, say, 20% is

changed to 25%, the fertile-to-fissile ratio changes from 4:1 to 3:1, and the Doppler coefficient changes accordingly. This uncertainty in the predictions of the Doppler coefficient had obviously nothing to do with uncertainties in basic nuclear data or calculational techniques, but is simply a result of a reactor configurational change. Furthermore, while a change in enrichment from 20% to 25% may be of little significance per se, a 25% reduction in the magnitude of the Doppler coefficient could indeed be of significance.

The importance of uncertainties of this third type can only be evaluated with reference to a specific design concept. As far as the uncertainties mentioned in items (1) and (2) are concerned, more general treatments are satisfactory. The present discussion addresses itself only to these latter kinds of uncertainties.

A. Survey and Summary of Recent Data

Many investigators have tried to relate specific uncertainties in cross section data to their effect on certain nuclear parameters. Typical of this approach is, for instance, the paper by Greebler et al, reference (8). While such a procedure is quite useful, an integral approach to the problem is of more interest from the nuclear design viewpoint. A number of attempts have been made along this direction. Two of the more important investigations in this connection were made by Okrent⁽⁹⁾ and Davey⁽¹⁰⁾.

In Okrent's "Intercomparison", a series of simple reactor configurations, covering a range of representative fast reactor compositions, was postulated. Various laboratories were then invited to calculate the

nuclear characteristics of these assemblies. As stated by Okrent in his paper, "In a crude way the intercomparison of calculations was expected to provide some measure of the current uncertainties in prediction of nuclear parameters for large power reactors".

The results of Okrent's survey showed rather startling divergences of results among the participating laboratories. While the original paper should be consulted for details, the results for the bare core calculations with a fertile-to-fissile ratio of 5:1 are given below:

Table III-1

Summary of Intercomparison of Calculations
for a bare, 5:1 core - Ref. (9)

	$B^2 \times 10^4$	% δk for 50% Na Rem.	Doppler Effect $-Tdk/dT$	$\frac{\sigma_f^{28}}{\sigma_f^{49}}$	$\frac{\sigma_c^{28}}{\sigma_f^{49}}$	$\frac{\sigma_c^{49}}{\sigma_f^{49}}$	% Fiss. below 25 kev	2.1 kev
Median	16.2	-3.64	.0052	.030	.14	.21	17.8	6.0
% Spread								
Plus	12%	23%	48%	7%	7%	9%	22%	87%
Minus	8%	24%	83%	10%	14%	24%	35%	69%

Integral results of this nature do not separate out the discrepancies due to differences in nuclear data from the discrepancies due to differences in calculational technique. It was, however, Okrent's opinion - in the discussion of his paper - that differences in cross section input were "the primary culprit". Some of the differences should really be ascribed to uncertainties typical of the third category. For given compositions, errors in buckling imply that the nuclear parameters were really compared for cores of different sizes, and, hence, the

reported discrepancies are at least partially due to the fact that they were really calculated for different systems. Nevertheless, the discrepancies are quite sizeable.

One deficiency of Okrent's "Intercomparison" was the absence of a "correct" answer. The range of the calculated values of physics parameters simply supplied a "feel" of the possible magnitude of discrepancies one might expect from design calculations on similar systems. Davey's paper, "Intercomparison of Calculations for a Dilute Plutonium-Fueled Fast Critical Assembly (ZPR-3, Assembly 48)", delivered at ANL in 1966 removed this deficiency by providing a set of "correct" experimental results which could be compared with the pre-calculated results submitted by various laboratories. As before, we have only extracted and summarized a few of the results from the reference.

Table III-2

Summary of Some Assembly 48
Predictions - Ref. (10)

	Crit. Mass (kg)	$k_{\text{eff}}-1$	Doppler Effect $\delta k(300-1100^\circ\text{K})$	$\sigma_{\text{p}}^{\text{Na center}}$ (mb)	$\sigma_{\text{p}}^{49 \text{ center}}$ (b)	$\sigma_{\text{p}}^{28 \text{ center}}$ (b)
Exp.	272	0	$-4.9 \pm .2 \times 10^{-6}$	-4.8	3.54	-.194
% Spread						
Plus	15.4%	2.4%	5.3%	119%	9.5%	17%
Minus	23.2%	3.6%	25.7%	117%	7.3%	22.6%

Comparison of Davey's and Okrent's results show very similar trends. The error bracket for critical mass obtained by Davey's survey compares with the deviation from the median value obtained in Okrent's survey.

Davey's results show a relatively small spread for the Doppler coefficient, but a larger spread for sodium worth is indicated by the large discrepancies in the perturbation cross section. There appear to be no startling changes in the "state of the art" of fast reactor calculations during the year's time that elapsed between the delivery of these two papers.

