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MASTER

**FAST BREEDER BLANKET FACILITY
(FBBF)**

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ABSTRACT

The work performed in the reporting period was primarily concerned with the continued preanalysis of the first experimental sequence (Task B). Progress on testing experimental equipment and performing check out tests is also summarized (Task C).

Further safety investigations of the initial loading and future double loading configurations are reported. Comparison of the fluxes generated by discrete sources and a line source are given. Spectra generated by the FBBF are compared to those obtained for a large fast breeder reactor. Further analysis of the gamma-ray heating rates have been performed using coupled neutron-gamma-ray transport calculations. Gamma-ray heating rates in uranium and UO_2 have been calculated; the effect of changing the extrapolated height for the transverse leakage correction to the one-dimensional calculations have been investigated.

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TASK B (R.C. Borg and R.H. Johnson)

The object of this task is to perform a detailed preanalysis of the first experimental sequence. Safety analysis of the FBBF is an integral part of Task B. Therefore, the latest developments in the safety investigations are presented and discussed. Investigation into the safety of 180° double loading configurations surprisingly indicates that the value of k_{eff} tends to be overly dominated by the most reactive 180° section. This problem is analyzed and discussed in Sec. B.2.

In Sec. B.3 additional preanalysis calculations are presented. Discrete and chopped cosine source distributions are investigated by looking at axial and radial tranverses of sample reaction rates and the total fluxes. A comparison of spectra calculated from the initial loading of the FBBF and a modified Clinch River Breeder Reactor (CRBR)¹ is also given. Summary of the most recent calculations of the coupled neutron-gamma-ray transport are presented in Sec. B.4, for the initial blanket loading.

B.2. Safety Investigations

(R.C. Borg, K.O. Ott, K.R. Koch, J. Paczolt and M. Sohn)

2.1. Introduction

Safety of the FBBF is primarily concerned with the hypothetical flooding accident. The maximum value for k_{eff} is currently specified in the license as $k_{eff}^{max} = 0.75$. Loading variations are investigated with respect to their flooded k_{eff} . The considered variations include the fuel pitch, type of secondary clad, blanket thickness and advanced loadings. The two types of secondary clad are aluminum and stainless steel. There is also a need to perform some experiments without any secondary cladding. In the advanced loadings some of the natural uranium pins are replaced by enriched fuel material in order to simulate the buildup of plutonium in a fast breeder reactor blanket. All configurations must be studied to assure that they are

substantially subcritical for hypothetical accidental flooding. Previous single loading investigations, with aluminum and stainless steel as secondary clad, no secondary clad and advanced loadings, [comp. Ref. 2], have shown that these 360° loading configurations remain substantially subcritical in case of hypothetical flooding. The investigations were performed for a designated type of loading, e.g., a given type of secondary clad, with the pitch increased from a minimum to a maximum value.

Further loadings will consist of two different 180° configurations. Therefore, the safety of these combined loadings must be investigated to show subcriticality in the case of a hypothetical flooding.

2.2. Initial Blanket Loading

The initial blanket loading has approximately two thirds of the innermost natural uranium pins surrounded by stainless steel secondary clad and the outer one-third surrounded by aluminum secondary clad. Two types of safety investigations were performed for the initial loadings. The procedure in both cases were to obtain group constants, from unit cell calculations performed by HAMMER³ and use then in a subsequent two-dimensional diffusion calculation with 2DB. See previous progress reports for unit cell descriptions and calculational models.²⁵

The first type of investigation was an evaluation of k_{eff} for the initial blanket pitch with different portions of the blanket loaded. The results of these investigations are given in Table B.1. They indicate that the initial FBBF blanket loading and partial loadings are substantially below the 0.75 k_{eff} limit set by the license. The value of k_{eff} increases as the number of natural uranium pins loaded increases. A full loading has the maximum value of 0.432.

