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SNAP SHIELD TEST EXPERIMENT
FISSION-PLATE POWER CALIBRATION

AEC Research and Development Report

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

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I. INTRODUCTION

The SNAP Shield Test Experiment (STE) was constructed for the radiation evaluation of shield materials and geometries for the SNAP reactor development program. A detailed description of the facility is given in References 1 and 2. The physics tests that were performed on the reactor system are presented in Reference 3.

The facility consists of: a 50-kw water-cooled zirconium-hydride uranium reactor; a neutron thermal column; a fission-source plate; and associated equipment. The uranium fission-source plate is normally located on a wall of the underground shield test room where all the shield measurements are performed. It is driven by the reactor through a graphite thermal column (see Figure 1). A shield test carriage is used in conjunction with the fission-source plate to support the shields while undergoing radiation evaluation. Both the fission-source plate and the carriage can be remotely positioned with the aid of a closed-circuit TV system. The fission plate can be driven into a lead shielding cask and the test shields can be retracted from the thermal column thereby allowing operating personnel to assemble the shield experiments in a low-radiation background. Since the fission-source plate serves as the radiation source for many shielding and radiation measurements, an accurate determination of its fission rate is required in order for accurate radiation measurements to be made. This report describes the techniques that were employed in obtaining these calibration measurements.

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Figure 1. STE Architectural Sections and Details

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II. SUMMARY AND CONCLUSIONS

Three techniques were employed in measuring the fission rate of the STE fission-source plate;⁴ (1) the thermal neutron absorption rate, (2) power dissipation (calorimetric method) and (3) the fission-neutron production technique. The results obtained, when weighted according to the reciprocal of the squares of the estimated errors, yielded a fission rate of 1.97×10^{12} fissions/sec $\pm 4\%$ for a STE reactor power level of 50 kw. Assuming that 190 Mev^{5,6} of usable energy is liberated per fission of U^{235} , this fission rate corresponds to a fission-plate power of 59.9 watts $\pm 4\%$.

The calorimetric calibration technique proved to be the most satisfactory of the three techniques used. The design of the STE fission source plate allowed the plate to be enclosed completely in fiberglass insulation and heated electrically as well as by nuclear means. The heating rate is sufficiently high that errors caused by ambient temperature changes are negligible. Details of the three calibration techniques employed are given in Section IV.

The average fission-plate power per unit area was determined to be 2.05×10^{-2} watts/cm² $\pm 4\%$, or about 14 times that of the Oak Ridge National Laboratory (ORNL) Lid Tank Facility fission plate.⁴ Although this is one-sixth that of the design value⁷ (apparently due to graphite contamination), it is nonetheless satisfactory for the initial shield measurements of the SNAP Reactor Development Program. The possibility of raising the power of the fission plate at least a factor of 5 through modification of the bismuth window and graphite thermal column is being investigated. In addition, the reactor power is capable of an increase from 50 to 500 kw for short-period operation by minor changes in the external piping and replacement of the primary coolant pump.⁸ Extended operation of 500 kw would necessitate the addition of a larger heat exchanger. Ultimately, however, the STE reactor fission plate is limited to a power of about 500 watts without supplementary radiation shielding.

Figure 2. Thermal Column Assembly

III. DESCRIPTION OF THE STE FISSION-PLATE SYSTEM

A. THERMAL COLUMN

The thermal column, 4-ft long and 5-ft square, is constructed of AGOT graphite, which is contained in an aluminum liner embedded in the concrete wall that separates the reactor from the shield test room (see Figures 1 and 2). The face of the thermal column in the shield test room is covered with a removable aluminum cover, sealed to the rest of the thermal column by means of bolts and a neoprene gasket. When the cover is in place, the cover and the bolt heads are flush with the wall of the shield test room. The four corner graphite stringers which extend the length of the thermal column are removable. Neutron-sensitive ionization chambers are placed within these stringers to continuously monitor the thermal-neutron flux in the column. A lead shutdown shield is placed between the reactor and the thermal column so that the graphite can be removed from the column without exposing operating personnel to a high gamma-radiation field from the irradiated reactor fuel. A bismuth window is located within the shutdown shield to permit the maximum number of neutrons to enter the column and still provide the reactor shutdown gamma-ray protection afforded by the lead shutdown shield. One-inch purge lines are provided at the top and bottom of the thermal column and are terminated with caps in the shield test room. These lines allow the entire thermal column and graphite to be purged with carbon dioxide or other suitable purge gases.

