**MEASUREMENT AND CALCULATION OF THE U-235 AND
U-238 FISSION DISTRIBUTIONS AND U-238 CAPTURE
DISTRIBUTIONS FOR THE ENRICO FERMI FAST BREEDER
REACTOR**

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FOREWORD

This is one of a series of reports on the low-power (up to 1 Mwt) and high-power (up to 200 Mwt) nuclear testing of the Enrico Fermi fast breeder reactor. The Nuclear Test Program is planned, directed, and evaluated by Atomic Power Development Associates, Inc. (APDA). The tests are conducted by Power Reactor Development Company (PRDC). The reactor proper is owned and operated by PRDC. The steam generators and electrical generating facilities are owned by The Detroit Edison Company (DECo).

Many individuals contributed to the nuclear testing of the Fermi reactor. Listed below are those, exclusive of the authors, who made significant contributions to the work reported in this document.

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SUMMARY

Experiments were conducted to determine U-235 and U-238 fission distributions and U-238 capture distributions in the core and blanket regions of the Enrico Fermi reactor by irradiating uranium foils in special core and radial blanket subassemblies. These special subassemblies were constructed from normal core and blanket subassemblies by replacing the central fuel pins with a stainless steel sheath. Foil tubes containing the desired uranium foils and stainless steel spacers were inserted in the steel sheaths. Loading (and unloading) of the foil tubes into the special subassemblies was performed in a glove box especially constructed for this purpose in the Fuel and Repair Building. The irradiated foils were counted with standard single channel scintillation counting equipment with fission product decay counting employed to establish fission distributions and Np-239 soft gamma counting employed to establish U-238 capture distributions. The techniques provided relative reaction-rate distributions. Absolute reaction rates were determined for selected foil pairs by radiochemical techniques at Argonne National Laboratory.

The measured distributions generally confirmed those obtained by multigroup techniques. Notable exceptions occurred mainly in soft spectrum regions far from the core; reasons for these exceptions are discussed in the body of the report and necessarily include an investigation of the sensitivity of the calculated results to various calculational parameters and techniques.

Duration of the foil irradiation experiment was about one month and took place on July 22, 1964; August 26, 1964 - September 8, 1964; and May 27, 1965 - June 18, 1965.

I. PURPOSE OF TESTS

The purpose of the tests was to experimentally determine the fission rates of U-235 and U-238 and the capture rates of U-238 at various points in the reactor core and blanket.

The data obtained from the experimental measurements are planned to be used to evaluate the ability to predict, by calculation, the experimental results, and hence other reactor parameters such as breeding ratio, power distribution, radiation damage, and plutonium buildup in reactor subassemblies.

II. DESCRIPTION OF THE ENRICO FERMI REACTOR

A. GENERAL DESCRIPTION

The Enrico Fermi fast breeder reactor and its associated structures are shown in perspective in Fig. 1. The reactor is contained in a stainless steel vessel sealed at the top by a rotating shield plug which supports the control mechanism, the holddown mechanism, and the fuel handling mechanism. The reactor vessel is surrounded by neutron shielding material of boronated and plain graphite which is contained inside the primary shield tank. The reactor is cooled by sodium and operated at essentially atmospheric pressure. The design maximum reactor power with the first core loading (core A) is 200 Mwt.

The core and blanket, located in the lower reactor vessel, consist of square (cross section) subassemblies which contain fuel pins and blanket rods arranged to approximate a cylinder 80-in. in diam and 70-in. high. The core, contained in the central portion of the core subassemblies, approximates a right cylinder 31-in. in diam and 31-in. high; it is axially and radially surrounded by breeder blankets. The fuel in core A consists of zirconium-clad pins containing U-10 w/o molybdenum alloy with the uranium enriched to 25.6 w/o U-235. Each fuel subassembly in the core contains 140 fuel pins which have a total mass of approximately 4.75 kg of U-235. The blanket is depleted U-3 w/o molybdenum alloy.

The reactor cross section, shown in Fig. 2, indicates the placement of individual components within the lower reactor vessel. Core and inner radial blanket subassemblies, the neutron source, and the 10 operating control- and safety-rod channels occupy 149 central lattice positions. The lattice positions surrounding the inner radial blanket comprise the outer blanket area, and when occupied with outer radial blanket subassemblies these positions form an annular region whose top and bottom are at the same elevation as the top and bottom of the inner radial blanket. Surrounding the outer radial blanket are lattice positions for stainless steel subassemblies which provide thermal and neutron shielding.

The reactor lattice is divided by two datum lines through the center of the reactor core. These lines have the same relationship as the X- and Y-axes in a coordinate system. The center subassembly location is designated P00-P00 and each subassembly location surrounding this location has a position designated by the distance from the axes.

The major deviation from the standard geometric coordinate system is that the quadrants are designated by P and N for the positive and negative

FIG. 1 PERSPECTIVE VIEW OF REACTOR

FIG. 2 REACTOR CROSS SECTION

signs, respectively. Each location is further identified by a set of two digits which indicates the distance from the axes. To identify a subassembly location in any quadrant, the number of subassembly spaces from the Y-axis is determined, and then the number of subassembly spaces from the X-axis. The subassembly is then identified with the proper position. For example, a fourth-quadrant position would be P16-N05.

All lattice positions are supplied with sodium coolant which flows upward from a plenum connected to the discharge lines of the three primary sodium pumps. The coolant flows upward through the individual core and inner radial blanket subassemblies into a large upper plenum and from there by gravity to the three intermediate heat exchangers and then to the suction side of the primary pumps. Sodium also is used in the secondary cooling system. The outer radial blanket and shielding lattice positions are supplied with sodium coolant from the low-pressure plenum. The sodium from the low-pressure plenum, after cooling the blanket and thermal shield subassemblies, is mixed in the upper plenum with the sodium from the core and inner radial blanket before flowing to the intermediate heat exchangers.

The reactor is controlled by two operating control rods and seven safety rods; provisions have been included in the design for an eighth safety rod. The rods are of the poison type, containing boron carbide (B_4C) in which the boron is enriched in boron-10 (B-10). One operating control rod is for regulating and the other is for shimming. Both rods have approximate reactivity worths of 46 cents. During operation of the reactor, the safety rods are held above the axial blanket section of the core so that they can be rapidly inserted into the core if a scram becomes necessary. The movements of the control rods and safety rods are actuated from the top and, during fuel reloading, they are detached from their drive extensions to allow them to remain in the core.

The neutron detectors (fission chambers and ion chambers) for normal at-power operation are located in six neutron-counter tubes embedded in the graphite neutron shield surrounding the reactor vessel (Fig. 3). There are 11 channels of nuclear instrumentation distributed throughout the six neutron-counter tubes in a manner which will cover the full power range during reactor operation. The detectors located in these channels are designed to operate at temperatures up to 500 F.

An antimony-beryllium (Sb-Be) neutron source is located in the reactor at the core-blanket interface (Fig. 2) to provide a neutron flux at the neutron detectors during reactor startup and to maintain a flux when the reactor is shut down. The radioactive antimony portion of the source is made as a separate unit for easy replacement and is in the form of a rod approximately 0.7-in. in diam and 25-in. long. The radioantimony rod fits inside a beryllium assembly which is in the form of a hollow cylinder inside a stainless steel can having the external dimensions of a normal core subas-

FIG. 3 LOCATION OF NEUTRON-COUNTER TUBES IN GRAPHITE SHIELD

sembly fuel can. The handling head on the antimony section allows it to be inserted and withdrawn from the beryllium by means of a normal fuel handling mechanism. The beryllium portion must be installed and removed from the reactor through a special access port in the reactor vessel shield plug.

Additional information concerning the reactor design may be found in the Enrico Fermi Hazards Summary Report.¹

B. REACTOR CORE AND BLANKET SUBASSEMBLIES

1. Reactor Core Subassemblies

The reactor core subassemblies consist of the outside wrapper tube which is a 2.642-in. square stainless steel tube of nominal wall thickness of 0.096 in. A combination handling head and holddown contact is attached to the top of the wrapper tube. To the bottom of the wrapper tube is attached a lower support section consisting of two concentric round tubes, the outer one of which fits into the reactor vessel lower support plate. A spring between the outer and inner tubes allows for thermal expansion of the subassembly structure. A strainer at the inlet of each core subassembly is provided to prevent particles from lodging between the fuel pins.

The fuel region of each core subassembly consists of 140 round U-10 w/o Mo alloy pins containing uranium enriched to 25.6 w/o U-235. The nominal overall length of the pins is 32.75 in., and the OD is 0.158 in. Each pin is clad with 5 mils of reactor-grade zirconium that is metallurgically bonded to the periphery of the fuel alloy and closed at the top and bottom with zirconium end-caps. The upper ends are free to accommodate changes in length resulting from temperature changes and growth due to irradiation effects.

There are 17 support grids, located at intervals along the length of fuel pins, which serve as the support structure for the pins. The support structure holds 140 pins on a square pitch of 0.199 in. The four corner positions are occupied by steel tie rods on which bear screws used to laterally restrain the fuel pin assembly within the subassembly wrapper tube.

The upper and lower axial blanket region in each core subassembly contains sixteen 0.395-in. -diam, U-3 w/o Mo alloy rods containing uranium depleted to 0.35 per cent in U-235. The rods are enclosed in a 10-mil-thick stainless steel tube having an OD of 0.443 in. The radial clearance between the uranium alloy and the tube is filled with sodium to provide a bond with low thermal resistance and to allow for an increase in diameter. Allowance for an increase in length of the alloy due to thermal expansion and radiation is provided inside the sealed tube by the sodium annulus and a gas space above the rod. A short stainless steel tube located within the gas space prevents the uranium from rising above the sodium.

The blanket rods are held in the subassembly and spaced around its periphery by lugged grids at the top and bottom of each axial blanket section. The center portions of the upper and lower axial blankets in each subassembly contain no blanket rod. An isometric view of a core subassembly is shown in Fig. 4.

2. Radial Blanket Subassemblies

There are two types of blanket subassemblies which are identical internally but differ in details of the support structure.

In the case of inner blanket subassemblies, the nozzle contains an accommodation spring for thermal expansion, and the support structure is properly sized to fit the holes in the lower support plates that are intended for core and inner radial blanket subassemblies.

The outer radial blanket subassemblies have a simple nozzle structure because, in the absence of a holddown plate, no special accommodation for thermal expansion is necessary. The nozzle for the outer radial blanket subassemblies is sized for the holes in this region of the lower support plates. The holes are smaller than those provided for core and inner radial blanket subassemblies; consequently, a core subassembly cannot accidentally be placed in an outer radial blanket position.

The radial blanket rods are the same as the axial blanket rods except that they are 71.5-in. long instead of 17 in., and they are wrapped spirally with wire. Further, the subassembly contains a complete lattice of 25 blanket rods supported by five parallel grids instead of the hollow lattice used in the axial blankets.

A removable orifice is located in the lower support plate for each outer radial blanket position. The nozzle of each inner radial blanket subassembly also has an orifice. An isometric view of a radial blanket subassembly is shown in Fig. 5.

FIG. 4 ISOMETRIC VIEW OF A CORE SUBASSEMBLY

III. EXPERIMENTAL PROCEDURE

A. GENERAL DESCRIPTION OF METHOD

The method used to obtain the U-235 and U-238 fission distribution and the U-238 neutron capture distribution was to irradiate small samples of enriched and depleted uranium at various locations throughout the reactor regions of interest. To determine the fission distribution patterns, the irradiated foils were analyzed by scintillation counting techniques to establish the relative radioactivity resulting from fission product decay. In addition, to establish the U-238 capture patterns, the depleted foils were analyzed to determine the relative activity of the gamma rays given off by Np-239, a reaction product of U-238 neutron capture. The method used to establish an activity proportional to the U-238 capture rate was developed at Argonne National Laboratory and is explained in detail in Reference 2.

A description of the foils, the method of irradiating the foils, and the determination of the information necessary for the analysis of the results are given in the following sections. Also described is the special equipment required to transport and locate foils in the reactor.

B. COUNTING EQUIPMENT

1. Equipment

The equipment for measuring the activity of irradiated foils was similar to standard scintillation counting equipment. The specific equipment* is shown in Fig. 6 and is listed below:

a. Scintillation Detector

The scintillation detector was a Hamner Electronic Co. A-18 scintillation detector consisting of a NaI (Tl) crystal (2-in. diam by 2-in. thick), optically coupled to a RCA 6342-A photomultiplier tube enclosed in a housing which contained an integral preamplifier.

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* Two channels of counting equipment were available for use. However, all the results reported herein were obtained using that equipment which constituted the channel designated "channel B".

FIG. 6 BLOCK DIAGRAM OF SCINTILLATION COUNTING EQUIPMENT

b. Linear Amplifier And Pulse Height Analyzer

The linear amplifier and pulse height analyzer was a Hamner Electronic Co. type N-302. It has a gain of 10,000 with single delay line clipping, a rise time of less than 0.2 μ sec, and a deviation from linearity of less than 0.2 per cent between 0 to 100 volts. The pulse height analyzer section could be operated in either differential or integral mode by means of a switch located on the front panel.

c. High-Voltage Supply

The high-voltage supply was Model 412A manufactured by the John Fluke Manufacturing Co. It has an output range of 500 to 2000 volts and a stability rating of 0.01 per cent for a 10 per cent change in line voltage. In addition to the regulation provided by the high-voltage supply, a Solar transformer was used to provide additional stability for the high voltage to the phototube and for the voltage to the linear amplifier.

d. Scaler

The scaler was a RIDL model 49-25 having three electronic decades and four mechanical decades. The resolving time of the electronic decades was one μ sec and the input limitation to the mechanical register is given by the manufacturer as 20 counts per sec. This results in an overall count-rate limitation of 20,000 counts per sec.

e. Shield

The shield was similar to the Technical Associates' type LS-66 lead shield which allowed the scintillation detector to be mounted vertically. The shield contained a stand for holding the sample tray at varying, highly-reproducible distances from the scintillation crystal. A planchet was used with the sample tray to provide accurate positioning of the foil within the sample tray.

2. Calibration Of Counting Equipment

The counting equipment was calibrated to establish the relationship between output of the linear amplifier, in volts, and the energy of gamma rays, in Mev, incident on the scintillation crystal. For simplicity, the gain of the scintillation counting system was adjusted so that the gamma-ray energy of cesium-137, 0.662 Mev, resulted in a voltage pulse of 66.2 volts from the linear amplifier. Large gain adjustments were accomplished by changing the gain setting of the linear amplifier, and smaller gain changes were made by adjusting the high voltage to the photomultiplier tube.

With the gain of the system adjusted to give an output pulse from the amplifier of 66.2 volts for a 0.662-Mev gamma ray, other sources having known gamma-ray energies were checked to verify the linearity of the system and the slope of the curve of gamma-ray energy versus amplifier output voltage. The isotopes used for this purpose and the energy of their gamma rays are listed in Table I. The result of the linearity determination is illustrated in Fig. 7.

TABLE I - LIST OF CALIBRATION ISOTOPES

<u>Isotope</u>	<u>Gamma-ray energy³, Mev</u>
U-235	0.096
	0.195
Ba-133	0.36
Cs-137	0.662
Co-60	1.17
	1.33

3. Gain Stability

To allow the comparison of the count rate observed on foils which were counted at significantly different times, it was necessary that the gain of the counting equipment be reproducible.

The equipment was initially calibrated to determine the standard activity of the Cs-137 source above 662 kev. To allow the gain of the counting equipment to be returned with a high precision to a standard value, the total activity above 662 kev (66.2 volt pulse height) of this source was measured approximately every 1/2 hour during the time periods over which foils were counted.* Since this base line lies within the peak of the pulse height distribution (Fig. 8), a small change in system gain would result in a relatively large change in Cs-137 counting rate, thus providing a sensitive measure of gain drift. When the activity observed during a gain-check differed from the

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* Decay of the 30-year half life standard source was taken into account where appropriate since the time between some of the foil irradiations approached one year.

FIG. 7 GAMMA-RAY ENERGY AS A FUNCTION OF LINEAR AMPLIFIER OUTPUT

FIG. 8 PULSE HEIGHT ANALYSIS OF STANDARD Cs SOURCE

standard source count rate by more than a specified amount, the system gain was adjusted by changing the high voltage to the photomultiplier tube until the standard counting rate was again observed. The observed relationship between count rate and change in the high voltage is shown in Fig. 9. It can be seen that an increase in high voltage of 1 volt results in a change in count rate of approximately 12 per cent. For the entire course of the experiment the total range covered by high-voltage corrections to the gain was 7 volts, which corresponds to a gain range of 5 per cent.

C. FOILS

The foils were designed to obtain comparable activities from the enriched and depleted foils after simultaneous irradiation at the same location. Simultaneous irradiation was utilized because the complexity of foil handling required that the number of irradiations be held to a minimum. The design was also influenced by the necessity of including in the foils enough material to provide an adequate count rate 35 hr after the irradiation without creating excessively high foil subassembly activity. Foils could then be removed manually from the subassembly without the use of shielding. The first requirement was satisfied by alloying the enriched uranium with aluminum and making these foils 1/4-in. shorter than the depleted foils. The second requirement was satisfied by making the foils much larger than foils used in other similar programs. The only drawback to this approach was the necessity of correcting for the difference in the gamma self-shielding between the solid uranium and Al-U alloy (see Section III K.).

1. Enriched Foils

The enriched foils consisted of Al-18 w/o U alloy, 0.080-in. in diam by 3/4-in. long. The uranium was enriched to approximately 93 w/o U-235. The alloying percentage and the length were selected so that the enriched uranium pin would have approximately the same activity as a depleted foil after simultaneous irradiation at core center (U-235 to U-238 fission ratio of approximately 26). With the enriched and depleted foils providing comparable decay rates at the core center, it was possible to obtain information regarding fission distribution of U-238 further into the blanket.

The mass of U-235 in each enriched foil was determined to a 2 σ accuracy of ± 0.9 per cent by the manufacturer, M and C Nuclear Division of Texas Instruments. This was accomplished by determining the proportionality constant between the U-235 content of a foil and the count rate observed for that foil on a scintillation counting channel. All enriched foils were counted with the use of the same scintillation equipment. The predetermined proportionality constant was applied to the observed count rate to give the mass of U-235 present in each foil.

FIG. 9 INTEGRAL Cs-137 COUNT RATE AS A FUNCTION OF HIGH-VOLTAGE DEVIATION FROM THAT REQUIRED TO OBTAIN STANDARD SYSTEM GAIN

2. Depleted Foils

The depleted foils consisted of uranium metal depleted to 0.22 per cent U-235. The foils were 0.080-in. in diam by 1.0-in. long. Their mass was determined on a laboratory-type balance to an accuracy of approximately 0.2 per cent.

3. Special Foils

Four special foils were obtained to determine the A/B ratio (see Section III-J)-the ratio of gamma-ray self-shielding to the count-rate efficiency corresponding to the different foil lengths. All types were uranium metal, (0.080-in. diam) enriched to 90 w/o U-235. Two of the foils were 0.75-in. long and two were 1.0-in. long. The 0.75-in. foils were identical in dimensions to the enriched Al-U foils but differed in U-235 content and density, and the 1.0-in. special foils were identical, dimensionally, to the depleted foils and differed only in enrichment.

