AN ANALYSIS OF THE AMOUNT AND TYPES OF NEUTRON
INDUCED ACTIVITY IN VARIOUS SOILS AND ROAD BUILDING MATERIALS

by

Oliver R. Dinsmore, Jr.

A Thesis Submitted to the Faculty of the
DEPARTMENT OF NUCLEAR ENGINEERING
In Partial Fulfillment of the Requirements
For the Degree of
MASTER OF SCIENCE
In the Graduate College
THE UNIVERSITY OF ARIZONA

1961
STATEMENT BY AUTHOR

This thesis has been submitted in partial fulfillment of requirements for an advanced degree at the University of Arizona and is deposited in the University Library to be made available to borrowers under rules of the Library.

Brief quotations from this thesis are allowable without special permission, provided that accurate acknowledgment of source is made. Requests for permission for extended quotation from or reproduction of this manuscript in whole or in part may be granted by the head of the major department or the Dean of the Graduate College when in their judgment the proposed use of the material is in the interests of scholarship. In all other instances, however, permission must be obtained from the author.

SIGNED: [Signature]

APPROVAL BY THESIS DIRECTOR

This thesis has been approved on the date shown below.

Keaton K. Keller
KEATON K. KELLER
Professor of Nuclear Engineering

Jan 23, 1961
Date
ABSTRACT

AN ANALYSIS OF THE AMOUNT AND TYPES OF NEUTRON INDUCED ACTIVITY IN VARIOUS SOILS AND ROAD BUILDING MATERIALS

by

Oliver R. Dinsmore, Jr.

In the design of mobile nuclear reactor power plants, the weight of the reactor and associated power conversion equipment will be a major problem. The use of directional shielding on such a plant for weight reduction was investigated in this paper. A direct problem associated with a minimum bottom shielded reactor is the neutron induced activity in the surface of the earth below. This gamma activity would be a serious problem for unprotected personnel entering the irradiated area after passage of the mobile reactor.

A series of samples of various soils and road building materials were irradiated in the University TRIGA reactor, and the resulting activity was measured. By developing scaling and geometry factors, these activities were converted to the expected activities which would be induced in specific types of ground surfaces by the mobile plant. Curves and graphs were developed for use of planning personnel in determining the feasibility of such directional shielding and to determine radiation dosages to be expected by personnel entering an area after the passage of such a reactor.
ACKNOWLEDGMENT

The assistance rendered to the author by Dr. Keaton K. Keller as thesis advisor is greatly appreciated.

The cooperation and advice given by Professor Marle H. Wittmeyer and Mr. Thomas W. Fern, as reactor operator, are acknowledged.

Thanks are extended to the personnel of the Departments of Mining Engineering and Civil Engineering for their assistance in the preparation of experimental samples.
# TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>Chapter</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>INTRODUCTION</td>
</tr>
<tr>
<td>2</td>
<td>PLANT GEOMETRY</td>
</tr>
<tr>
<td>3</td>
<td>SCALING FACTOR</td>
</tr>
<tr>
<td>4</td>
<td>CONDUCT OF THE EXPERIMENT</td>
</tr>
<tr>
<td>5</td>
<td>SAMPLE DATA</td>
</tr>
<tr>
<td>6</td>
<td>EXPERIMENTAL DATA</td>
</tr>
<tr>
<td>7</td>
<td>PERMISSABLE DOSE RATES</td>
</tr>
<tr>
<td>8</td>
<td>DEVELOPMENT OF DOSE RATES</td>
</tr>
<tr>
<td>9</td>
<td>CONCLUSIONS</td>
</tr>
</tbody>
</table>

**ANNEX**

| I       | DEFINITIONS OF TERMS    | 50   |
| II      | SAMPLE SPECTRA          | 53   |
| III     | TRIGA REACTOR           | 54   |
| IV      | COPY OF REVISED AEC REGULATIONS | 58   |

**BIBLIOGRAPHY** 62
CHAPTER 1
INTRODUCTION

Much study has been conducted by the Atomic Energy Commission and the Armed Services regarding neutron induced activity in various types of soil, equipment and other material. The source of these neutrons has been from the detonation of nuclear weapons and devices, and more recently from various prototype and test reactors. The emphasis on the soil studies so far completed and in unclassified form has not been on the induced activity, but instead on the shielding effect of such materials, with the induced activity study as a by-product of the work.

It is the purpose of this paper to discuss and analyze the subject of artificially induced radioactivity from a continuous neutron source, such as a reactor. For instance, a vehicle mounted reactor will be used to illustrate the activation of the roadbed over which the vehicle will operate. This type of neutron source would subject the roadbed or ground surface beneath the reactor to lower fluxes than a nuclear weapon detonation, but in many instances the duration of the neutron bombardment would be many times that of the weapon generated neutrons. Times of the order of several seconds to hours of irradiation would result if such a vehicle were stopped and the reactor kept operating to supply power for necessary housekeeping, maintenance, loading, and communications operations.

The various agencies conducting reactor design for mobile plants
have stated that many problems will exist in the development of the reactor core with its associated moderator, reflector, shielding and power conversion equipment. Two of the most important are the size and weight of the complete reactor power plant. Vehicles over eight feet in width require special bridge and road clearances at the present time. If such a vehicle could be made to conform to current road and bridge limitations, it would have maximum mobility. Keeping the size of the reactor core within a reasonable range is possible, but to reduce the shielding size and weight for such a plant is extremely difficult. One possible solution to the weight problem will be discussed herein.

Based upon this weight problem, the use of a directionally shielded reactor will be investigated. Shielding will be so placed that a substantial neutron flux will be directed into the ground surface below the vehicle. By thus shielding the reactor, a considerable weight saving would be realized. The discussion will be restricted to a possible vehicle design to be used by the U. S. Army as a logistical cargo carrier. However, the conclusions and recommendations would apply to any power plant having similar design and operating characteristics; i.e., locomotive, tank, or mobile power station, where weight and size of plant are primary considerations for mobility.