B. FISSION-SOURCE PLATE

The fission plate is composed of a 24-in. dia, 60-mil thick highly enriched (93.17% U^{235}) uranium disc with a total U^{235} content of 7.8 kg. The uranium disk consists of three 8-in. wide pieces of uranium and was fabricated by Metals and Controls Nuclear Incorporated, a subsidiary of Texas Instruments. The uranium disk is recessed in a 1-1/8 in. aluminum plate. An anodized aluminum heater coil is embedded in the aluminum plate in contact with the uranium. On the opposite side of the uranium plate are nine highly sensitive resistance thermometers* (see Figure 3). When the plate was

*Remote-reading temperature transducers having a range of 50 to 300°F are fastened directly to the uranium plate. These transducers and their recorders have a rated overall system accuracy of $\pm 0.3^\circ\text{F}$ and were obtained from the Foxboro Instrument Company, Foxboro, Mass.

Figure 3. Fission Plate

assembled, a boron sheet was placed over the uranium plate, and the whole assembly was heliarc-welded within a 1/8-in. aluminum plate. The whole assembly was outgassed at 250°F, evacuated, and sealed. Because the reactor was designed ultimately for higher power operation, the fission plate was supplied with a water-cooling system making use of the same water used to cool the reactor. All the electrical and cooling lines penetrate the top of the edge-supported aluminum plate through a 4-in. thickness of lead and Mallory 2000 shot* shielding which forms part of the fission-plate radiation shielding cask.

The fission plate is remotely positioned on the centerline of the thermal column by the fission plate drive mechanism. This operation can be performed from the reactor console and can be viewed continuously by television. The fission-plate drive is interlocked with the shield test carriage drive and the fission-plate cask lid. Therefore, the plate can be moved only when the cask lid is open and the shield test carriage is at the rear limit. Interlocks also prevent movement of the fission plate and shield test carriage during reactor operation.

C. FISSION-PLATE CASK

The fission-plate cask is normally located in a pit directly under the front face of the thermal column. The cask is aligned with the fission plate and its drive mechanism. The cask lid is remotely operated from the reactor console by a bottle of compressed nitrogen and a 4-way solenoid valve. The cask consists of a 4-in. thickness of poured lead surrounding a 3-ft-square by 2-in. - deep recess which receives the fission plate. The cask weighs about 5 ton and is positioned with the aid of the 7-1/2-ton shield test room crane. Once the fission plate is in the cask, all the electrical and cooling lines can be removed and the cask taken to a hot cell for repairs or alterations that may be necessary on the fission plate. Entry into the shield test room is normally made when the fission plate is fully within the cask. A radiation-activated interlock on the shield test room door prohibits entrance into the shield test room when the radiation in the room exceeds a preset level.

*Mallory 2000 shot is a tungsten alloy shot having a density of 10.0 gm/cm³ and was obtained from the P.R. Mallory Co., Inc., Indianapolis, Ind.

Figure 4. Shield Test Carriage

D. SHIELD TEST CARRIAGE

The shield test carriage is a self-propelled cart 8 ft 6 in. in width and 10 ft in length. It is designed to hold shields up to 5 ft square and 8 ft 6 in. in length. The carriage can be remotely operated from the reactor console and viewed continuously on closed circuit TV (see Figure 4).