D. SPECIAL SUBASSEMBLIES

It was not possible to put foils in a normal Core-A fuel or blanket subassembly. To allow the foils to be irradiated in the reactor, it was necessary to design and build special subassemblies so that foils could be located within the reactor core and blanket without creating the possibility of loss of the foil or significant distortion in the flux level and spectrum at the foil positions. These special subassemblies are described in the following:

1. Core Foil Subassembly

The core foil subassembly differed from a normal core subassembly (Section II-B-1) in that the four centrally located pins in the fuel section of the subassembly were removed and replaced by a square sheath. The sheath extended from the lower end of the adjacent fuel pins to a point below the bottom of the handling head. The handling head was hollow and had a cap which could be unscrewed to allow insertion or removal of foils from the sheath. The details of the two core foil subassemblies are shown in Fig. 4. Both subassemblies were designed and manufactured so they could be used as normal core subassemblies upon completion of the foil irradiation program.

2. Blanket Foil Subassembly

The two blanket foil subassemblies were identical to the normal outer radial blanket subassemblies* with the exception that the central blanket pin was replaced by a tube which extended to a point below the handling head of the subassemblies. The tube was plugged at its lower end so that the effective length of the tube in the blanket foil subassemblies was the same as the effective length of the sheath in the core foil subassemblies. As in the core foil subassemblies, the handling head was hollow and capped by a threaded plug which could be removed to allow the insertion or removal of foil specimens. Details of the two blanket foil subassemblies are shown in Fig. 5.

3. Removable Shield Bars

The removable shield bars (Fig. 2), used to make foil irradiations in the reactor thermal shield, differed from a normal shield bar in four respects: (1) the handling head was threaded so that it could be removed, (2) below the handling head there was a cylindrical hole which extended the length of the subassembly, (3) the removable shield bars had camming surfaces, (4) the four shield bars to which they were adjacent had cams which enabled the removable shield bars to be replaced in the thermal shielding.

To adapt the removable shield bars for use during the foil irradiation program, a plug was fabricated which, when installed in the shield bar, sat on a neck created by a change in diameter of the cylindrical hole. The plug was pinned in place to prevent it from floating in the sodium. The length of the plug was designed so that the effective distance from the top of the plug to the shield bar handling head was the same as the effective length of the sheath in the core foil subassemblies and the tube in the blanket foil subassemblies. One of the removable shield bars is illustrated in Fig. 10.

E. OTHER EQUIPMENT

1. Foil Tubes And Spacers

To prevent sodium from coming into contact with the uranium foils during their irradiation, all of the foils were put into stainless steel tubes which were closed before being inserted into a foil subassembly. To maintain the proper elevation of the foils in the foil tubes, stainless steel spacers were inserted between the foil as each foil was being loaded into a foil tube.

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* There were no special inner radial blanket subassemblies made; thus, foil measurements in blanket subassemblies were confined to the outer blanket positions.

SECTION A-A

FIG. 10 REMOVABLE THERMAL SHIELD BAR

The foil tubes were made of 304 stainless steel tubing having an ID of 0.101 in. and an OD of 0.125 in. The tubes were 54-in. long, and were closed at the bottom with a 5/8-in.-long stainless steel plug which was welded in place. After loading, the foil tubes were closed at the top by welding in place a cap which served both to close-off the tube and to provide a convenient means of retrieving the foil tube from the foil subassembly after irradiation. The foil tube and its cap are illustrated in Fig. 11. This system for containing the foils required that the welded cap be removed from the foil tube before the foils could be removed and analyzed.

The spacers used to maintain the elevation of the foils within the foil tubes were made of 1/16-in. diam 308 stainless steel wire. The wire was cut to the proper length and manually filed to within ± 0.001 in. of the nominal length. Both the foil tubes and the spacers were used several times as a means of holding the cost of the program to a minimum.

2. Glove Box

To allow foil tubes to be inserted and removed from the foil subassemblies without allowing air to come in contact with residual sodium clinging to the subassemblies, such transfers were made in a glove box. The glove box was located in the cask-car maintenance pit of the Fuel and Repair Building at a level which allowed a man convenient access to the glove ports when standing in the pit. A valve was supported on the top of the glove box by a thimble piece positioned at the proper elevation to allow it to be mated to the seal flange of the cask car. An illustration of the glove-box installation, and its manner of use is shown in Fig. 12.

Tools were available in the glove box to facilitate removal of the cap from the foil subassembly handling head and removal of the foil tubes from the subassemblies. Foil tubes were put into and removed from the glove box by means of an air lock which was located 180 deg from the viewing glass and gloves. An insulated, induction-heated tube formed the lower section of the glove-box assembly. The tube was required to accommodate the entire subassembly in the inert atmosphere of the glove box during foil-tube exchanges.

The lower section of the glove-box assembly was heated to prevent freezing of the sodium around the subassembly lower support section and the sodium bonding of the upper and lower axial blanket pins in the core foil subassembly. If the subassemblies froze they would not have been useable under full-power conditions because of the possibility of a nonbonded area being created by the freezing and remelting of the sodium bond. Such a nonbond in a blanket pin could lead to a hot spot and possible melting of the pin. The refreezing of the sodium bond in the blanket foil subassemblies was not considered to be a problem because it was not intended that they

FIG. 11 FOIL TUBE, CAP, AND FOIL LOADING

FIG. 12 GLOVE BOX

would be used at full power. Argon gas was supplied from the argon header in the Fuel and Repair Building to provide an inert atmosphere in the glove box.

3. Temporary Instrument Thimble

A temporary instrument thimble (TIT) was used during the tests to provide direct access to the reactor core when the standard foil method was used for power normalization. The TIT occupies the P03-P00 position. This location was designed as a safety-rod position.

The TIT is a gastight, stainless steel tube which is sealed at the lower end. It penetrates the reactor rotating plug, extends downward through the core, and bottoms against the lower guide-tube dashpot. During the tests, which were normalized using a standard foil, an aluminum foil holder was used to locate the standard foil in the TIT. The foils were placed into holes drilled in the holder, thus assuring highly reproducible positioning of the standard foil. At the end of each irradiation, the foil holder was removed from the TIT, the standard foils were removed from the holder, and a new set of standard foils was inserted for use during the next irradiation.

4. Power Monitoring Channels

One method of obtaining a power normalization between irradiations involved the use of the integrated counts accumulated on three neutron-detector channels during the irradiation. Three channels of detection equipment were used. Two channels consisted of BF_3 proportional counters located one each in neutron-counter tubes numbers 3 and 4 (see Fig. 3). The third channel was a fission counter also located in a neutron-counter tube.

The electronics of the three channels were identical and consisted of a preamplifier, amplifier, discriminator, and scaler. The elevations of the neutron detectors in the guide tubes were adjusted so as to obtain a count rate at 10kw of approximately 2500 counts per minute. This resulted in a total integrated count for a 1-hr irradiation at the highest power level used of approximately 6×10^6 counts. This value is within the total count capacity of 1×10^7 for the scalers which were used.

F. FOIL HANDLING AND IRRADIATION

Before the foils could be irradiated in the reactor, it was necessary to load them into foil tubes separated by spacers, to weld the tubes shut, and to perform various handling procedures. The following sections describe these procedures.

1. Loading Of Foil Tubes

The foil tubes were loaded in such a manner that the center of a foil was located at the core central plane, at 5, 10, 15, 16.5, 18.75, 20, 25, 30 and 32 in. above the central plane and at 10 in. below the central plane. Foils were located at 16.5- and 18.75-in. elevations because these elevations were near the top of the fuel region and the bottom of the upper axial blanket, respectively. Several foil tubes were loaded so as to place foils every 2.5 in. axially instead of every 5 in. Also, foils to be chemically analyzed were located at 1 in. above and below the central plane.

The location of the foil array with respect to the geometrical core central plane was determined by the length of the first stainless steel spacer inserted in the tube. To obtain the proper axial positioning of the foil it was calculated that the length of the bottom spacer should be 5.186 in. for the enriched foils and 5.061 in. for the depleted foils.

The analysis of some of the early irradiations (3 - 11) indicated that the results obtained from the foils located 10 in. above and below the core central plane did not agree as they should have if they were located symmetrically with respect to the nuclear center of the reactor. It was thus concluded that either the length of the first spacer was wrong and did not locate the foils at the assumed distances from the core central plane, or, that the flux distribution was skewed slightly from the core midplane. In later irradiations (12 through 19 and the position P00-P05 of irradiation 11) the lengths of the first spacers were changed to 4.556 in. for the enriched foils and to 4.431 in. for the depleted foils (see Table II for a list of the irradiations).

When loading of the foil tube had been completed, the overall length of the column of spacers and foils was measured to ensure against any loading error and to allow an estimate of the inaccuracy in foil location due to the additive errors of the foil and spacer lengths. The overall error in column length was generally within 0.050 in. of the nominal length, thus indicating the error in the results due to mislocation of the foils in the foil tubes was negligible.

To allow identification of individual foil tubes during any particular irradiation, stainless steel spacers of different lengths were the last things placed in each foil tube. The result was that each foil tube in any irradiation could be identified by the length of the first pin removed from the tube.

The length of the stainless steel spacers used to locate foils in 5-in. increments above the core center (with the exception of those located at the end of the fuel and blanket regions) is shown in Fig. 11 for loaded U-235 and U-238 foil tubes. All foils were assigned identification numbers when they were received from the manufacturer. The identification numbers of the foils inserted into each foil tube were recorded on a sheet entitled "foil tube loading diagram". One of these diagrams was made for each foil tube.

2. Handling Of Special Subassemblies

It was necessary that several conditions be considered with respect to the foil subassemblies. The possibility of galling had to be minimized to prevent the possible formation of sodium oxide in the lower support section, and to prevent the sodium bond in the blanket pins of the upper and lower axial blanket of the core foil subassemblies from freezing. The prevention of oxide formation in the lower support section of the core foil subassembly was necessary because of the possibility that the sodium oxide would prevent the spring contained in this section from flexing as the holddown column was lowered, with the possible result that a holddown finger could be bent. The result of the freezing of the sodium bond in the core foil subassembly blanket pin was discussed in Section III-E-2.

The chief concern regarding the use of the blanket foil subassemblies was that galling could occur between the support plate and the nozzle of the subassembly, making it impossible to remove or insert the subassembly. Galling was possible because the nozzles were made of the same metal as the support plate (304 stainless steel) and the nozzles had not been specially treated to prevent galling as were the nozzles of the core subassemblies.

To reduce the possibility of a blanket subassembly becoming frozen in the support plate because of galling, each blanket subassembly was inspected after each irradiation by lifting it out of the finned pot (see below) and visually inspecting the lower support section for signs of the beginning of galling. This inspection was accomplished in the glove box by means of a special tool which enabled the cask-car crane to lift a subassembly out of a finned pot.

Two types of finned pots were used to transport the foil subassemblies between the Fuel and Repair Building and the reactor. The core foil subassemblies were transported in drainable finned pots which when drained, kept the nozzle of the core foil subassembly immersed in sodium. Keeping the nozzle of the subassembly constantly immersed in sodium prevents the formation of sodium oxide around the nozzle spring. The sodium bond in the blanket pins of core foil subassemblies was prevented from freezing by means of induction heating on the lower portion of the glove-box assembly.

The blanket foil subassemblies were transported in finned pots which had holes in the bottom that allowed the sodium in the pots to drain completely as the pot was withdrawn from the reactor sodium by the cask car.

The finned pots were handled between the reactor and the glove box by the cask car. The specific procedure used for handling the foils and subassemblies during this program is listed below:

- (1) Locate foil subassemblies in finned pots by placing a finned pot in the glove box, using the overhead crane and then placing the subassembly in the finned pot using the same crane.
- (2) Purge glove box with argon until a sample of the atmosphere indicated less than 30 ppm of oxygen.
- (3) Put the foil tubes in the foil subassembly by means of the gloves on the front of the glove box. (If the foil subassembly were being loaded into the box for the first time the foil tubes would already have been in the subassembly.)
- (4) Locate the cask car over the glove-box valve, and connect the cask car to the valve by lowering the cask-car seal flange.
- (5) Purge the space between the cask-car ball valve and the glove-box valve.
- (6) Open the cask-car ball valve, and then slowly open the glove-box valve to equalize the cask-car and glove-box pressures.
- (7) Lift the finned pot into the cask car, using the cask-car crane.
- (8) Disconnect the cask car from the glove box and move the cask car to the reactor exit port.
- (9) Connect the cask car to the reactor exit port and close the equipment door.
- (10) Remove the plug from the reactor exit port and lower the finned pot into the transfer rotor.
- (11) Using normal fuel handling procedures, locate the foil subassembly in the proper reactor lattice position.
- (12) Following normal procedures, prepare the reactor for operation.

- (13) Irradiate the foils for 1 hour.
- (14) Following the end of the irradiation, prepare the reactor for fuel handling.
- (15) Using normal fuel handling procedures, remove the foil subassembly from the reactor lattice and put it in the finned pot in the reactor transfer rotor. Return the normal fuel subassembly to the position from which the foil subassembly was removed.
- (16) Hoist the finned pot containing the foil subassembly into the cask car.
- (17) Open the reactor equipment door and disconnect the cask car from the exit port.
- (18) Transfer the cask car to the Fuel and Repair Building and connect it to the glove-box valve.

The finned pot is then lowered into the glove box, and the irradiated foil tubes are removed and are replaced by unirradiated ones. If a blanket foil subassembly is being handled, it would be lifted out of the finned pot and inspected for any signs of galling on the nozzle. The process then begins again by lifting the finned pot into the cask car. The irradiated foil tubes are then removed from the glove box via the air lock and a new set of unirradiated foil tubes are put into the glove box in preparation for the next exchange.

The total time required for the cycle described above was approximately 24 hr. Generally, foils were irradiated at two subassembly positions during each of these cycles, but where possible, more were irradiated. During one irradiation, subassemblies were irradiated at four positions.

3. Irradiation Of Foils

Normal reactor operating procedures and the methods described in Reference 4 were followed during the irradiation of the foils. The irradiation time was held constant at 1-hr and the reactor was always scrammed at the end of the irradiation to keep the power shape as nearly rectangular as possible. The reactor power was increased as nearly as possible on a 60-sec period so that the deviation in the power shape from a rectangle would be as nearly constant as possible. Operating power levels varied from 2 kw for core position irradiations to 500 kw for outer blanket irradiations.

Power level determinations were made from a power-calibrated Keithley micro-microammeter. This ammeter signal was also used by the reactor operator to maintain the reactor power constant.

G. LOCATION OF IRRADIATIONS

During this program a total of 20 separate irradiations was made in 39 lattice positions, the transfer rotor, and the temporary instrument thimble. The results of 18 of the 20 irradiations (17 lattice positions and a special irradiation in the temporary instrument thimble to measure foil self-shielding ratios) are given in this report. The remaining two were preliminary irradiations to verify required reactor powers and equipment operation. The lattice locations at which the foil subassemblies were located during the 17 lattice irradiations are illustrated in Fig. 13. The location of foil subassemblies, the nominal reactor power, the core loading, and the method used for power normalization during each of the lattice irradiations are summarized in Table II.

All irradiations were not conducted with the same core loading. In some cases core changes were made to measure the effects of core edge loading on the fission rate within core and blanket subassemblies, and in some cases changes were made because of reactivity requirements or the convenience of fuel handling. Because the edge loading of the reactor has significance to many of the irradiations, Figs. 14 through 21 have been included to record the loading of the reactor during each of the irradiations.

H. DETERMINATION OF FOIL ACTIVITY

The technique employed to obtain the U-238 capture patterns required that foil activities be determined in two energy regions. The higher energy region was designated band B and included gamma rays from fission product decay and the natural activity of uranium. The band-B activity was measured for both the enriched and depleted foils and was assumed to be proportional, (after a correction for natural background), to the number of fissions which had occurred in a foil. The lower energy range was designated band A and was measured only for the depleted foils. This energy included gamma rays which originated from fission products, natural uranium activity, and from the decay of the precursors of plutonium formed by neutron capture in U-238. The band-A activity was assumed to be proportional to the U-238 capture rate after corrections had been applied for fission product activity and natural background. The energy ranges of the bands and how they were defined are discussed in the following sections.

1. Band B

The two criteria which governed the determination of the energy limits of band B were:

FIG. 13 LATTICE POSITIONS AT WHICH FOILS WERE IRRADIATED

TABLE II- - SUMMARY OF IRRADIATIONS*

<u>Irradiation no.</u>	<u>Subassembly locations</u>	<u>Nominal reactor power, kw</u>	<u>Core loading illustration (Fig. no.)</u>
<u>STANDARD FOIL NORMALIZATION</u>			
3	P00-P01	2.5	14
4	P04-P00, N01-P01	2.5	14
5	P05-P00, P06-P00	4.0	15
6	P00-P00, P02-P00	2.0	14
7	P00-P03, N03-P03	2.0	14
8	N04-P04, N04-N04	3.5	14
9	N05-N05, N05-N03	4.5	16
10	N06-N05, N06-P05	5.5	17
<u>INTEGRATED- COUNT NORMALIZATION</u>			
11	N04-P04, N06-P05, P00-P05	4.0	18
12	P00-P10, P00-N10	60	18
13	P00-P14, N10-P10, N11-P09	500	18
14	N06-P04, P08-P00	7.5	19
15	P14-P00, P14-P01, P16-P00	400	19
16	P14-P00, P16-P00, P00-N16, T.R.	200	20
17	P12-P00, P00-P12	120	19
18	N08-P08, N06-P06	75	21
19	P00-P07, P00-P09	30	20

* Length of irradiation - 1hr

FIG. 14 CORE LOADING DIAGRAM FOR IRRADIATION NOS. 3, 4, 6, 7, 8

FIG. 15 CORE LOADING DIAGRAM FOR IRRADIATION NO. 5

FIG. 16 CORE LOADING DIAGRAM FOR IRRADIATION NO. 9

FIG. 17 CORE LOADING DIAGRAM FOR IRRADIATION NO. 10

SR Safety Rod	S Shim Rod	CS Core Shim Subassembly	Inner Radial Blanket Subassembly
R Regulating Rod	TIT Temporary Instrument Thimble	Core Subassembly	S Neutron Source

FIG. 18 CORE LOADING DIAGRAM FOR IRRADIATION NOS. 11, 12, 13

FIG. 19 CORE LOADING DIAGRAM FOR IRRADIATION NOS. 14, 15, 17

FIG. 20 CORE LOADING DIAGRAM FOR IRRADIATION NOS. 16, 19

FIG. 21 CORE LOADING DIAGRAM FOR IRRADIATION NO. 18

SR Safety Rod
S Shim Rod
R Regulating Rod
OR Oscillator Rod

Core Subassembly
Inner Radial Blanket Subassembly
S Neutron Source

- (1) The counts obtained on an irradiated uranium foil should be as nearly as possible proportional to the number of fissions which had occurred in a foil during the irradiation.
- (2) A change in gain of the counting system should cause the smallest change in the count rate from a fission product gamma-ray spectrum.

The first condition was satisfied anywhere in the energy region above the highest-energy gamma ray resulting from the nonfission capture of U-238. The highest-energy gamma ray from this reaction is the 0.335-Mev gamma ray originating from Np-239. Because a Cs-137 source was used to check the counting system for gain stability, it first appeared that the most convenient place to locate the lower energy limit of band B would be at 0.662 Mev, the energy of the gamma ray from Cs-137. This energy was very close to a peak in the fission product gamma-ray spectrum. However, a valley existed in the fission product gamma-ray spectrum at 0.600 Mev. A study was conducted on irradiated uranium foil as a fission product gamma source to determine which of these energy settings provided the greater counting stability for a change in the gain of the counting system.