In order to develop a set of dimensions and numbers on which to base the calculations, a hypothetical vehicle has been developed to give sizes and weights and operating powers which will be appropriate.

The vehicle is to be an overland train consisting of an operations and control car, ten 15-ton cargo cars and the power car. The power car will be located on the rear of the train with the control car at the
front. See Fig. 1.1. Each car of the train is self-propelled by a fifty horsepower electric motor located in each of its four pneumatic tired wheels, similar to the power arrangement now used on some large pieces of earth moving equipment. Wagon steer on the cars makes them track directly behind the car ahead. The total delivered motive power for propulsion of the train alone would be approximately 2500 HP (50 wheels at 50 HP each) or 1870 kW. Using a power conversion efficiency of 25%, the reactor would have to produce approximately 7500 KW (thermal) for motive power alone.

With the above arrangement of cars within the train, the danger to the crew from direct and indirect irradiation will be a minimum when in normal over the road operation; however, such a directionally shielded reactor could be a potential danger to repair and maintenance personnel approaching from the sides of the vehicle. In addition to the direct neutron and gamma irradiation from the operating reactor, the induced gamma activity in the ground surface could be a hazard to unprotected and uninformed personnel entering the area previously swept by the neutron stream emitted from the opening in the bottom of the reactor vessel as the vehicle moved over the ground. The determination as to the actual danger to personnel from the induced activity is the major objective of this paper.

The approach to the problem outlined above has been an experimental one in the main. Calculations are used to substantiate the results where possible and to cover those areas for which experimental results could not be obtained. The TRIGA reactor at the University of Arizona was used to irradiate the various samples of soils and road
building materials, such as Portland cement concrete and asphalt concrete. Neutron induced activity was then measured immediately after irradiation and at various time intervals thereafter to determine the duration and intensity of the radiation from the isotopes formed. Readings of dose rates were taken with a standard cutie-pie meter in milliroentgens per hour. Annex I contains definitions of terminology used in this paper. In order to determine the approximate strength of the gamma radiations being emitted and to possibly identify the isotopes, an analysis was run on the 200 channel analyzer in the nuclear engineering department. See Annex II.

Data so gathered was then translated, using the proper scaling and geometry factors onto graphs and charts, which can be used as a reference guide for personnel to determine the amounts of radiation to be expected after exposure of the ground surface to specific neutron fluxes for given periods of time. Dangerous dose rates have been indicated on the graphs. Scatter and absorption of neutrons in the air gap between the reactor and the ground surface are not considered in the study as they are not too significant with the particular distances chosen.

Conclusions and recommendations have been developed concerning the feasibility of this type of directional shielding on mobile reactor installations and the possible hazards resulting therefrom.
The principal types of radiation associated with nuclear reactors are Betas, Gammas and Neutrons. Fission fragments can be very dangerous because of their size; however, these seldom travel far within the reactor. Beta particles can be stopped by a few thousandths of an inch of solid material. High energy or hard gammas, on the other hand are very penetrating and may require many inches of lead to stop them. High energy or fast neutrons are much more massive particles, which, strangely enough, can be slowed down and stopped effectively by the use of light materials. If we consider a typical fission type reactor, we find that approximately two and one-half fast neutrons and about a dozen hard gammas are released per fission. There are approximately \(7 \times 10^{10}\) neutrons per second emitted per watt of power and approximately \(3 \times 10^{11}\) gammas per second emitted per watt. Roughly, two-thirds of the neutrons and gammas are absorbed in the reactor core, and the balance escape into the shield.

For the purpose of this discussion, a specific geometry was chosen in order to permit power and flux calculations. The reactor has a spherical shape to provide the largest volume with the least surface area. The maximum outside diameter of the shield is eight feet (8') in order to keep the plant to reasonable size for highway operations. The core is three feet (3') in diameter surrounded by a one foot (1') reflector and a one and one-half foot (1.5') shield. The shield has been
left off of the bottom ten percent of the reflector surface, leaving a square hole in the shield of approximately three feet (3') on a side. Figure 2.1 shows the geometry as developed. The ground clearance of the reactor is about four feet (4'), and the irradiated area to be considered is a square of approximately thirty-six (36) square feet.

Using the geometry described above, the following calculations show the neutron flux which would be passing through the hole and striking the ground surface below:

Total surface area of reflector is

$$A = \pi d^2 = \pi \times 5 \times 5 = 78.5 \text{ sq. ft.}$$

Area of hole in shield at reflector boundary is

$$A = 2\pi rh$$

$$h = .5'$$

$$A = 2\pi(2.5')(.5') = 7.85 \text{ sq. ft.}$$

Ratio of size of hole to total outside area of reflector is

$$\frac{7.85}{78.5} = .10 \text{ or } 10\%.$$ 

Power calculations are based upon the following:

(a) 3 x 10^{10} fissions per watt.

(b) 7 x 10^{10} neutrons per second per watt.