To assess uncertainties of more specific interest to FTR configurations, a fairly extensive effort of theory-experiment correlation studies and calculations was undertaken by PNL on available fast critical experiments information germane to the FTR program. The results of this survey⁽¹⁾ were also presented at the 1966 ANL conference, together with a discussion of the proposed critical experiment program for FTR. This calculational survey spanned a rather wide range of neutron spectra and covered 14 fast critical experiment configurations. The calculations for these assemblies were carried out with the then current PNL 26 group cross section set (MOD-1). Detailed results of cross section ratios, reactivity coefficients and k_{eff} 's are given in reference (1). As soon as more detailed data became available on assembly 48, many of the calculations were repeated for this and other assemblies, with additional modifications made in the cross section set (MOD-2). These later results were published in BNWL-347.⁽¹¹⁾ In the following tables, some of the more important results contained in references (1) and (11) have been reproduced. In Table III-3, the k_{eff} 's for several assemblies obtained with the MOD-1 and MOD-2 cross section set are compared.

Table III-3

Comparison of ($k_{\text{eff}-1}$) for Several Critical Assemblies
Using Two Cross Section Sets - Ref. (1) and (11)

Assembly	$k_{\text{eff}-1}$ (MOD-1)	$k_{\text{eff}-1}$ (MOD-2)
ZPR-3 - 35	-.012	-.007
44	+.008	+.004
46B	-.015	-.019
47	+.014	+.006
48	+.012	+.002
ZEBRA - 2	-.024	-.024
3	-	-.009
6A	-	-.002

With the single exception of assembly 46B, use of the MOD-2 cross section set has resulted in a considerable improvement in the k_{eff} calculation for a number of assemblies.

With regard to sodium worth and sodium coefficients, the results of reference (11) show a discrepancy of 39% for central Na reactivity coefficients in assembly 47 (SEFOR), a discrepancy of $\sim 20\%$ in σ_p^{Na} for assembly 48, and discrepancies of 10-20% in σ_p^{Na} for the ZEBRA 1,2,3 assemblies. For ZEBRA 6A, the total sodium worth was calculated 14.3% less than measured.

While PNL did not participate in Okrent's original "Intercomparison", nuclear parameters for Okrent's 5/1 bare assembly were obtained with the MOD-2 data in reference (11). The results are shown below.

Table III-4

Summary of MOD-2 Calculations for Okrent's Bare,
5/1 Fast Assembly - Ref. (11)

	$B^2 \times 10^4$	% δk for 50% Na Rem.	Doppler Effect - Tdk/dT	$\frac{\sigma_f^{28}}{\sigma_f^{49}}$	$\frac{\sigma_c^{28}}{\sigma_f^{49}}$	$\frac{\sigma_c^{49}}{\sigma_f^{49}}$
Median	16.2	-3.64	.0052	.030	.14	.21
PNL Result	14.3	-3.05	.0054	.032	.15	.21
% Spread						
Plus	12%	23%	48%	7%	7%	9%
Minus	8%	24%	83%	10%	14%	24%
PNL	-11.7%	-16.2%	+4%	+6.5%	+7%	0%

The comparisons shown in Table III-4 demonstrate that the PNL calculational methods and cross section routines fit in quite well with what is considered to be the present "state of the art".

Both Davey's and Okrent's surveys include contributions from foreign installations. The information from European installations has been recently augmented by the IMFBR Physics Team visit to European Atomic Energy Installations.⁽⁷⁾ For instance, the French power reactor RAPSODIE went critical recently, and in spite of previous ZPR mockup experiments, the predicted and actual fuel loadings differed by $\sim 10\%$. A similar discrepancy (in the opposite direction) was also observed during the loading of the critical experiment MASURCA. Comparison of RAPSODIE's isothermal temperature coefficients (which largely consist of Na and structural expansion effects) also showed a discrepancy of $\sim 10\%$ between the predicted coefficient (based on ZPR data) and the measured coefficient.

In Germany, the recent SNEAK critical loaded with 504 kg of U-235 followed very closely the predicted value of 502 kg. This first SNEAK

loading was, of course, patterned on an existing experimental configuration (ZPR-3/41).

The present critical experiment in the British ZEBRA facility is ZEBRA-7 which is in direct support of PFR. Extensive experiments and comparison of calculations with experiments in the area of control rod worth and spatial reaction rate distributions have been done for ZEBRA-7. For central B_4C rods, errors in the rod worth of 6% to 14% were quoted by the British. For peripherally located rods, errors as high as 32% were mentioned. These error bands compare with the 10%-20% error quoted in reference (12). Analysis, to date, of the FTR Phase A critical experiments appears to give better agreement between measured and calculated rod worths.

A large number of reaction rate scans were performed in ZEBRA-7, and calculations have been done to compare with the experimental results. These comparisons are shown in Table III-5 for data normalized at the reactor center. The largest discrepancies occur fairly deep in the blanket regions.