Variations in gain from the standard gain were obtained by varying the high voltage to the photomultiplier tube. Two sets of count rates as functions of high-voltage deviation from that required to produce the standard system gain were obtained. The first was obtained with the energy cut off the system set so that all gamma rays with energy above 0.662 Mev were counted, and the second was obtained with the energy cut off the system set so that all gamma rays with energy above 0.600 Mev were counted. The results of the two sets of data are illustrated in Fig. 22. This shows that appreciable counting stability of fission product gammas is to be obtained by using a gamma-ray energy cutoff of 0.600 Mev. Moreover, it should be noted that the accompanying fraction of change count rate for a given change in gain is a factor of nine less than that for the standard Cs-137 source used to stabilize the gain (Section III-B-3).

From the above information it was concluded that band B would include all gamma rays with an energy above 0.600 Mev.

2. Band A

U-238 capture patterns were determined by the analysis of the radiation which lay in band A of the gamma-ray spectrum. This activity originated principally from gammas emitted by Np-239, Pu-239^m, and Pu-239. This decay chain emits both beta and gamma rays, some of the latter of which are internally converted to give plutonium X rays. The assignment of specific gamma rays to specific nuclei is complicated and has not been fully established. However, the origin of the gamma rays is not important

FIG. 22 INTEGRAL FISSION PRODUCT COUNT RATE AS A FUNCTION OF HIGH-VOLTAGE DEVIATION FROM THAT REQUIRED TO OBTAIN STANDARD SYSTEM GAIN

as long as they are proportional to the neptunium decay rate. Analysis of the decay of neptunium has shown that a large number of gamma rays from the nuclei listed above combines to give a large peak in the gamma-ray spectrum at approximately 106 kev. Hence, it was concluded that band A would be defined so as to straddle this peak.

The actual upper and lower limits of band A were determined by work done at Argonne National Laboratory² to optimize the variables involved. The results of this work indicated that the optimum limits on band A were 87 to 125 kev. A pulse height analysis obtained 38 hr after irradiation of a depleted foil verified that these settings were adequate.

3. Shelf Ratios

During some of the irradiations made at locations near the outer edge of the outer radial blanket, the reactor power level at which foils were irradiated was governed by the desire to get a usable band-B count rate from U-238 fission products. To accomplish this, the irradiations were made at extremely high power levels. This resulted in band-A count rates for the depleted foils which were too high to allow them to be counted on the top shelf (the shelf closest to the detector) of the shielding geometry. The count rate of these foils was reduced to a suitable value by moving them to the bottom shelf of the shielding geometry during counting. This approach necessitated the determination of the shelf ratio for band-A gamma rays.

The shelf ratio for gamma rays between 87 and 125 kev was determined by measuring the band-A activity of a depleted uranium foil on both the top and bottom shelf of the counting geometry several times after the end of the irradiation. The result of this work is given in Table III. Because it was evident that the shelf ratio varied with time, only those values obtained between 34 hr - 40 min, and 37 hr - 30 min after scram were averaged to obtain a value for the band-A shelf ratio. The value obtained from averaging five determinations was 7.122. The value obtained by averaging all determinations was 7.104. The former value was used to correct band-A activities obtained with a foil located on the bottom shelf of the shielding geometry.

TABLE III - SHELF RATIO DATA - BAND A

<u>Time, hr-min</u>	<u>Count rate top shelf,* cpm</u>	<u>Count rate bottom shelf,* cpm</u>	<u>Shelf ratio</u>	
31:30.0	635,734	90,867	6.996	
31:50.0	633,101	89,242	7.094	
32:15.0	624,723	88,845	7.032	
33:50.0	617,479	87,166	7.084	
34:40.0	614,689	86,284	7.124	} Avg = 7.122
35:34.0	609,446	86,183	7.072	
36:12.0	602,234	84,453	7.131	
36:35.0	600,214	84,019	7.144	
37:30.0	596,859	83,634	7.137	
46:00.0	535,158	74,111	7.221	

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* Counts corrected for background

I. REACTOR POWER LEVEL NORMALIZATION

To allow the comparison of results obtained from foils irradiated at different power levels and counted at different times after the end of an irradiation, it was necessary to normalize all results for power to a common shutdown time. For irradiations 3 through 10, these normalizations were accomplished by irradiating a standard foil in the TIT (temporary instrument thimble) simultaneously with the foils it was to normalize. After irradiation 10, the TIT was removed from the reactor and was not available to accommodate a standard foil for the normalization of irradiations 11 through 19. The power normalization of these irradiations was accomplished by means of an integrated count rate obtained during the irradiation on the three neutron-counting channels. To allow all results to be compared on a common basis, an intercalibration was obtained between the standard foil and integrated-count method of power normalization. The details of the two power normalization methods and of the intercalibration of the two methods are described below.

1. Standard Foil Method

Power normalization by means of standard foils was accomplished by locating a depleted and an enriched foil in the TIT during each irradiation. These foils were identical to the ones being irradiated in the reactor lattice. The foils were held within the TIT by means of a foil holder (see III-E-3). At the end of an irradiation the standard foils were removed and counted at regular intervals (approximately every 30 min) during the time period required to count the foils which had been irradiated in lattice positions. From the standard foil data, three curves were constructed for each irradiation. One gave the activity as a function of decay time of the enriched standard foil in band B for all times during which enriched foils were counted in band B; a similar curve gave the activity as a function of decay time of the depleted standard in band B; and a third curve gave the activity as a function of decay time of the depleted standard in band A. Power and time normalization were accomplished by dividing the count observed for any lattice irradiated foil by the count obtained from the plot of the standard foil activity at the time the activity of the lattice-irradiated foil was obtained.

As foils were irradiated in lattice positions farther away from the core center, it was necessary to increase the reactor power to obtain a count large enough to be accurate. As the reactor power was increased, it was necessary to move the standard foils to holes in the TIT foil holder which were farther away from the center of the reactor. This was necessary to prevent the standard foils from becoming too active to be of use. During the course of these irradiations, enriched standard foils were located at three elevations in the TIT and the depleted standard foils were located at two elevations.

To maintain the validity of the standard foil count rate as a power normalization among all irradiations, it was necessary to correct all standard foil counts of each type to a common elevation. The common elevation chosen was the one closest to the core center. As stated above, enriched standard foils were located in holes No. 5, No. 11, and No. 13 of the TIT foil holder (see Fig. 23). Depleted standard foils were located at two elevations, hole No. 6 and hole No. 10 of the TIT foil holder. Correction factors to the hole closest to core center were obtained for all elevations above the closest to core center by simultaneously irradiating a foil at each elevation. In the case of the enriched foils, two irradiations were required because the large difference in count rate between a foil irradiated in hole No. 6 and hole No. 13 did not allow simultaneous irradiation at three elevations. After being irradiated, the foils were counted several times between approximately 20 and 50 hr after the end of the irradiation. The resulting count rates were corrected for mass and background and plotted as a function of time. The curves obtained for the band-B count rate of enriched foils irradiated in holes 5 and 11 of the TIT (temporary instrument thimble) foil holder are illustrated

FIG. 23 TEMPORARY INSTRUMENT THIMBLE WITH FOIL HOLDER

in Fig. 24. Similar curves were drawn for the relationship between the count rate in band A and band B for depleted foils irradiated in holes 6 and 10 of the TIT foil holder, and for the band-B count rate of enriched foils irradiated simultaneously in holes 11 and 13 of the TIT foil holder.

From these curves the ratios of the count rates were obtained as a function of time, for identical foils which were simultaneously irradiated at different elevations in the TIT. These ratios for each foil type, in each counting band for each hole, were averaged to determine a correction factor to be applied to all standard counts obtained at elevations other than those nearest to the core center. The resulting correction factors are given in Table IV.

TABLE IV - CORRECTION FACTORS* APPLIED TO
STANDARD FOILS IRRADIATED AT
ELEVATED POSITIONS IN THE
TEMPORARY INSTRUMENT THIMBLE

Elevated position (hole no.)	Enriched foil, band B	Depleted foil	
		Band B	Band A
10	-----	2.54 (4, 5, 8) [†]	1.504 (4, 5, 8, 9, 10) [†]
12	-----	12.737 (12) [†]	-----
11	2.146 (4, 5, 8) [†]	-----	-----
13	4.468 (10) [†]	-----	-----

* Response at the standard elevation (hole Nos. 5 and 6 for depleted and enriched foils, respectively) relative to that obtained at elevated positions indicated.

† Numbers in parentheses are the irradiation numbers for which it was required to apply the corresponding correction factor.

The appropriate correction factors were applied to the standard foil count rate from a given irradiation before a graph was drawn of the standard foil activity versus time. This enabled values to be taken directly from the graph when applying the power level and time correction during the analysis of the irradiated foil data.

FIG. 24 ELEVATION NORMALIZATION DATA FOR ENRICHED STANDARD FOILS

2. Integrated-Count Method

Two channels (channel 3 and channel A) of neutron-counting equipment and a Keithley micro-microammeter (see Section III-E-4) were available for use with the integrated-count method. In its simplest form this method consists only of starting the neutron-monitoring channels at the beginning of the irradiation and allowing them to continue to accumulate counts throughout the entire length of an irradiation. In theory, the number of counts accumulated during this time should be directly proportional to the integrated power of the irradiation. In practice, however, it was necessary to make some adjustments to the integrated counts to correct for obvious anomalies of uncertain origin (i.e., voltage variation, amplifier gain, power drift, etc.).

The accumulated counts obtained on the two neutron-counting channels are shown in Table V, along with the average Keithley reading obtained for each irradiation for which the integrated-count method was used for power normalization. Also given in Table V are three types of ratios for each irradiation. These are: (1) the ratio of the average Keithley reading to the integrated count on channel 3; (2) the ratio of the average Keithley reading to the integrated count on channel A; and, (3) the ratio of the integrated count on channel A to that on channel 3. If there were no irregularities in the method, each ratio would have a constant value for all irradiations, but they do not. In an attempt to obtain the best value for use in power normalization, the counts observed on channels 3 and A and the observed current from the Keithley micro-microammeter were empirically adjusted with the aid of additional irradiation and power-monitoring data to bring each of the three ratios for each irradiation as close as possible to the average value obtained during all irradiations. The adjusted counts and current values are given in Table VI. For the purpose of power normalization, the adjusted values for the channel-3 count rate were used because they were least affected.

3. Intercalibration of Normalization Methods

To compare the results of all irradiations on a common basis it was necessary to intercalibrate the two power normalization methods. In practice, a function $R(t)$ is obtained which, when multiplied by the integrated count, gives the time-dependent count rate that would have resulted from irradiation of a standard foil in the temporary instrument thimble. Thus all foil count rates obtained using the integrated-count method can be made relative to standard foil count rates. The desired function $R(t)$ was obtained by using both the integrated-count method and the standard foil method, to normalize irradiations with foils at the same lattice position. (Both normalization methods could not be used simultaneously because the TIT had been removed by the time it was decided that the integrated-count method of normalization would have to be used.) The derivation of $R(t)$ is given below.

TABLE V - INTEGRATED COUNTS, CURRENTS, AND RATIOS FOR EACH
IRRADIATION NORMALIZED BY THE INTEGRATED-COUNT METHOD

Irradiation no.	Integrated counts		Keithley current, amp.	Ratio of channels		
	Channel 3	Channel A		$K/3 \times 10^{-13}$	$K/A \times 10^{-13}$	A/3
11	65,392	45,643	0.244×10^{-7}	3.73	5.34	0.698
12	868,190	600,000	0.329×10^{-6}	3.79	5.48	0.691
13	6,616,734	4,196,960	0.233×10^{-5}	3.52	5.55	0.634
14	*	68,070	0.398×10^{-7}	*	5.85	*
15	6,192,105	3,945,800	0.234×10^{-5}	3.78	5.93	0.637
16	3,434,432	2,265,260	0.124×10^{-5}	3.61	5.47	0.660
17	1,763,769	1,125,783	0.650×10^{-6}	3.69	5.77	0.638
18	1,086,910	697,814	0.400×10^{-6}	3.68	5.73	0.642
19	461,908	307,468	0.168×10^{-6}	3.64	5.46	0.666

---

* Channel 3 inoperative during irradiation no. 14

TABLE VI - ADJUSTED COUNTS AND CURRENTS FOR
EACH IRRADIATION NORMALIZED BY
THE INTEGRATED-COUNT METHOD

Irradiation no.	Adjusted integrated counts		Adjusted Keithley current, amps.
	Channel 3	Channel A	
11	66,250	43,000	0.240×10^{-7}
12	875,000	570,000	0.320×10^{-6}
13	6,600,000	4,250,000	0.238×10^{-5}
14	107,200	69,200	0.387×10^{-7}
15	6,200,000	4,000,000	0.224×10^{-5}
16	3,450,000	2,230,000	0.125×10^{-5}
17	1,770,000	1,140,000	0.640×10^{-6}
18	1,087,000	698,000	0.390×10^{-6}
19	462,500	298,000	0.167×10^{-6}

Since the ratio of the foil activity at any lattice position to the activity of the standard foil in the TIT is a constant, the following relationship may be established.

$$\frac{A_p^s}{A_f^s} = \frac{A_p^i}{A_f^i} \quad (1)$$

where,

s = an irradiation normalized for power by means of a standard foil

i = an irradiation normalized for power by means of an integrated count

A_p = the activity of a foil irradiated at any position, p

A_f = the activity of the standard foil irradiated in the TIT simultaneously with A_p

In the above equation, the activities are corrected for mass, background, and to a common time from the end of the irradiation.

It was not possible to obtain a value for A_f^i directly because the TIT was not available for the irradiation of a standard foil during irradiation i. However, A_f^i can be calculated as a function of time, $A_f^i(t)$, by measuring the values of A_p^i as a function of time and solving Eq. (1) using the value for the

ratio obtained during the irradiation normalized by the standard foil method. If during irradiation, i , a value is obtained for the integrated count, the following equation can be solved for $R(t)$.

$$R(t) = \frac{A_f^i(t)}{I} \quad (2)$$

where I is the integrated count used for normalization, in this case the adjusted value of the integrated channel-3 count. Note that $R(t)$ is a function only of decay time relating the adjusted integrated count on channel 3 of the neutron-monitoring equipment to what the count rate would have been on a standard foil if one had been irradiated in the TIT.

This technique was used to obtain three $R(t)$ functions. One gave $R(t)$ for band B of the enriched foils, $R(t)_B^{25}$; one gave $R(t)$ for band B of the depleted foils, $R(t)_B^{28}$; and one gave $R(t)$ for band A of the depleted foils, $R(t)_A^{28}$. These curves are illustrated in Figs. 25, 26 and 27.

A typical pseudo-standard foil activity curve, based on Table VII, is illustrated in Fig. 28; it is constructed of values of $R(t)_B^{25}$ obtained from Fig. 25 and the value of the integrated count used to normalize irradiation number 12. The curve of the deduced standard foil activity as a function of time was applied to the analysis of all foil data in the same manner as the curve obtained using standard foil activity.

The data necessary for defining $R(t)$ was obtained by repeating the irradiation of foils at lattice position N06-P05 while obtaining an integrated count on the neutron-counting channels. Lattice position N06-P05 was selected because it appeared to contain the least perturbations of any position previously irradiated for which a standard foil had been used for power normalization.

J. DETERMINATION AND VERIFICATION OF RATIO OF BAND A TO BAND B

As discussed under the general description of the method, before the counts observed in band A of a depleted foil can justifiably be assumed to be proportional to the U-238 capture rate, the fission product gamma-ray component of the count must be subtracted. The method for determining the magnitude of this fission product component was developed and verified at Argonne National Laboratory and is discussed in Reference 2. It is based on the fact that for times greater than 35 hr after fission, the ratio of U-235 fission product gamma rays in band A to the U-235 fission product gamma rays in band B is the same as the ratio of the U-238 fission product gamma rays in band A to the U-238 fission product gamma rays in band B. Hence,

FIG. 25 $R(t)$ FOR ENRICHED FOILS IN BAND B

FIG. 26 $R(t)$ FOR DEPLETED FOILS IN BAND B

FIG. 27 $R(t)$ FOR DEPLETED FOILS IN BAND A

TABLE VII - CALCULATION OF STANDARD FOIL ACTIVITY
FROM $R(t)_B^{25}$ DURING IRRADIATION NO. 12

Time*	$R(t)_B^{25} \times 10^{-2}$	Standard count rate, $875,000 \times R(t)_B^{25}$
35:00	4.638	40,582
35:20	4.575	40,031
35:40	4.513	39,489
36:00	4.453	38,963
36:20	4.392	38,430
36:40	4.333	37,914
37:00	4.277	37,424
37:20	4.222	36,943

NOTE: The adjusted channel-3 integrated count for irradiation number 12 was 875,000.

* Time in hours and minutes from the end of the irradiation

if the band A-to-band B ratio (A/B ratio) can be determined for U-235, it is also known for U-238. When it is known for U-238 it can be used to determine the fission product component of the band-A count of the depleted foil by multiplying the band A-to-band B ratio by the counts obtained for the depleted foil in band B.

It is necessary to determine the ratio of the counts for band A to those for band B for each foil and counting geometry. Hence, an evaluation was made of this ratio applicable to the foil and counting geometry used during this experiment.

To measure the above ratio and the self-shielding ratio (discussed in Section III-K), a special irradiation was made during which three foils were irradiated at the same elevation in the TIT. One of these foils was used to determine the ratio of the counts for band A to those for band B, one to determine the self-shielding ratio, and the third was used for both determinations. One of the two foils used to measure the A/B ratio was a normal, depleted uranium pin; the other was identical in geometry and material to the depleted pins but differed in that the uranium was enriched to 90w/o U-235, instead of being depleted to 0.22 w/o. The TIT foil holder was modified so that the above foils could be situated so that they would receive identical neutron irradiation. The foils were irradiated for 1 hr at 100 watts,

FIG. 28 COUNT RATE OF STANDARD FOIL DEDUCED FROM INTEGRATED-COUNT METHOD OF POWER NORMALIZATION

after which the reactor was scrammed, the foils removed from the TIT, and their activity analyzed periodically up to 120 hr after the scram. Both the enriched and the depleted foil activities were determined in both band A and band B. The activity data corrected for background and mass for these two foils are given in Table VIII.

TABLE VIII - COUNT RATES FROM FOILS IRRADIATED
TO EVALUATE A/B RATIO

Time*	Enriched foil [†]		Depleted foil [‡]	
	Band A, cpm	Band B, cpm	Band A, cpm	Band B, cpm
4:00	177,540	825,385	32,308	34,487
4:50	156,137	619,225	29,719	25,083
9:00	100,956	268,768	24,444	10,753
10:50	88,735	216,031	23,003	8,782
13:00	76,557	169,556	22,619	7,252
14:55	68,076	141,777	20,798	5,995
24:00	45,394	76,515	18,053	3,207
28:00	36,649	62,775	16,807	2,685
32:40	30,449	51,167	15,209	2,293
35:00	30,531	46,690	15,028	1,877
35:50	28,869	44,698	15,412	1,858
37:20	28,695	43,147	14,526	1,834
38:50	26,277	40,957	14,538	1,990
39:50	25,267	39,524	14,316	1,660
49:10	19,934	30,344	13,595	1,282
55:50	16,426	25,467	11,238	1,032
76:55	12,027	16,672	9,007	751
99:40	8,369	13,030	6,737	541
121:35	5,264	10,650	4,793	506

* Hours and minutes after scram

† Wt of enriched foil is 1.5403 g

‡ Wt of depleted foil is 1.5768 g

In determining the A/B ratio, it was necessary to separate from the band-A counts of the enriched foils those counts which originated from U-238 capture in the 10 per cent of the foil material that was U-238. To accomplish the reduction of the raw data, a set of simultaneous equations was developed which allowed the various components of the band-A enriched foil count rate to be determined. These equations are:

$$I_e^B = m_e^{25} x + m_e^{28} y \quad (3)$$

$$I_d^B = m_d^{25} x + m_d^{28} y \quad (4)$$

$$I_e^A = m_e^{25} u + m_e^{28} v + m_e^{28} z \quad (5)$$

$$I_d^A = m_d^{25} u + m_d^{28} v + m_d^{28} z \quad (6)$$

where,

I_e^B = total fission product activity in band B of the enriched foil

I_d^B = total fission product activity in band B of the depleted foil

I_e^A = total activity in band A of the enriched foil

I_d^A = total activity in band A of the depleted foil

x = fission product activity per g from U-235 in band B

y = fission product activity per g from U-238 in band B

u = fission product activity per g from U-235 in band A

v = fission product activity per g from U-238 in band A

z = neptunium activity per g from U-238 in band A

m_e^{25}

m_d^{28}

m_d^{25}

m_e^{28}

= the mass in grams of U-235 in the enriched foil, the mass of U-238 in the depleted foil, etc.