(c) Approximately one-third of the neutrons born within the core reach the shield.
FIGURE 2.1
REACTOR GEOMETRY
(section through center)

GROUND CLEARANCE
\sim 4'

1 e^{3/4''} = 1'

ACTIVATED AREA
6' x 6' = 36 sq ft

GROUND SURFACE
For maximum power of 7500 kW (thermal), the number of neutrons reaching the shield is

\[
\frac{7 \times 10^{10} \times 7500 \text{ kW} \times 10^3}{3} = 1.75 \times 10^{17} \text{ neutrons/sec.}
\]

So at maximum power, the number of neutrons passing out through the hole in the shield will be

\[
1.0 \times 1.75 \times 10^{17} = 1.75 \times 10^{16} \text{ neutrons/sec.}
\]

At the ground surface, assuming that the surface is uniformly irradiated and that no air scattering of neutrons occurs, the neutron flux at the ground is

\[
\frac{1.75 \times 10^{16}}{(6)^2(12 \times 2.5 \text{ cm})^2} = \frac{1.75 \times 10^{16}}{3.35 \times 10^4} = 5.22 \times 10^{11} \text{ neut/cm}^2/\text{sec}
\]

The TRIGA reactor produces a flux of $10^{11}$ neutrons per cm$^2$ per sec in the lazy susan sample area when operating at a power of 10 kW. Therefore, the flux from our hypothetical plant operating at 7500 kW is approximately

\[
\frac{5.22 \times 10^{11}}{2.5 \times 10^{11}} = 2.1
\]

or 2.1 times as great as that produced in the TRIGA when operating at 25 kW.

Sample activities will, therefore, be scaled upward or downward by an appropriate factor depending upon the power selected.
CHAPTER 3

SCALING FACTOR

As shown in Fig. 2.1, the ground irradiated area to be considered is approximately 36 sq. ft. It is known that neutron scattering in the four feet of air between the reactor reflector and the ground surface will cause a larger area to receive a substantial neutron dose; however, the surface directly below the opening will receive the highest dosages and consequently will have the greatest induced activity. As an approximation, only the 36 sq. ft. of surface area has been considered, and this to receive a uniform dose per unit area, and all of the neutrons leaving the reflector are assumed to reach this surface.

It was necessary to develop a scaling factor to relate the readings taken by the cutie pie meter from the small samples irradiated in the TRIGA to the actual readings that would be received from a road surface of similar material which had been irradiated by the mobile reactor operating at some specified power level.

When the vehicle is standing still with the reactor operating, the 36 sq. ft. area on the ground is considered to be irradiated with a uniform dose per unit area. Second, if the vehicle is moving at some given rate of speed, a path 6' wide is swept by the neutron stream emitted from the hole in the reactor shield.

Considering first the case with the vehicle remaining stationary. It has been assumed above that (1) the ground surface is uniformly
irradiated, and (2) all the neutrons passing out through the hole in the shield reach the ground.

It is noted that in order to simplify the geometry calculations regarding the induced activity in the ground surface, a square area one cm. thick or deep was considered to be irradiated by neutrons passing out through the square hole; this results in uniform ground surface irradiation. Then, to develop a scaling factor to relate the sample measurement to the actual condition, that is, relate the sample reading to that which would be given on the cutie pie meter held about two feet above the earth surface, the gamma activity was considered to emanate from a circle of 6.8\textsuperscript{1} diameter. This circle has approximately the same area as the 6\textsuperscript{1} square we have been referring to.

It is realized that the above assumptions are only an approximation of the actual conditions which would exist; however, it is believed that they are valid and constitute a method of solution. Actual experimentation with a prototype reactor would be required to substantiate the data developed.

Following are the geometry calculations which have been developed to give the approximate meter reading from a uniformly irradiated 1 cm. deep surface 6.8\textsuperscript{1} in diameter. The meter is considered to be held two feet above the ground surface.

Using the geometry shown in Fig. 3, a determination was made of the scaling factor to relate gamma emission readings from TRIGA irradiated samples to expected readings from ground surface irradiated by the mobile plant.
FIGURE 3.1

**K** - Source strength in Mr/hr/CC

**D** - Perpendicular distance to ground surface from center of sensitive volume of C.P. detector.

**R** - Radius of neutron irradiated surface

CENTER OF CUT TIE-PIE METER SENSITIVE VOLUME
Expected meter reading (predicted) = Dose = mr/hr

\[ \text{Dose (mr/hr)} = K \int_{\text{area}} \frac{1}{\rho^2} \, d\rho \]

\[ d\rho = rdrd\theta \]

\[ \rho = \sqrt{D^2 + r^2} \]

\[ K \int_{0}^{R} \int_{0}^{2} \frac{1}{D^2 + r^2} \, rdrd\theta \]

Integrating first with respect to \( \theta \) gives \( 2\pi \) so

\[ 2\pi K \int_{0}^{R} \frac{1}{D^2 + r^2} \, rdr = \text{Dose} \]

using \( \int \frac{du}{u} = \ln |u| + C \)

\[ \pi K \int_{0}^{R} \frac{2rdr}{D^2 + r^2} = \pi K \ln \left| \frac{D^2 + r^2}{D^2} \right|_{0}^{R} \]

\[ = \pi K \left[ \ln(D^2 - R^2) - \ln D^2 \right] \]

\[ = \pi K \ln \frac{D^2 - R^2}{D^2} \]

For our calculations:

\[ R = 3.4 \times 12^2 \times 2.54 \text{ cm} = 103.7 \text{ cm} \]
\[ D = 2^3 \times 12^2 \times 2.54 \text{ cm} = 61 \text{ cm} \]
Substituting in above:

\[
\text{Dose} = \nu K \left[ \ln(14.47 \times 10^3) - \ln(37.2 \times 10^2) \right]
\]

\[= \nu K \ln 3.89 \]

\[= \nu K (1.35341) \]

\[
\text{Dose} = 4.27 K
\]

For the case when the vehicle is moving, the same approximation is used. The K factor will vary depending upon the reactor power and speed of movement of the vehicle over the ground. It is realized that the cutie pie meter will receive gammas which originate outside the 6.8' circle along the irradiated path below the reactor shield opening; however, the $1/d^2$ rule rapidly reduces the intensity of the gamma beam reaching the instrument. As previously stated, solution to this problem requires complicated mathematical manipulation to express the complete situation with calculations and definitely requires additional experimental confirmation.

As the sample capsules being used for irradiation in the TRIGA are approximately 1 cm x 1 cm with a volume of about 3.14 cm³, the sample readings will be divided by three to give approximate K values in mR/hr/cm³.