A 7-ft-square by 1-ft-thick borated ordinary concrete "donut" can be fastened to the front of the carriage if so desired by the experimenter. A stepped 3-ft-square opening exists in the center of the donut, and a 1-3/8 in. - deep recess on the front of the donut provides space for the fission plate and drive mechanism. The purpose of the donut is to remove the degraded neutrons that emerge from the edges of the fission plate and thermal column; these neutrons might otherwise interfere with a given experimental measurement. Test slabs can be placed within the center of the donut without interfering with the other test slabs which may be on the shield test carriage.

IV. FISSION-SOURCE PLATE CALIBRATION TECHNIQUES

The three techniques that were used to calibrate the ORNL Lid Tank Shielding Facility source plate⁴ were adopted for calibrating the STE fission-source plate. The fission source plate was designed and constructed with these three measurement techniques in mind. Each technique deals with a different property of the fission reaction, viz., (1) the rate of neutron absorption, (2) the power dissipation, and (3) the rate of fission neutron production within the fission-source plate. Each of these methods and the corresponding fission rate estimates are discussed below. The neutron absorption and power dissipation techniques were used for calibrating the shielding-facility fission plate at the Battelle Memorial Institute⁹ also.

A. NEUTRON-ABSORPTION RATE TECHNIQUE

The actual STE Fission-Source Plate is a gas-tight welded assembly. This does not permit access to the actual inside of the plate, which would allow measurement of the neutron-absorption rate of the uranium. External neutron-absorption rate measurements would be greatly distorted by the presence of the internal boral sheet. This sheet serves the purpose of reducing the number of escaping neutrons re-entering the plate as low-energy neutrons and altering the power of the plate as the test shield composition is changed. A dummy fission-source plate, identical in geometry and composition to the original plate, was fabricated with a few necessary modifications; (1) the plate could be mechanically disassembled and contained no heater wires, transducer wires, or uranium. Internal space was provided for inserting gold and uranium foils for the measurement of neutron fluxes.

1. Measurement of the Thermal Utilization of the Fission Plate

Three uranium disks were obtained from scrap trimmed from the edge of the rolled fission plate during the fabrication process. Two of these were rolled to thinner thicknesses yielding foils of the following thicknesses: 61.2 ± 1.5 , 46.3 ± 1.0 , and 18.5 ± 0.5 mils. These disks were then used to determine experimentally the thermal-neutron attenuation of the incident neutrons from the reactor. Bare and cadmium covered gold foils were placed in the dummy fission-source plate assembly as thermal-neutron detectors. The boral sheet which forms the back (the side away from the thermal column) of the

fission-source plate protected the foils from low-energy neutrons leaking into the fission plate from the water tank shield. Uranium thicknesses of 18.5, 46.3, 61.2, 79.7 and 107 mils were used in the measurements. The measurements were performed at 5 radial positions on the source plate. The attenuation data from the 107 mil-thickness were used as a background monitor.

Two additional bare gold foils were exposed during each run as normalization foils. The resultant centerline data are plotted in Figure 5. It can be noted that the relaxation length of the neutrons measured with a gold-activation detector becomes longer as the thickness of the uranium is increased. This effect is probably due to hardening of the neutron spectra as the uranium thickness is increased, a normal result in an experiment of this type. Radial traverses across the uranium foils did not indicate appreciable distortion of the central data points by neutrons scattered around the edges of the foils.

From the experimental data a value for the thermal utilization of the fission source plate was determined to be $0.987 \pm 5\%$.