Equations(3)-(6) can be simplified by assuming that the fission product activity in bands A and B of the enriched foil from U-238 fission products can be neglected. With this assumption and by inserting the proper masses in place of the symbols, the following four equations in five unknowns result:

$$I_e^B = 1.3841 x \quad (7)$$

$$I_d^B = 0.00347 x + 1.5733 y \quad (8)$$

$$I_e^A = 1.3841 u + 0.1562 z \quad (9)$$

$$I_d^A = 0.00347 u + 1.5733 v + 1.5733 z \quad (10)$$

These equations can be solved for the five unknowns (u, v, x, y, z) by using the experimental observation that the ratio between x and y equals the ratio between u and v (for times greater than 35 hr after irradiation). The A/B ratio is then obtained by taking the ratio of u to x.

The values for the band-A and band-B activities for the enriched and depleted foils used to solve the above equations were obtained from the smooth curves drawn through actual count rate values. The calculated values for x, x/y, u, z, and the A/B ratio as a function of time are given in Table IX. The A/B ratio is plotted in Fig. 29, indicating that time dependence of the A/B ratio becomes quite small after about 35 hr. Since the depleted foil data were taken between 35 and 38 hr after irradiation, a single value of 0.636 for the A/B ratio was used in the determination of the U-238 capture rate.

To verify the values obtained for the A/B ratio, the data obtained for band B and band A of the depleted foil were analyzed to yield the relative neptunium activity as a function of time. The results are illustrated in Fig. 30. The estimated best-fit straight line through the empirical points represents a half life of 58.0 hr. This agrees well with the commonly accepted value of 56.4 hr, indicating that the A/B ratio serves its desired purpose.

K. DETERMINATION OF THE SELF-SHIELDING AND LENGTH-CORRECTION FACTOR

To obtain the U-238 fission distribution from the counts observed in band B for the depleted foils, it was necessary to first subtract the portion that originated from fission products generated by the fissioning of the U-235 impurity present in the depleted foils. To obtain an estimate of the magnitude of the correction required for the U-235 component it was necessary to know four parameters: (1) the mass of the U-235 in each depleted foil, (2) the fission rate per unit mass of U-235 at each location at which a depleted foil was irradiated, (3) the difference in self-shielding for band-B fission

TABLE IX - NEPTUNIUM-239 ACTIVITY AND A/B
RATIO AS A FUNCTION OF TIME

<u>Time*</u>	<u>x,</u> <u>cpm/g</u>	<u>x/y</u>	<u>u,</u> <u>cpm/g</u>	<u>z,</u> <u>cpm/g</u>	<u>A/B</u>
5	447,944	30.05	108,529	14,827	0.242
10	120,507	29.28	67,832	12,407	0.563
15	101,149	28.16	48,634	11,323	0.481
20	70,443	27.87	37,466	10,491	0.532
25	53,103	28.64	30,222	9,874	0.569
30	41,904	28.60	25,188	9,329	0.601
35	33,957	29.00	21,586	8,869	0.636
40	28,538	28.80	18,175	8,386	0.637
45	24,203	28.29	15,741	7,831	0.650
50	21,097	28.20	13,787	7,362	0.654
55	18,785	28.00	12,337	6,937	0.657
60	16,762	27.99	11,124	6,506	0.664
75	12,282	28.13	8,620	5,427	0.702
80	11,560	27.77	8,001	5,132	0.692
90	9,826	27.80	6,960	4,565	0.708
100	8,381	27.81	6,079	4,090	0.725

* Hours after the end of the irradiation

product gamma rays in the enriched and depleted uranium pins and, (4) the effect of the depleted pin being 1-in. long compared to the 3/4-in. length of the enriched pin.

The mass of U-235 in each pin was obtained, using the reported value of 0.22 w/o for the amount of U-235 in the depleted foils in conjunction with the weight of each foil. The 0.22 w/o was the isotopic analysis reported by the vendor who fabricated the pins. During the irradiation program, depleted and enriched foils were irradiated as pairs. This allowed the relative U-235 fission product activity per unit mass to be determined for the enriched foils at every location at which depleted foils were irradiated. However, because the enriched foils were uranium-aluminum alloy 3/4-in. long and the depleted foils were uranium metal 1-in. long, a factor had to be obtained to correct for the difference in the gamma-ray self-shielding and the effect of length for band-B fission product gamma rays in the two types of foils.

FIG. 29 A/B RATIO FOR FISSION PRODUCTS

FIG. 30 MEASURED NEPTUNIUM-239 DECAY

A straightforward technique for obtaining the factor required to correct for the difference in self-shielding and length was to simultaneously irradiate foils whose only differences were their densities and lengths. One of the foils was a normal, highly enriched U-Al alloy pin. The other was the same special highly enriched uranium metal pin used to measure the A/B ratio. (See Section III-C-3 for details of the special pin.)

The two foils were irradiated simultaneously at the same location in the temporary instrument thimble during the special irradiation discussed in Section III-J. The length of irradiation was 1 hr at a power level of 100 watts. The foil activities were measured at time intervals between 1 and 120 hr after scram, at the end of the irradiation. The resulting activities were analyzed by correcting for material background and mass. A combined factor for both effects was obtained by taking the ratio of the counts per minute per mg of U-235 in the 1-in. long enriched uranium foil to the same quantity obtained for the enriched uranium-aluminum foil. These quantities for both foils as a function of time, as well as the correction factor resulting from differences in self-shielding and length, are listed in Table X.

Independent observations of the effect on the counting efficiency due to the difference in length between the 3/4-in. and 1-in. foils indicated that the effect is essentially negligible. Thus, the correction obtained above is due almost entirely to the difference in the gamma-ray self-shielding between the alloy and uranium metal foils.

Some of the U-238 fission distribution data was analyzed prior to the availability of the special, highly enriched uranium metal foils used to determine the self-shielding correction. An alternate method was used to obtain the correction factor used in the data analysis. The more straightforward method described above served as a check on the earlier technique. The earlier technique involved the simultaneous irradiation of a normal, depleted uranium foil and a normal, highly enriched U-Al alloy foil. Measurement of the resulting band-B activities, use of independently obtained values of the U-235 to U-238 fission ratio, and the ratio of the gamma-ray activity per unit fission of U-235 to that of U-238 as a function of time after fission, allowed a determination of the desired gamma-ray self-shielding correction. The required fission ratio was obtained by radiochemical analysis (Section IV-B), and a constant gamma-per-fission ratio, valid for long periods after shutdown, was obtained from Fig. 7 of Reference 5. The resulting value for the gamma-ray self-shielding correction, assumed constant for long periods after shutdown, was 0.853. This compares well with the values in Table X for long periods after shutdown and leads to an error of ~5 per cent for the worst case of foils irradiated near the handling heads of subassemblies, far out in the radial blanket where the U-235 fission component in the depleted foil is a maximum.

TABLE X - GAMMA-RAY SELF-SHIELDING AND
LENGTH-CORRECTION FACTOR

<u>Decay time*</u>	<u>Activity</u>		<u>Correction factor (col. 2/col. 3)</u>
	<u>Enriched U metal foil, cpm/g</u>	<u>Enriched U-Al foil, cpm/g</u>	
0:57.0	3100	4123	0.752
1:24.0	2100	2836	0.740
1:55.0	1500	2021	0.742
2:26.5	1170	1560	0.750
2:59.0	905	1207	0.750
3:54.5	635	853	0.744
4:26.5	530	711	0.745
4:48.5	470	635	0.740
6:39.5	303	405	0.748
8:55.0	207	278	0.745
10:44.5	163	219	0.744
12:58.5	126	168	0.750
14:51.0	110	141	0.759
24:01.5	56.5	69.7	0.810
27:54.5	46.0	56.0	0.821
32:54.5	38.0	45.0	0.844
34:55.0	34.8	41.4	0.840
34:43.5	34.0	40.5	0.840
37:25.5	32.0	38.0	0.842
38:36.0	31.0	37.0	0.838
38:59.5	30.5	36.2	0.843
39:59.5	29.5	35.3	0.836
40:00.0	29.2	34.8	0.839

---

* In hours, minutes, and tenths of minutes from the end of the irradiation.

IV. ANALYSIS OF EXPERIMENTAL RESULTS

The measured activity of a foil required corrections and normalizations before it could be meaningfully compared with the activities obtained from other foils. It was the purpose of the experimental analysis to apply, in an orderly manner, the necessary corrections and normalizations. These are described in the following.

A. NUMERICAL ANALYSIS

1. Background Count Rate

The background count rate was that observed when an unirradiated foil was analyzed with the same counting equipment used to analyze an irradiated foil. The count rate represented the sum of two components. The first was the environmental background, originating from radioactive contaminants in the lead shield, cosmic radiation, and irradiated foils located in the counting room. The second source was the radiation originating from the natural decay of the uranium in the foils.

The background count-rate corrections for both the enriched and depleted foils were obtained periodically by counting unirradiated, enriched, and depleted foils. The uranium contained in the enriched foil added very little count to the natural room background. Hence, the background correction for the enriched foils was applied without considering the mass of the U-235 in the foils. However, the uranium in the depleted foil added a considerable amount to the room background activity in band A and band B. For this reason, both the depleted irradiated foil activity and the depleted background foil activity were corrected for the differences in the foil mass before the background count was applied. The background count was applied by subtracting it from the gross count in the case of the enriched foils and from the gross count divided by the foil mass in the case of the depleted foils.

2. Mass Correction

The uranium content of the enriched foils and depleted foils varied a maximum of 8 per cent and 5 per cent, respectively. A correction for mass differences was accomplished by dividing the total count observed for each foil by the uranium mass of each foil.

The masses of the depleted foils were determined with the use of a laboratory-type analytical balance. The uranium content of the enriched foils was determined by the fabricator who used a scintillation counting system

which had been calibrated to determine the relationship between the counts per minute observed on a foil and the U-235 content of that foil. Each foil was counted and the indicated U-235 content was reported.

3. Power Level and Time Correction

As explained in detail in Section III, a correction had to be applied to all results to compensate for the effect of counting the foils at different times after the end of an irradiation. Also required was a normalization to compensate for the different reactor power levels at which foils were irradiated. These two corrections were applied by the use of a single factor that represented the activity of a standard foil at a specific time after the end of each irradiation.

This factor, which was plotted as a function of time, was obtained by measuring the activity of the standard foil at approximately 1/2-hour intervals during the time required to determine the activity of the lattice-irradiated foils for any particular irradiation. The counts thus obtained for the standard foils were corrected for background and mass and then plotted as a function of time. The standard foil count rate was selected from the plot for the time at which the activity of a lattice-irradiated foil was determined. The time correction and power normalization were accomplished by dividing the adjusted lattice-irradiated foil count by the adjusted standard foil count. In all cases, foil activities were obtained for one minute. Thus, total count is actually counts per minute.

This same method was used to correct the results of the U-235 fission distribution data, the U-238 fission distribution data, and the U-238 capture distribution data. However, each adjusted standard foil activity was obtained by analyzing the proper data from the proper standard foil.

4. Effect of U-235 in the Depleted Foils

To obtain a value for the U-238 fission distribution it was necessary to correct the depleted foil band-B activity for the portion which originated from the fission products of U-235. This correction ranged from approximately 6 per cent at the core center to approximately 65 per cent at the farthest position in the outer radial blanket.

The magnitude of the correction was obtained using information from the enriched foil which was irradiated simultaneously at the same location as the depleted foil. The mass and background-corrected activity of the enriched foil was corrected for the differences in the self-shielding between the two foils (See Section III-K); the resulting activity per unit mass of U-235 was then multiplied by the mass of U-235 in the depleted foil. The resulting value was subtracted from the observed depleted foil band-B activity to obtain an activity which was assumed to be proportional to the U-238 fission rate.

5. Effect of Fission Product Gamma Rays in Band A of the Depleted Foils

To obtain a count from band A of a depleted foil proportional to the neutron capture in U-238, it was necessary to subtract that portion of the band-A count which originated from either U-235 or U-238 fission products. Section III-J explained how a value was obtained for the ratio of the band A to-band-B activities for fission products in the depleted uranium foils. The band-A activity of the depleted foils was corrected for fission product gamma rays by subtracting from the observed band-A count the product of the band A-to-band-B ratio and the count observed in band B for each depleted foil. The band-B activity was determined as part of the technique for obtaining the U-238 fission distribution. The remaining activity is assumed to be proportional to the number of neutron captures which occurred in U-238 during the irradiation.

6. Numerical Analysis Procedure

Table XI illustrates the format of the procedure used for analyzing the enriched foil activity. Table XII illustrates the format of the procedure used to analyze the depleted foil band-B activity, and Table XIII illustrates

TABLE XI - DATA REDUCTION FORM, U-235 FISSION DISTRIBUTION

(1)	(2)	(3)	(4)	(5)	(6)	(7)
Foil no.	Time counted, t_{25}	Gross counts, cpm	Back- ground, cpm	Net foil count, cpm (3)-(4)	Foil weight, mg U-235	Net specific 25-count, cpm/mg U-235 (5) $\div$ (6)
287	37:58.5	49.004	360	48,644	32.09	1,516
(8)		(9)		(10)		(11)
Net specific* 25-standard count at t_{25} , cpm/mg U-235		Normalized net specific 25-count, (7) $\div$ (8)		Net 25-count [†] corrected for U-self-shielding, cpm/mg U-235 $0.853 \times (7)$		Elevation
1262		1.201		1293		Core cen- tral plane

* Obtained from separate analysis procedure.

[†] Required for the analysis of the depleted foil band-B activity.

TABLE XII - DATA REDUCTION FORM, U-238 FISSION DISTRIBUTION

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)
Depleted foil no.	Enriched foil no.	Time counted, t ₂₈	Depleted gross counts, cpm	Foil weight. g	Specific gross counts, cpm/g U (4) ÷ (5)	Specific background, cpm/g U	Net specific foil count, cpm/g U (6)-(7)
874	287	38:04.5	81,820	1.5940	51,330	1960	49,370
(9)	(10)	(11)	(12)	(13)	(14)		
25-count per mg, [†] cpm/mg U-235	U-235 count per g U. cpm/g U 2.2 x (9)	Net specific U-238 count, cpm/g U (8)-(10)	Net specific U-238 standard count at t ₂₈ ,* cpm/g U	Normalized net specific U-238 count, (11) ÷ (12)	Elevation		
1293	2845	46,525	35,200	1.322	Core cen- tral plane		

* Obtained from a separate analysis procedure.

† From col. (10), Table XI.

TABLE XIII - DATA REDUCTION FORM, U-238 CAPTURE DISTRIBUTION

(1)	(2)	(3)	(4)	(5)	(6)	(7)
Depleted foil no.	Time counted, A t_{28}	Gross counts, cpm	Foil weight, g	Specific gross counts, cpm/g $(3) \div (4)$	Specific background, cpm/g U	Net specific foil count, cpm/g U $(5)-(6)$
874	38:09.0	498,704	1.5940	312,863	6739	306,124
(8)	(9)	(10)	(11)	(12)	(13)	
U-band-B counts, $\ddagger$ cpm/g U	$(8) \times 0.636^*$ cpm/g U	Net specific Np count, cpm/g U $(7)-(9)$	Net specific Np std. count at $t_{28}^{\dagger}$, cpm/g U	Normalized net specific Np count, $(10) \div (11)$	Elevation	
49,370	31,399	274,725	255,700	1.074	Core cen- tral plane	

* 0.636 is the Band A-to-Band B ratio.

† Obtained from a separate analysis procedure.

‡ From col. (8), Table XII.

the format of the procedure used to analyze the depleted foil band-A activity counts. All of the values given in the illustration are for the foils irradiated at the core central plane in core position P00-P00.

B. CHEMICAL ANALYSIS

Eight enriched and eight depleted foils were radiochemically analyzed at Argonne National Laboratory after being irradiated, to determine the absolute number of fissions which had occurred. The depleted foils were also analyzed to determine the number of neutron captures which had occurred during their irradiation. The pertinent data for the foils which were chemically analyzed are listed in Table XIV.

TABLE XIV - RADIOCHEMICAL ANALYSIS OF SELECTED FOILS

Irradiation no.	Foil exposure, kwhr ^e	Core location	Foil no.	Foil weight, mg U-235	Foil weight, g U	Foil elevation, ^f	Normalized fission count ^c		Normalized capture count ^d	Radiochemical Analysis	
							U-235	U-238		Total no. of fissions in foil	No. of U-238 captures in foil
6	1.60	P00-P00	875 ^a		1.589	+ 1		1.306	1.075	4.13 x 10 ¹⁰	9.71 x 10 ¹⁰
			885 ^a		1.597	+ 1		1.299	1.066	3.56 x 10 ¹⁰	9.93 x 10 ¹⁰
			286 ^b	32.92		- 1	1.162			1.77 x 10 ¹⁰	
			297 ^b	32.27		- 1	1.164			1.73 x 10 ¹⁰	
7	1.77	P00-P03	851 ^a		1.613	+ 1		1.222	0.950	2.83 x 10 ¹⁰	7.24 x 10 ¹⁰
			861 ^a		1.584	+ 1		1.243	0.968	2.90 x 10 ¹⁰	7.06 x 10 ¹⁰
			262 ^b	32.82		- 1	1.045			1.36 x 10 ¹⁰	
			273 ^b	32.74		- 1	1.054			1.41 x 10 ¹⁰	
12	46.3	P00-P10	761 ^a		1.581	+ 1		0.0101	0.136	1.26 x 10 ¹⁰	3.64 x 10 ¹¹
			771 ^a		1.576	+ 1		0.0099	0.132	1.30 x 10 ¹⁰	3.41 x 10 ¹¹
			172 ^b	33.22		- 1	0.111			5.42 x 10 ¹⁰	
			183 ^b	32.29		- 1	0.110			5.17 x 10 ¹⁰	
13	353	N11-P09	1011 ^a		1.579	+15		0.00446	0.0153	3.09 x 10 ¹⁰	2.93 x 10 ¹¹
			423 ^b	31.92		+15	0.0129			4.45 x 10 ¹⁰	
		N10-P10	1024 ^a		1.589	+ 5		0.00240	0.0220	7.88 x 10 ⁹	2.62 x 10 ¹¹
			444 ^b	31.82		+20	0.0134			4.63 x 10 ¹⁰	

a Foil nos. 100-699 are enriched to 93 w/o U-235.

b Foil nos. 700-1299 are depleted to 0.22 w/o U-235.

c Values from Tables XV and XVI.

d Values from Table XVII.

e Exposure values utilize a conversion factor of 6.15×10^{-12} amps/watt for the Keithley micro-microammeter.

f Inches above core midplane.

