-COO-2826-2

DNB-76-116

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ABSTRACT

The work performed in the reporting period was primarily concerned with the development of the experimental program (Task A) and with the pre-analysis of future loadings and the impact upon the permanent loading of the two converter regions, which contain 4.8% enriched UO_2 rods.

It appears necessary that a neutron poison (B_4C) be placed in the converter (transformer) regions in order to hold, also for future loadings, the k_{eff} of a hypothetically flooded FBBF well below 1. Since it is planned to use the same welded converter regions for all experiments, the required B_4C loading needs to be determined prior to the first blanket loading.

Further, the equipment needs have been identified (Task D), the ^{252}Cf -source has been requested on a loan basis (Task E). First discussions with ANL on blanket experiments have been initiated (Task F).

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Task A (K. O. Ott, F. M. Clikeman, R. H. Johnson, and R. C. Borg)

EXPERIMENTAL PROGRAM FOR THE FAST BREEDER BLANKET FACILITY, FBBF

I. INTRODUCTION

The "detailed" experimental program for the FBBF is being worked out in two phases. In a first phase, the overall structure of the detailed program is being established. This phase is described in this Report. The second phase, which stretches out over most of the program, is concerned with the detailed specifications of each loading, with the specifications of the experiments, and with the sequencing of the various loadings. The second phase will be strongly influenced by pre-analysis and by the experience from previous measurements and their theoretical interpretation. Sufficient flexibility in the experimental program is probably a requirement for achieving the General Objective "to reduce or eliminate the rather large uncertainties found at present in the predicted physical parameters of fast reactor blankets." This flexibility will be needed during the second phase of the determination of the detailed experimental program.

II. STRUCTURE OF THE EXPERIMENTAL PROGRAM

A) Categories and Guidelines

The overall experimental program includes the two basic categories of integral experiments:

- (1) "Clean" experiments, which are laid out for unquestionable comparability with respective theoretical calculations.
- (2) "Engineering mock-up" type of experiments, which are laid out to closely simulate actual complicated configurations as they may occur e.g. in rotating subassemblies with skewed plutonium distribution.

Emphasis is first placed on the clean integral experiments, which have the character of "benchmark experiments."

The detailed experimental program is being developed under three guidelines:

- (1) The experiments must be safe in the sense that the k_{eff} of a hypothetical complete flooding of the FB8F, remains well below 1, including computational errors.
- (2) The search for possible causes of the current inaccuracies in predictions of blanket parameters is facilitated by the investigation of various loadings which differ in only one compositional component.
- (3) The differences between the various FB8F configurations should be maximized so that they provide experimental information with a good "resolution," suitable for theoretical interpretation.

B) The Basic Variations

There are basically four compositional components which can be varied in the FB8F:

- (1) the pitch of the fuel rods,
- (2) the secondary cladding,
- (3) the fuel enrichment (possibly by regions),
- (4) the blanket thickness.

These three basic variations have a hierarchical order which results from the degree of "effort" for each change. This effort includes the "time" required for a change and thus the labor costs, as well as possibly some additional mechanical components:

- (1) The most complicated change, a change of the highest (first) level of the hierarchy, is a variation of the pitch of the fuel rods. It requires a new grid plate and a complete dismantling and reconstruction of the blanket. The converter regions remain unaffected.

(2) The second level of the hierarchy consists of secondary cladding changes. It also requires a complete reconstruction of the blanket, but for a given grid plate.

(3) On the third level, only the U-enrichment is changed, within the same secondary cladding.

(4) A change of the blanket thickness is the easiest variation (fourth level).

C) Classification of the Basic Variations

The following variations of the basic compositional parameters are considered. The guideline of maximum variation has been applied in defining these variations. A convenient notation scheme is introduced to identify the specific components in the various loadings.

(1) Pitch Variations

P1 denotes the "normal" pitch, the pitch in the first loading.

P0 denotes a "close" pitch, for which the fuel rods are in close contact with each other, without secondary cladding tubes.

P2 denotes a "wide" pitch (the value will be determined later).

(2) Variations of Secondary Cladding

C1 denotes the "normal" secondary cladding, i.e. the stainless steel tubes of the current design (C1 = SS).

C2 represents an aluminum cladding of the same dimensions as the SS-cladding of C1.

C3 denotes the case without secondary cladding.

(3) Variations of Fuel Enrichment

The variations of the fuel may be performed on a regional basis. This gives the changes in fuel composition a large variability (although also the secondary cladding could be changed on a regional basis if so desired).

Three fuel enrichments are available: 4.8%, 1.3% enriched and natural uranium. Only the natural uranium is available in sufficient quantity to construct an entire blanket. The small number of 4.8% enriched UO_2 rods can only be used to fill an inner blanket region (which suffices for practical purposes). The 1.3% enriched UO_2 rods can fill a large blanket ring or a fuel blanket-half.

U1 denotes the "normal" blanket of natural UO_2 rods.

U2 denotes the 1.3% blanket of maximum thickness.

UX denotes collectively more complicated loadings, including also 4.8% enriched UO_2 rods. The specifications of these loadings is to be devised later.