2. Neutron Flux Measurement

Two techniques were used to determine the thermal neutron flux incident upon the fission plate; the first consisted of gold foils and the second utilized uranium foils enriched to 93.17% in U^{235} . The first measurement was an absolute flux determination while the second was relative in nature.

a. Gold Foils

The gold foils used in this experiment were 5 mil thick and 3/4-in. in diameter. The foils were irradiated in aluminum and cadmium boxes (20 mil thick) and cross-calibrated with the ORNL Lid Tank Shielding Facility, the ORNL X-10 reactor thermal column, and the ORNL standard high-pressure ionization chamber. The high-pressure ionization chamber had previously been cross-checked with standard gold solutions obtained from the National Bureau of Standards.* A flux depression and self-shielding correction of 31.1% was applied to the irradiations performed in the ORNL thermal column and within the dummy STE fission-source plate assembly, based upon experiments performed at ORNL.^{10,11} The foils were counted in a well-type scintillation counter.

*This work was performed by L. Jung of ORNL.

Figure 5. Thermal Utilization Experiment

A value of the $2200 \text{ m/sec Au}^{197}$ cross section of 98.8 barns was used in the neutron flux calculations. A Maxwell-Boltzmann neutron distribution was assumed that was corrected for an effective neutron temperature of 297°K . An isoflux plot based upon the corrected gold data normalized to unity in the center of the fission-source plate is shown in Figure 6. The average value of the thermal-neutron flux (area weighted) was $7.40 \times 10^8 \pm 5\% \text{ n/cm}^2\text{-sec}$ was obtained by graphical integration of the measured data at the aluminum-uranium interface of the fission-source plate.

b. U^{235} Foils

Uranium foils of the same enrichment as the fission-source plate were irradiated in a similar manner to the gold foils described above. The foils were counted on a well-type gamma-ray scintillation spectrometer using monitoring foils to correct for the decay of the fission products during the decay cycle. The counter was biased for a gamma-ray energy of 0.5 Mev. The data obtained by this technique duplicated the shape of the gold-foil data; however, no calibrations on the U^{235} foils were available so that no absolute flux determinations were performed.

3. Fission-Rate Calculation

The intensity or power of the fission-source plate expressed in terms of fissions per second was determined from the following relation:

$$P = \phi U f S \quad \dots (1)$$

where

P = source plate fission rate (fissions/sec)

ϕ = area weighted incident thermal-neutron current ($\text{n/cm}^2\text{-sec}$)

U = thermal utilization (unitless)

f = fraction of neutrons absorbed that result in fission (fissions/neutron)

S = source plate area (cm^2)

The numerical values of these terms are given in Table I.

Figure 6. Fission-Source Plate Isoflux Plot

TABLE I
DATA USED IN DETERMINING THE SOURCE PLATE POWER
USING THE NEUTRON ABSORPTION RATE TECHNIQUE

Term	Numerical Value	Estimated Error
ϕ	$7.40 \times 10^8 \text{ n/cm}^2\text{-sec}$	$\pm 10\%$
U	0.987	$\pm 5\%$
f	0.838	$\pm 2\%$
S	2919 cm^2	$\pm 0.3\%$
P	$1.79 \times 10^{12} \text{ fissions/sec}$	$\pm 11\%$

On the basis of 190 Mev of useful energy emitted per fission, the fission rate measured corresponds to 54.4 ± 6.0 watts. The estimated error in the measurement of P is based upon the square root of the sum of the squares of the individual estimated errors.

B. POWER DISSIPATION TECHNIQUES

1. Temperature-Rise Measurements

The STE fission plate contains an internal electrical heater composed of 200 ft of anodized aluminum wire to simulate the fission heating of the uranium plate (see Figure 3). The heater is in contact with the uranium on the thermal column side of the fission plate. Nine temperature-sensitive resistance transducers are located on the opposite side of the uranium plate from the heater. By thermally insulating the fission-source plate assembly, the temperature response of the plate during fission heating could be compared with that due to artificial heating for several different electrical powers.