The second type of investigation considered variations of the pitch. The calculations were formulated identical to those presented in Ref. 2

TABLE B.1

Values of k_{eff} for Flooded Initial
Blanket Loadings

	k_{eff}
Full Blanket	0.432
Half Blanket	0.382
No Blanket	0.341

for the unclad, aluminum secondary clad, stainless steel secondary clad and advanced loadings, [comp. Sec. B.3 of Ref. 2]. The calculational procedure is also identical. In Fig. B.1 and Table B.2, the value of k_{∞} for both the stainless steel and aluminum secondary clad portions and the value of k_{eff} of the FBBF are plotted for different values of the pitch. Since the blanket thickness is constrained to 51 cm, as the pitch is increased from the minimum where adjacent secondary clad tubes touch, the number of aluminum secondary clad pins is reduced. After a pitch of approximately 2.2 cm the number of pins have been reduced significantly enough to eliminate all aluminum secondary clad tubes. The values of k_{eff} from the 2.2 cm pitch to the maximum pitch are identical to those presented in Fig. B.13 of Ref. 2, which considered stainless steel as secondary clad. Initially the smallest value of k_{eff} corresponds to the smallest pitch. A maximum value of approximately 0.617 is obtained for a pitch of about 2.2 cm. As was also illustrated in Ref. 2, the value of k_{eff} follows the trend of the value of k_{∞} for the stainless steel secondary clad until there is a trade off between moderation and the amount of fuel. This occurs for a pitch of about 4.6 cm. Then the value of k_{eff} for the system is greater than the value of k_{∞} for the blanket. The values of k_{eff} shown in Fig. B.1 are very similar to those given in Fig. B.13 of Ref. 2. Since the aluminum secondary clad pins are on the outer parameter of the blanket, they do not significantly influence the value of k_{eff} .

2.3. Double Blanket Loadings

Double blanket loadings consist of two 180° sections which have different physical characteristics. For example, one half of the facility could have stainless steel as secondary cladding and the other half could have aluminum. These double loadings enable more experimental information to be generated in a shorter period of time. As indicated in a previous progress report² there is

FBBF Safety Study Double Clad Blanket

Nat. UO_2 Blanket Pitch (cm)

Figure B.1: FBBF Safety Study Initial Loading

TABLE B.2

INITIAL BLANKET LOADING RESULTS

Pitch		Number Density	R Interface	k_{∞} SS Blanket	k_{∞} Al Blanket	k_{eff}
(in)	(cm)	atoms/barn-cm	(cm)			
0.668	1.697	0.0139233	57.1237	0.45104	0.57767	0.41261
0.673	1.709	0.0137172	57.4878	0.47048	0.60706	0.43150
0.700	1.778	0.0126799	59.4610	0.55073	0.72395	0.51021
0.740	1.880	0.0113457	62.4031	0.61903	0.81885	0.57262
0.800	2.032	0.0097077	66.8529	0.66602	0.87970	0.60626
0.962	2.189	0.0083615	71.4893	0.68237	0.89625	0.61525
0.900	2.286	0.0076703	---	0.68305	---	0.61679
1.000	2.540	0.0062129	---	0.66882	---	0.60750
1.100	2.794	0.0051346	---	0.64031	---	0.58338
1.200	3.048	0.0043145	---	0.60568	---	0.55234
1.500	3.810	0.0027613	---	0.49745	---	0.45609
1.650	4.191	0.0022821	---	0.44746	---	0.41714
1.750	4.445	0.0020287	---	0.41666	---	0.40297
1.850	4.699	0.0018153	---	0.38803	---	0.39509
2.000	5.080	0.0015532	---	0.35129	---	0.38606
2.250	5.715	0.0012272	---	0.29647	---	0.37953
2.500	6.350	0.0009941	---	0.25237	---	0.37462

a window for the single clad and aluminum double clad blankets where the value of k_{eff} would be slightly greater than that specified by the license, [comp. Figs. B.10, B.12, Tables B.4, B.4 of Ref. 2]. For all pitches the value of k_{eff} for the stainless steel double clad blanket is substantially below the specified value of 0.75, [comp. Fig. B.13 and Table B.6 of Ref. 2]. Thus, based on the reported values for k_{eff} of these three single loadings one would expect that all aluminum and unclad loading pitches could be experimental investigated if half of the facility consisted of a stainless steel secondary clad loading. The value of k_{eff} for double loadings intuitively should be between the values of the two single loadings and thus below the 0.75 limit. This intuitive consideration does not hold and is discussed in the following paragraphs.