(4) Variations of the Blanket Thickness

The first loading consists of a blanket on the low end of the range of blanket thickness values considered. Space is provided between the minimal blanket thickness (34 cm) and the reflector to enlarge the blanket up to a maximum thickness of 51 cm. A further increase would require the replacement of the current reflector cans. The letter B is used to label the blanket thickness variations.

B1 denotes the minimal (first) blanket thickness.

B2 denotes the enlarged blanket thickness.

Further outer blanket variations, such as a more detailed variation in thickness or the "moderation" of outer blankets may be considered later.

D) Basic Loading

The combinations of the compositional variations listed above give the various blanket loadings. The number of straightforward combinations is large. However, most of those combinations can be eliminated by

- physical constraints,
- safety constraints, or

- by other considerations.

Safety considerations require the converter regions to be blackened for thermal neutrons for the loading without secondary cladding. Appropriate additions of a neutron poison to the converter regions are currently under consideration.

(1) Pitch and Cladding Variations

The normal pitch, P1, is of interest with all three cladding variations:

(P1, C1), (P1, C2) and (P1, C3).

The close pitch, P0, allows apparently only the option C3, without secondary cladding:

(P0, C3).

The wide pitch, P2, allows in principle all three claddings. But C3 (without secondary clad) might be eliminated by safety calculations.

Thus, the following pitch-cladding combinations appear to be of interest:

(P1, C1), (P1, C2), (P1, C3), (P0, C3), (P2, C1) and (P2, C2).

(2) Fuel and Blanket Thickness Variations

All pitch and cladding variations are considered only in combination with the simplest fuel loadings in order to simplify the interpretation of the experiments. Thus, U1 and U2 are combined with P1 and P0. But, all of the more complicated fuel loadings (UX), appear to be advisable only together with the normal pitch-cladding combination.

The same holds for the blanket thickness variations which are considered in the "clean experiments" part of the program only for the normal blanket (P1, C1, U1, B1). All other combinations of P, C and U are considered only for a single blanket thickness.

Blanket thickness variations are clearly of practical interest for the blanket which simulates bred plutonium. But such experiments are considered to be part of the "mock-up experiments" phase of the program.

E) Summary and Discussion of Tentative Clean Experiment Loadings

The Table summarizes the clean-experiment loadings as they evolved from the application of the above guidelines. All loadings are part of a direct comparison scheme with just one parameter varied. Some of the combinations may be eliminated later on the basis of the pre-analysis if shown that certain effects appear essentially in linear combinations. The combinations are briefly discussed in the following:

1. $(P_x, C1, \dots)$ vs. $(P_x, C2, \dots)$ gives the impact of substituting Al for steel. This is important since all natural and 1.3% enriched UO_2 rods have Al cladding. The Al vs. steel substitution experiment gives the impact of this deficiency of all FBBF blanket simulations.

2. The same substitution is investigated for U2 and U3 to see if its impact can be linearly superimposed upon the change in fuel, or if there are nonlinear effects.

3. The $(P1, C2, \dots)$ vs. $(P1, C3, \dots)$ series gives added information on the impact of Al alone, i.e. not in its substitution with steel. These first three experimental comparisons should allow a correction for the Al cladding.

4. The $P_x, C_x, U1, B1$ vs. $(P_x, C_x, U2, B1)$ experiments show the combined effect of the 1.3% vs. natural ^{235}U content, unfortunately together with a slight change in smeared UO_2 density (due to the different diameter of both types of rods). Therefore a separate comparison with the variation of the smeared density alone is required:

5. The $(P1, \dots)$ vs. $(P0, \dots)$ vs. $(P3, \dots)$ experiments show the impact of the variation of the pitch. This information is of general importance, and not merely as completion of No. 4.

6. The blanket thickness variations are obvious difference experiments.

Table of Tentative Clean-experiment Loadings

(P1,C1,U1,B1), (P1,C1,U2,B1), (P1,C1,U3,B1)	(P0,C3,U1,B1) (P0,C3,U2,B1)	(P2,C1,U1,B1), (P2,C1,U2,B1)
(P1,C2,U1,B1), (P1,C2,U2,B1) (P1,C2,U3,B1)		(P2,C2,U1,B1), (P2,C1,U2,B1)
(P1,C3,U1,B1), (P1,C3,U2,B1) (P1,C3,U3,B1)		
(P1,C1,U1,B2), (P1,C1,U2,B2) (P1,C1,U1,B2)		

III. DESCRIPTION OF THE EXPERIMENTS

A. Introduction

The basic philosophy governing the design of experiments in the FBBF is twofold. First, the experiments should yield information useful in verifying and improving the calculational methods used in the blanket regions of fast reactors. Second, the experiments should yield some basic data or provide a means of evaluating basic nuclear data used in theoretical calculations. Within the guidelines of this general philosophy, three general types of experiments are being planned for the FBBF.