Table III-5

Comparison of Calculated and Experimental
Spatial Reaction Rates, ZEBRA-7 - Ref. (7)

Axial Reaction Rates

Type of Reaction Rate	Location	Dev. from Exp.
U-235 (n,f)	Axial Breeder	-3%
Pu-239 (n,f)	" "	-8%
U-238 (n,f)	" "	+1%
(n, γ)	" "	-6%
B-10 (n, α)	" "	+1%

Radial Reaction Rates

Type of Reaction Rate	Location	Assembly	Dev. from Exp.
U-235 (n,f)	Core-Refl. Interface	Z-7B	-25%
Pu-239 (n,f)	" "	"	- 3%
U-238 (n,f)	" "	"	+ 4%
B-10 (n, α)	" "	"	-40%
U-235 (n,f)	Radial Breeder	"	-19%
Pu-239 (n,f)	" "	"	-40%
U-238 (n,f)	" "	"	-15%
B-10 (n, α)	" "	"	-30%
U-235 (n,f)	Core-Breeder Interface	Z-7C	- 4%
Pu-239 (n,f)	" "	"	- 3%
U-238 (n,f)	" "	"	+10%
B-10 (n, α)	" "	"	-10%
U-235 (n,f)	Radial Breeder	"	-12%
Pu-239 (n,f)	" "	"	-30%
U-238 (n,f)	" "	"	-12%
B-10 (n, α)	" "	"	-22%

B. Conclusions and Recommendations

The preceding survey of available data and commentaries on uncertainties in fast reactor physics parameters can be used as a basis for making some general judgements concerning the expected uncertainties in nuclear design calculations. By the nature of the data, such judgements contain a considerable amount of arbitrariness, and lack of precision.

It is fairly obvious that nuclear design parameters predicted at an early stage of design, based on simple calculational models, could differ greatly from the final measured design numbers. Consequently, the adoption of these large uncertainty ranges would be unrealistic. On the other hand, it would also be quite unrealistic to consider as typical, design numbers which had been generated after a long series of critical experiments and normalizations of calculational methods and nuclear data.

The assessment of uncertainties with which we are concerned here pertains to an intermediate situation where calculational paraphernalia and nuclear data have been brought up to a reasonable degree of sophistication and have been checked - and modified if necessary - against pertinent - though not specific - critical experiment information. Furthermore, our assessment obviously pertains only to uncertainties introduced via nuclear data and calculational methods, and are not due to reactor configurational changes brought about by design iterations.

With these provisions kept in mind, the adoption of the following typical nuclear design uncertainties that could be expected is proposed.

Table III-6

Recommended, Expected Nuclear Design Uncertainties

Parameter	Expected Uncertainty
k_{eff}	$\pm 2\%$
Doppler Coeff.	$\pm 25\%$
Sodium Worth	$\pm 25\%$
Local Power Density	$\pm 10\%$
Rod Worth	$\pm 20\%$

The expected deviations in k_{eff} of $\pm 2\%$ may be on the low side, particularly for cores like the FTR, which have very asymmetric, heterogeneous features.

The estimated 25% Doppler coefficient uncertainty may be somewhat high. However, very large discrepancies appeared in the Okrent Inter-comparison survey, and these should be accounted for. The estimated 25% uncertainty on the sodium coefficient is probably better established.

The 10% and 20% discrepancies for local power densities and rod worth estimates are based largely on the British data.

It is expected that the estimated uncertainties will be reduced with time, but the present values can serve as a convenient point of departure for estimating the presently expected uncertainties in nuclear design calculations.

IV. Effects of Calculational Uncertainties on the FTR Nuclear Design

In this section the problem of assessing the effects of calculational uncertainties on the FTR design will be considered. An attempt is made to include all the more important effects. However, since the FTR design is still undergoing substantial changes, additional uncertainties might come to light, later on, which could require further evaluation. This is particularly true in view of the fact that the recent conceptual design selection resulted in the adoption of the split-cone design. Most of the calculated parameters available at this time are specifically applicable to the previous, more compact, skewed FTR configuration, and apply to the split-cone design in only an approximate way. Since the split-cone design is more complex than a compact core design, one would expect the calculational uncertainties to be generally larger for the split-cone configuration.

A. Effects of Uncertainties in k_{eff}

Uncertainties in the initial fissile fuel inventory, i.e. in fuel enrichment, can be expressed as uncertainties in k_{eff} . The expected uncertainty of $\pm 2\%$ in k_{eff} , quoted in the previous section, is probably somewhat low for the split-cone geometry. A more conservative estimate might be nearer 3 to 4%.

If, at the time of the first FTR fuel loading, this uncertainty has been substantially reduced, one would proceed with the fuel loading as planned. A slight decrease in excess reactivity would simply reduce the duration of the first fuel operating cycle. If the excess reactivity is

slightly higher than expected, the reactor shutdown margins would be reduced somewhat. The reactor will have adequate safety margins built-in to absorb small discrepancies in the initial k_{eff} . It is expected that any secondary effects introduced by modest errors in initial k_{eff} can also be absorbed by the reactor design margins. For instance, errors in initial k_{eff} will affect the initial control rod positioning. This, in turn, will have an effect on the initial power distributions, and, hence, power and flux levels. The change in rod position will also have modest effects on the initial temperature and power coefficients. All these effects are expected to be small, for small errors in k_{eff} , and, hence, no appreciable change in the expected mode of reactor operation is expected.