Physically, a double loading configuration in the FBRF is three-dimensional. Since there is not a three-dimensional computational capability available at Purdue University, a number of two-dimensional calculations are necessary to obtain the desired information about the safety of the double loadings. First, group constants are obtained from HAMMER³ by the procedure and methods reported previously.^{2,5} Next, two r-z calculations are performed to determine the value of k_{eff} for each single loadings based on the physical description of each half. A buckling search is performed in r- θ geometry for each physical description with convergence specified by the value of k_{eff} calculated from the initial r-z problem. The value of k_{eff} for a double loading is obtained by mocking up the facility in r- θ geometry and describing the leakage by input buckling values calculated from the previous r- θ problems.

A number of different ways to describe the leakage of the double loading r- θ calculations were considered. The bucklings were input

as group and zone dependent values, zone dependent values, one value for each half, and one average value. The results of this investigation indicated an insensitivity of the leakage to the buckling description indicating that most of the neutrons leak out radially. The calculated value of k_{eff} is also insensitive to the method of inputting the buckling which describes the axial leakage.

Since the basic concern of the safety analysis is to keep k_{eff} as low as possible, the stainless steel double clad half of the facility was first specified with a pitch of 1.88 cm. The value of k_{eff} for a single loading with this pitch and stainless steel as secondary clad is 0.544. Aluminum as secondary cladding was specified in the other half. A pitch of 2.286 cm, which corresponds to the point near the maximum value of k_{eff} , 0.796, of the aluminum double clad single loading, was chosen. Surprisingly, the calculated value of k_{eff} for the double loaded facility is almost the same as a single loaded facility with aluminum as secondary clad. Further investigations were performed in order to obtain an understanding of this seemingly inconsistent finding. First, the portion of the aluminum loading was reduced from 360° to 0°. Simultaneously, the portion of the facility with stainless steel as secondary clad was increased. The results are shown in Fig. B.2 and Table B.3. The value of k_{eff} stays fairly constant and does not start to substantially decrease until the aluminum portion of the facility is less than 15 per cent. These results do not seem to be consistent with the physical understanding of the problem.

In order to understand the results presented in the previous paragraph, the problem was slightly modified so that the pitch of each half of the facility was set to 1.709 cm. This constrained the water and uranium content to be the same for each half and removes interpretation difficulties. A value of k_{eff} for this double loading was again essentially the same as

K_{eff} vs. Angular Loads

Figure B.2: Double loading safety investigation, value of k_{eff} vs the angular per cent of stainless steel secondary cladding.

TABLE B.3

Value of k_{eff} for Various Fractions of Aluminum and Stainless Steel as Secondary Clad.

Fraction of Stainless Steel (pitch = 1.793cm)		Fraction of Aluminum (pitch = 2.286 cm)		k_{eff}
0.0		1.0	(360°)	0.797
1/2	(180°)	1/2	(180°)	0.785
2/3	(240°)	1/3	(120°)	0.774
5/6	(300°)	1/6	(60°)	0.739
11/12	(330°)	1/12	(30°)	0.674
17/18	(340°)	1/18	(20°)	0.629
35/36	(350°)	1/36	(10°)	0.557
1.0	(360°)	0.0		0.489