The first broad category of experiments is the systematic determination of reaction-rate data throughout the blanket region. This broad category of experiments will be carried out using two different techniques. The first technique is the activation of foils followed by the measurement of the foil activities. Among the more important activities included in this technique is the ^{239}Np activity from the irradiation of ^{238}U foils since this is a direct measurement of the ^{239}Pu breeding rate. The second technique is the use of solid state track recorders (SSTR) for the measurement of the fission rates of ^{235}U , ^{238}U , ^{237}Np , and ^{239}Pu . The ^{238}U and ^{239}Pu fission rates will give important information about the blanket fission rates and, hence, the heat generation rate in fast breeder blankets. These two fission rates together with the ^{235}U and ^{237}Np fission rates and the foil reaction rate data will provide input to computer codes such as SAND-II⁽¹⁾ for an indirect determination of the neutron flux.

The second broad category of experiments is a more direct determination of the energy dependent neutron flux in the blanket using proton recoil neutron spectroscopy to determine the neutron spectra. Higher energy regions

of the spectra will also be measured using a ^3He proportional counter. The neutron spectra determined with these gas counters will be compared with spectra determined from the computer code SAND-II (and from an analogous Purdue program, to be developed), using the foil activities and fission rates as input. Both of these experimentally determined spectra will be compared with the theoretically calculated spectrum providing another check of the calculational techniques and basic assumptions used in the computer codes.

The third main category of experiments will involve the use of thermoluminescent dosimeters for the measurement of gamma heating. This information will contribute to a better understanding of the spatial dependence of the heat generation in the breeder blankets.

Combined, these experiments will form a large pool of information for comparison with theoretical calculations. The comparison will either serve to verify calculational techniques and the basic nuclear data for the blanket and shield-type materials or will point out areas where improvement is needed. In addition, the experiments will provide accurate information concerning the distribution of plutonium breeding, fission rates, and gamma heating throughout the blanket as well as information on the neutron spectrum emerging from the breeder blanket.

The following sections will deal with the details of the experimental techniques for the FBBF, including the expected reaction rates.

8. Activation Experiments

One of the major experimental efforts with the FBBF will be the determination of neutron activation rates. The activation rates, determined as a function of radial and axial position in the fast breeder blanket mock-ups will be used in two ways. First, the experimentally determined rates will

be compared with theoretically determined activation rates giving a good test of the basic assumptions, including the spatial dependence and the basic nuclear data that are used for the theoretical calculations. This comparison with theoretical results will be on both the basis of relative activation distributions throughout the blanket and the absolute values of the neutron activation rates. The second use of the experimentally determined activation rates will be as input to computer codes such as SAND-II that determine neutron spectra from activation information. The CRYSTAL BALL code of Kam⁽²⁾ will also be investigated for possible use. A method for incorporating fine structure in the unfolded spectra based on neutronics calculations will also be investigated (see Ref. 3, for example). Any unfolding methods developed at Purdue for use with reaction rates in the FBBF will be evaluated for accuracy and effectiveness by comparing with the SAND-II code. The neutron spectra from the SAND-II type programs will be compared directly with spectra determined from proton recoil and ^3He spectrometers as well as with theoretically determined spectra.

Isotopes to be used in the neutron activation experiments were chosen to meet one of two criteria besides the obvious ones of detectable radioactive decay rates, reasonable half-lives, useable activation rates and availability. The first of the two criteria is that the neutron capture reaction be one that will normally occur in the blanket of a fast reactor. The most important of the reactions meeting this criterion is probably the $^{238}\text{U} (n, \gamma) ^{239}\text{U}$ reaction which is the process for the production of ^{239}Pu in the blanket. In addition, neutron capture in ^{238}U is also the largest capture rate in the blanket regions due to the large concentrations of ^{238}U .

in the blanket. Other neutron capture rates in which one might be interested would be capture in sodium and stainless steel (iron, chromium, and nickel). Measurements of reaction rates in these materials will be difficult because either the neutron capture reaction does not lead to a usable activity or the reaction rate is too small to produce detectable quantities of the radioactive isotope. In some cases, some information on the reaction rates could possibly be determined if separated isotopes of the element were available. However, at the present time we do not have enough cross section information available to be able to calculate whether or not the activities produced would be measurable in the FBBF.

The second criterion for selecting materials for activation measurements is the neutron spectrum dependence of the activation rates. Since the neutron energy dependence is expected to vary strongly with position in the blanket, a good test of theoretical predictions would be the spatial dependence of reaction rates with different neutron energy dependences. These reaction rates, together with other measured reaction rates, will be used in the SAND-II type computer codes. A list of the reactions currently being considered appears in TABLE I.

One reaction, the ^{238}U capture rate has been the subject of a M. S. Thesis⁽⁴⁾. In this case, the foil is naturally radioactive and contributes strongly to the background counting rate. In addition, ^{238}U will also fast fission producing radioactive fission products that complicate the analysis. Trace amounts of ^{235}U in the ^{238}U will also contribute to the fission products present. A common method of detecting the ^{239}Np decay from the reaction is to count the "103 keV" gamma peak and then subtract the background determined by measuring the foil before the irradiation. The fission product contribution is removed by simply letting the foil decay for 12 hours between the

irradiation and counting.⁽⁴⁾ In reality, the "103 keV" peak consists of three closely spaced peaks, the 106 keV gamma ray from ^{239}Np and the 99 and 100 keV X rays from ^{239}Pu . The ^{239}Pu X rays are caused by internal conversion of ^{239}Np gamma rays. In actual practice an energy window is set over the "103 keV peak" to count everything between 85 keV and 121 keV so as to detect the whole peak. However, such a broad window also includes counts from the 95 keV X ray peak from uranium.