V. ANALYSIS OF EXPERIMENTAL ERRORS

A. INTRODUCTION

The errors associated with the normalized values which comprise the U-235 fission, U-238 fission, and the U-238 capture distributions result from the combination of many component errors. No attempt has been made to determine the magnitude of these individual components and combine them to obtain values for the overall error, except for a few points in the U-238 fission distribution. This was not attempted because of the large number of components and the complex nature of their interrelationship.

However, estimates of the overall errors were obtained by analyzing the distributions of the individual observations of $R(t)$ about the three $R(t)$ curves obtained to intercalibrate the integrated-count method of power normalization to the standard foil method of power normalization (see Section III-I-3). The individual observations of the $R(t)$ values were obtained by counting each of three foils 12 times and, in addition, by counting nine foils once each to obtain the data used to specify each of the $R(t)$ curves.

The observations obtained to define each $R(t)$ curve were sufficiently numerous and were obtained in a manner which justified the assumption that they were a representative sample of all data. Hence, an error analysis generally valid for all points not too distant from core midplane could be calculated using only the results of the observation made to define the $R(t)$ curves. Errors significantly greater than the representative values described here can occur for the U-238 fission distribution when the foils are irradiated in a soft spectrum such as occurs far above the core midplane elevation. This special case is discussed in Section V-F.

A description of the individual error components and the analyses for determining the average error associated with each of the distributions are given in the following sections.

B. DESCRIPTION OF COMPONENT ERRORS

1. Gross-Count Error

The gross-count error arises from the random nature of the radioactive decay process. The magnitude of this error is inversely related to the activity of the foil being counted.

2. Mass Error

The mass error arises from the uncertainty of the mass of the principal isotope contained in each foil. The uncertainty of the mass is limited by the accuracy of the balance on which the depleted foils were weighed and the accuracy with which their depletion is known. The uncertainty of the mass of the U-235 content of the enriched uranium foils is limited by the accuracy of the method used by the manufacturer to measure the U-235 content of the individual foils.

3. Background Error

The background error also results from the random nature of the radioactive decay process. In this case, the background count rate was the sum of the natural room background and the radioactive decay of the uranium and uranium daughters present in the foils (see Section IV-A-1). The sum of the room and foil background observed for the depleted foils was approximately 3200 counts per minute in band B and 11,000 counts per minute in band A. The background count observed for the enriched foils was approximately 400 counts per minute. The magnitude of the background error was determined in the same manner as was the gross-count error.

4. Standard Foil Position Normalization Error

As the reactor power was increased to suitably activate foils located at increasing distances from the core center, it was necessary to increase the elevation above the core center line of the standard foil in the temporary instrument thimble, to keep its activity low enough to be usable (see Section IV-I-1). During the analysis of the experimental results, the standard foil activities were normalized to obtain the activities at the core center line. This normalization, where used, introduced an additional statistical error into the results.

5. Error Due to Depleted Foil U-235 Activity

To obtain the U-238 fission distribution, it was necessary to adjust the observed activity to compensate for that portion which originated from the fission products of the U-235 impurity in the depleted foil. The magnitude of the error introduced by the use of this adjustment is based on the errors associated with the three quantities used to make the adjustment: (1) the U-235 fission rate at the position the depleted foil was irradiated, (2) the factor used to adjust the U-235 fission rate for the difference in the self-shielding between the enriched and depleted foils (see Section III-K), and (3) the error associated with the mass of the U-235 in the depleted foil (see Section IV-A-4).

6. Band-A Fission Product Component Error

To determine the portion of the count rate in band A of the depleted foil which was proportional to the neptunium decay activity it was necessary to subtract the component of the total observed count which originated from fission products (see Section IV-A-5). The error associated with this adjustment resulted from the errors associated with the two quantities used to determine the magnitude of the adjustment: (1) the depleted foil Band-B count rate and, (2) the error in the A/B ratio (see Section III-J). Because the depleted foil band-A activity was always much larger than the product of the depleted foil band-B activity and the A/B ratio, the error in the U-238 capture distribution is rather insensitive to the error in the A/B ratio.

C. MAGNITUDE OF U-235 FISSION DISTRIBUTION ERROR

The error associated with a mass- and power-normalized value for the relative U-235 fission rate was estimated, as indicated above, by analyzing the data obtained to determine the $R(t)_B^{25}$ curve (Section III-I-3). The curve (Fig. 25) is an estimated least-mean-square deviation curve about a dispersion of empirically determined points. The error associated with each of the points results from the combination of the following component errors:

(1) Specimen Foil Errors - Irradiation No. 11

- a. Gross-count error
- b. Background-count error
- c. U-235-mass error

(2) Errors in Normalized Values - Irradiation No. 10

- a. Specimen foil errors
  1. Gross-count error
  2. Background-count error
  3. U-235-mass error
- b. Standard foil errors
  1. Gross-count error
  2. Background-count error
  3. Standard foil position normalization error
  4. U-235-mass error

To determine the magnitude of the combined errors the standard deviation of the points about the $R(t)_B^{25}$ curve was calculated using a least-square method. The standard deviation, σ , of the points about the curve

was calculated to be 1.40 per cent.* It was next assumed that the three major error components, represented in items (1), (2)-a, and (2)-b above, were equal, so that (the only difference in principle between the three major error components is (2)-b, which can be demonstrated to be small):

$$\sigma^2 [\text{for (1)}] = \sigma^2 [\text{for (2)-a}] = \sigma^2 [\text{for (2)-b}] = \sigma_T^2/3 \quad (11)$$

where σ_T was calculated to be 1.4 per cent. Based on this assumption, each of the three major components of the total error was calculated to be 0.81 per cent. Because a typical normalized fission rate is the quotient of items (2)-a and (2)-b, the standard deviation of the error associated with these values was assumed to be twice** 0.81 or 1.6 per cent.

This computation results in a 95-per cent confidence limit of ± 3.2 per cent for the relative U-235 fission rate when normalized by the standard foil method (irradiations 3 - 10).

To determine the same quantity for values normalized using an integrated-count method (irradiations 11 - 19), the error associated with the integrated-count normalization factor must replace the error associated with the standard foil. The integrated-count normalization factor is obtained by multiplying the integrated count obtained during an irradiation by the $R(t)_B^{25}$ value for the time at which the foil activity to be normalized was obtained. Hence, the error associated with the integrated-count normalization factor results from the combination of the error associated with the $R(t)_B^{25}$ curve and the adjusted channel-3 integrated count obtained during the irradiation being analyzed. The former quantity was obtained by dividing the standard deviation of the distribution of the points about the $R(t)_B^{25}$ curve (Fig. 26) by the square root of the number of points used to determine the $R(t)_B^{25}$ curve. The standard deviation of the points about the curve was given before as 1.4 per cent. The number of points used to determine the curve was 44. Hence, the standard deviation which represents the error associated with any value of $R(t)_B^{25}$ was 0.21 per cent ($1.4\%/\sqrt{44}$).

The standard deviation of the error associated with the adjusted channel-3 integrated count was estimated by calculating the standard deviation of the ratio of the Keithley current to the channel-3 count listed in Table VI. The value obtained was 2.5 per cent. The combination of the 2.5 per cent and the 0.21 per cent by addition resulted in a value of 2.7 per cent for the standard deviation representing the error associated with the integrated-count normalization.

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* Standard deviation throughout this report is expressed as a percentage of the arithmetic average of the individual values.

** Due to the nature of the assumptions used in this analysis it appeared best to use the conservative approach of combining the errors by addition.

The 95-per cent confidence limit for the relative U-235 fission rate distribution, when normalized by the integrated-count method (irradiations 11 - 19), was obtained by adding the standard deviations computed for item (1) above (0.81 per cent) to the standard deviation computed for the error in the integrated-count normalization factor (2.7 per cent). This combination resulted in an estimated 95-per cent confidence limit of ± 7.0 per cent.

D. MAGNITUDE OF U-238 CAPTURE DISTRIBUTION ERROR

The error associated with the mass- and power-normalized values for the relative neutron capture rate in U-238 was estimated by analyzing the data obtained to determine the $R(t)_A^{28}$ curve (Section III-I-3). This least-mean-square deviation curve, together with the dispersion of the empirically determined points, is given in Fig. 29. The error components present in each of the points are:

(1) Specimen Foil Errors - Irradiation No. 11

- a. Gross-count error
- b. Mass error
- c. Background-count error
- d. Band-A fission product component error

(2) Error in Normalized Values - Irradiation No. 10

- a. Specimen foil errors
  1. Gross-count errors
  2. Mass error
  3. Background-count error
  4. Band-A fission product component error
- b. Standard foil errors
  1. Gross-count error
  2. Mass error
  3. Background-count error
  4. Band-A fission product component error
  5. Standard foil position normalization error

As in Section C, it was assumed that the magnitudes of the errors represented in (1), (2)-a, and (2)-b above were approximately equal so that again,

$$\sigma^2[\text{for (1)}] = \sigma^2[\text{for (2)-a}] = \sigma^2[\text{for (2)-b}] = \sigma_T^2/3 \quad (11)$$

where σ_T is assumed to be the standard deviation of the points about the best-fit curve in Fig. 27. This standard deviation was calculated to be 2.5 per cent. This results in a value for the standard deviation of the error associated with items (1), (2)-a, and (2)-b of 1.4 per cent.

Because the normalized capture rate is the quotient of items (2)-a and (2)-b the standard deviation of the normalized value results from the combination of the standard deviation of items (2)-a and (2)-b. This combination, for the reason given in Section V-C, was accomplished by adding the two sigmas. The result was an estimated 95-per cent confidence limit of ± 5.6 per cent for the values comprising the U-238 neutron capture distribution when normalized by the standard foil method.

The error associated with the integrated-count normalization factor is obtained in the same manner as in Section V-C with the exception that the $R(t)_A^{28}$ curve is based on 47 points. Hence, the standard deviation of the error associated with any value of $R(t)_A^{28}$ is 0.32 per cent ($2.5\%/\sqrt{47}$). When this is combined by addition with the value obtained in Section IV-C for the error in the adjusted channel-3 integrated count, a value of 2.8 per cent is obtained for the standard deviation of the error of the integrated-count normalization factor.

The 95-per cent confidence limit for the relative U-238 capture rate, when normalized by the integrated-count method, was obtained by adding the standard deviations computed for item (1) above (1.4 per cent) and the value computed for the standard deviation of the integrated-count normalization factor (2.8 per cent). This resulted in a 95-per cent confidence limit of ± 8.4 per cent.

E. MAGNITUDE OF U-238 FISSION DISTRIBUTION ERROR

The error associated with each of the mass- and power-normalized values of the U-238 fission distribution results from the combination of the following component errors:

(1) Specimen Foil Error - Irradiation No. 11

- a. Gross-count error
- b. Background-count error
- c. Mass error
- d. Depleted foil U-235-activity error

(2) Errors in Normalized Values - Irradiation No. 10

- a. Specimen foil errors
  - 1. Gross-count error

2. Background-count error
3. Mass error
4. Depleted foil U-235-activity error

b. Standard foil errors

1. Gross-count error
2. Background-count error
3. Mass error
4. Standard foil position normalization error
5. Depleted foil U-235-activity error

As has been done for the analyses of the error associated with the other two distributions, the magnitude of the error resulting from the combination of the components given in items (1), (2)-a, and (2)-b above was estimated by calculating the standard deviation of the points about the $R(t)_B^{28}$ curve established to analyze the U-238 fission data. These data and their dispersion about the best-fit $R(t)_B^{28}$ curve are illustrated in Fig. 26. The standard deviation of the points about the curve was calculated to be 4.2 per cent. Again, assuming that the standard deviations representing the errors associated with items (1), (2)-a, and (2)-b are equal, a value of 2.4 per cent is obtained for each standard deviation. The value of the standard deviation of the quotient of items (2)-a and (2)-b is then 4.8 per cent. Hence, the 95-per cent confidence limit is ± 9.6 per cent for the U-238 fission rate distribution when normalized by the standard foil method.

The comparable quantity for a value normalized by the integrated-count method is obtained by combining the standard deviation of the error associated with item (1) above, shown to be 2.4, with the standard deviation of the error associated with the integrated-count normalization factor. The latter standard deviation was obtained by combining the standard deviation of the error associated with a $R(t)_B^{28}$ value, which is equal to 0.6 per cent ($4.2\%/\sqrt{44}$), with the standard deviation of the error associated with the adjusted channel-3 integrated-count calculated in Section V-C to be 2.5 per cent. When these two values are combined by addition, a value of 3.1 per cent is obtained for the standard deviation of the error associated with the integrated-count normalization factor. Hence, the 95-per cent confidence limit is ± 10.5 per cent for the U-238 fission distribution when normalized by the integrated-count method.

The error estimated for the U-238 fission distribution is strongly dependent on the relative magnitude of the U-235 fission product activity in the depleted foils. The error computed above is typical for foils irradiated near the midplane elevation. The next section discusses specific cases where this error is significantly greater due to the large fraction of the foil activity from the U-235 component.

F. DETAILED U-238 FISSION ERROR ANALYSIS FOR SPECIFIC FOILS

In regions where the neutron spectrum is relatively soft, the fraction of gamma activity from an irradiated depleted foil due to U-235 fission products can be quite large. Since the desired U-238 fission rate is essentially obtained by subtracting the U-235 fission product activity from the total foil activity, nominal-size errors in either of these terms can be reflected as relatively large uncertainties in the implied U-238 fission rate. For this reason, a rather detailed error analysis was performed for the measured U-238 fission rate near the tops of the subassemblies for the lattice positions included in the plots of Figs. 38 and 39.

Uncertainties associated with essentially all of the components affecting the measured U-238 axial fission distribution for a given subassembly were examined.* A number of these could be dismissed as having a negligible effort on the final result; the remainder were assigned one standard deviation (one σ) uncertainty values and these were combined in a manner consistent with the formulation of the U-238 fission rate from the various measured components and assuming all errors to be random and independent. The components to which uncertainties were assigned and the magnitudes of the uncertainties are listed below.

<u>Component</u>	<u>1 σ Uncertainty (%)</u>
Depleted foil count, (C)	$\pm (1/\sqrt{C}) \times 100$
Background for depleted foil count, (B)	$\pm (1/\sqrt{B}) \times 100$
Associated U-235 foil count, (C ₂₅)	$\pm (1/\sqrt{C_{25}}) \times 100$
U-235 content of depleted foil	± 1.0
U-235 content of associated enriched foil	± 0.5
Difference in gamma self-shielding for U-235 in depleted uranium and enriched alloy foil	± 1.0

The resulting total uncertainty is computed for a 95-per cent confidence (2σ) and is expressed as an error band about the appropriate plotted point in Figs. 38 and 39. The largest 2σ uncertainty is ± 47 per cent for subassembly P08-P00 at an elevation of 81.3 cm.

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* Uncertainties in power normalization among different runs do not apply since only the axial distribution in a given subassembly is of interest here.

The greatest weakness of this approach lies in the difficulty of assigning an appropriate uncertainty to the U-235 content of the depleted foil. The one σ value of 1.0 per cent lies within the accuracy of a careful enrichment analysis and would also encompass statistical variation among pins. However, a recent investigation of the actual source of the depleted material indicates that an enrichment assay of the specific depleted material used for the foils was not made. Thus, the deviation of the actual average enrichment of the depleted foils from the nominal 0.22 w/o listed by the supplier could be considerably in excess of the 1.0 per cent used in the error analysis.

VI. MEASURED REACTION-RATE DISTRIBUTION DATA

The resulting U-235 fission, U-238 fission, and U-238 capture-rate distributions as determined by foil activation are given in Tables XV, XVI and XVII, respectively. Each of the three types of reactions is independently normalized to a reference position in the temporary instrument thimble. Thus, the values shown in these Tables cannot be used to establish absolute reaction-rate ratios. Such measured ratios must be obtained from the values given by radiochemical analysis in Table XI or from fission ratio results obtained from absolute fission counter measurements in the temporary instrument thimble.⁶ Graphical representations of a portion of these measured reaction-rate distributions are included in Section VII, together with calculated distributions and detailed comparisons between measurement and calculation.

TABLE XV - U-235 FISSION DISTRIBUTION NORMALIZED

		ELEVATION - IN. (cm)																							
Type of Sub-assembly*	Position	Radius (in)	12.5 (-01.75)	-10.0 (-25.40)	-7.5 (-19.05)	-5.0 (-12.70)	-2.5 (-6.35)	Core Center	-1.0 ^f (-2.54)	2.5 (6.35)	5.0 (12.70)	7.5 (19.05)	10.0 (25.40)	12.5 (31.75)	15.0 (38.10)	16.50 (41.91)	16.75 (47.62)	20.0 (50.80)	22.5 (57.15)	25.0 (63.50)	27.5 (69.85)	30.0 (76.20)	32.0 (81.28)		
CF	P00-P00	0		0.933				1.201	1.162	1.164			1.104		0.883		0.622	0.566	0.510	0.455		0.283		0.163	0.137
CF	P00-P01	6.24		0.879				1.187					1.070		0.853		0.600	0.561	0.494	0.446		0.266		0.158	0.127
CF	N01-P01	3.67		0.897				1.130					1.026		0.801		0.552	0.511	0.448	0.398		0.240		0.142	0.117
CF	P02-P00	13.66		0.937				1.086					1.011		0.801		0.573	0.541	0.472	0.408		0.249		0.147	0.127
CF	P00-P03	20.52		0.827				1.083	1.045	1.054			0.970		0.802		0.547	0.507	0.466	0.418		0.236		0.141	0.118
CF	P04-P00	27.36		0.696				0.876					0.833		0.684		0.471	0.440	0.398	0.349		0.202		0.123	0.117
CF	N03-P03	29.00		0.734				0.894					0.810		0.640		0.442	0.421	0.356	0.335		0.184		0.112	0.104
CF	P00-P05	34.20		0.588				0.774					0.706		0.567		0.403	0.391	0.339	0.290		0.168		0.116	0.110
CF	P05-P00	34.20		0.571				0.753					0.530		0.376		0.357	0.318	0.277			0.160		0.104	0.102
CF ^b	N04-P04	38.72		0.484				0.619					0.569		0.461		0.339	0.316	0.230	0.246		0.144		0.102	0.095
CF ^a	N01-P04	38.72						0.672					0.510							0.264					
CF	N04-N04	38.72		0.553				0.669					0.62*		0.491		0.342	0.324	0.279	0.244		0.147		0.104	0.103
CF	N05-N03	39.95		0.536				0.667					0.510		0.344		0.337	0.299	0.256		0.155		0.106	0.103	
CF	P06-P00	41.04		0.447				0.547					0.513		0.407		0.300	0.285	0.232	0.209		0.131		0.091	0.092
BF	P00-P07	47.58		0.302				0.366					0.307		0.227		0.202	0.171	0.154		0.106		0.118	0.119	
CF	N01-N05	48.36		0.376				0.460					0.433		0.343		0.236	0.226	0.189	0.170		0.110		0.097	0.099
BF	N06-P04	49.32		0.326				0.422					0.400		0.333		0.248	0.231	0.159	0.176		0.112		0.104	0.102
BF	N06-N05	53.42		0.264				0.339					0.320		0.255		0.187	0.158	0.135	0.120		0.090		0.104	0.106
BF	N06-P05	53.42		0.242				0.309					0.289		0.236		0.176	0.159	0.133	0.119		0.088		0.100	0.102
BF ^c	N06-P05	53.42		0.246				0.306					0.292		0.237		0.174	0.160	0.133	0.121		0.085		0.100	0.100
BF	P08-P00	54.72		0.199				0.257					0.247		0.209		0.155	0.138	0.119	0.108		0.076		0.098	0.102
BF	N06-P06	58.14		0.182				0.239					0.22		0.188		0.144	0.131	0.108	0.097		0.074		0.095	0.096
BF	P00-P09	61.56		0.136				0.173					0.168		0.141		0.101	0.094	0.078	0.071		0.060		0.095	0.098
BF	P00-P10	68.40		0.089				0.113	0.110	0.111			0.107		0.086		0.066	0.059	0.051	0.048		0.050		0.083	0.087
BF	P00-N10	68.40		0.086				0.120					0.118		0.098		0.078	0.069	0.060	0.055		0.051		0.080	0.086
BF	N08-P06	77.29		0.051				0.063					0.063		0.053		0.042	0.038	0.033	0.031		0.033		0.064	0.063
BF	P00-P12	82.08		0.033				0.041					0.039		0.034		0.026	0.025	0.022	0.021		0.027		0.057	0.061
BF	P12-P00	82.08		0.034				0.041					0.039		0.033		0.026	0.025	0.022	0.021		0.025		0.055	0.059
BF	P14-P00	95.76		0.018				0.027					0.026		0.023		0.020			0.014		0.016			0.039
BF ^d	P14-P00	95.76						0.020					0.017							0.015		0.020			
BF	P00-P14	95.76	0.018	0.020	0.022	0.024	0.025	0.025	0.0251	0.0250	0.0250	0.024	0.0230	0.022	0.0205	0.016	0.018	0.016	0.016	0.016	0.0172	0.021	0.0320	0.040	0.043
CF	P14-P01	96.04						0.019					0.016							0.018		0.027			
BF ^e	N10-P10	96.72		0.017				0.021					0.020		0.017		0.015	0.014	0.0135	0.013		0.017		0.037	0.042
CF	N11-P09	97.13		0.014				0.016					0.017		0.015		0.013	0.015	0.015	0.015		0.018		0.031	0.035
SB ^d	P16-P00	109.40						0.016					0.025							0.029		0.027			
SB	P16-P00	109.40		0.020				0.033					0.035		0.033		0.030			0.026		0.023		0.025	
SB	P00-N16	109.40		0.018				0.029					0.031		0.029		0.026			0.023		0.021			0.024
CF	T.R.			0.0000110				0.0000245					0.0000327		0.0000546		0.0000874			0.000286				0.000436	