the value with only aluminum. The reason for this result can be better understood by investigating Fig. B.3 which illustrates the log of the normalized flux vs radial position. The flux was normalized to the maximum value of the total flux on the aluminum side. The points cut through the middle of each half of the facility. For all groups the flux is between two and three orders of magnitude higher on the aluminum side when compared to the stainless steel side. The flux shown in Fig. B.3 is physically not realized since it results from an eigenvalue-problem solution. One would expect a flux difference of three or four times, reflecting the difference of the absorption in each half for a physically realized flux. To check this latter point, the flux solution was determined, based on a source calculation (see Fig. B.4). The differences of the fluxes for each half of the blanket, given in Fig. B.4, should be representative of the differences which would be expected in the flooded facility if a source were present, rather than the λ -mode flux differences presented in Fig. B.3. The difference in the fluxes in each half are not orders of magnitude, as depicted in Fig. B.3, but rather three to four times greater on the aluminum side as shown in Fig. B.4. This difference depicted in Fig. B.4 is what one would expect from physical considerations.

The general calculational procedure to obtain the value of k_{eff} is to artificially modify the balance equation in order to obtain criticality, i.e., in one-group notation:

$$\frac{\lambda k_{\infty}}{1 + L^2 B^2} = 1 \quad (1)$$

where λ is the eigenvalue identical to $1/k_{eff}$; the second term in the denominator describes the leakage. Thus, for each region a modified value of k_{∞} can be considered as

$$k_{\infty}' = \lambda k_{\infty} \quad (2)$$

Figure B.3: Log of the λ -mode group fluxes.

Figure B.4: Group fluxes based on a source calculation.

The values for k_{∞} , for a pitch of 2.286 cm., in the aluminum and stainless steel secondary clad halves, are 0.893 and 0.683, respectively. Multiplying these values by the eigenvalue λ , which is approximately 1.25, modified infinite multiplication constants of approximately 1.12 and 0.86 are obtained. Thus, the eigenvalue artificially modifies the aluminum side to such an extent that it is treated as super productive medium but the stainless steel side is still an absorbing medium. The flux shapes illustrated in Fig. B.3 are exactly what one would expect from a medium with one side super productive and the other side an absorbing medium. Similar calculations were performed for the initial loading pitch and the same type flux shape differences were obtained.

The investigations presented in this subsection indicate that the λ -mode flux, for the double loading subcritical problems needed to verify the safety of the FBBF, are not physically realized.

B.3. Comparison of Neutron Fluxes (R.C. Borg, M. Sohn and J. Paczolt)

3.1. Source Axial Distribution

3.1.1. Fluxes and Reaction Rates

Previously, axial and radial reaction rate traverses were calculated from fluxes generated by a line source with an axial chopped cosine strength distribution.⁶ These initial investigations showed that sample blanket region axial reaction rate traverses fit a chopped cosine extremely well. The ^{252}Cf source in the FBBF is not a line source but actually consists of four point sources of different strength. Their positions in the source rods predetermined to give a chopped cosine flux distribution in the blanket region, with the maximum value at the axial mid-plane. In this subsection, comparison of the total flux and sample reaction rates generated from both the line source and discrete point sources are investigated.

The calculational procedure and two-dimensional model needed for the source distribution investigations are the same as those used for the sample reaction rate calculations presented in Ref. 6. The same 21-group constant set is also used, since the 26-group library mentioned in Ref. 6 is presently still not available. The inherent problem of the 21-group library lower energy group structure will not hamper the interpretation of results since the contributions of the lowest energy group are neglected as was suggested.⁶

The axial shape of the flux and reaction rates are directly influenced by the strength and position of the four discrete sources. In Fig. B.5 and B.6, the axial shape of the total flux, at the converter blanket and blanket midplane, is illustrated. The solid line represents the flux generated from a line source. The smallest dashed lines indicate the flux generated from the calculation with four discrete sources. The maximum value of

Figure B.5: Axial flux shape at the converter-blanket interface.