This technique, although simple, does not completely correct for the fission products. Experiments using natural uranium foils irradiated in a relatively hard neutron spectrum, indicate that several percent of the activity attributed to ^{239}Np decay is coming from longer lived fission products. An improved version of this technique was developed that will permit the $^{238}\text{U}(n,\gamma)$ reactors to be measured in regions with a high capture to fission ratio. This technique will be used in the outer regions of the blanket where the neutron spectra is expected to be relatively soft.⁽⁵⁾

A second technique for determining the ^{239}Np decay rate is to use a Ge(Li) detector to detect the 210 and 277 keV gamma rays from the ^{239}Np decay. Bresesti et al.,⁽⁶⁾ using this technique report that these peaks are free of the interference of gamma rays from both fission products and the decay of ^{238}U . They found good agreement in comparing absolute activities of ^{239}Np using the Ge(Li) detector and a 4π β counter and the $\beta^- - \gamma$ coincidence technique. Using the Ge(Li) detector permits corrections for both the natural ^{238}U decay and fissioning in the sample. The problem with this technique is the relatively low efficiency of Ge(Li) detectors. This low detector efficiency may hinder its use as a routine method of determining relative ^{239}Np activities, but the technique is still the simplest one for determining the absolute activity of the more active foils. However, if a

12% efficient Ge(Li) (relative to a 3" x 3" NaI(Tl)) is available, calculations show that it could be used for making the relative and absolute measurements even in the outer regions of the core.⁽⁵⁾

C. Measurement of Fission Rates

The knowledge of fission rates through the blanket is basic to the determination of the power distribution. In addition fissions occurring in the outer regions of the blanket strongly influence the spectrum of neutrons leaking out of the system. Three techniques are commonly used in the determination of fission rates. The first technique is the measurement of specific fission products, the second is the use of small fission chambers, and the third is the use of solid state track recorders (SSTR).

The technique of determining the fission product activity involves the measurement of the gamma rays from specific fission products. This technique can be used over a very wide range of neutron fluxes and indeed is the only technique that can be used in the high neutron fluxes found in large reactors. The use of the technique implies a knowledge of the fission product yield, a quantity that is not well known for all fissioning isotopes and is also known to be dependent on neutron energy. This latter point makes its use very questionable, especially in regions such as the blanket where the neutron energy spectrum is known to have a strong spatial dependence. There is some indication that the yield of some fission products, e.g., ^{140}Ba and ^{103}Ru , may be constant with neutron energy, but more work is required.⁽⁷⁾ Because of the uncertainties in the yields and the poor sensitivity of the technique in low fluxes, the technique is not being considered for use in the FBBF at the present time.

Solid state track recorders (SSTR) employ the technique of sandwiching thin ($\sim 10\mu\text{ g/cm}^2$) fissile sources with a track recording material. Recoil

fission products then embed themselves in the recording material. In the past this technique has been used with photographic film and the tracks were counted in the developed film or with aluminum plates and the activity of the fission products is counted with radiation detectors. The use of film is usually precluded by the sensitivity of the film to gamma radiation. The counting of fission product activity has the uncertainty in fission product yields as has already been discussed. More recently the technique of recording the fission product tracks in mica or plastic films such as Mackrofol E has been developed. In this process, the irradiated plastic or mica is chemically etched. The microscopic areas near the fission tracks, having undergone radiation damage, are preferentially etched, enlarging the tracks so that they can be seen with the use of a microscope. This technique has proved to be very sensitive and has a detection efficiency approaching 95% for the detection of fissions in the source plates.⁽⁸⁾ The technique is insensitive to gamma radiation and has the added advantage of having no time dependence. This means that for very low fission rates the SSTR's may be left in the facility for long times until the desired number of tracks have been recorded. The disadvantage of the system lies in the fact that the tracks must be counted by eye with a microscope. This process could certainly be automated as proven by the techniques used in processing tracks in high energy physics, although these systems are usually very expensive, involving the use of medium sized computers. Simpler and less expensive systems have also been developed. A spark counting system similar to ones already developed at Purdue⁽⁹⁾ and other places has been constructed. This system will eliminate the major disadvantage of the SSTR technique.

Calculations have shown that by using $50 \mu\text{g/cm}^2$ ^{238}U sources, and a 10^{10} n/s ^{252}Cf source approximately 30 fissions per cm^2 -day will be recorded

at the blanket reflector interface. Under these conditions, the reaction rates can be determined to be about 1% in 35 days using a 10cm^2 fission source. Measurement of other fission rates of interest, such as ^{235}U , ^{237}Np and ^{239}Pu will be easier.