If, on the other hand, uncertainties in k_{eff} of $\pm 2\%$ (or larger) persist to the time of the first fuel loading of the FTR, changes in the FTR fuel loading pattern may become necessary. A two-zone driver fuel loading pattern is being considered for the FTR. Consequently, two batches of driver fuel with two different enrichments will be available at the time of loading. Further, it is expected that excess driver elements of both enrichments will also be available. The difference in the fissile enrichment of the two driver fuels can now be used to shim the reactor so that the initial excess reactivity is as expected. While fuel shimming can be used to correct errors in k_{eff} , this procedure is not without penalty, because fuel shimming affects initial power distribution patterns and, hence, the flux level at which the reactor will operate.

Suppose a two-zone reactor has been fuel zoned to flatten the radial power distribution. Assuming that the two zones are equal, and that about 90 elements are involved, each zone contains 45 fuel elements. The enrichment of the two batches has been chosen in such a way that the initial excess reactivity requirements have been met, and the peak power densities are the same in each fuel zone. The resulting radial power distribution is sketched in Fig.: 1, Case A.

If the initial k_{eff} value had been miscalculated, say 2% too low, more of the highly enriched Zone II fuel would now be required and the interface would move toward the center. (Case B, Fig.: 1.) A 20% zone size change would be required approximately (i.e. 9 elements out of 45). The power density at the interface would become limiting. For a 2% change in k_{eff} , the power density change would be about 4%. (See Fig.: 2.) Consequently, the power level would have to be lowered, and the peak flux at the center would drop by about 7%.* (See Fig.: 2.) If, on the other hand, k_{eff} was calculated 2% too high, the inner, Zone I would have to be increased by about 20% (~ 9 elements) and the power density at the center would become limiting (Case C, Fig.: 1). Reduction of the power level would, in this case, not affect the central peak flux value to any appreciable extent.

* It is assumed here that the error in k_{eff} is not directly traceable to an error in σ_f .

Case A: Power Flattened

Case B: Interface Power Limiting

Case C: Central Power Limiting

FIGURE 1: POWER DISTRIBUTIONS FOR DIFFERENT FUEL LOADING PATTERNS

FIGURE 2: RADIAL DEPENDENCE OF POWER DENSITY AND PEAK FLUX

Another effect of the change in the fuel loading pattern would be a change in the Doppler coefficient. Case B would result in a reduction of the magnitude of the Doppler coefficient, while Case C would tend to increase it. Additional minor changes in the other temperature coefficients could also be expected.

Critical experiments are expected to reduce these uncertainties in k_{eff} , and thus reduce the uncertainties in the initial power and flux patterns and temperature coefficients. As the uncertainty decreases, it will be possible to more precisely define the optimum enrichments in the two fuel zones, and thus more closely approach optimum operating conditions in the reactor.

B. Effects of Uncertainties in the Doppler Coefficient

The magnitude of the negative Doppler coefficient is an important consideration in the safety analysis of the FTR. The relationship between fertile-to-fissile fuel ratio and the Doppler coefficient is almost linear (See Fig.: 3). Since the FTR is relatively small, and since the split-cone driver fuel arrangement leads to rather unfavorable fuel distributions, a relatively high enrichment fuel results and the Doppler coefficient tends to be small. For typical fertile-to-fissile fuel ratios in the FTR, one would conclude from Fig.: 3 that Tdk/dT is about $-.004$. The real FTR Doppler coefficient may, however, be less than this value, since it can be affected adversely by test fuel and control rod patterns.

To illustrate these effects a series of calculations were performed on an 800 lit zoned FTR core for a variety of control rod and test fuel patterns.

FIGURE 3: DOPPLER COEFFICIENT DEPENDENCE ON FERTILE-TO-FISSILE FUEL RATIO

For the clean, initial rod-free case the Tdk/dT value was established at $-.004$. This is obviously an idealized reactor model which, because of the absence of rods, has a $k_{eff} > 1$. Assuming that peripheral shim rods will be used to control the reactor, $k_{eff} = 1$ was established by inserting a poison ring between the core and reflector. This model now represents the reactor in the initial critical condition. Recalculation of the Doppler coefficient for this case resulted in a value of $-.0034$ for Tdk/dT . In general, test elements would tend to be less reactive than driver fuel, and consequently such low reactivity, high U-238 elements would improve (increase) the Doppler coefficient. However, it is possible that high reactivity test elements might be introduced, and the in-core test shim rods are provided to compensate for this type of test. This situation was simulated in another calculation where test elements (with a composition similar to that of Zone II drivers) were introduced into Zone I. The reactivity increase was about $.018 \Delta k$, and criticality was reestablished by inserting an in-core rod bank located in Zone II. Again, the Doppler coefficient was recalculated for this reactor model, and, as expected, a sharp decrease in Tdk/dT to a value of $-.0028$ was observed. In the table below, these three calculations are recapitulated.