Figure B.6: Axial flux shape at the blanket midplane

the total flux, generated by the discrete source, at the converter blanket interface is shifted slightly above the center line, (comp. Fig. B.5). Further out into the blanket the maximum value of the total flux generated by the discrete sources is much closer to the axial midplane. As indicated in both figures, the discrete point sources generate a flux shape which is very close to the flux shape associated with cosine shaped line source.

A second investigation considered sample reaction rate traverses at the blanket midplane. The ^{238}U capture and ^{239}Pu fission rates calculated from fluxes generated by the line source and discrete point sources are illustrated in Figs. B.7 and B.8. A cosine fit of the discrete source result is also given. Reaction rates match the cosine fit very well for the two source versions. In Figs. B.9 and B.10 radial traverses of the ^{238}U capture rate and ^{239}Pu fission rate for the two source version are illustrated. The difference between the two source specifications becomes smaller as the distance from the converter-blanket interface is increased. In the converter regions the match is not extremely good as would be expected. Since most experiments are performed in the blankets this slight mismatch should not matter. It will be corrected for comparisons with actual experiments.

3.1.2. Conclusions

The material presented in this subsection indicate that the axial flux distribution in the blanket generated with the discrete sources follow a chopped cosine distribution and gives a flux shape comparable to shapes generated by a line source. The difference between the flux generated by the line source and the discrete sources decreases as comparisons are made further into the blanket.

Figure B.7: Axial traverse of ^{238}U capture rate at the blanket midplane.

Figure B.8: Axial traverse of ^{239}Pu fission rate at the blanket midplane.

Figure B.9: Radial traverse of ²³⁸U capture rate at the blanket midplane.

Figure B.10: Radial Traverse of ^{239}Pu fission rate at the blanket midplane.

3.2. Comparison of Blanket Spectra

One of the basic objectives of the FBBF is to generate spectra in the blanket which are similar to spectra found in a large fast breeder reactor blanket. The conceptual design study⁸ indicated a good match between spectra generated by the FBBF and a typical 1000 MWe fast breeder reactor. A number of modifications have been implemented into the original design. Therefore, it is desirable to compare spectra generated from the final design of the FBBF with typical fast breeder reactor blanket spectra.

The physical and geometrical characteristics of the FBBF and a modified 1000 MWe CRBR were mocked-up with the 1-dimensional computer code LAZARUS.⁷ The desired source and eigenvalue calculations were performed to determine the spectrum. The blanket of the modified CRBR was identical to the initial loading of the FBBF.

The normalized spectrum from each calculation for the blanket interface, middle of the inner blanket, inner and outer blanket interface and middle of the outer blanket are plotted in Figs. B.11 to B.14. The midpoint of each group is connected for better comparison. For all illustrated space points the spectrum calculated for the FBBF is very representative of the spectrum calculated for the modified CRBR. Thus, one of the basic objectives of the FBBF to generate spectra characteristics of large fast breeder reactors can be accomplished.

Figure B.11: FBBF and modified CRBR spectrum at the inner blanket interface.

Figure B. 12: FBBF and modified CRBR spectrum at the midplane of the inner blanket.

Figure B.13: FBBF and modified CRBR spectrum at the inner and outer blanket interface.

Figure B.14: FBBF and modified CRBR spectrum at the midplane of the outer blanket.

B.4. Coupled Neutron-Gamma-Ray Transport Calculations
for Initial FBBF Loading
(R.H. Johnson, J. Paczolt)

4.1. Introduction

One-dimensional discrete ordinates calculations for the preanalysis of the gamma-ray heating rates have been reported previously.⁶ Further analysis of the gamma ray heating rates has been performed. Heating rates in uranium and UO_2 have been calculated and are presented below.

4.2. Calculations

ANISN-W⁹, a CDC version of ANISN¹⁰, was used for the one-dimensional discrete ordinates calculations. A collapsed version of the DLC-37/EPR cross section set^{11,12} was used. The collapsing procedure and the one-dimensional model of the initial FBBF loading have been reported previously.⁶ S_4, P_3 calculations were performed. Seven outer iterations gave good convergence; the first four outer iterations used the diffusion approximation.