As a check to the geometry calculations, the following experimental check was made. An irradiated sample was placed on the floor and the cutie pie meter was held two feet above the sample. The reading was taken in this position, and then the meter was lowered until it touched the sample. Readings were taken in this position. According to the $1/d^2$ rule, the reading at 2 ft from the center of the sensitive volume
should be \( \frac{2.5 \times 2.5}{2.4 \times 2.4} = 1.085 \times 10^{-2} \), or 1/90 of the reading at 2.5\( ^{\circ} \) (the closest distance to the center of the sensitive volume). Data regarding the readings from the experiment is shown below:

Sample irradiated 8 min at 25 KW \( (2.5 \times 10^{11} \text{ neut/cm}^2/\text{sec}) \)

<table>
<thead>
<tr>
<th>Time</th>
<th>Dist. from sample to chamber</th>
<th>Reading (mr/hr)</th>
<th>Ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>130h</td>
<td>2.5( ^{\circ} )</td>
<td>1750</td>
<td>( \frac{25}{1750} = 1/70 )</td>
</tr>
<tr>
<td></td>
<td>2.4.0( ^{\circ} )</td>
<td>25</td>
<td></td>
</tr>
<tr>
<td>131h</td>
<td>2.5( ^{\circ} )</td>
<td>100</td>
<td>( \frac{2}{100} = 1/50 )</td>
</tr>
<tr>
<td></td>
<td>2.4.0( ^{\circ} )</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

It is noted that the two readings had ratios of 1/70 and 1/50 as compared to the 1/90 calculated. It is believed, however, that these indicate that the geometry calculations are fairly good, especially with the large number of variables involved in such measurements: internal absorption in sample, air scattering of gammas, inaccuracies in distances, and meter readings.

The G.P.o or cutie pie meter was chosen to make the measurements in this experimental work as it is a common type of meter used in radiation monitoring. It consists of an air filled ionization chamber made of Bakelite, which is approximately air equivalent.

The dose rate, proportional to the ion current, is indicated directly in mr/hr on a familiar type of pointer micro-ammeter. Its range is 5-10,000 mr/hr. Its accuracy is \( \pm 10\% \), and it is robust, of light weight and may be carried and read easily.
CHAPTER 4

CONDUCT OF THE EXPERIMENT

The experimental irradiation of the samples was accomplished using the TRIGA reactor at the University of Arizona. A description of the reactor is contained in Annex III.

Samples of the various materials to be irradiated were placed in plastic vials approximately 1 cm in diameter by 4 cm long. Vials were sealed with a plastic press fit cap. These vials were then carefully washed to remove all traces of foreign material and then placed in larger plastic tubes with screw fit tops. The large tubes were then lowered into the lazy susan area of the reactor for exposure to a specified neutron flux.

Samples were exposed for periods up to 15 minutes in fluxes to $2.5 \times 10^{11}$ neutrons/cm²/sec. Upon removal from the reactor test cells, readings were taken with the cutie pie meter by placing the sample directly against the end of the sensitive volume of the meter. Gamma and Gamma plus Beta doses were read, but only the gamma readings were considered in this work.

By applying the geometry and scaling factors developed previously, it was possible to predict the meter readings from an actual ground surface being irradiated by a mobile reactor having the approximate characteristics described. Use of these graphs and charts is covered in Chapter 8.
CHAPTER 5
SAMPLE DATA

A total of ten different soil and road surface samples were tested. Five were samples of road surfacing materials consisting of Portland cement and asphalt concrete, and five were naturally occurring soils. These materials are representative of the Southwest and the Northwest United States areas. As the hypothetical logistical carrier will be capable of cross country operation, as well as movement over surfaced roadways, it was necessary to consider both types of materials. It was realized that in most cases the natural occurring aggregates used in the manufacture of the concretes would contain the same materials as the natural soils found in the area. Samples of ocean beach sand were tested because the vehicle might be unloaded from a ship over the beach. The beach sand contains salts and other minerals from the salt water, which produces some very active isotopes.

The samples are described below, and a general description of the chemical makeup of each is discussed in general. No chemical analysis of the samples tested was made. The amount of induced activity was the important consideration; the chemical content of the material was less important. General data regarding the samples is from handbooks and test reports from the AEBC.

17
In general, Portland Cement has the following materials included in varying quantities:

<table>
<thead>
<tr>
<th>Material</th>
<th>Symbol</th>
<th>Percent</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lime</td>
<td>CaO</td>
<td>60-65</td>
</tr>
<tr>
<td>Silica</td>
<td>SiO₂</td>
<td>20-25</td>
</tr>
<tr>
<td>Alumina</td>
<td>Al₂O₃</td>
<td>5-10</td>
</tr>
<tr>
<td>Magnesium Oxide</td>
<td>MgO</td>
<td>0-4</td>
</tr>
<tr>
<td>Ferric Oxide</td>
<td>Fe₂O₃</td>
<td>0-3</td>
</tr>
<tr>
<td>Sulphur Trioxide</td>
<td>SO₃</td>
<td>1-2</td>
</tr>
<tr>
<td>Sodium Oxide</td>
<td>Na₂O</td>
<td>--</td>
</tr>
<tr>
<td>Potassium Oxide</td>
<td>K₂O</td>
<td>--</td>
</tr>
</tbody>
</table>

The aggregates used in forming the Portland Cement concrete are those normally found near the construction site and are usually one or
a combination of the following materials: Natural — gravel, limestone, granite, quartzite, traprock; Artificial — cinders, slag, haydite or tufa.

The asphalt concrete has the same aggregates as those listed above, mixed with varying amounts and types of asphalt.

Graphs of the gamma spectrums produced by several of the samples are included for information in Annex II. The approximate energies of the activity peaks are indicated. These were determined from comparison with known samples.