Table IV-1

Doppler Tdk/dT as a Function of Core Configuration

	<u>Tdk/dT</u>
Rod-Free Core	$-.004$
Ref1. Rods Inserted	$-.0034$
Test Fuel, In-Core & Ref1. Rods Inserted	$-.0028$

At the end of life, the control rods would, of course, be further withdrawn, to compensate for driver fuel depletion and the presence of fission products. It is expected that, under these conditions, a further decrease in the Doppler coefficient would occur. In addition, if we now postulate errors of $\pm 25\%$ in the Doppler coefficient calculation, the Tdk/dT value might be as low as $-.002$. This is the value of Tdk/dT that one might want to use in reactor safeguards and dynamics studies.

A fairly extensive study has been carried out for a series of postulated accidents and feed-back coefficients. The details of this study are described in references (13) and (14). Some of the more important results will be briefly described here, in order to evaluate the effects of the uncertainties in the Doppler coefficient on the FTR design.

For purposes of the excursion studies, fuel pin "failure" is defined to be the point at which melting just begins in the hottest region of the reactor. Since the kinetics code used for these studies determines only the centerline temperature of the average fuel pin, a correlation between this temperature and the reactor peak temperature is required. While this correlation is not well known at present, a 300°C rise in the average temperature might be a fairly good representation of the failure point. Proceeding on this basis, Fig.: 4 represents the reactor response to various reactivity ramp rates. In constructing this figure, it was assumed that the scram rods contain $\$20$ of negative reactivity, and that they fall with $1g$ acceleration. The rods are assumed to be positioned about 10 cm above the core and top reflector interface, and that they

FIGURE 4: VARIOUS ACCIDENTS FOR NOMINAL ROD CONDITIONS

start falling 200 msec after the start of the accident. For a T_{dk}/dT value of $-.004$, it is seen that a ramp of $3.5 \text{ \$/sec}$ just grazes the failure line. Hence, for the specified conditions, the core would not fail under these circumstances.

The effect of a change in the Doppler coefficient is shown in Fig.: 5 for the $3.5 \text{ \$/sec}$ ramp. A decrease in T_{dk}/dT to $-.002$ would result in core failure. Assuming an overpower ratio of 1.2, a $3.5 \text{ \$/sec}$ ramp requires about 50 msec to be detected, so that a 200 msec total delay corresponds to an "electronic delay", t_d of 150 msec.

In order to prevent fuel failure in the case of the $3.5 \text{ \$/sec}$ ramp, and $T_{dk}/dT = -.002$, three alternatives might be considered: (1) increase the rod acceleration, (2) decrease the electronic delay time, or (3) position the scram rods deeper into the core, so that negative reactivity can be added at a faster rate initially. These three possibilities are identified on Fig.: 6 as Cases I, II and III, respectively. A $2g$ acceleration, a reduction of the electronic delay time to 120 msec, or an initial rod position 7 cm inside the core (i.e. a 17 cm displacement from the original position), all result in about the same fuel temperature excursion and equally prevent core failure. Of course, all three of these alternatives imply some design disadvantages compared to the reference core and illustrate the interrelations of Doppler coefficient uncertainties and FTR design choices.

While there are a variety of other situations which may be affected by uncertainties in the Doppler coefficient and which may be of lesser importance, the effect of the Doppler coefficient in case of a

FIGURE 5: 3.50 \$/SEC ACCIDENT FOR VARIOUS DOPPLER COEFFICIENTS

FIGURE 6: THREE POSSIBLE DESIGN CHANGES TO ALLOW SURVIVAL OF A 3.50 \$/SEC ACCIDENT IF THE DOPPLER COEFFICIENT IS -0.002

catastrophic core failure should be mentioned. In case of complete coolant loss and major core meltdown, the Doppler coefficient would be further reduced because of the spectral hardening that occurs in the absence of sodium. (T_{dk}/dT might perhaps be as low as $-.0015$.) Energy release studies for this situation have been discussed by various authors (see for instance ref. (15)). Under certain conditions, if the Doppler coefficient is beyond a certain threshold value, the available energy of the excursion is relatively low, and fairly insensitive to the Doppler coefficient. If, however, the coefficient is below the threshold value, the available work increases sharply for smaller coefficients.

Again, for this reason, a knowledge of the Doppler coefficient, with some precision, is quite important for carrying out FTR safeguards analysis.

C. Effects of Uncertainties in Sodium Worth

Total sodium worth, and the spatial dependence of the sodium worth curve, can have an important effect on reactor safety. In large breeder reactors, the leakage component of the total sodium worth is minimized, and the possibility of having positive sodium coefficients in a design is enhanced. Even if the overall sodium coefficient is negative, the central regions of the reactor can make large positive contributions which are then compensated by the even larger negative contributions of the peripheral sodium.