Gamma-ray heating rates were calculated for various elements at all spatial mesh points. A response matrix containing gamma-ray kerma factors K_m^g for each energy group g and material (element) m was input to ANISN-W. The code calculated the gamma-ray heating rate for material m at mesh point 1 as

$$H_{m,1} = \sum_{g=1}^G K_m^g \phi_1^g$$

where ϕ_1^g is the calculated gamma-ray flux for group g at mesh point 1 and the summation is over all gamma-ray groups. However, kerma factors for uranium were not available with DLC-37/EPR. Therefore, gamma-ray heating rates have been performed for each spatial mesh point within the blanket regions; the results for one mesh point are shown in Fig. B.15.

Meshpoint 54

Fig. B.15: Gamma-ray heating rate per atom for several different elements. The points shown for $Z \leq 82$ were calculated using the kerma factors for DLC-37/EPR and the results of the ANISN-W calculation for the blanket height of 139.52 cm. The point for $Z = 92$ was obtained by extrapolation. Meshpoint 54 is in the inner blanket.

The one-dimensional model used for the preanalysis had an extrapolated height of 129.04 cm. (The extrapolated height is used for a transverse leakage correction.) This height is appropriate for the converter regions, but the blanket regions are better represented by an extrapolated height of 159.52 cm.

4.3 Results

Gamma-ray energy deposition rates in type 316 stainless steel and natural UO_2 are shown in Figs. B.16 and B.17. The TLD measurements in the FBBF will give, to a first approximation, the energy-deposition rate in the material surrounding the dosimeter. The energy deposition rate in natural UO_2 is seen to be larger than in type 316 stainless steel. The radial dependences within the blanket region are approximately the same for both materials.

Fig. B.16: S_4, P_3 ANISN-W calculation of gamma-ray energy deposition rate in type 316 stainless steel for blanket height of 159.52 cm.

$2\pi \times \text{Radius} \times \text{Gamma-Ray Heating Rate in Nat UO}_2$ (cm*rads/hr)

FIG. B.17: S_4, P_3 ANISN-W calculation of gamma-ray energy deposition rate in natural UO_2 for blanket height of 159.52 cm.

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TASK C
(F.M. Clikaman)

Introduction

The objectives of this task are: (1) loading the first configuration of the FBBF; (2) testing experimental equipment and techniques on the FBBF; and (3) performing check-out tests required for licensing purposes and for safe operation of the FBBF. Progress on this task is summarized below.

C.1. Testing Experimental Equipment and Techniques

(F.M. Clikeman, R.H. Johnson, D.W. Vehar,
G.A. Harms, H.P. Chou and K.R. Koch)

Preliminary testing of equipment and techniques for neutron spectroscopy foil activation, solid state fission track recorder measurements, and gamma dosimetry measurements continue. Although the FBBF is not yet available for measurements, testing of techniques and equipment is being carried out using the PUR-I Reactor as well as radioactive sources including a small ^{252}Cf source.

During the progress period a Canberra Scorpio 3000 data acquisition system was delivered and installed in the laboratory. This system, modified for 2 parameter data acquisition, provides capabilities for data acquisition, graphics display, and data analysis. Also delivered during this period was a Canberra Model 7229 Ge(Li) detector with the resolution of 2 keV (FWHM) and a relative efficiency of 13%. The software provided with the SCORPIO system permit the determination of the absolute quantity of radioactivity in samples analyzed with the Ge(Li) detector system. Similar software is also available with the system for analysis of spectra measured with a NaI(Tl) detector. The testing and the process of familiarizing the personnel with the system is now under way. Energy and efficiency calibrations of the detector and analyzer system are now being performed. Following the calibration process the adequacy of the detector and analyzer and software programs for determining the absolute level of radioactivity in a sample will be determined. If it is found that the software programs are not adequate, new programs based on known analysis techniques will be written.