Since the total and fractional energy release for the fission process is known,⁵ and the fractional energy deposition in the fission-source plate assembly could be determined analytically (see Appendix A), the fission rate of the plate could be estimated. The temperature-indicating system had a rated accuracy of $\pm 0.3^\circ\text{F}$ which yielded a maximum uncertainty in the temperature of $\pm 0.4\%$ in the range of temperatures encountered during the experiment. The electrical power was determined within an accuracy of $\pm 2\%$. Eight electrical power runs were performed ranging from 10.4 to 98.3 watts. Three different

fission runs were made at a reactor power level of 50 kw. Before each run, the fission-source plate was cooled overnight. The temperature data points were automatically recorded at each transducer location every 30 sec. During the electrical runs the power was measured every hour. Each run required about 7 hr.

In order to reduce the effects of ambient temperature changes in the shield test room air, and heat loss effects of the fission-source plate, the assembly was surrounded on all sides by a 2-in. thickness of aircraft-fiberglass insulation. This necessitated moving the fission-plate assembly 2 in. away from the thermal column for both the electrical and the fission-heating runs.

Time-temperature plots of the central resistance temperature transducer for the electrical heating runs are shown in Figure 7. An expanded version of the electrical and the fission runs are shown in Figure 8. The best fit of fission-run data yielded a value of 30.3 watts of heat energy deposited in the fission-source plate. This result is subject to the corrections described below.

2. Attenuation and Geometry Corrections

Since it was necessary to remove the fission-plate assembly from the face of the thermal column about 2 in. to insulate the assembly for the electrical and fission runs, geometrical and radiation attenuation corrections had to be applied to the heating results to obtain a true power-generation rate. The structure that held the fiberglass insulation in place was composed of 1/4-in. plywood. Gold foils were placed on each side of the thermal-column/insulation interface and on the inner side of the insulation. The result of these measurements indicated a gross thermal-neutron flux reduction due to plywood and geometry effects of $1.96 \pm 3\%$, thereby yielding a power dissipation in the plate assembly of $59.4 \pm 3\%$.

3. Fission-Rate Calculation

Calculations were performed to determine what this measured power dissipation means in terms of the actual fission rate of the uranium. To obtain this information the following "non-productive" reactions which deposit energy in the fission-source plate assembly were considered; (1) gamma rays absorbed by the source plate arising from thermal-neutron absorption in the source-plate structure and U^{238} , (2) direct thermal-neutron absorption of reactor neutrons

Figure 7. Temperature vs Time Curves for Fission Plate, STE Reactor

Figure 8. Time vs Temperature for Fission Plate, STE Reactor

that have penetrated the uranium, (3) thermal-neutrons that are reflected by the water tank and reabsorbed by the boral plate, and (4) incident thermal neutrons absorbed by the boral plate that miss the uranium source plate. All of the calculations are based upon thermal neutron measurements made either within the dummy source plate assembly or within the water-tank shield. The energy deposition from the thermal-neutron absorption in the boral plate was obtained from the following reaction:¹²

A summary of these non-productive reactions are presented in Table II.

TABLE II
NON-PRODUCTIVE ENERGY DEPOSITION IN THE STE
FISSION-SOURCE PLATE

Source	Energy Deposition (Watts)
(n, γ) captures	0.184
Neutron absorption in boral:	
(a) uranium penetration	0.013
(b) water reflected	0.352
(c) other*	0.005
	0.554 watts

By subtracting the 0.6 watts of non-productive energy deposition from the 59.4 watts measured during the experiment, a net power of 58.8 watts is deposited by the fission-associated reactions. In order to determine the actual fission rate to which this energy corresponds, the IBM 7090 program GRACE I¹³

*Thermal neutrons from thermal column which miss the circular uranium disk but are absorbed in the square boral of the source plate assembly.

was employed. The data on the energy available per fission were obtained from compilations by Maienschein⁵ and Roos¹⁴ and are presented in Table III. The details of the calculation are given in Appendix A.