The relatively compact, skewed FTR design did have an overall net negative sodium coefficient. However, near the core center the coefficient was positive. The split-cone FTR design is expected to display a

more complicated sodium coefficient pattern which will be affected by the kind of test fuel located in the central gap. The importance of uncertainties in the sodium-worth curve can, however, be understood quite well by examining the sodium worth behavior for the 800 lit FTR design. In Fig.: 7, a cumulative sodium worth curve for a typical FTR configuration is shown. This curve was taken from reference (16). The curve was generated, using a spherical reactor model, incorporating a depleted reactor condition including a modest BeO inventory. It can be seen that the total reactivity change for a fully voided core would be $\sim -2.5\%$, while the maximum positive void value would be $\sim +1.3\%$. The actual mechanisms that might lead to this type of central void formation are not evaluated here, and the probability of such an event actually occurring is judged to be negligible. Nevertheless, some quantitative assessments of the effects of uncertainties in this spatial worth function are instructive. For instance, to generate 1\$ of positive reactivity would require the voiding of about 14% of the core. If the uncertainties in the sodium worths were such that the positive values in the sodium void curve were 25% higher, then, to obtain 1\$ of positive reactivity would require the voiding of only $\sim 9\%$ of the core volume. (This is a decrease of $\sim 35\%$ compared to the 14% value.) A much larger core volume would have to be voided if a 25% negative error margin were postulated.

Clearly, a knowledge of the spatial sodium worth pattern is essential to carrying out meaningful safety analyses on the FTR design.

FIGURE 7: REACTIVITY CHANGE AS A FUNCTION OF PERCENT CORE VOIDED

Experimental measurements are expected to improve our knowledge of the spatial sodium coefficients to a considerable extent.

D. Effects of Uncertainties in Local Power Densities

Errors in the calculated local power distribution can have an important effect on the achievable flux level. If the limiting power density turns out to be too high, the reactor power level must be reduced correspondingly and, hence, the operating flux level is reduced. If reactor power is not reduced, the reactor is simply operating closer to the failure point, and in an excursion fuel failures could occur which otherwise might have been avoided.

Early knowledge of discrepancies in the calculated and measured spatial power distributions provide an opportunity to make design changes which will remove these difficulties. Reaction rate measurements in a critical experiment can reduce these uncertainties.

E. Effects of Uncertainties in Rod Worth

An analysis of FTR control rod patterns for an 800 lit, zoned FTR design was recently completed.⁽¹⁷⁾ The results of this study can be used to assess the effects of uncertainties in control rod worth calculations on the overall FTR design.

The total control requirements for the FTR are about 12% Δk . Shim requirements are about 4%, test compensation requires about 2% Δk , and 6% Δk is provided for the safety shutdown margin. Test compensation and scram requirements will probably be met by in-core rods, while the shim requirements can be covered by rods placed in the side reflector.

In the 800 lit core, 36 reflector positions were available for rod placement. As shown in Fig.: 8, the rod bank worth for reasonable B_4C rod compositions is shown. Curves for separated, single rods, as well as rods bunched in groups of two, are shown. The use of 12 reflector rods would supply adequate shim control, $\sim .047 \Delta k$. Since the rod worth curve begins to saturate in this region, an overestimate in rod worth by 20% would imply an increase in the number of required rods of about 25%. I.e. 15 rods, instead of 12 rods would be needed. Another solution would be to use B^{10} enriched B_4C instead of natural B_4C . Referring to Fig.: 9, it is seen that an increase in required rod worth of 20% would change the B^{10} enrichment of a 12 rod bank from 20 atom % B^{10} to ~ 26.5 atom %.

The Phase A, FTR Critical Experiments have provided important anchor points for the evaluation of FTR rod worth, and it is now expected that as a result of these experiments, the calculation of reflector rod worths is on a reasonably sound basis. Nevertheless, it should be recognized that the relatively large split-cone design introduces sufficient complexity into the calculation of worth of eccentrically located rods, so that more detailed rod experiments during the Phase C mockup series will become necessary.

FIGURE 8: REFLECTOR CONTROL ROD WORTHS

-45-

FIGURE 9: REFLECTOR ROD BANK (12 RODS) WORTH vs B¹⁰ ENRICHMENT

V. Outline of Proposed FTR Critical Experiment Program

The schedule for the proposed FTR Critical Experiment Program is contained in reference (18). In this reference the overall scope of the program is delineated, and the relationship of the experiments to the FTR design effort are indicated. Since the issuance of this document, some FTR schedule changes have occurred. However, the objectives and proposed timing of the experimental program remain substantially unchanged.

As stated in reference (18), the proposed critical experiment program is divided into four phases, Phases A through D. Phase A of the program is essentially complete,⁽¹⁹⁾ though the analysis of the experiments is not yet completed. A brief summary of the scope of these experiments is given below.

1. Phase A - The main purpose of the Phase A experiments was to assess the validity of nuclear design methods relating to control rod worth calculations. Specific topics covered by these experiments were control rod worth measurements, control rod interaction and rod heterogeneity effects.

The results of the Phase A experiments directly affect FTR conceptual and Title I design efforts, and the results are also expected to be of use in the preparation of the Preliminary Safeguards Analysis Report (PSAR). In addition, the Phase A experimental results will be of utility in the design of the NPTF.

2. Phase B - The main purpose of the Phase B experiments is to provide experimental verification of the nuclear design information for Title II design and for use in the Preliminary Safeguards Report. Phase B experimental information is also expected to be used in the NPTF design.