The asphalt itself is made up of hydrocarbons and only the small quantities of impurities, such as sulphur, would produce important isotopes under neutron bombardment.

Isotopes which have been identified in tests previously conducted using sand, clay, limestone, and mixtures of these materials consist of

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Half Life (Hrs.)</th>
<th>Gamma Energy (Mev)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mn$^{56}$</td>
<td>2.60</td>
<td>2.130, 1.310</td>
</tr>
<tr>
<td>Na$^{24}$</td>
<td>15.00</td>
<td>2.754, 1.380</td>
</tr>
<tr>
<td>Pr$^{142}$</td>
<td>19.20</td>
<td>1.572</td>
</tr>
<tr>
<td>Si$^{31}$</td>
<td>2.65</td>
<td>1.260</td>
</tr>
<tr>
<td>Sb$^{122}$</td>
<td>67.20</td>
<td>0.56, 0.7, 1.2</td>
</tr>
<tr>
<td>Fe$^{59}$</td>
<td>1080.00</td>
<td>1.289, 1.098</td>
</tr>
<tr>
<td>Sc$^{46}$</td>
<td>2040.00</td>
<td>1.119, 0.885</td>
</tr>
<tr>
<td>Lu$^{177}$</td>
<td>160.80</td>
<td>0.206</td>
</tr>
<tr>
<td>Ca$^{45}$</td>
<td>3940.00</td>
<td>No Gamma</td>
</tr>
<tr>
<td>Cr$^{51}$</td>
<td>667.20</td>
<td>0.325</td>
</tr>
</tbody>
</table>
From the AEC tests in which the above isotopes were identified, only Na$^{24}$, Fe$^{59}$, Ca$^{45}$ could be identified from chemical analysis of the test samples. The remainder were identified from activation analysis alone.

Using the general analysis of the Portland Cement previously given, there is a possibility of the following listed isotopes being formed under neutron irradiation: Si$^{31}$, Al$^{28}$, Mg$^{27}$, Fe$^{59}$, Na$^{24}$ and K$^{42}$. During the experimental work for this paper, no attempt has been made to identify the specific isotopes formed, as the measurement of the total induced activity was the important result. The total gamma dose received by personnel entering the irradiated area was desired.
CHAPTER 6

EXPERIMENTAL DATA

The raw data developed from the experimental work using the TRIGA reactor and counting equipment in the Nuclear Engineering Department have been reduced to graphical form and are included in the following pages. Data shown on the graphs has been converted to one (1) cc samples irradiated at 25 kW for ten minutes as a standard.

Included are the area alpha meter readings taken at various times after irradiation. The geometry and scaling factors will be applied to these readings to convert them to the predicted readings for the large reactor.
TIME AFTER EXPOSURE - FRS

SAMPLE #2
DESSERT SAND (TUCSON, ARIZONA)

PICTURE 6.2
FIGURE 6.3

PLOT: SAMPLE # 3 - PORTLAND CEMENT CONCRETE (ARIZONA)

METER READING - MR/HR/CC
Figure 6.14

Sample #4 - Portland Cement Concrete (Arizona)
METER READING: MR/HR/CC

TIME AFTER EXPOSURE - HRS

SAMPLE #5 - ASPHALT CONCRETE (ARIZONA)
METER READING: NR/HR/CL

FIGURE 6.6

SAMPLE #6 - BEACH SAND (PUGET SOUND)
SAMPLE #7  -  BEACH SAND (CALIFORNIA)
METER READING - MR/HR/CC

SAMPLE #9 - PORTLAND CEMENT CONCRETE (WASHINGTON)
METER READING - MR/HR/CC

SAMPLE #10 - ASPHALT CONCRETE (WASHINGTON)
CHAPTER 7
PERMISSABLE DOSE RATES

Much work and experimentation has been conducted to determine what are the permissible dose rates of radiation which can be received by personnel and the effects of same.

This paper does not attempt to discuss the validity of these rates. The graphs have included the various rates so that persons using them may select a desired rate on which to enter the graphs. Various rates of dosage now used by the AEC, and the military will be included. It has been stated that a total lifetime dose of 25r ought not to be exceeded for planned operations, and that the LD-50 (i.e., the lethal dose for 50% of the exposed personnel) is approximately 400r, if the dose is given in a single relatively short period of time. It has also been stated that if the dose is spread out over a period of weeks or years, then a total dose of perhaps twice or three times as great can be absorbed. Unfortunately, many deleterious effects of radiation appear after radiation doses of much less than the LD-50 dose. These effects include superficial burns, eye cataracts, leukemia, sterility, genetic effects, and shortening of the normal life span.

The LD-50 (400r), 25r, and the latest published AEC dose rates of 300 mr/week are indicated where possible on the graphs. The user will then be able to select the dose rate which is of particular interest to him and examine the data from this viewpoint.
The latest published AEC permissible dose rates are shown in Annex IV.
CHAPTER 8
DEVELOPMENT OF DOSE RATES

This portion of the paper applies the geometry and scaling factors developed in Chapters 2 and 3 to the experimental data as given in Chapter 6 to develop curves and charts which may be used to predict dose rates of radiation which might be expected.

Figure 8.1 is used to determine the geometry factor between the TRIGA samples irradiated for 10 minutes at 25 KW and the large mobile reactor operating between 10 and 10,000 KW.

Figures 8.2 through 8.11 are the TRIGA sample readings from Chapter 6, multiplied by the scaling factor of $k_{27}$ developed in Chapter 3. This is the reading to be expected on the cutie pie meter held two feet above the ground from a 6.8' diameter circle which has been uniformly irradiated to a depth of 1 cm. Each cubic centimeter of the surface was exposed to the same irradiation as the one cc test sample.