Development of a new preamplifier for use with the proton recoil neutron spectrometer system has continued. A preamplifier small enough to fit, together

with the proton recoil detector, inside a fuel rod has been built. The preamp and the detector, using improved pulse shape discrimination techniques, have been tested with satisfactory results in the Purdue (PUR-I) Reactor. A new detector with filling gas and dimensions optimized for neutron spectroscopy measurements between the energy ranges of 1 and 100 keV has been ordered. The new detector will first be tested using PUR-I reactor and then will be moved to the FBBF facility for incorporation into the 2-parameter data acquisition system. In the meantime, the necessary data analysis programs for use with the proton recoil detector system are being converted for use with the computer in the SCORPIO data acquisition system.

During the past quarter experiments with solid state fission track recorders (SSTR) for measuring fission rates have lead to the decision to use glass as the recording media. Experiments using the variety of recording films including polycarbonate film and glass as well as a variety of etching techniques have shown that glass gives the sharpest images and best contrast. Both of these features are important if an automatic counting system based on a scanning optical microscope is to be developed. Work is now being carried out on the renovation of a scanning optical microscope for use in the proposed automatic counting system. In the meantime, the fission tracks on the SSTR will be counted manually. Using microphotographic techniques photographs of the tracks are produced, made into a large mosaic of the full fission track area, and then counted.

During the past quarter, a thermoluminescent dosimeter readout unit has been acquired. The unit consists of a Harshaw model 2000-A readout and a 2000-B integrating picoammeter. The analyzer system provides for a wide range of heating rates, a high temperature and high sensitivities that will allow a wide range.

of TLD dosimeters to be analyzed. The built in radium-activated NaI(Tl) light source and provisions for flowing dry nitrogen through the sampling chamber permit a high degree of reproducibility in the analyzer system. The system permits all of the common geometries of TLD dosimeters, including powders, to be analyzed. In addition to the TLD analyzer a furnace for both pre and post annealing of the dosimeter is also available. The furnace is capable of temperatures in excess of 300°C, allowing it to be used with all of the more common TLD dosimeter material. A rotating exposure holder has been built which will allow all of the TLD dosimeters to be exposed to a uniform exposure and then read out. The dosimeters will then be grouped on the basis of uniformity of response. Calibrated gamma sources are also available to determine the absolute calibration of the TLD's and the analyzer system.

C.2. Installation and Testing of Facility Equipment and
Preperation of Operating Procedure
(F.M. Clikeman)

During the progress period most of the systems for the FBBF facilities were completed, checked out and placed in operation. These include the security system, the source drive system and area monitor safety systems. The ventallation system is also installed and checked out but because of a high noise level near the blowers some modification of the system may be required in the future.

Following the installation and checkout of the security system the 4.8% enriched stainless steel clad fuel for the inner and outer transformer regions were moved from the storage area to the FBBF facility. The fuel was placed directly into the inner and outer transformer cans, the sodium pins were also added to the outer transformer region and the interstitial regions of both the inner and outer transformer regions were filled with boron carbide. The inner and outer transformer cans were then sealed by welding and placed in position in the FBBF facility. Measurements of the neutron multiplication were made for each of the transformer regions seperately and for the two regions together. The results agreed reasonably well with the calculated values. On October 29th the californium source was delivered and installed in the facility. Reflector cans to surround the blnaket region of the FBBF facility are now being fabricated and upon their completion the blanket regions will be loaded.

C.3. Experimental Program for the FBBF
(F.M. Clikeman, R.H. Johnson and K.O. Ott)

The experimental program for the FBBF facility was outlined in the previous progress report. Detailed plans for the initial blanket loading has also been reported in previous reports. Starting on or about January 1, 1978 loading of the first blanket will commence. The loading of the first blanket is expected to take about a month and will be followed by six months of measurements on the first blanket.