* CF is core foil subassembly, BF is blanket foil subassembly, and SB is shield bar.

a During this irradiation there was a blanket subassembly on only one side of N04-P04.

b During this irradiation there was a blanket subassembly on two sides of N04-P04.

c An irradiation was repeated at N06-P05 to intercalibrate power normalization using a standard foil and using the integrated count on several nuclear counting channels.

d This irradiation was made with core subassemblies stored at P14-P01 and P14-N01.

e This irradiation was made with core subassemblies stored at N11-P09 and N09-P11.

f Two foils located at this elevation.

TABLE XVI - U-236 FISSION DISTRIBUTION $\times 10^2$ NORMALIZED

ELEVATION - IN (cm)

Type of Subassembly	Radius (cm)	-12.5 (-31.75)	-10.0 (-25.40)	-7.5 (-19.65)	-5.0 (-12.70)	-2.5 (-6.35)	0 (0)	2.5 (6.35)	5.0 (12.70)	7.5 (19.65)	10.0 (25.40)	12.5 (31.75)	15.0 (38.10)	16.5 (41.91)	18.75 (47.62)	20.0 (50.80)	22.5 (57.15)	25.0 (63.50)	27.5 (69.85)	30.0 (76.20)	32.0 (81.28)
CF	P00-P00	0	101.3					130.6	122.0	97.0	48.0			30.7	20.1	15.1		4.5		1.4	1.1
CF	P00-P01	6.4	105.7					136.6	124.7	99.0	47.6			31.7	20.5	14.9		3.83		1.22	0.456
CF	N01-P01	9.67	100.5					133.2	123.1	96.9	47.0			30.7	20.0	14.6		4.33		1.42	0.89
CF	P02-P00	13.68	94.8					123.6	113.9	91.5	45.3			28.2	19.2	14.6		4.4		1.6	0.907
CF	P00-P03	20.32	93.5					123.2	114.3	90.4	41.3			28.8	18.4	13.2		3.7		1.2	0.521
CF	P04-P00	27.86	78.9					103.7	95.4	75.6	36.8			23.7	14.8	10.7		2.96		1.02	0.52
CF	P03-P03	29.00	79.6					103.5	97.3	77.2	37.0			23.7	15.4	11.1		3.3		1.1	0.682
CF	P00-P05	34.20	55.5					75.0	72.9	55.9	33.4			20.2	12.7	3.3		2.76		0.857	0.483
CF ^b	P05-P00	34.20	66.4					82.4	75.2	57.9	38.4			15.9	9.7	6.9		2.1		0.559	0.222
CF ^b	N04-P04	38.72	49.9					65.5	60.3	47.8	23.5			15.1	9.6	7.0		1.9		0.531	0.170
CF ^a	N04-P04	38.72						68.2		50.2						6.7					
CF	N04-N04	38.72	52.0					69.1	63.2	49.8	24.5			15.3	10.0	7.2		2.1		0.571	0.458
CF	N05-N03	39.95	49.9					64.6	60.2	47.5	23.0			14.4	9.2	6.9		1.7		0.500	0.433
CF	P06-P00	41.04	41.6					53.0	48.1	37.5	14.9			10.5	6.69	4.6		1.25		0.390	0.081
BI	P00-P07	47.88	54.1					11.9	11.1	8.6	5.4			4.4	3.0	2.5		0.925		0.463	0.327
CF	N05-N05	48.86	25.1					32.6	31.0	24.5	12.4			7.3	4.7	3.5		1.2		0.365	0.258
BF	P04-P04	49.32	17.47					21.2	20.1	16.2	9.2			6.8	4.6	3.6		1.10		0.442	0.282
BF	N05-N05	53.42	3.4					7.1	6.5	5.2	3.0			2.5	1.3	1.5		0.588		0.204	0.163
BF ^c	N06-P05	53.42	7.1					9.4	9.0	7.0	4.1			3.25	2.34	1.85		0.70		0.24	0.15
BF	N06-P05	53.42	9.8					9.8	3.1	7.2	4.0			3.24	2.32	1.82		0.615		0.227	0.158
BF	N06-P06	58.14	3.96					5.5	5.2	4.1	2.6			2.03	1.54	1.22		0.496		0.194	0.150
BF	N06-P06	58.14	5.3					5.3	4.9	2.8	2.4			1.85	1.49	1.22		0.510		0.340	0.343
BI	P00-P03	61.46	1.52					2.1	1.9	1.5	1.0			0.86	0.69	0.56		0.232		0.199	0.177
BF	P00-P10	64.40	0.74					0.98	0.9	0.725	0.494			0.404	0.346	0.285		0.16		0.15	0.156
BF	P00-N10	65.40	0.99					1.38	1.3	1.06	0.88			0.61	0.46	0.42		0.23		0.17	0.14
BF	N08-P08	77.29	0.352					0.474	0.437	0.356	0.251			0.214	0.180	0.152		0.103		0.080	0.096
BF	P00-P12	82.08	0.169					0.236	0.240	0.197	0.143			0.119	0.107	0.094		0.065		0.102	0.113
BI ^d	P12-P00	92.08	0.201					0.351	0.262	0.226	0.151			0.129	0.102	0.107		0.073		0.100	0.128
BI ^d	P14-P00	95.76						0.391	0.327	0.327	0.151			0.129	0.102	0.107		0.045		0.069	0.204
BF	P14-P00	95.76	0.052					0.095	0.089	0.080	0.057			0.052	0.045	0.046		0.041		0.187	0.221
BF	P00-P14	95.76	0.067					0.096	0.090	0.082	0.071			0.054	0.045	0.046		0.041		0.187	0.221
CF	P14-P01	96.04						0.846	0.703	0.703	0.064			0.098	0.070	0.076		0.050		0.150	0.188
BF ^e	N10-P10	96.72	0.191					0.248	0.240	0.206	0.130			0.098	0.070	0.054		0.089		0.050	0.188
CF	N11-P04	97.13	0.616					0.779	0.743	0.648	0.446			0.183	0.085	0.064		0.041		0.052	0.053
SP ^d	P16-P00	100.40	0.184					0.184	0.141	0.141	0.086			0.086	0.085	0.026		0.027		0.008	0.019
SB	P16-P00	108.40	0.026					0.082	0.085	0.086	0.044			0.044	0.026	0.026		0.027		0.008	0.019
SB	P00-N16	109.40	0.020					0.053	0.064	0.066	0.064			0.064	0.042	0.042		0.023		0.008	0.019
CF	T.R.		0.00154					0.00154	0.00167	0.000859	0.00146			0.00146	0.000675	0.000675		0.023		0.008	0.019

* CF is core fuel subassembly, BI is blanket fuel subassembly, and SB is shield blanket.

† During this irradiation the reactivity was a blanket subassembly on only one side of N04-P04.

‡ During this irradiation there was a blanket subassembly on two sides of N04-P04.

§ An irradiation was reported at N06-P05 to indicate the power normalization using a standard source and using the uncalibrated count on several nuclear counting channels.

|| This irradiation was made with one subassembly set to 10 at P14-P01 and P14-N01.

¶ This irradiation was made with one subassembly set to 10 at P14-P01 and N09-P11.

|| This irradiation was made with one subassembly set to 10 at P14-P01 and P14-N01.

TABLE XVII - U-238 CAPTURE DISTRIBUTION X 10², NORMALIZED

Name of Subassembly*	Position	Radius, cm.	Elevation - IN. (cm)	Core Center	1.0 ^f (2.54)	2.5 (6.35)	5.0 (12.70)	7.5 (19.05)	10.0 (25.40)	12.5 (31.75)	15.0 (38.10)	16.5 (41.91)	18.75 (47.62)	20.0 (50.80)	22.5 (57.15)	25.0 (63.50)	27.5 (69.85)	30.0 (76.20)	32.0 (81.29)
CF	P00-P00	0		107.4	107.5	106.6	100.9		85.3		68.2	69.9	64.1	54.9		34.6		20.5	16.9
CF	P00-P01	6.84		105.4			98.3		82.1		65.2	67.6	62.5	53.6		32.0		21.5	15.7
CF	N01-P01	9.67		99.4			94.0		78.7		62.9	65.2	61.1	52.1		32.8		17.6	15.8
CF	P02-P00	13.68		103.0			95.9		81.2		66.1	69.3	63.9	54.2		32.9		19.5	16.4
CF	P00-P03	20.52		95.9		95.0	90.2		75.3		60.3	62.7	57.1	49.2		29.4		16.9	14.1
CF	P04-P00	27.36		67.8			77.9		65.2		55.3	50.9	43.6	23.4		25.6		15.3	13.9
CF	N03-P03	29.00		66.6			63.5		55.2		52.7	54.6	50.6	42.6		25.2		15.1	12.7
CF	P00-P05	34.20		56.1			66.8		56.8		46.7	47.9	45.7	40.9		23.9		15.1	13.4
CF	P05-P00	34.20		58.4			64.5		53.6		47.8	47.7	40.8	35.1		21.1		13.6	13.6
CF ^b	N04-P04	38.72		63.9			60.1		50.2		42.0	43.6	39.6	33.6		20.0		13.4	12.6
CF ^a	N04-P04	38.72		62.9			50.7		56.5		44.3	45.8	41.7	35.6		21.1		13.9	12.9
CF	N05-N03	39.95		57.2			67.2		56.5		44.3	46.2	41.8	35.2		21.4		14.2	13.3
CF	P06-P00	41.04		48.7			55.6		55.6		44.3	46.2	41.8	35.2		21.4		14.2	13.3
BF	P00-P07	47.86		42.3			33.7		44.7		39.7	39.6	33.1	27.9		17.1		12.2	13.2
CF	N05-N05	48.36		33.4			39.9		34.1		26.2	23.8	20.4	18.4		11.9		17.2	17.4
BF	N06-P04	49.32		43.9			51.5		42.8		32.5	33.2	29.8	27.9		16.0		12.7	12.3
BF	N06-P05	53.42		32.0			41.2		34.9		28.0	26.5	23.6	21.5		13.5		14.7	16.2
BF ^c	N06-P05	53.42		30.4			38.2		31.4		22.8	20.6	17.4	15.8		11.1		17.9	18.2
BF	P08-P05	53.42		29.7			34.3		29.5		23.0	20.9	17.8	16.0		10.7		16.8	17.2
BF	N06-P06	54.72		24.6			36.6		30.4		22.4	20.8	17.7	16.0		10.9		17.0	17.3
BF	N06-P06	58.14		19.9			29.6		25.3		19.5	17.8	15.0	13.7		9.4		16.0	16.7
BF	P00-P09	61.56		16.3			23.5		20.4		15.8	14.4	12.4	11.3		7.9		13.8	14.0
BF	P00-P10	66.40		13.3			19.7		16.6		12.5	11.4	9.6	8.8		6.5		13.9	14.8
BF	N08-N10	68.40		10.9			12.3		10.5		8.1	7.3	6.2	5.8		4.8		12.3	13.0
BF	P00-P08	77.29		15.1			14.5		12.5		9.6	8.9	7.6	7.0		5.4		12.0	12.6
BF	P12-P00	82.08		6.4			7.5		6.5		5.0	4.6	3.6	3.7		3.3		9.4	10.2
BF	P12-P00	82.08		4.1			4.9		4.3		3.4	3.1	2.7	2.5		2.6		8.4	9.2
BF	P14-P00	95.76		1.9			2.7		2.4		2.0	2.0	2.6	2.6		2.5		8.1	8.9
BF ^d	P14-P00	95.76		2.2			2.2		1.9		2.0	1.7	1.5	1.5		1.9		7.1	7.1
BF	P00-P14	95.76	1.90	2.1	2.3	2.4	2.5	2.5	2.2	2.0	1.8	1.8	1.6	1.6	1.6	1.7	4.3	5.6	6.0
CF	P14-P01	96.04		2.2			2.2		1.9		1.6	1.5	1.4	1.3		2.0		5.0	5.6
BF ^e	N10-P10	96.72		2.3			2.2		1.9		1.6	1.5	1.4	1.3		1.5		2.6	2.9
CF	N11-P09	97.13		1.7			2.0		1.7		1.5	2.1	2.1	2.1		1.7		3.1	3.1
SB ^d	P13-P00	109.40		4.0			4.0		4.0		4.3	4.3	4.3	3.9		3.5		3.8	3.8
SB	P13-P00	109.40		4.9			4.5		4.8		4.3	4.3	4.3	3.5		3.2		3.5	3.5
SB	P00-N16	109.40		4.3			4.4		4.3		3.9	3.9	3.9	3.5		3.2		3.5	3.5
CF	T.R.			0.00317			0.00474		0.00590		0.0107			0.03				0.027	0.027

* CF is core foil subassembly, BF is blanket foil subassembly, and SB is shield bar.

a During this irradiation there was a blanket subassembly on only one side of N04-P04.

b During this irradiation there was a blanket subassembly on two sides of N04-P04.

c An irradiation was repeated at N06-P05 to intercalibrate power normalization using a standard foil and using the integrated count on several nuclear counting channels.

d This irradiation was made with core subassemblies stored at P14-P01 and P14-N01.

e This irradiation was made with core subassemblies stored at N11-P09 and N09-P11.

f Two foils located at this elevation.

VII. COMPARISON OF EXPERIMENTAL RESULTS WITH CALCULATIONS

A. INTRODUCTION

There has been a continuing effort to predict pertinent reaction-rate distributions in the Enrico Fermi fast breeder reactor and to correlate measured distributions with analysis. Essentially all computations have utilized multigroup diffusion theory.* Preliminary foil activation levels were predicted using 10-group, one-dimensional calculations with the code PROD to aid in setting reactor power levels for the experimental program.⁹ Subsequent to the foil irradiations, the bulk of the multigroup analyses has been performed with the CRAM code in both one and two dimensions. Computations were made that provided information on the effects of geometry, cross sections, group structure, mesh points, etc. on the computed U-235 fission, U-238 fission, and U-238 capture reaction rates.

Only the more significant aspects of these results are reported here, together with samples of the more meaningful calculations for comparison with measurements. A thorough, systematic analysis of reaction-rate measurements is beyond the scope of this project and would have been difficult to accomplish immediately following the measurements due to the rapid evolution of cross-sectional sets and basic data. Such a study can be done on a more consistent basis and with more meaningful results when a more sophisticated and standardized cross-sectional data system becomes available.

B. CALCULATED AND MEASURED DISTRIBUTIONS

Calculated and measured reaction-rate distributions are presented in three general formats: (1) radial distribution of midplane values in cylindrical geometry, (2) axial distributions for selected subassembly positions, and (3) XY geometry presentation for midplane values in specific core subassemblies.

1. Radial Distribution of Midplane Values in Cylindrical Geometry

The best agreement with a general radial distribution, as measured by foil activation, was obtained with a 24-group, one-dimensional

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* Brief studies comparing reaction rates computed by the diffusion theory code, AIM, and the transport theory code, NIOBE, have been performed in Fermi Core-B (cermet) geometry.^{7,8}

CRAM calculation for a 103 subassembly core using space-dependent axial buckling (this problem is designated 24 GP 1D CYL). Buckling values in the blanket were based on the axial shape of the flux as obtained with an 8-group, two-dimensional RZ problem. Axial bucklings are given in Table XVIII. The cross-sectional library used for both the flux and reaction-rate calculations is a 24-group interim set in use at APDA designated LIB SEVEN 24, employing a self-shielded option for U-235 and U-238 (see Section VI-C). In addition, infinite dilute cross sections for U-238 were used in computing the capture-rate distributions. The significance of the cross sections is discussed in Section VII-C. The resulting calculated radial reaction-rate distributions are shown in Fig. 31 together with the experimental points obtained from Tables XV, XVI and XVII.

TABLE XVIII - SPACE-DEPENDENT AXIAL BUCKLING
USED IN ONE-DIMENSIONAL PROBLEMS*

<u>Region</u>	<u>Buckling, cm⁻²</u>
Core	0.000893
Radial blanket:	
41.03 - 52.66 cm	0.0008
52.66 - 64.29 cm	0.00075
64.29 - 75.92 cm	0.0007
75.92 - 87.55 cm	0.00065
87.55 - 99.18 cm	0.0006
Small sodium gap	0.0006
Thermal shield bars	0.0004

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* These values were used in APDA CRAM problem 030466.

2. Axial Distributions for Selected Subassembly Positions

Axial distributions have been computed with four two-dimensional RZ CRAM calculations. An early 24-group interim cross-sectional set in use at APDA and designated LIB-APDA-24 was used for the flux calculations in all four problems (see Section VII-C). The CRAM problems are briefly described below, following their identifying initials.

FIG. 31 RADIAL REACTION-RATE DISTRIBUTIONS AT CORE MIDPLANE ELEVATION

8GP RZ: This is an 8-group problem with 99.5 clean core sub-assemblies designed primarily for accurate mockup of core and blankets at the time of initial criticality. Control- and safety-rod channels contain no boron and are shown as rings. Detailed use of this problem should be restricted to the core and core midplane elevation due to an error in the height of the fueled portion of the radial blanket and in the length of the end-gap between core and axial blanket fuel. Definition of the energy group structure is given in Table XIX and the problem geometry is shown in Fig. 32.