3. Phase C - The main purpose of the Phase C experiments is to verify detailed nuclear design information needed for fuel specification and Title II design. In addition, Phase C experimental results will be used in the preparation of the Final Safeguards Analysis Report (FSAR) and to a lesser extent for the resolution of NPTF safeguards questions.

4. Phase D - The purpose of the Phase D experiments is to supply whatever additional, specific nuclear safeguards information that may be needed for the final safeguards report, and to formulate reactor startup procedures.

The tentative schedule for the entire proposed program is shown in Fig.: 10. Key dates in the overall FTR and NPTF schedule are also identified on this program layout.

A. Phase B Experiments

The importance of the Phase B experiments has been enhanced by the recent split-cone concept design choice. The gross asymmetry introduced by the vertical test zone in the split-cone FTR design appears to be at the root of the major physics problem of this design. The geometric features of the core require a full-fledged 3-D analysis. At

FIGURE 10: SCHEDULE INTERRELATIONSHIPS

the present time no adequate 3-D transport codes exist and, hence, a complete, rigorous, nuclear analysis of the FTR core is not possible. The only available analytical approach to the problem involves sequential 2-D analyses, or possibly synthesis methods. Both of these approaches involve many approximations and, therefore, extensive verification critical experiments are essential.

It is proposed that the forthcoming Phase B experiments consist of the following stages:

- (I) Uniform Cylinder
- (II) Uniform Conical Core (simulated by stacked disks)
- (III) Split Cone
- (IV) Split Cylinder
- (V) Non-uniform Conical Core (simulated by stacked disks)

These experiments would be carried out in the ZPR-3 facility. This is the only facility available in the U.S. at the moment which can accommodate reasonably large Pu inventories.

Since the split-cylinder geometry contains the salient FTR gap feature, and at the same time affords the opportunity to do at least some meaningful calculations, most of the detailed experiments are expected to be carried out for this geometry. A considerably less extensive series of measurements is proposed for the other core geometries.

The following experiments are proposed:

Stage I - Uniform Cylinder

1. At least two Doppler coefficient measurements at two different core locations.
2. Sodium void effects, via removal of large sodium volumes at select core locations.
3. Spatial reaction rate measurements.
4. Spatial reactivity coefficient measurements of core materials.
5. Rossi alpha.
6. Spectrum measurements.

Stage II - Uniform Conical Core

1. Critical mass determination for at least two different combinations of disk sizes.

Stage III - Split Cone

1. Critical mass determination.

Stage IV - Split Cylinder

1. Doppler coefficient measurements at two different core locations.
2. Measurements to test the effect of gap size, shape and composition on core neutronics.
3. Sodium void effects, via removal of large sodium volumes at selected core locations.
4. Spatial reaction rate measurements.

5. Spatial reactivity coefficient measurements of core materials.
6. Rossi alpha.
7. Spectrum measurements in both the core and gap.
8. Control rod measurements.
9. Major fuel relocation and replacement effects.
10. Rod-test interaction experiments.

Stage V - Non-uniform Conical Core

1. Critical mass determination for at least a two-region conical simulation. Each zone is to have a different volume fraction of fuel.
2. Doppler coefficient measurements in each zone.
3. Spectrum measurements in each zone.

In order to make immediate plans for the ZPR-3 experiments, a reference core of about 1000 liters with a 600 kg (Pu-239) critical mass has been postulated. Inconel would be used in the side reflectors rather than a nickel-SS mockup. An attempt will be made to avoid extensive simulation of Na with low density Al.

Preliminary experiment design calculations have been completed on the reactivity decrement and critical mass increases associated with conicality and with the introduction of a test zone in a critical right cylinder. The 950 liter (150 liter test zone and 800 liter driver zone) split-core reactor presently being considered in the nuclear design calculations was taken as the point of departure.

Using the driver fuel region composition, the size of a uniform cylindrical core volume was found. The removal of the gap results in a very sharp reduction of core size and, hence, fissile fuel inventory. This core configuration will be the first proposed Phase B critical geometry.

A further modification was then made to represent a conical core by means of two stacked disks. This configuration represents the second proposed Phase B critical geometry. In Table I the two geometries are compared.

Table I

	<u>Right Cyl.</u>	<u>Two Disks</u>
k_{eff}	1.0005	1.0005
Top Core Radius cm	39.75	46.75
Bot. Core Radius cm	39.75	34.75
Refl. Radius cm	68.674	68.674
Height cm	85.00	42.5, 42.5
Core Volume liter	422	453
Critical Mass Pu ²³⁹ kg	268.5	288.20
Mass Pu ²⁴⁰ kg	47.4	50.85
Total Mass Pu kg	315.9	339.05
Equivalent angle of conicality (degrees)	0	8

In view of the continuing evolution taking place in the FTR conceptual design effort, it is important to emphasize the prelimi-

nary nature of the above calculations. More detailed experiment design calculations are presently under way.

B. Phase C and Phase D Experiments

As a result of the Phase B experiment-theory correlation, methods of calculation will be selected for development of the final nuclear design. The resultant design will be verified experimentally in Phase C for inclusion in Title II, for specification of fuel composition and for the final safeguards report. Therefore, the experiment period is proposed as being from 1/69 to 10/69.