TABLE III
ENERGY DEPOSITION IN THE STE FISSION-SOURCE PLATE

Energy Source	Energy Available (Mev/fission)	Energy Disposition in Source Plate (Mev/fission)
Fission fragments	165 ± 2	165 ± 2
Prompt neutron	4.87	0.77
Delayed neutron	0.0068	negligible
Prompt & fission- product gamma rays	12.1 ± 2	9.9 ± 2.6
Fission-product beta rays	8.0 ± 0.7	8.0 ± 0.7
Fission-product neutrons neutrinos	10.5 ± 0.8	negligible
TOTAL	200.48 ± 5	183.67 ± 5

It can be seen from Table III that only 183.7 Mev out of a possible 190 Mev of usable energy per fission are deposited in the fission-source plate assembly. Since there are 3.29×10^{10} fissions per watt on the basis of 190 Mev/fission,⁶ the actual fission rate of the STE source plate is:

$$P = \frac{(58.8 \text{ watts})(190 \text{ Mev/fission})(3.29 \times 10^{10} \text{ fission/sec-watt})}{183.7 \text{ Mev/fission}}$$

$$= 2.00 \times 10^{12} \pm 5\% \text{ fissions/sec} \quad \dots (3)$$

C. FISSION-NEUTRON PRODUCTION-RATE TECHNIQUE

The third technique used to determine the power of the fission-source plate is based upon the fact that the number of neutrons emitted by thermal fission of U^{235} is a well known quantity.¹⁵ The rate of neutron production within the

fission-source plate was estimated by measuring the thermal-neutron flux from the surface of the fission plate opposite the thermal column through an aluminum water tank which served as a temporary radiation shield during the calibration measurements. During the measurements four of the six surfaces of the water tank were covered with 3/8-in. -thick sheets of polyethylene containing 2% natural boron by weight. * The two sides of the water tank which were not covered were the top and the side toward the fission plate. The addition of the borated polyethylene sheet prevented the neutrons which leaked out of the tank as fast neutrons, from re-entering as thermal neutrons and affecting the measurements within the water tank.

1. Water-Tank Thermal-Neutron Flux Integration

In this measurement technique the following assumptions were made: (1) each incremental volume of uranium in the source plate emits neutrons isotropically; (2) the fraction of the fission-energy neutrons interacting or colliding within the uranium can be neglected; and (3) since the source plate is thin compared to the energy of the fission neutrons, half the neutrons are emitted from each of the two faces of the plate.

In the measurement, only neutrons leaving the source plate in the direction of the water tank are considered. Disposition of these neutrons is as follows:

- 1) Some neutrons are scattered or absorbed within the source-plate assembly without ever entering the water-tank shield.
- 2) Some neutrons enter the water tank but are scattered out before they are thermalized.
- 3) Some neutrons enter the water tank and remain long enough to become thermalized, after which they are (a) absorbed by the water in the tank, (b) diffuse into the source-plate assembly and are absorbed by the boron sheet, or (c) diffuse to the edge of the water tank and are absorbed by the borated polyethylene sheet.

Due to the large distance from the top of the water tank to the ceiling of the shield test room, neutron leakage out the top of the tank will be treated

*The polyethylene sheets were obtained from Gilman Bros. Co., Gilman, Conn.

similarly to an absorption in the boral or borated polyethylene as in the case of the other 5 sides of the water tank.

When each of these possibilities is considered, the fission rate of the source plate can be estimated from the following equation:

$$P = 2 \frac{n_1 + n_2}{\nu p} \quad \dots (4)$$

where

n_1 = rate of thermal-neutron absorption in water tank (n/sec)

n_2 = rate of thermal-neutron leakage from water tank (n/sec)

ν = number of neutrons produced per thermal fission = (2.45 ± 0.04)
neutrons/fission¹⁵

p = fraction of neutrons that leave the fission site in the uranium disk
in the direction of the water tank and escape the source plate
assembly

Calculations indicate that the fast-neutron contributions due to neutrons leaving the tank and re-entering the tank were not of sufficient magnitude to be considered in the fission-rate estimation.