In order to determine the expected dose rate at a given time after passage of the land train over a specific type of terrain, the following procedure is used in conjunction with the above listed curves:

A. Select a mobile reactor power from Fig. 8.1. Divide flux at the ground surface as indicated from the figure by $2.5 \times 10^{11}$ (TRIGA flux at 25 KW) to obtain the geometry factor.
FIGURE 8.2

PREDICTED CURVE PIE METIE READINGS

MR/HR

TIME AFTER EXPOSURE HAS

SAMPLE #1 DESERT SOIL (ARIZONA)
PREDICTED CUTIE PIE METER READINGS

SAMPLE #2 DESERT SAND (ARIZONA)
PREDICTED CURVE: PIEMETER READINGS

MR/HR

TIME AFTER EXPOSURE:

SAMPLE #3 PORTLAND CEMENT CONCRETE (ARIZONA)
PREDICTED CUT IN PIE METER READINGS

MR/HR

FIGURE 8-5

SAMPLE #4 - PORTLAND CEMENT CONCRETE (ARIZONA)
FIGURE 8.6

PREDICTED CUM-DIE METER READINGS

MR/HR

SAMPLE #5 - ASPHALT CONCRETE (ARIZONA)
FIGURE 8.7

PREDICTED CUTOFF PIE METER READINGS

MR/HR

TIME AFTER EXPOSURE - HOURS

SAMPLE #6  BEACH SAND (POUGET SOUND)
FIGURE 8.8

SAMPLE 17  BEACH SAND  (CALIFORNIA)
SAMPLE #8 - DESERT SAND (YUMA, ARIZONA)
B. Go to the curve of the ground surface material selected. See Figs. 8.2 through 8.11.

C. Select time after irradiation on ordinate of graph and enter the graph stopping at the curve representing the time of irradiation or speed of vehicle.

D. Read predicted dose in mr/hr on abscissa. Multiply this reading by the geometry factor in A above for the final predicted reading in mr/hr.

As an example, consider the determination of the predicted dose rate thirty minutes after the reactor has moved on from an area which was subjected to ten minutes of irradiation from the reactor operating at 3500 KW. The soil material is Washington beach sand (See Sample 6). Following the procedure given above,

A. Reactor power is 3500 KW; therefore, the flux at the ground surface from Fig. 8.1 is \(2.5 \times 10^{11}\). Divide this by \(2.5 \times 10^{11}\) to get a geometry factor of .96.

B. Go to Fig. 8.7 (Sample 6), read up the ordinate to thirty minutes (.5 hr.) after exposure, and enter the graph proceeding along the abscissa until you intersect the ten minute irradiation curve.

C. Read the dose rate of 185 mr/hr from abscissa.

D. Multiply this dose rate by the geometry factor of .96 to get a final predicted reading of 177 mr/hr.

Curves may then be plotted for varying speeds and reactor powers to determine the levels of activity which will be built up in various types of road surfaces and soils.
CHAPTER 9

CONCLUSIONS

The purpose of this paper has been to investigate and analyze the amount and types of neutron induced activity in soils and road building materials. The means of analysis was to irradiate the materials of interest in the University of Arizona TRIGA reactor and measure the gamma activity present after various times and amounts of neutron activation.

In order to present a practical approach to the problem, a hypothetical reactor powered land train was developed for the purpose of establishing realistic numbers and sizes for the system.

To establish a means of irradiating the ground surface below the operating reactor and a possible weight reduction in the reactor shielding, a directionally shielded reactor was used. A hole was left in the shielding covering the bottom of the reactor reflector, and a percentage of the neutrons born within the core was allowed to escape and penetrate the ground surface below. The gamma activity given off from the decay of the radioactive isotopes formed was then estimated.

Based upon the above hypothetical vehicle and the experimentation with soil and road building materials, and the combination of the resulting data with appropriate scaling and geometry factors, the following conclusions are presented:
A. The directional shielded reactor will produce relatively high levels of induced activity in the ground surface, especially if the vehicle is standing still for periods of over five minutes and the reactor is operating at high powers.

B. The gamma activity from the radioactive isotopes formed is relatively short lived. The major activity has half lives ranging from 15 to 43 hours. This activity is caused by the decay of single isotopes or combinations of several isotopes.

C. As the isotopes have relatively short half-lives, the irradiated areas will be safe for unprotected personnel in about five minutes after passage of the vehicle which is traveling at speeds of one mile per hour or greater. If the vehicle is standing still for periods of ten minutes or longer, the area should be surveyed before entry of personnel. For example, the AEC recommended dose of 300 mr/week and the planned operation dose of 25r are shown plotted on Figures 8.2 through 8.11 in the preceding chapter. This plot is only valid for a scaling factor of one (1), as the curves would move to the right or left depending upon the reactor power. A scaling factor of one indicates a reactor power of 3600 KW. Both doses have been converted to a forty hour basis and show on the graphs as 75 mr/hr and 625 mr/hr for the AEC and planned operation doses, respectively.
D. The radioactive isotopes formed from the ten samples tested have characteristic curves with the major variances occurring in the period up to one hour after irradiation. From one to one hundred hours after the irradiation, all curves are similar; however, they are displaced right or left depending on their location at the one hour period.

E. The characteristic knee, which is most pronounced in samples 3, 5, 6, 7, 9, and 10 at approximately 38 minutes (0.63 hr.) after irradiation is probably caused by the build-up of relatively energetic daughter elements. This situation can cause a temporary increase in the gammas reaching the survey meter.

Further experimentation with prototype reactors and varying types of directional shielding should be conducted to determine if the weight savings indicated as possible herein are structurally and design-wise feasible.

The induced activity in the normal types of soils and road building materials over which such a vehicle would operate will not pose a problem as far as dosages to be expected by exposed personnel. However, the direct irradiation of personnel by neutrons and gammas escaping through the hole in the shielding is a serious problem which must be investigated further.
ANNEXES
DEFINITION OF TERMS

Curie — Unit of radioactivity equal to 37 billion \((3.7 \times 10^{10})\) disintegrations/sec and is represented by the symbol \(\text{c}\).