The relatively small size of the SSTR's is a definite advantage for neutron flux and spectrum measurements. The small size minimizes the flux perturbation. In addition it will be possible to design experiments to be placed in specially designed fuel rods to measure the fission rates both in the rods and in the regions between the rods. Such experiments will provide information on the self-shielding and heterogeneity effects in the blanket, especially near the blanket-reflector interface where these effects are expected to become important. This is desirable since FBBF mock-up the correct heterogeneity, having approximately the same diameter rods as in the blanket of a large LMFBR.

The third important technique for determining fission rates involves the use of fission counters. In this technique, ionization chambers are loaded with fissile material, usually enriched in the particular isotope of interest. The technique measures fission counting rates which can be directly related to the fission rate. This technique is probably the fastest of the three techniques described. The disadvantage is that a separate detector must be fabricated for each of the fissile materials of interest. In addition, because of the small quantities of material that can be used in these detectors, counting rate for ^{238}U and ^{232}Th would be very low and not feasible for use in the FBBF. However, reaction rates in ^{235}U , ^{239}Pu and possibly ^{237}Np fission chambers would be adequate for measurement.

Work at Argonne National Lab^(7,10) has shown that fission chamber measurements and SSTR measurements are equal within 2-3% for these isotopes.

Another major disadvantage of the fission chambers is their size relative to the SSTR's. Although the size of the fission chambers is small, the chamber must be connected to its electronics with a cable. Together the chamber and cable perturb the neutron flux. The smaller size of the SSTR's will perturb the flux less.

The results of the fission rate measurements will be compared directly with theoretically predicted reaction rates and will also be used as input to neutron spectra codes such as SAND-II.

D. Neutron Spectrum Measurements

The energy dependence of the neutron spectra in the FBBF will be determined as a function of position throughout the blanket regions of the facility. Direct measurements of the neutron spectra will be made using proton recoil spectrometers and a ^3He proportional counter. Indirect measurements of the spectra will use the fission rate and neutron activation rate data previously discussed, together with the computer codes such as SAND-II.

The technique of making neutron spectrum measurements using proton recoil proportional detector has been well established in the literature.^(11,12) The technique has been used at Purdue for measuring neutron spectra in shielding experiments since 1969.⁽¹³⁻¹⁶⁾ The technique analyzes the proton recoil energy distribution from neutron scattering events. The proton energy distribution is then corrected for end and wall effects and is unfolded to yield the neutron energy distribution.⁽¹⁷⁾ For neutron energies greater than 100 keV, gamma interactions in the detector can be neglected or eliminated by simple pulse height analysis. In the neutron energy range from 1 keV to 100 keV, pulse shape discrimination (PSD) techniques must be used to eliminate the gamma-ray events. Bennett and Yule^(11,12) perform

PSD using the specific ionization of each event. This technique is adequate for measurements made in the cores of fast-reactor mock-ups. In the blanket region of the FBBF, however, the gamma-ray background will be very high compared to the neutron count rate. Therefore, a two-parameter data acquisition system (pulse height versus pulse rise time) will be used to achieve better PSD (see Ref. 18, for example). This improved PSD technique is necessary for extending the neutron spectrum measurements to as low an energy as practical.

The proton recoil neutron spectrometer system requires the use of two different detectors to cover the neutron energy range from 1 keV to 10 MeV. A hydrogen filled detector is used in the low energy range up to 1 MeV, while a second methane filled detector is used in the region from 100 keV to 10 MeV. A ^3He neutron spectrometer proportional detector has been ordered. Experiments are now being planned to study the characteristics of the ^3He detector and possibly to use it for neutron measurements in the 100 keV to 2 MeV range. If the experiments with the ^3He detector are successful, it will probably replace the methane filled proton recoil detector for measuring neutron spectra above 100 keV.

Bennett and Yule use the differentiation method of spectrum unfolding. Their unfolding code PSNS-N utilizes a least-squares method for numerical differentiation. A more accurate numerical differentiation method (which also gives a better-defined energy resolution) has been developed⁽¹⁹⁾ and is being implemented in the Purdue version of PSNS-N. This improved version of PSNS-N will be used to unfold the proton-recoil spectra measured in the FBBF.

The experimentally determined neutron spectra at various positions throughout the blanket and reflector will be compared directly with theoretically calculated neutron spectra. This comparison will be an extremely sensitive test of the theoretical results. Comparison of the reaction rate data is in reality the comparison of the energy integrated product of the cross section and flux and results in a single number. Although a great deal of information can be gained by the comparison of the relative and absolute counting rates with theory, a better test of the theory is the comparison with the experimentally determined neutron flux. This of course depends on the resolution with which the neutron energy spectrum can be determined.

The one disadvantage of the neutron spectrometers is their size. The detector and their cables tend to perturb the neutron flux, although Bennett has shown this to be small. Because of the size of the detectors, comparison of the energy spectrum inside the fuel rods with the flux between the rods will be impossible with the spectrometers. For the measurements of the heterogeneity effects in the blanket, integral reaction rate data will have to be used.