9 GP RZ: This is a 9-group problem with 99.5 clean core sub-assemblies designed primarily for computing temporary instrument thimble reaction ratios far up into the sodium pool.¹⁰ The safety-rod channels are homogenized with nearby core subassemblies and safety-rod boron is included in the full-out position. There is no thermal shield exterior to the radial blanket. Definition of the energy group structure is given in Table XIX while the calculational geometry is shown in Fig. 32.

9 GP RZ THRESHOLD: This is a 9-group problem with 99.5 clean core subassemblies designed primarily for computing threshold-type reaction rates in the temporary instrument thimble.¹¹ The problem is similar to 9 GP RZ but does not extend so far into the sodium pool since meaningful threshold reaction-rate measurements cannot be made far from the core. Also, the energy group structure is chosen to give reasonably good resolution at high energies, while relatively course resolution is used at lower energies. Definition of the energy group structure is given in Table XIX while the calculational geometry is shown in Fig. 33.

20 GP RZ: This is a 20-group problem with 99.5 clean core sub-assemblies designed primarily for improving the calculated axial reaction-rate distributions in radial blanket subassemblies.¹¹ Control rods were homogenized with core material. Definition of the energy group structure is given in Table XIX while the problem geometry is shown in Fig. 33.

Due to the large number of subassembly positions chosen for foil irradiations, only representative samplings were chosen for comparing measured with calculated axial distributions of reaction rates. Further, since there are numerous calculations employing various geometric models, neutron energy structure, and reaction-rate cross sections, only a limited number were included in the comparison study. These were carefully chosen to afford a demonstration of the sensitivity of the calculation to a given parameter or assumption* (see Section VII-D-2). Graphs of these selected axial reaction-rate distributions are given in Figs. 34 through 37.

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* The 8 GP RZ problem is not used for the axial distribution comparison due to the geometric errors mentioned above; however, it is this problem from which the midplane axial buckling was obtained for use in the one-dimension, 24-group problems discussed in Section VI-B-1.

TABLE XIX - ENERGY GROUP STRUCTURES FOR
TWO-DIMENSIONAL CRAM PROBLEMS

<u>8 GP RZ</u>			<u>9 GP RZ</u>		
Group no.	Equivalent LIB-APDA-24 groups	Lower energy boundary	Group no.	Equivalent LIB-APDA-24 groups	Lower energy boundary
1	1 - 3	1.35 Mev	1	1 - 2	2.225 Mev
2	4 - 6	0.30	2	3 - 4	0.825
3	7 - 8	0.11	3	5 - 7	0.18
4	9 - 10	40.7 kev	4	8 - 11	25 kev
5	11 - 12	15	5	12 - 14	5.5
6	13 - 14	5.5	6	15 - 16	2.04
7	15 - 17	1.01	7	17 - 19	0.249
8	18 - 24	thermal	8	20 - 22	2.77 ev
			9	23 - 24	thermal

<u>9 GP THRESHOLD</u>			<u>20 GP RZ</u>		
Group no.	Equivalent LIB-APDA-24 groups	Lower energy boundary	Group no.	Equivalent LIB-APDA-24 groups	Lower energy boundary
1	1	3.678 Mev	1	1 - 3	1.35 Mev
2	2	2.225	2	4	0.825
3	3	1.35	3	5	0.50
4	4	0.825	4	6	0.30
5	5	0.50	5	7	0.18
6	6	0.30	6	8	0.11
7	7 - 8	0.11	7	9	67 kev
8	9 - 11	25 kev	8	10	40.7
9	12 - 24	thermal	9	11	25
			10	12	15
			11	13	9.1
			12	14	5.5
			13	15	3.36
			14	16	2.04
			15	17	1.01
			16	18	0.502
			17	19	0.249
			18	20	55.6 ev
			19	21	12.4
			20	22 - 24	thermal

FIG. 32 CRAM CALCULATIONAL GEOMETRY FOR RZ PROBLEMS -8GP RZ AND 9GP RZ

FIG. 33 CRAM CALCULATIONAL GEOMETRY FOR RZ PROBLEMS—9GP RZ AND 20GP RZ

FIG. 34 U-235 FISSION AXIAL DISTRIBUTIONS

FIG. 35 U-238 FISSION AXIAL DISTRIBUTIONS , CORE SUBASSEMBLIES

FIG. 36 U-238 FISSION AXIAL DISTRIBUTIONS , RADIAL BLANKET SUBASSEMBLIES

FIG. 37 U-238 CAPTURE AXIAL DISTRIBUTIONS

3. XY Geometry Presentation for Midplane Values in Specific Core Subassemblies

The rather complex and irregular geometry of the core necessitates a two-dimensional calculation in XY geometry to effect a detailed comparison with the measured values. Such a calculation has been run with eight energy groups employing the LIB SEVEN 24 library and is designated 8 GP XY. The energy group structure is the same as for the 8 GP RZ problem (Table XIX), and the problem geometry is shown in Fig. 38. Energy group-dependent axial bucklings were obtained from flux shapes computed near the core midplane in the 8 GP RZ problem. Buckling values are given in Table XX. These buckling values are not critical for radial reaction-rate distributions in the core. The requirement that specific reaction rates be available for more than 100 subassemblies dictated a moderately crude mesh for the inner portion of the radial blanket and a very crude mesh for the outer portion. However, all core subassemblies contained four mesh points.

It would be impossible for a single XY problem to accurately portray the core during all of the core foil irradiations since those irradiations encompassed numerous core geometries. (See Table II.) A compromise model was chosen that best represented the local geometry used when foils were irradiated in a given area. A diagram of the calculated geometry is given in Fig. 38 with X's denoting core foil subassembly locations. The calculated and measured midplane reaction rates for their positions are given in Table XXI.

TABLE XX - ENERGY GROUP-DEPENDENT AXIAL BUCKLING
USED IN 8 GP XY PROBLEM

Group	Core, 10^{-3} cm^{-2}	Control-rod channels, 10^{-3} cm^{-2}	Source, 10^{-3} cm^{-2}	Radial blanket, 10^{-3} cm^{-2}
1	0.806	0.809	0.805	0.797
2	0.823	0.824	0.805	0.815
3	0.800	0.799	0.772	0.797
4	0.748	0.746	0.733	0.755
5	0.664	0.661	0.687	0.688
6	0.564	0.517	0.641	0.594
7	0.435	0.373	0.584	0.493
8	-0.022	-0.409	0.485	0.231

(See Also Table XXI)

FIG. 38 CALCULATIONAL MODEL USED FOR COMPARISON WITH MIDPLANE CORE FOIL MEASUREMENTS

TABLE XXI - CALCULATED AND MEASURED MIDPLANE
REACTION RATES IN CORE SUBASSEMBLIES*

(8 GP XY, CRAM No. 0110-67)

	U-235 Fission			U-238 Fission			U-238 Capture		
	calc.	meas.	$\Delta(\%)^{\ddagger}$	calc.	meas.	$\Delta(\%)^{\ddagger}$	calc.	meas.	$\Delta(\%)^{\ddagger}$
P00-P00	10.16	10.47	3.0	10.52	10.43	-0.9	10.11	9.95	-1.6
P00-P01	10.04	10.18	1.4	10.53	10.78	2.4	9.96	9.77	-1.9
N01-P01	9.99	9.86	-1.3	10.24	10.51	2.6	9.90	9.21	-7.0
P02-P00	9.50	9.47	-0.3	9.85	9.75	-1.0	9.44	9.54	1.1
P00-P03	9.04	9.44	4.4	9.69	9.72	0.3	8.89	8.89	0
P04-P00	7.98	7.64	-4.3	8.46	8.18	-3.3	7.89	7.76	-1.6
N03-P03	7.63	7.80	2.2	8.20	8.32	1.5	7.51	7.74	3.1
P00-P05 [†]	6.40	6.75	5.5	6.17	5.92	-4.1	6.41	6.38	-0.5
P05-P00	6.57	6.57	0	6.70	6.50	-3.0	6.53	6.43	-1.5
N04-P04	5.52	5.40	-2.2	5.04	5.17	2.6	5.56	5.92	6.5
N04-N04	6.24	5.83	-6.6	5.45	5.45	0	6.48	6.64	2.5
N05-N03	5.87	5.82	-0.9	4.91	5.10	3.9	6.15	6.39	3.9
P06-P00	5.05	4.77	-5.5	4.24	4.18	-1.4	5.17	5.37	3.9

* Each reaction-rate type is normalized so that the total for all subassemblies is 100.

[†] This is the only core portion for which the integrated-count method was used to normalize the power (Section III-I).

[‡] Per cent by which the measured value exceeds calculated value.

4. Special Effects

Foil measurements were made at various locations to determine the effect of local geometric or composition perturbations on nearby reaction rates. These are discussed briefly below and the results are summarized and compared to expected results where calculations are available.

a. Effect of Sb-Be Neutron Source - The large Sb-Be neutron source located in position N05-N04 was expected to have a rather large local effect on reaction rates, due principally to the moderating effect of the beryllium (Be occupies ~ 55 per cent of the subassembly volume). The effect of the neutron source on reaction rates at the center of core position N04-N04 was obtained by comparing the response of foils irradiated in N04-N04 to those irradiated in the symmetric position, N04-P04. The latter irradiation was made twice, once with a core subassembly and once with a blanket subassembly in position N05-P04, to obtain the effect of the source relative to both core and blanket subassemblies, respectively. Calculated effects were obtained from the 8 GP XY problem. A summary of the effects for the mid-

plane of subassembly N04-N04 is given in Table XXII. Unfortunately, meaningful measurements could not be performed in a blanket subassembly adjacent to the neutron source since there were no IRB foil subassemblies.

TABLE XXII - EFFECT OF THE ANTIMONY-BERYLLIUM
NEUTRON SOURCE ON FOIL RESPONSE
IN CORE SUBASSEMBLY N04-N04

	<u>Response Ratios*</u>	
	<u>Measured</u>	<u>Calculated[†]</u>
U-235 fission	1.08	1.13
U-238 fission	1.05	1.08
U-238 capture	1.12	1.16

* Response (N04-N04) divided by response (N04-P04).
Position N04-N04 is adjacent to the antimony-beryllium
neutron source. See Fig. 38.

[†] Based on 8-group full Core-A XY problem designated
CRAM 0110-67.

b. Effect of Stored Core Subassemblies - There are a number of positions in the outer radial blanket where core subassemblies may be placed for storage. Such placement was expected to affect the heat-generation rates in adjacent blanket subassemblies. A measurement of this effect was obtained by comparing the response of foils irradiated in blanket subassembly P14-P00 when positions P14-P01 and P14-N01 contained blanket subassemblies, to the response obtained in P14-P00 when positions P14-P01 and P14-N01 contained stored core subassemblies. A calculated result was obtained by comparing computed reaction rates in a two-dimensional XY problem (equivalent to the 8 GP RZ problem discussed earlier) to those from a similar problem but containing stored core subassemblies at positions P14-P01 and P14-N01.* A summary of the results is given in Table XXIII.

c. Effect of Missing Radial Blanket Subassembly - The radial blanket lattice position P00-N09 is vacant, forming a source of local flux perturbations relative to conditions that would exist for a uniform homogeneous radial blanket. The magnitude of such perturbations as they affect reaction rates was measured by comparing the response of foils irradiated

* Both problems utilized quadrant symmetry and the LIB SEVEN 24 library. Problem numbers are CRAM 0084-66 and 0090-66 for the stored core subassemblies and base case, respectively.

TABLE XXIII - EFFECTS OF ADJACENT STORED CORE
SUBASSEMBLIES ON FOIL RESPONSES
IN BLANKET SUBASSEMBLY P14-P00

	<u>Response Ratios*</u>	
	<u>Measured</u>	<u>Calculated</u> [†]
U-235 fission	0.741	1.04
U-238 fission	4.12	5.09
U-238 capture	0.786	0.952

* Response with adjacent stored core subassemblies divided by response with adjacent blanket subassemblies. Foils were irradiated in position P14-P00 during two runs. For one run there were normal blanket subassemblies in the adjacent positions P14-P01 and P14-N01. For the other run these positions contained stored core subassemblies.

† Based on two 8-group Core-A problems utilizing quadrant symmetry and designated CRAM 0084-66 and CRAM 0090-66 for the stored core subassembly and base case, respectively.

in P00-N10 with those irradiated in the symmetric position P00-P10. The effect has been computed in terms of heating-rate perturbation by using one-dimensional problems depicting the missing subassembly as a ring and applying geometric correction factors.¹² A summary of the results is given in Table XXIV.

d. Effect of Large Core - As burnup increases, the number of core subassemblies required for criticality increases. This growth in core size will create an increase in reaction rates for blanket subassemblies located near the core. A measure of this effect was determined by obtaining foil responses in radial blanket subassemblies N06-P04, N06-P05, and N06-P06, with the core extended in the quadrant where these subassemblies are located. No attempt was made to compare these responses with those that would be obtained from a smaller core, nor have computations been performed to obtain reaction rates at the above positions for the core geometry that existed in the experiment. Results are therefore limited to the measured relative reaction rates given in Tables XV - XVII.

C. CROSS SECTIONS

A sensitive ingredient in the analysis of foil-measured, reaction-rate distributions is the cross-sectional data. Such data are logically divided into two categories: (1) cross sections used for flux calculations and, (2)

TABLE XXIV - EFFECT OF THE VACANT BLANKET
SUBASSEMBLY POSITION P00-N09 ON
FOIL RESPONSES IN BLANKET
SUBASSEMBLY P00-N10

	<u>Measured Response Ratios*</u>
U-235 fission	1.06
U-238 fission	1.40
U-238 capture	1.14
Subassembly power	1.19 [†]

* Response in position P00-N10 divided by response in position P00-P10. Foils were irradiated simultaneously in position P00-N10 adjacent to the vacant lattice position and in the symmetric position P00-P10.

† A calculated value of 1.25 is obtained, based on a geometrically corrected, one-dimensional, cylindrical calculation for a blanket corresponding to 75 Mw-years of reactor operation.¹²

cross sections used for foil response. While there has been relatively little effort expended in this area, some attempts have been made to ascertain the sensitivity of computed reaction rates to the choice of various cross sections. Essentially all of the effort has been in the latter category, ie., cross sections used in foil response. A brief documentation and description of the cross-sectional work is given below.

1. Cross Sections Used for Flux Calculations

Most of the fluxes calculated by multigroup techniques used either the LIB APDA 24 or LIB SEVEN 24 libraries. These are temporary 24-group libraries constructed to update and extend an earlier 16-group library¹³ to lower energies. A brief description of the LIB APDA 24 library may be found on pp. 35 - 37 of Reference 14, while a more detailed description of the LIB SEVEN 24 library may be found in Reference 15. Some of the earlier work such as that reported in References 7 and 8 used an 18-group cross-sectional set designated 18 R2, based on values given in Reference 16. Specific data for the 18 R2 set is given in Reference 17. Still earlier work such as that reported in Reference 9 was based on a 10-group library originated at Argonne National Laboratory. There has been no attempt, for the foil response work reported here, to perform a parametric variation of cross sections or to incorporate the very latest data for the purpose of determining the effect such changes would have on computed flux distributions.

2. Cross Sections Used for Foil Response

Experimental U-235 reaction rates were based on the response of Al-18 w/o uranium alloy pins (0.080-in. diam by 3/4-in. long) with the uranium enriched to 93 w/o U-235. Experimental U-238 reaction rates were based on the response of depleted uranium pins (0.080-in. diam by 1-in. long) with an enrichment of 0.22 w/o U-235. At positions where the neutron flux is relatively soft, the U-235 fission and U-238 capture response of these foils become dependent on the composition of the local environment and on the composition and geometry of the foils, due to the phenomenon of neutron self-shielding. In the core, the spectrum is sufficiently hard so that these self-shielding effects are negligible; however, far out in the blanket the effects become appreciable.

A detailed determination of the proper self-shielded foil-response cross sections for all the environments where measurements were made is beyond the scope of this work and inconsistent with the accuracy level of some of the data contained in the temporary libraries used for most of the analyses. Such a study will be more appropriate when the new ENDF/B data system and its auxiliary provisions for routinely-computing, self-shielding effects become available. However, preliminary analyses of self-shielding effects have been made (1) to estimate the sensitivity of reaction rates to various assumptions with regard to self-shielding, and (2) to obtain a reasonable estimate of the proper foil response.

Uncertainties also exist in the basic data, but a detailed study of the possible variation of basic data and their effect on foil measured reaction rates has not been made. The major reported uncertainties in U-235 fission and U-238 capture cross sections that are important for reactor criticality and internal breeding-ratio calculations have little effect on the reaction-rate distribution of interest here. They will affect the corresponding reaction-rate ratios.

A brief outline of the foil response cross-sectional effort used in this study is given below, according to cross-sectional type.

U-235 Fission: The two sets of U-235 fission cross sections, provided in the LIB SEVEN 24 library for groups 1-14 (10 Mev - 5.5 kev), are identical to that given by Yiftah in Reference 13 and are essentially unaffected by either energy or space self-shielding considerations. Values below 5.5 kev in the LIB SEVEN 24 library are based on a 33-group set from the United Kingdom for a nonresonant cross section, (σ_p), of 400 barns per atom at 300 K for one set, while infinite dilute values are provided as a second set. Self-shielding need not be considered until the neutron energies become less than about 500 ev where the major resonances begin. Significant fluxes below 500 ev exist only in the outermost parts of the blanket and beyond. Furthermore, the U-235 concentration in the blankets, enriched foil, and depleted

foil is extremely low so that conditions approaching infinite dilution exist.¹⁸ Thus, the appropriate response cross sections for U-235 fission should be those corresponding to essentially infinite dilution. However, auxiliary analyses also utilized the nominally self-shielded ($\sigma_p = 400$) U-235 fission cross-sectional set of LIB SEVEN 24. The use of both of these self-shielding options are correspondingly indicated on the axial distribution curves shown in Fig. 34. The radial distributions shown in Fig. 31 were computed with the nominal self-shielded option for convenience; however, the maximum difference in reaction rate resulting from the use of this option rather than the infinite dilute cross section is two per cent, occurring at the maximum radius ($r = 102.6$ cm).

U-238 Fission: The U-238 fission cross sections in the LIB SEVEN 24 library (as well as the LIB APDA 24 library) are identical to those given by Yiftah in Reference 13. An additional source of U-238 fission cross sections that were used in the 9 GP RZ THRESHOLD problem was BNL-325, weighted with a nominal calculated Core-A spectrum. However, the difference in U-238 fission distributions calculated by these two cross-sectional sets in a given flux are negligible, so that no distinction is made regarding the choice of basic U-238 fission cross sections for foil responses. However, uncertainties in the U-238 fission cross-sectional values (e.g., as implied by the data scatter shown for U-238 fission in Reference 19) will be reflected in calculated fission ratios involving U-238.

U-238 Capture: The two sets of U-238 capture cross sections provided in the LIB SEVEN 24 library for groups 1-12 (10 Mev - 15 kev) are identical to that given in Reference 13, with the exception of minor revisions in groups 7-9. Values from group 13-21 (15 kev - 12.4 ev) for one set come from the self-shielded dependent values given by Fischer in Reference 20 for a nonresonant cross section of 80 barns per atom at 300 K, while the other set was the infinite dilute values. Values for groups 22-24 (12.4 ev - thermal) are based on the 33-group United Kingdom set for the same two self-shielding options.