The quantity n_1 was obtained by three-dimensional traverses of the water shield tank using gold foil irradiations and BF_3 counter traverses. The foils were standardized by techniques previously described,^{10,11} while the BF_3 traverses were adjusted to the gold foil traverses after BF_3 flux depression corrections had been performed on the data. The thermal neutrons captured in an infinitesimal volume of water was

$$dn_1 = \sum_a \phi_{th} dV \quad \dots (5)$$

where the thermal Σ_a for water was assumed to be 0.022 cm^{-1} and ϕ_{th} is the measured thermal-neutron flux.¹⁶ A plot of the thermal neutrons absorbed in the X-Y plane of the water shield tank as a function of Z is shown in Figure 9. The area under the curve is equal to 1.40×10^{12} n/sec for a reactor power of 50 kw. In order to perform these measurements the STE reactor power level

Figure 9. Thermal Neutron Absorption Rate in the X-Y Plane of Water Shield Tank as a Function of Z

had to be varied from 50 watts to 50 kw to maintain statistical count rates on the BF_3 counters. Similar power variations were necessary to perform the various gold foil irradiations.

The thermal-neutron current leaking into the boral sheet of the fission plate and the borated polyethylene sheet at the other edges of the water shield tank assumed a Fermi angular distribution.¹⁷ The Fermi distribution is

$$f(\theta) \sim \cos \theta + \sqrt{3} \cos^2 \theta \quad \dots (6)$$

where $f(\theta)$ is the angle-differential neutron current (neutrons $\cdot \text{cm}^{-2} \cdot \text{sec}^{-1}$) crossing one square centimeter of the interface and moving in the direction having an angle θ to the normal. The neutron current for a Fermi distribution is related to the integral flux according to the following equation

$$J = \phi_{\text{th}} / \sqrt{3} \quad \dots (7)$$

and the total thermal-neutron leakage out of the tank which is defined as n_2 is obtained by integrating the thermal-neutron current, J , for each face of the water tank shield over surface area of each respective face:

$$n_2 = \sum_{n=1}^6 \phi_{i, \text{th}} d A_i \quad \dots (8)$$

where $\phi_{i, \text{th}}$ is the average thermal-neutron flux measured over each of surfaces of the water tank shield. A value of 0.80×10^{12} neutrons/sec was obtained for n_2 .

The value of p (see Equation 4) is dependent upon the composition of source-plate assembly and the one end of the water tank as well as their respective dimensions. By replacing the gamma-ray cross sections normally used in the GRACE I¹³ code with fast-neutron removal cross sections (which best describes the absorption of neutrons in the source-plate assembly and the end of the water tank facing the fission plate) the fast neutron absorption and scattering due to these sources was calculated to be 15.8%. This yielded a value of 0.842 for p .

2. Fission-Rate Calculation

The values used in the fission-rate calculation for the fission-neutron production-rate technique are shown in Table IV.

TABLE IV
NUMERICAL VALUES USED IN THE NEUTRON-
PRODUCTION TECHNIQUE

Term	Numerical Value	Estimated Error
n_1	1.40×10^{12} n/sec	$\pm 14\%$
n_2	0.80×10^{12} n/sec	$\pm 14\%$
ν	2.45 n/fission	$\pm 1.6\%$
p	0.842	$\pm 2\%$
P	2.13×10^{12} fissions/sec	$\pm 15\%$

The fission rate determined using this technique yielded a value of 2.13×10^{12} fissions/sec $\pm 15\%$. It can be seen that this technique has three times the expected error of the power dissipation method, 50% greater than that estimated by the neutron absorption-rate technique. It should also be mentioned that this calibration method consumed twice as many man hours as the other two techniques combined. Therefore, it is not recommended as a measurement technique in future power calibrations.