E. Gamma-Ray Dosimetry Experiments

Gamma-ray dosimetry measurements taken throughout the blanket region will yield valuable information about the transport of energy by gamma rays. The measurements will be taken using thermoluminescent dosimeters (TLD). Similar experiments have been made at Argonne National Laboratory using the ZPR-3 and ZPPR facilities. (20-24)

The use of TLD's is simple and straight forward, involving the irradiation of material such as LiF and then measuring the light from the dosimeter as it is heated. In practice the interpretation of the measurements made

with TLD's is considerably more complicated. First, corrections must be made for the neutron response of the TLD's. Approximately 10% of the response of LiF dosimeters in the core regions of fast reactors is attributed to fast neutrons.⁽²³⁾ Correction for the fast neutron response can be made if the fast neutron spectrum is known. Measurements of the neutron spectrum in the blanket have already been discussed. Calibration of batches of the TLD detectors will be made using the Co-60 irradiation facilities of the Bionucleonics Department at Purdue.

The remaining difficulty in analyzing the results of the TLD measurements is relating the absorbed dose in the TLD to the different materials present in the reactor. Work carried on at Argonne National Laboratory^(21, 22, 24) has compared the results of TLD experiments in a variety of materials ranging from low to high Z material and using both small and large cavity theory. The results indicate that TLD dosimetry measurements agree well with theoretical calculations.⁽²⁴⁾

The first experiments at Purdue will use sets of $\text{CaSO}_4:\text{D}_y$ TLD detectors. Investigations with higher-Z TLD devices have been carried out and results indicate that such devices better match the high-Z material in the blanket and will require a smaller correction factor.⁽²³⁾

IV. CALIFORNIUM-252 SOURCE SIZE

The ^{252}Cf source which will be used with the first loading of the FBBF will have a strength of approximately 10^{10} neutrons/sec. This source strength will suffice for most of the planned experiments in the inner blanket. For measurements in areas of the blanket where the neutron flux is very low (the outer part of the axial blanket, for example), a larger source strength may be necessary. Neutronics calculations cannot resolve this question, however, because various uncertainties⁽²⁶⁾ in calculated and measured neutron fluxes. Neutron flux mappings for the first FBBF loading will determine whether a larger ^{252}Cf source (with strength of approximately 10^{11} neutrons/sec) may have to be procured.

TABLE I

Preliminary set of activation foils for FBBF

<u>Foil</u>	<u>Reaction</u>	<u>Remarks</u>
Depleted Uranium	$^{238}\text{U}(n,\gamma)^{239}\text{U}$	-
Gold	$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	Resonance ~ 5 eV
Cobalt	$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	Resonance ~ 132 eV
Iron	$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	Resonance ~ 230 eV
Manganese	$^{55}\text{Mn}(n,\gamma)^{56}\text{Mn}$	Resonance ~ 337 eV
Molybdenum	$^{98}\text{Mo}(n,\gamma)^{99}\text{Mo}$	Resonance ~ 480 eV
Copper	$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	Resonance ~ 580 eV
Sodium (salt)	$^{23}\text{Na}(n,\gamma)^{24}\text{Na}$	Resonance ~ 2710 eV
Scandium	$^{45}\text{Sc}(n,\gamma)^{46}\text{Sc}$	Resonance ~ 5000 eV
Neptunium*	$^{237}\text{Np}(n,f)$	Threshold ~ 0.5 MeV
Depleted Uranium*	$^{238}\text{U}(n,f)$	Threshold ~ 1.45 MeV
Thorium*	$^{232}\text{Th}(n,f)$	Threshold ~ 1.75 MeV
Iron	$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	Threshold ~ 2.2 MeV
Nickel	$^{60}\text{Ni}(n,p)^{60}\text{Co}$	Threshold ~ 2.9 MeV
Aluminum	$^{27}\text{Al}(n,p)^{27}\text{Mg}$	Threshold ~ 3.3 MeV
Magnesium	$^{24}\text{Mg}(n,p)^{24}\text{Na}$	Threshold ~ 6.3 MeV
Aluminum	$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	Threshold ~ 8.7 MeV
Tantalum	$^{181}\text{Ta}(n,\gamma)^{182}\text{Ta}$	Approximately 1/v
Enriched Uranium*	$^{235}\text{U}(n,f)$	
Plutonium*	$^{239}\text{Pu}(n,f)$	