Final FTR core and reflector design is to be simulated as closely as possible consistent with the fuel geometry and composition limitations of ZPR-3. Phase C experiments will encompass detailed control and safety rod simulation, in both size and location. Doppler and sodium void coefficients are to be measured for a variety of reactor configurations. The effects of test loops and test materials on the reactor neutronics will be established.

The Phase D experiments are quite unspecified at the present time. Their major function will be to supply information needed for the resolution of the FSAR and for the planning and initiation of the FTR startup tests. If the tight NPTF schedules can be met, it is possible that the entire Phase D program can be performed in this proof test facility. This procedure might be preferable, since the NPTF is expected to be a much better mockup of the FTR than what can be achieved in ZPR-3. Whether or not a parallel Phase D program will be needed, and other details of the Phase D program have not yet been resolved.

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Atomic Power Development Associates

BDV Farris

Babcock & Wilcox Company

SH Esleeck
TP Farrell

General Atomic Division (SAN)

D. Coburn (2)

General Electric Company

Dr. K Cohen (3)
Bertram Wolfe

Idaho Nuclear Corporation

DR deBoisblanc

Idaho Operations Office

CW Bills, Director

Office of Sr. RDT Site Rep (AEC)

Atomics International
Argonne National Laboratory
Atomic Power Development Assoc.
General Electric
Pacific Northwest Lab. (3)

Richland Operations Office

CL Robinson, Director (2)
JH Krema

San Francisco Operations Office

Director, Reactor Division

Stanford University

Dr R Sher

Westinghouse Electric Corp.

Dr JCR Kelly

Battelle Northwest

	<u>Bldg</u>	<u>Area</u>		<u>Bldg</u>	<u>Area</u>
GE Akre	703	700	PL Hofmann (20)	703	700
WG Albert	703	700	JE Irvin	703	700
ER Astley	703	700	RL Junkins	Fed Bldg.	
WJ Bailey	308	300	JH Kinginger	703	700
QL Baird	3702	300	RL Knecht	321	300
RE Bardsley	308	300	DJ Lagrou	703	700
EF Barrett	324	300	DD Lanning	3702	300
AL Bement	326	300	HD Lenkersdorfer	703	700
RA Bennett	703	700	WR Lewis	703	700
DE Bloomfield	703	700	CW Lindenmeier	703	700
JR Boldt	703	700	HE Little	703	700
CL Boyd	703	700	WW Little	703	700
DC Boyd	703	700	RJ Lobsinger	308	300
CL Brown	209E	300	MK Mahaffey	703	700
WL Bunch	703	700	WB McDonald	703	700
CA Burgess	308	300	MH Meuser	703	700
MP Burns	308	300	MK Millhollen	308	300
CP Cabell	703	700	DM Nero	703	700
JR Carrell	703	700	RE Nightingale	703	700
WL Chase	703	700	LJ Nitteberg	703	700
TD Chikalla	308	300	WM Noble	703	700
TT Claudson	326	300	A Padilla	703	700
PD Cohn	308	300	MG Patrick	703	700
DL Condotta	703	700	CE Peckinpugh	703	700
RR Cone	703	700	LA Pember	308	300
KA Creek	703	700	JA Perry	703	700
TJ Crocker	314	300	RE Peterson	703	700
RE Dahl	3701L	300	HG Powers	308	300
JM Davidson	703	700	FG Richards	703	700
FG Dawson	326	300	WE Roake	308	300
DR Doman	703	700	JR Samuel	703	700
GE Driver	703	700	JD Schaffer	703	700
RV Dulin	703	700	DP Schively	703	700
SL Engstrom	703	700	FH Shadel	703	700
JF Erben	703	700	DW Shannon	328T	300
PL Farnsworth	308	300	BO Shaver	703	700
LM Finch	703	700	DE Simpson	703	700
JK Flickinger	703	700	RJ Squires	703	700
CD Flowers/MT Jakub	703	700	WF Stevenson	703	700
CJ Foley	703	700	GH Strong	703	700
GL Fox	308	300	CD Swanson	703	700
EE Garrett	703	700	G Thieme	703	700
RL Gibby	308	300	JP Thomas	703	700
SM Gill	703	700	GL Tingey	326	300
JE Hanson	308	300	WL Townsend	314	300
H Harty	328	300	KG Toyoda	321	300
RA Harvey	703	700	N Vieli	703	700
BR Hayward	308	300	DM Walley	703	700
JW Helm	703	700	AE Waltar	703	700
LG Henke	703	700	JH Westsik	703	700
RJ Hennig	703	700	LA Whinery	703	700
JD Herb	703	700	RD Widrig	703	700
CW Higby	703	700	MF Wiitala	703	700
NA Hill	326	300	MR Wood	703	700

Battelle Northwest, Continued

	<u>Bldg.</u>	<u>Area</u>
FW Woodfield	703	700
JM Yatabe	321	300
HH Yoshikawa	326	300
FFTF Record File	703	700
FFTF Reference File	703	700
Tech. Info. File (3)	3760	300