A millicurie (mc.) is one one thousandth of a curie.

Mev — Abbreviation for million electron volts, a unit of energy equal to \(1.6 \times 10^{-6}\) ergs. The energy of most of the normally encountered nuclear radiations lie between 0.01 and 3.0 mev.

Roentgen — (Milliroentgen) is a measure of X or gamma radiation in a region. One roentgen produces one electrostatic unit of charge (esu) as a result of ionization in one cc of dry air at standard temperature and pressure. Table A indicates the number of gammas per roentgen for each of several gamma energies.

Rep — (Milli rep) the roentgen equivalent physical dose, is a measure of both gamma and neutron radiation damage to organic material, one rep being the amount of radiation required to deposit 93 ergs per gram of animal tissue. For practical purposes, one roentgen is considered equal to one rep.
Rem

- The roentgen equivalent dose for man is a measure of both gamma and neutron radiation dose. While it depends on the type of tissue involved, neutron energy, etc., Table A shows the amount of neutron or gamma radiation for a unit of dose.
<table>
<thead>
<tr>
<th>Energy Mev</th>
<th>Gammas/Cm²</th>
<th>Neutrons/Cm²</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>5</td>
<td>5</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>0.5</td>
<td>0.5</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td>0.1</td>
</tr>
<tr>
<td>Roentgen</td>
<td>5.4 x 10⁹</td>
<td>2 x 10⁹</td>
</tr>
<tr>
<td></td>
<td>3.6 x 10⁹</td>
<td>8 x 10⁹</td>
</tr>
<tr>
<td>Rep</td>
<td>5.4 x 10⁹</td>
<td>2 x 10⁹</td>
</tr>
<tr>
<td></td>
<td>3.6 x 10⁹</td>
<td>8 x 10⁹</td>
</tr>
<tr>
<td>Rem</td>
<td>5.4 x 10⁹</td>
<td>2 x 10⁹</td>
</tr>
<tr>
<td></td>
<td>3.6 x 10⁹</td>
<td>8 x 10⁹</td>
</tr>
<tr>
<td></td>
<td>2.8 x 10⁸</td>
<td>4 x 10⁸</td>
</tr>
<tr>
<td></td>
<td>4 x 10⁸</td>
<td>1.05 x 10⁹</td>
</tr>
<tr>
<td></td>
<td>1.2 x 10⁷</td>
<td>2.8 x 10⁷</td>
</tr>
<tr>
<td></td>
<td>4 x 10⁷</td>
<td>1.05 x 10⁸</td>
</tr>
</tbody>
</table>
ANNEX III

TRIGA REACTOR

TRIGA stands for Training Research Isotope Production Reactor. It is manufactured by the General Atomic Division of the General Dynamics Corporation.

The TRIGA was designed as a multi-purpose tool for those academic institutions and other groups working in various areas of nuclear science. The 10 to 30 KW reactor is simple in concept, inherently safe in operation and flexible in use. Built in a 20 foot pit, the surrounding earth and 16 feet of water above the core provide shielding. Since no special containment building is required, the installation cost is substantially reduced.

The main components of the TRIGA are (See sketch on following page)

1. Lifting Mechanism
2. Hand Crank
3. Specimen Container
4. Loading Tube
5. Rabbit Tube
6. Rotary Rack Drive Shaft
7. Cooling Coil
8. Control Rod Guide Tubes
9. Glory Hole
CUTAWAY VIEW OF TRIGA REACTOR
10. Rotary Specimen Rack

11. Core

12. Graphite Reflector

At 10 KW power, the thermal neutron flux at the Lazy Susan is approximately $10^{11}$ neutrons/cm$^2$/sec.

The reactor core has a prompt negative temperature coefficient. Zirconium hydride, the moderating material, is homogeneously mixed with approximately 2 Kg. of $^{235}$U in the form of 20% enriched uranium in the aluminum-clad fuel elements on the TRIGA core.

The prompt negative temperature coefficient is achieved by the presence of appreciable amounts of $^{238}$U in the core, which, with an increase in fuel-element temperature, leads to a prompt decrease in reactivity due to the Doppler broadening on the $^{238}$U resonances and the subsequent greater parasitic absorption of neutrons.

The intimate mixture of moderator and fuel provides a prompt self-regulating mechanism since there is no time delay in fuel and moderator heating. The average energy of the thermal neutrons immediately increases, leading to a prompt decrease in reactivity resulting from a decrease in the effective fission cross-section.

**Performance and Design Data**

Power = 10–30 KW → up to 100 KW intermittently

Neutron Flux at

- Rotary Specimen Rack = $10^{11}$ neutrons/cm$^2$/sec
- Critical Mass = 2 Kg. $^{235}$U
- Excess Reactivity = 0.7%
- Cooling = natural convection of refrigerated water
Inlet Water = 50°F
Reactor Pit = 20' deep - 6' diameter (underground)
Diameter of Core = 17''
Height of Active Lattice = 14''
Fuel Cladding = 0.030'' aluminum
Reflector = 12'' graphite
Shielding = 16'' water
Instrumentation = 1 fission counter, 1 compensated ion chamber
2 uncompensated ion chambers
Control Rods = 3
Control Rod Drives = winch type
Glory Hole = 1'' ID into center of core
Isotope Production Facilities:
  Rotary Rack = 40 positions 3'' from core
  Rabbit Tube = 1'' ID at edge of core
  Rotary Rack Specimen Containers = 50 cc, 1'' diameter x 1/4''
    long
ANNEX IV

COPY OF REVISED AEC REGULATIONS

UNITED STATES
ATOMIC ENERGY COMMISSION
Washington 25, D. C.