As in the case of U-235, neutron self-shielding considerations make little difference in the computed distributions in the core. However, beyond the core, neutron self-shielding effects have a significant effect on the foil response for U-238 capture. This is due to the relatively large capture resonances in combination with the high concentration of U-238 in the radial blanket as well as in the 80-mil depleted uranium foil. There are two main conditions to consider. First is the determination of the effective U-238 capture response of the foil within the radial blanket environment; the second is the determination of the U-238 capture response within a sodium-stainless steel environment such as that which occurs for measurements made in the end-gap, handling head, and thermal shield-bar regions.

U-238, as it exists in the 80-mil foil-- virtually surrounded by depleted U-238 blanket pins-- is heavily self-shielded. An approximate, effective, nonresonant cross section is calculated to be only 65 barns per atom, using the recipe for a simple lattice given in Reference 21. When the foil is removed from the depleted uranium blanket environment and placed in an essentially infinite scattering medium the effective, nonresonant cross section is estimated to be 120 barns per atom.¹⁸

The average U-238 capture cross sections equivalent to the dilutions estimated above are appropriate for computing the actual U-238 capture rate within the entire foil. However, the measurement involves the counting of very soft gamma rays (Section III-H-2) with a mean free-path in uranium of 0.015 cm, which is small relative to the mean chord length of the foil. Thus, only surface gammas are counted and the experiment is measuring neutron capture in U-238 near the surface of the foil. This consideration would result in a greater effective nonresonant cross section and hence a larger effective group cross section than conventionally computed for the entire foil if the experimental conditions are to be matched. Since the magnitude of this correction would involve an energy-dependent interrelation between gamma and neutron escape probabilities, it would be quite difficult to calculate and will not be formally done as part of this report. However, with this surface effect included for the cases of the foil in a blanket environment and outside the blanket, upper limits to the effective nonresonant cross section of 600 barns per atom and infinite dilution, respectively, have been established. The values previously calculated (65 and 120 barns per atom) that neglect the gamma self-shielding effect would represent lower limits to the range encompassing the proper value.

The foil response cross sections used for the calculated radial and axial U-238 capture distributions shown in Figs. 31 and 37 use the two self-shielded options provided in the LIB SEVEN 24 library, namely 80 barns per atom at 300 K and infinity for the effective nonresonant cross section. The former value lies between the lower and upper limit, which is appropriate for use in the blanket and thus represents a consistent estimate in the face of a lack of bases for a more precise value.* The choice of extent of self-shielding becomes important (>10 per cent effect) in the outer 10-15 cm of blanket and beyond. The range of uncertainty in σ_p in the blanket (65-600 barns per atom) reflects a 30 per cent variation in the computed U-238 capture rate at the outer edge of the radial blanket at core midplane elevation ($R =$

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* Actually, the 80 barns per atom at 300 K cross-sectional option in the LIB SEVEN 24 library gives about the same reaction rate as 65 barns per atom at 500 K, the appropriate temperature for the experiment. Thus, the calculation using the self-shielding cross sections may be considered to lie very close to the lower side of the σ_p uncertainty range.

98.7 cm in Fig. 31). Unfortunately, the variation in computed U-238 capture rate, at points exterior to the blanket due to uncertainties in the value of σ_p , are considerably larger due to the greater difference between maximum and minimum values of σ_p (120 and ∞ barns per atom) in combination with the softer spectrum. However, the calculated U-238 capture curves for points exterior to the blanket resulting from the two cross-sectional options (Figs. 31 and 37) would bracket the calculation utilizing the proper choice of effective self-shielding for the experimental conditions.

D. DISCUSSION OF RESULTS

1. Radial Distribution of Midplane Values in Cylindrical Geometry

There is general agreement between the measured and calculated reaction-rate distributions shown in Fig. 31 except for points near the outer portion of the blanket. Some of the general scatter of points most evident in the U-238 fission distribution can be explained by the experimental 95-per cent confidence limit of ± 10.5 per cent. The fact that essentially all the measured U-238 capture points are greater than the calculated distribution except for core center, where agreement is forced, places suspicion on the validity of the measured point at core center. However, no error in data reduction could be found. At the outer edge of the blanket there is evidence that the lower energy flux is greater than that calculated since both the measured U-238 capture and U-235 fission points are higher than calculated, even using the infinite dilute cross sections.* At the same location, the measured value for U-238 fission is greater than calculated. This is consistent with the above conclusion since a substantial fraction of the U-238 fission rate at this position has been shown by analysis to be due to fission neutrons from U-235 in the outer row of radial blanket subassemblies.

2. Axial Distributions for Selected Subassembly Positions

The nature of the axial distribution of reaction rates is highly dependent upon radial position (see Figs. 34 through 37). Here the term "radial position" must include the degree of proximity to core and shield-bar boundaries as well as absolute radial distance from core center. This phenomenon, coupled with the fact that the selection of mesh-point positions in the various two-dimensional problems could not in general be chosen to correspond precisely with foil positions, introduces a basic geometrical uncertainty when comparing measurements with calculations in the radial blanket. However,

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* Though only one calculated curve is shown for U-235 in Fig. 31, it may be taken as representing results for both the infinite dilute and $\sigma_p = 400$ barns per atom U-235 fission cross-sectional capture, since there is negligible difference in the resulting reaction rates over the region calculated.

typical axial distributions within core subassemblies are not so sensitive to radial position.

Two core subassembly positions, P00-P03 and N01-P01, have been chosen for comparison. The U-235 axial distributions for these two positions are similar (Fig. 34a). There seems here to be a depression for N01-P01 relative to P00-P03 in the vicinity of the end-gap. This is consistent with the position of the boron-loaded regulating rod during the experiment. Such an operating control rod is expected to have a greater effect on the U-235 fission rate in the softer spectrum location of the end-gap than in the harder spectrum location of the core where no effect was observed (Section VII-D-3). Agreement with calculations is quite good over the core and most of the axial blanket. The same discrepancy as noted in the radial distribution may be observed near the top of the axial blanket, with measured values being greater than calculated values. The effects of the proximity of the safety rods may be seen in both measured and calculated values. Positions P00-P03 and N01-P01 are adjacent to a safety-rod channel while position P05-P00, for which two measured points are included, is far removed from such channels. A radius of 8.6 cm in the calculation is interior to a region that included homogenized safety-rod material, while a radius of 19.7 cm is exterior to this region.

Essentially the same comments may be made concerning the U-238 capture distributions in the core subassemblies (Fig. 37a). However, there is no depression evident for the foils adjacent to the regulating rod as was the case for the U-235 fission. This is surprising since U-238 capture is more responsive to low energy neutrons than U-235 fission.

There is also reasonably good agreement between the measured and calculated distributions for U-238 fission in the two core subassemblies (Fig. 35). However, the difference in calculated distribution through the region homogenized with safety-rod channel material ($R = 8.6$ cm) and that in pure core material ($R = 19.7$ cm) is greater than for the other distributions. Further, the uncertainty associated with points high in the axial blanket is relatively large because of the large fission component in the depleted foil due to U-235. For the point for which the largest uncertainty band is shown in Fig. 35, 64 per cent of the fissions in the foil take place in U-235.

The blanket subassembly positions most suitable for comparison with calculation are P00-P12 and P12-P00. They are not close to either the core or outer blanket boundaries, and calculated axial distributions exist for essentially the same radius from core center, thus minimizing the uncertainty discussed at the beginning of this section. Moreover, better counting statistics existed by virtue of the reactor power level that was used for the irradiation, and the use of two symmetric positions at the same radius provides data redundancy that confirms the good statistics of the experiment. With only a few exceptions, there is excellent agreement between measured points for P00-P12 and P12-P00, as shown in Figs. 34d, 36c, and 37d.

Agreement with the 20-group calculation is quite good for the U-235 fission distribution (Fig. 34d). The inadequacy of the 9-group calculation is evident, though it may be a little exaggerated, since the radius is somewhat smaller than for the 20-group problem. While it appears the self-shielded cross sections give better agreement than the infinite dilute values, contrary to theoretical expectation (see Section VII-C), the difference is too small to form a basis of selection of cross sections in view of the other uncertainties calculated.

Agreement between calculation and measurement for U-238 fission distribution is reasonable over the fueled portion of the radial blanket, though the data scatter is somewhat greater than the measurement statistics would suggest (Fig. 36c). However, the rather large disagreement for the two measured points in the end-gap is surprising. Both the estimated experimental uncertainty (2σ value for the last point is indicated in Fig. 36c) and the agreement between measurements for the two symmetric subassembly locations would indicate that the disagreement is not due to poor experimental accuracy. The single exception to this logic would be if the enrichment of the depleted foils was larger than assumed.* A 17-per cent increase in assumed enrichment (from 0.22 to 0.257 w/o) would bring agreement between the measured point at 81.3 cm and the 20-group calculation. If this enrichment error were assumed for all pins, it would have an adverse effect on measurement - calculational agreement for the upper points of N06-P04 (Fig. 36c) and P00-P03 (Fig. 35). However, the enrichment effect would be smaller than for the P00-P12 position, and it should be noted that the uncertainty bands at these positions are quite large due to poorer counting statistics. Further, geometrical complexities affect the relationship of measurements at N06-P04 and P00-P03 to calculations because of proximity to the core and safety rods, respectively. Finally, use of more dilute U-235 cross sections in the flux calculations would provide some improvement, but the effect would be small as indicated by the comparison of the $\sigma_p = 400$ barns per atom and infinite dilute values used for foil cross sections in Fig. 34d.

The curves for U-238 capture (Fig. 37d) indicate the strong dependence on calculated U-238 capture rate on both cross-sectional dilution and number of energy groups in regions of soft neutron spectrum. The two measured points in the end-gap lie between the calculated curves corresponding to cross-sectional options that encompass the correct value (see Section VII-C).

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* The 2σ uncertainty assigned to the enrichment was ± 15 per cent.

The progression of axial distributions in the radial blanket, starting near the core and moving outward, display the expected flattening in the fuel portion and increased magnitude in the end-gap, the latter particularly characteristic of the nonthreshold reactions. The four measured points for U-238 capture, in subassembly N06-P04 at an elevation equivalent to the end-gap of a core subassembly, are greater than calculations. This discrepancy is probably because the actual core configuration places the core end-gap region on two sides of the foil subassembly in question and within 3.4 cm of the foil, while the calculated traverse is about 5 cm from the boundary of the equivalent cylindrical core. The moderating effect of the end-gap on the U-238 capture rate is thus more pronounced for the actual configuration. The last measured point in the fueled portion of each radial blanket subassembly for the nonthreshold reactions appears inconsistent with the calculated trends for all the subassemblies shown. The most likely explanation is the difficulty in calculating steep gradient fluxes in the vicinity of a material boundary coupled with the geometric complexity of the region itself. For example, the fuel pin immediately above the fueled portion contains a small layer of sodium followed by a gas-gap. The composition used in all calculations employs none of this detail, using a homogenous composition typical of the entire handling head region which would tend to exaggerate the moderating influence of the region.

3. XY Geometry Presentation for Midplane Values in Specific Core Subassemblies

Agreement between measurement and calculation for the radial midplane distribution of reaction rates in the core as given in Table XXI is almost within the average 2σ uncertainty ranges given in Section IV. Some discrepancies may be due to the fact that the calculational model (Fig. 38) does not exactly correspond to any of the reactor configurations used during the core portion of the foil program (Figs. 14, 15, 16, 18). For example, the reason that all measured values for P04-P00 are less than calculated values may be due to the inclusion of a core subassembly at P06-P00 in the calculation that was not present in the reactor during that particular irradiation. Two of the larger discrepancies appear in P00-P05. This is the only core position for which the integrated-count method was used to normalize the power (Section III-1), and thus additional statistical uncertainties are introduced.

The greatest discrepancy relative to the measurement of uncertainties as determined in Section V is for U-235 fission in N04-N04. This may be partially due to the fact that this position is adjacent both to the core boundary and neutron source where the flux gradients are rather steep, thus aggravating both the consequences of the foil position uncertainty and errors due to lack of geometrical rigor in the calculational model. There appears to be no measurable effect on reaction-rate

measurements adjacent to position P01-P00 due to the presence of the boron-loaded shim rod.

4. Special Effects

a. Effect of Sb-Be Neutron Source

Comparison between the measured and calculated effects of the Sb-Be neutron source in position N05-N04 on reaction rates in subassembly N04-N04 is within expected agreement (see Table XXII). A major difficulty in the analysis stems from the complex geometry. The beryllium in the square source subassembly exists as a solid ring occupying 55 per cent of the subassembly. The calculation homogenizes the entire subassembly. Since flux gradients close to the beryllium can be quite steep, such a procedure will introduce error in computed reaction rates close to the source.

The moderating effect of the source is sufficiently strong to increase the U-238 capture response in the adjacent subassembly over that which would be obtained if a normal core subassembly occupied the source position. Unfortunately, the reaction-rate gradients are steepest adjacent to the source, and the construction of the foil subassembly²² does not allow foils to be placed at such a position. Thus, there is still some uncertainty in translating the measured reaction rates to accurate, local maximum, and total subassembly power and plutonium buildup rates.

b. Effect of Stored Core Subassemblies

The storing of core subassemblies for all three reaction-rate types leads to less-than-calculated reaction rates in an adjacent blanket subassembly (Table XXIII). The measured U-235 fission and U-238 capture rates are reduced for the case of the adjacent, stored core subassemblies. This is due to the large neutron sink afforded by the highly enriched core subassemblies in a poor moderating environment. The U-238 fission rate is increased due to the increase in fission neutrons. The calculation was somewhat hampered by a lack of energy groups and mesh points in the area of interest since the problems had not been optimized to study this particular effect.

E. REACTION RATES AND REACTION-RATE RATIOS

The only absolute measurement of reaction rates in this program was obtained by means of radiochemical analysis performed on selected foils. These results are given in Table XIV. Since the ratio of absolute reaction rates are useful as spectral indices, the ratios of U-238 fission and U-238 capture to U-235 fission were formed from the experimental results reported in Table XIV. Ratios were determined only for the two

core subassembly positions and are given in Table XXV. Since there was a pair of enriched foils and a pair of depleted foils at equivalent elevations in each of the two subassemblies (Table XIV), the ratios utilized the average of the data for each pair. However, the data for one of the foils (No. 875) were rejected since they appeared inconsistent with the remaining data. The stainless steel foil tube and sheath would tend to artificially depress the U-238 fission rate,²³ but this effect should be quite small as the combined wall thickness is only 0.024 in.

TABLE XXV - CALCULATED AND MEASURED
REACTION-RATE RATIOS *

	<u>Measured</u>	<u>Calculated †</u>	
		<u>24 GP ID CYL</u>	<u>8 GP XY</u>
<u>P00-P00:</u>			
f_{28}/f_{25}	0.0421	0.0407 (3.4)	0.0408 (3.2)
c_{28}/f_{25}	0.116	0.118 (-1.7)	0.119 (-2.6)
<u>P00-P03:</u>			
f_{28}/f_{25}	0.0429	0.0430 (-0.2)	0.0422 (1.7)
c_{28}/f_{25}	0.107	0.117 (-9.3)	0.118 (-10.3)

* Ratios on a "per atom" basis.

† Values in parentheses are the percentages by which the measured value exceeds the calculated value.

Calculations of these reaction-rate ratios were made with the 24-group, one-dimensional cylindrical problem (24 GP ID CYL) and the 8-group problem in XY geometry (8 GP XY), with the results included in Table XXV.

Agreement between calculated and measured ratios is reasonable though the 10-per cent discrepancy in the U-238 capture-to-U-235 fission ratio at P00-P03 is somewhat large. Unfortunately, the uncertainty in the radiochemical analysis is not known so that it becomes difficult to make inferences concerning the accuracy of the calculation. However, the U-238 fission-to-U-235 fission ratio measurements appear consistent with a value of 0.0379, obtained from absolute fission counters in the temporary instrument thimble. The latter value should be less than those in positions P00-P00 and P00-P03 due to the nonfueled moderating environment of the temporary instrument thimble. Also, there is a slight spectral hardening

in going from position P00-P00 to P00-P03, as evidenced by both the measured and calculated increase in the U-238 fission-to-U-235 fission ratio.

It is interesting to compare the results in Table XIV for absolute number of U-235 fissions at core center (P00-P00) with that inferred by the response of an absolute U-235 fission counter at core midplane in the temporary instrument thimble and used in conjunction with the power calibration of the reactor 11 months previous to the foil irradiations. For a given reactor power, the U-235 fission rate implied by the absolute foil response was 0.5 per cent less than that obtained with the fission counter.* This agreement (the extremely close agreement is fortuitous) lends confidence to the experimental determination of local absolute U-235 fission rate required for power calibration of the reactor by nuclear techniques. However, it should also be noted that good agreement is not obtained for a similar comparison between the radiochemical data for U-235 fissions in position P00-P03 and the absolute fission counter. Further, a consistency check, performed by comparing enriched foil count rate for the foils listed in Table XIV with the radiochemical determination of absolute number of fissions in the same foils, indicated a corresponding discrepancy in the P00-P03 enriched foil data.²⁴ The conclusion derived from such comparisons is that the absolute number of fissions in the enriched foil irradiated in P00-P03, as determined by radiochemical techniques, appears to be low by approximately 26 per cent. It would appear that the fissions and captures in the depleted foil at the same location must also be too low by a comparable amount since the appropriate reaction-rate ratios as reported in Table XXV for position P00-P03 are essentially consistent with the calculated ratios and with the ratios for position P00-P00.

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* This comparison utilizes a fission counter efficiency of 0.98 and a U-235 fission response ratio between core center and the temporary instrument thimble of 1.16, as determined from the 8 GP XY multigroup calculation.

VIII. CONCLUSIONS

The foil irradiation technique described in this report successfully provided the U-235 and U-238 fission distributions and the U-238 capture distributions throughout the core and blanket of the EFFBR. These experimental distributions are in general agreement with those obtained by multi-group diffusion techniques. To obtain acceptable analytical values, however, care must be taken regarding group structure, geometry make-up, and basic cross sections at points near sharp boundaries of materials and in regions of softer neutron spectrum. It would be well to perform a parametric study on reaction-rate calculations and investigate some of these parameters, particularly now that significant advances have been made in the acquisition and standardization of basic cross sections. Recent measurements indicating increased inelastic scattering cross sections for U-238 may affect calculated reaction rates in the blanket, as may the development of techniques for generating space-dependent group constants. With such improvements, the true limits of diffusion theory in reaction-rate calculations could be better established.

The analysis of data and applicability of results would have been aided if additional measurements had been made below the core midplane, especially in the radial blanket where the lengths of the fuel above and below core midplane elevation differ markedly. Further, it has been demonstrated here that the lack of accuracy with which the enrichment of the depleted foils is known, continues the weakest link in the U-238 fission distribution measurements in soft spectrum regions. Future foil measurements should include more effective measures to both reduce and clearly determine the uncertainty of the depleted uranium enrichment. If use of commercial grade depleted uranium greatly increases this task relative to that required for natural uranium, then the latter should be used. Any benefits obtained through the use of nominally depleted uranium can be completely negated by a small increase in enrichment uncertainty. Finally, any future foil work should use, if at all possible, the standard foil techniques for normalizing reactor power among different runs rather than rely on external power monitoring, as was required for the latter part of the program.

The most practical conclusion of the work reported here with regard to the present core in the Enrico Fermi reactor is that careful multigroup diffusion calculations can in most cases be used to determine reliable power distributions in and around the core. It is necessary to examine computed blanket powers, particularly at large distances from the core, with regard to their consistency with measured reaction-rate distributions, and to apply corrections based on such an examination.

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