List of References

1. W. N. McElroy, S. Berg, T. B. Crockett, and R. J. Tuttle, "Measurement of Neutron Flux Spectra by a Multiple Foil Activation Iterative Method and Comparison with Reactor Physics Calculations and Spectrometer Measurements," Nucl. Sci. Eng., 36, 15 (1969).
2. F. B. K. Kam and F. W. Stallmann, "CRYSTAL BALL, A Computer Program for Determining Neutron Spectra from Activation Measurements, Oak Ridge National Laboratory, ORNL-TM-4601 (June 1974).
3. Y. D. Harker, "An Application of Semi-Empirical Modeling and Non-Linear Regression to Unfolding Fast Neutron Spectra from Integral Reaction Rate Data," to be published in the Proceedings of the RSIC Seminar-Workshop on Unfolding Radiation Energy Spectra, Oak Ridge National Laboratory, 12-13 April 1976 (to be published as ORNL-RSIC-40).
4. T. C. Leung, M. J. Driscoll, I. Kaplan and D. D. Lanning, "Neutronics of an LMFBR Blanket Mock-Up," COO-3060-1 (1962).
5. K. R. Boldt, "Preliminary Analysis of the FBBF Experiment to Determine Uranium-238 Capture Rates by the Detection of Neptunium-239 Decays," M. S. Thesis, Purdue University (December 1976).
6. A. M. Bresesti, M. Bresesti, and H. Neumann, J. of Nucl. Eng. 23, 379 (1969).
7. C. E. Crouthamel and N. D. Dudey, "Reactor Development Program Progress Report," ANL-RDP-14, 6.7 (Feb. 1973).
8. K. W. MacMurdo, R. M. Harbour and R. W. Benjamin, J. Inorg. Nucl. Chem. 33, 1241 (1971).
9. Phillip H. Jenkins, "A Comparison of the Use of Photographic Emulsions and Plastic Materials for Autoradiography of Californium-252 in Bone," M. S. Thesis, Purdue University (1972).
10. C. E. Crouthamel and N. D. Dudey, "Reactor Development Programs Progress Report," ANL-RDP-9 (Sept. 1972).
11. E. F. Bennett, Nucl. Sci. Eng. 27, 16 (1967).
12. E. F. Bennett, Nucl. Sci. Eng. 27, 28 (1967).
13. C. K. Paulson, "Fast Neutron Spectral Measurements in Water Using Proton-Recoil Detectors," Ph.D. Thesis, Purdue University (1970).
14. C. D. Livengood, "Development of a Fast Neutron Spectrometer and Measurement of Neutron Spectra Transmitted Through Iron Slabs," Ph.D. Thesis, Purdue University, (1970).

15. G. M. Ault, "An Investigation of Heterogeneity Effects on the Transmitted Neutron Spectrum of Ordinary Concrete," M. S. Thesis, Purdue University (1971).
16. J. T. Owens, "An Investigation of Neutron Streaming in a Two-Legged Duct," M. S. Thesis, Purdue University (1972).
17. E. F. Bennett, R. Gold, and I. K. Olson, "Analysis and Reduction of Proton-Recoil Data," ANL-7394 (1968).
18. M. Obu, T. Ichimori and K. Shirakata, Nucl. Inst. Methods **89**, 131 (1970).
19. R. H. Johnson, "Improvements in Differentiation Unfolding of Radiation Spectra," to be published in the Proc. of the RSIC Seminar-Workshop on Unfolding Radiation Energy Spectra, Oak Ridge National Laboratory (April 1976).
20. N. D. Dudey, R. R. Heinrich, R. S. Popek, R. P. Larsen, and R. D. Oldham, "Activation-Rate Measurements in the ZPR-3 Mock-up Critical Experiments," ANL-7781 (1971).
21. G. G. Simons, A. P. Olson, G. L. Brasseschi, and T. S. Huntsman, "Analysis and LiF TLD Measurements of Proton Heating in ZPPR Assembly 2," Argonne National Laboratory, ZPR-TM-122 (Sept. 1972).
22. G. G. Simons and T. J. Yule, "Gamma-Heating measurements in Zero-Power Fast Reactors with Thermoluminescent Dosimeters," Argonne National Laboratory, ZPR-TM-128 (Dec. 1972).
23. T. J. Yule, "A Survey of Techniques, Including the Gas-Filled Ionization Counter, For Measuring Gamma-Ray Heating in Low-Power Fast-Reactor Environments," Argonne National Laboratory, ZPR-TM-140 (April, 1973).
24. G. G. Simons and L. L. Emmons, "Experimental Evaluation of a Model for Calculating the Gamma-Ray Response of Encapsulated Solid-State Dosimeters," ZPR-TM-238, Argonne National Laboratory (1976).
25. J. L. Westbrook, "An Investigation of the Use of Thermoluminescent Dosimeters to Measure Gamma Heating in the Fast Breeder Blanket Facility," M. S. Thesis, Purdue University (August, 1976).
26. K. D. Ott, K. R. Boldt, and F. M. Clikeman, "Review of Nuclear Physics Uncertainties in Fast Reactor Blankets," Department of Nuclear Engineering, Purdue University, PNE-75-103 (March 1975).

Task B (R. C. Borg)

The object of this task is to perform a detailed pre-analysis of the first experimental sequence. This task has not been completed because design modifications are necessary for future enriched blanket loadings. The insertion of a high thermal neutron absorber into the transformer regions is needed to keep k_{eff} of the hypothetical flooding accident as low as possible. Since the welded transformer regions will be the same for all blanket loadings, this design modification must be determined prior to the further detailed pre-analysis of the initial loading.