D. SUMMARY OF RESULTS

The STE fission-plate power calibrations are summarized below:

- 1) The first technique, thermal-neutron absorption in the fission-plate assembly, yielded a value of 1.79×10^{12} fissions/sec $\pm 11\%$.
- 2) The second technique, power dissipation in the fission-plate assembly, yielded a value of 2.00×10^{12} fissions/sec $\pm 5\%$.
- 3) The third technique, fission-neutron production in the fission-plate assembly, yielded a value of $2.13 \times 10^{12} \pm 15\%$.

A weighted average of these three values, with the weighing function proportional to the inverse of the squares of the estimated errors, yielded a value of 1.97×10^{12} fissions/sec $\pm 4\%$. With an assumed usable energy of 190 Mev/fission, this fission rate corresponds to a fission-plate power of 59.9 watts $\pm 4\%$. The source plate has an area of 2919 cm^2 , thereby yielding an average power per unit area of 2.05×10^{-2} watts/cm² $\pm 4\%$. This is approximately 14 times the specific power of the ORNL Lid Tank Facility fission plate. It is, however, one-sixth that which was originally calculated by Clark⁷ based upon a modified one-group diffusion theory. The most likely reason for this discrepancy is the possibility of a small high cross-section impurity in either the bismuth window or the graphite column. A calculation has shown that a few parts per million cadmium impurity could in itself account for the discrepancy between the calculated and the measured STE fission-source-plate power estimates.

Experimental measurements obtained on the thermal columns of the water-boiler reactors fabricated by Atomics International for the Japanese AEC and Armour Research Institute of Chicago, Illinois were used to "adjust" the flux calculations. The difference by a factor of six between the measurements and the calculation is outside the range of expected error in the calculation. For this reason, future flux mapping of the STE reactor core, bismuth window, and graphite thermal column are planned in order to resolve this discrepancy.

V. APPENDIX

FISSION-ENERGY ABSORPTION CALCULATIONS

The "productive" energy dissipation in the STE fission-source plate, i. e., the fraction of the available fission energy that is deposited in the STE fission plate in the form of heat, was determined by using measured thermal-neutron flux data and the GRACE I¹³ radiation-heating code programmed for the IBM 7090 computer. The energy available per fission which is shown in Table III was used in the calculations. Fission fragments and fission-product beta particles were assumed to be completely absorbed within the source plate while the neutrino energy was assumed to escape absorption entirely. Spectra for the prompt and fission-prompt gamma rays were obtained from a summation by Roos.¹⁴ Non-productive (n, γ) reaction spectra were obtained from the GRACE I library data. The thermal-neutron data used were obtained from experimental measurements previously described in Sections IV-A and C of this report.

GRACE I is designed for calculating heat-generation rates along the axis of infinite-plane slab shields, i. e., it is a one-dimensional code. The heating rates are expressed in watts/gm of shield material, and calculations are performed for the number of mesh points and materials desired by the analyst. A typical plot of the heat deposition along the centerline of the uranium fission-source-plate due to prompt and fission-product gamma rays is shown in Figure 10. Similar curves were obtained for the (n, γ) reactions in the uranium and aluminum support structure as well as prompt and fission-product gamma-ray energy deposition in the aluminum.

The best values of the energy deposited in the source plate (quoted in Tables II and III) were obtained by geometrically weighting the centerline heating-rate values according to the angular penetration thickness which is proportional to $\sec \theta$ as measured from the axial centerline of the fission plate.

The energy deposition of the fast neutrons generated as part of the fission process assumed a generation rate as a function of uranium thickness similar to that shown in Figure 5. The neutron energy spectra were obtained from a summation by Goldstein.⁶ Absorption and inelastic scattering cross sections¹⁵ were used to determine the amount of energy deposited by these neutrons in the

Figure 10. Energy Deposition of Prompt Fission Gamma Rays in the Uranium Region of the Source Plate

source-plate assembly. Again, GRACE I was used in performing the actual calculation. Because the delayed neutron contribution was so small compared to other sources of radiation, it was considered negligible compared to the errors expected in the final results.

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