To enhance the safety features of the FBBF for the hypothetical flooding accident, a large thermal neutron absorber in the form of B_4C powder will be vibrated into the inner and outer transformer regions. The impact of B_4C on the fast neutron spectrum, during normal operation must be evaluated in order to assure minimal distortion.

To investigate the effect of the thermal neutron absorber on the fast neutron spectrum the 1-dimensional model given in Ref. 1 was slightly modified by inserting various concentrations and enrichments of B_4C into the voided volume of the transformer regions. Reference 1, Task B describes the calculational model and gives the number densities and volume fractions used for the investigation. The boron and carbon number densities were determined based on B_4C at 80% theoretical density. Four enrichments were initially considered, natural boron, 35%, 50% and 100% B-10 enriched B_4C . Number densities for the first and last case are given below.

Natural B_4C

(20% B-10, 80% B-11)

C-12	2.19805×10^2 atoms/barn cm
B-10	1.74085×10^2
B-11	7.05133×10^2

Enriched B_4C

(100% B-10)

C-12	2.19805×10^{-2} atoms/barn cm
B-10	8.79218×10^2

The voided volume fraction available for B_4C in the inner and outer transformer regions is 0.116 and 0.051 respectively. B_4C was assumed to occupy all of the available space for the enrichment variations investigated.

Figures 1 and 2 illustrate the spectral comparison of the FRBF transformer-blanket interface for five different B_4C enrichments. The behavior of groups 11 thru 16 in Figure 1, for curve five can be attributed to the increased enrichment of B-10 which proportionately lifts the $1/v$ macroscopic cross section. As the enrichment of B-10 increases, the contribution of boron to the total removal cross section extends further into the higher energy range. Increasing the amount of B-10 by a factor of five i.e. natural compared to 100% enriched boron, lifts the boron macroscopic cross section by a factor of five due to the increase in the number density of B-10. Since the boron cross section is proportional to $1/\sqrt{E}$, increasing the number density of B-10 by a factor of five causes boron to dominate the total removal macroscopic cross section for energy range twenty-five times greater than that of natural boron. This causes the spectrum to be increasingly hardened with increasing B-10 enrichment, as indicated in Fig. 1.

The value of k_{eff} for each of the five cases is given in Table 1. Note that use of natural B_4C reduces k_{eff} from 0.396 to 0.338 and use of 100% B-10 enriched B_4C further reduces k_{eff} to 0.276. Increasing the amount of B-10 in B_4C from 20% to 100% reduces k_{eff} by essentially twice the amount of the 20% case, and demonstrates the self-shielding effect of B-10.

The series of calculations described previously indicate that vibrating natural B_4C into the voided volume of both the inner and outer transform has minimal effects upon the fast neutron spectrum. The value of k_{eff} , as given in Table 1 drops from 0.396 to 0.338 when both converters have natural B_4C in them. For experimental reasons, it is beneficial to have the neutron

multiplication as large as possible. Therefore a second set of calculations was performed to further investigate and help determine the best value of k_{eff} for normal operation. Again the calculational model was identical to the one given in Ref. 1. The concentration of natural boron in the inner converter was reduced to enhance the neutron multiplication. The resulting values for k_{eff} are given in Table 2 for various volume fractions of B_4C . Table 1 indicates that any value of k_{eff} between 0.338 and 0.396 can be obtained by varying the amount of B_4C . To determine the best placement and concentration of B_4C , the hypothetical flooding assembly must also be evaluated for the various configurations. These calculations are presently being performed for a number of transformer B_4C concentrations. When these investigations are completed, the optimum B_4C strategy for the normal operation of the FBBF and the hypothetical flooding accident can be properly evaluated.

Figure 1

Figure 2

TABLE 1

FBBF k_{eff} Value for B_4C Enrichment

Boron Enrichment	k_{eff}
No B_4C	.396
20% B-10	.339
35% B-10	.316
50% B-10	.302
100% B-10	.276

Table 2
 k_{eff} for Different
Natural B₄C Volume Fractions

Inner Transformer	Outer Transformer	k_{eff}
0.116	0.051	.338
0.087	0.051	.347
0.051	0.051	.362
0.0	0.051	.387
0.0	0.0	.396

REFERENCES

1. FBBF Quarterly Progress Report for period January 1, 1976 - August 30, 1976, Purdue University, PNE-76-115 (1976).

Task C (F. M. Clikeman, R. H. Johnson and P. J. Fulford):

Further details concerning the loading of the FBBF blanket have been worked out. The blankets will be loaded from the side; the rods will be placed between upper and lower grid-plates. Different adapter-pieces will be required to adjust the three types of fuel rods to the same length.

Task D (F. M. Clikeman and R. H. Johnson)

The equipment needs for the FBBF have been identified, and have been reported to the EPRI/ERDA Project Group.

Task E (P. J. Fulford, F. M. Clikeman and K. O. Ott)

The ^{252}Cf source for the first operational phase has been requested on loan from ERDA. A strength of 10^{10} neutrons/sec will be used in this phase of the program.

Task F (K. O. Ott)

Discussions on planning of blanket-reflector experiments and analysis have begun with the Applied Physics Division of the Argonne National Laboratory.