No. C-173
Tel. Hazelwood 7-7831
Ext. 3146

FOR USE IN MORNING PAPERS OF
WEDNESDAY, SEPTEMBER 7, 1960

AEC REVISES REGULATIONS TO SET LOWER LIMITS ON RADIATION EXPOSURE

The Atomic Energy Commission has revised its regulations for the protection of employees in atomic energy industries and the general public against hazards arising out of the possession or use of AEC-licensed radioactive materials.

The revisions are embodied in amendments to Title 10, Chapter 1, Part 20, of the Code of Federal Regulations entitled "Standards for Protection against Radiation." The amendments become effective on January 1, 1961.

Radiation exposure levels in the present regulations, which were issued in January, 1957, were based on the recommendations published by the National Committee on Radiation Protection in National Bureau of Standards Handbook 52 entitled "Maximum Permissible Concentrations in Air and Water" and in Handbook 59 entitled "Permissible Dose from External Sources of Ionizing Radiations."

The regulations comprise the basic AEC document for radiation safety in AEC-licensed operations. Among other things they prescribe the limitations which govern exposure of personnel within licensed atomic installations and industries to both radiation and concentrations of radioactive material; concentrations of radioactive material which may be discharged into air or water; disposal of radioactive wastes, and limits on levels of radiation outside atomic installations and industries.

(more)
Proposed amendments, designed to bring the Commission's regulations into accord with the current recommendations of the National Committee on Radiation Protection and Measurements, were published by the AEC in the Federal Register on May 2, 1959, for public comment. Many comments and suggestions were received by the Commission and these have been taken into consideration in drawing up the amendments in their final form.

In addition to receiving written comments, the AEC staff has had meetings with a committee designated by the AFL-CIO and with industry representatives and others at a meeting called by the Atomic Industrial Forum to discuss the amendments.

On May 13, 1960, the President approved recommendations on exposure made to him by the Federal Radiation Council for the guidance of agencies in the executive branch of the government. The numerical values contained in the guides recommended by the Council are substantially the same as the corresponding values contained in the current basic recommendations of the National Committee on Radiation Protection and Measurements and as those incorporated in Part 20 by these amendments.

Radiation Limits One-Third Present Levels

The principal effect of the amendments will be to limit the life-time accumulated dose of radiation workers to approximately one-third the limits permitted under the regulation as it now stands. The amendments will limit the total external radiation exposure that any worker may accumulate beyond the age of 18 to an average of five rems per year and to not more than three rems in any one quarter. Present limits for radiation workers are 0.3 rem per week, or approximately 15 rems a year, without further restrictions as to accumulated dose.

A rem (roentgen equivalent man) is a radiation dose of any ionizing radiation estimated to produce a biological effect equivalent to that produced by one roentgen of x-rays.

Radiation limits as set in the regulations are not to be regarded as absolute limits below which no hazard from radiation exists and above which an individual automatically receives a dangerous dose of radiation. The limits may be compared to speed limits, which are established to promote traffic safety.

A speed limit, for example, is not an absolute safety limit. It represents the best judgment of a number of qualified people as to a reasonable limit of speed for the safe operation of an automobile.

(more)
In establishing the limits now incorporated in the amendments to Part 20 of the Commission’s regulations, the National Committee on Radiation Protection and Measurements pointed out that the lowering of the limits should not be interpreted as indicating that exposures at levels currently permitted by the regulations have caused damage. NCRP said changes, rather, were based on a desire to bring radiation standards into accord with new trends of scientific opinion and to reflect awareness of the probability of a large future increase in radiation uses.

**Basic Table Provided**

The amendments provide a basic table showing quarterly levels of radiation to which all workers may be exposed — 1-1/4 rems per calendar quarter for whole body, head and trunk, active blood-forming organs, lens of eyes or gonads; 18-3/4 rems for hands, forearms, feet or ankles; 7-1/2 rems for the skin.

The licensee may permit an employee to receive a greater dose than that listed in the basic table provided that the quarterly dose to the whole body from radioactive material and other sources of radiation in the employer’s possession does not exceed three rems and the dose to the whole body — when added to the employee’s accumulated dose of radiation in previous jobs — does not exceed that arrived at by a formula based on the individual’s age.

To permit exposure above the limits in the basic table, the employer first must make reasonable attempts to obtain reports of the employee’s previous occupational exposure to radiation and must take these exposures into account in determining how much additional exposure an employee may receive.

In the absence of exposure records for an employee, the employer is to be guided by a table of assumed occupational exposure for the worker.

**Notice to Workers Required**

Employers will be required to notify the individual, as well as the Commission, of any exposure of the individual to radiation or to radioactive material above the established limits. Individuals who receive exposures in excess of quarterly limits (in the case of external radiation) or weekly limits (in the case of concentrations of radioactive materials) must be removed from further exposures during the remainder of the period involved.
The Commission may also require that, where appropriate, the employer provide an employee with an examination to determine the extent of his exposure to radioactive material. The employer must make the results of the examination known to the Commission.

Employers are required to periodically advise employees of their exposures if such information is requested by them. Forms are incorporated in the regulation for the recording of occupational exposures and for the recording of histories of exposure. These forms should be of considerable assistance to employers in complying with the regulations.

Employers must also post in conspicuous places an informative poster which will advise employees as to the nature and purpose of the Commission's regulatory requirements, how they may get information as to these requirements, the reports which must be furnished to employees by employers, and similar matters.

Each licensee is also required to post or to make available for examination by employees upon request a copy of the Commission's Part 20 regulations and a copy of his AEC license, including any operating procedures applicable to the work permitted under the license.

Limits on Concentrations of Material Reduced

The amendments include a comprehensive revision of the concentrations of radioactive material to which employers may expose persons in areas under their control or which may be released by the employer into the environment without specific approval by the Commission.

With respect to most isotopes listed, the principal changes in the values set forth in the new tables are a reduction to one-third in the concentrations of those radioisotopes having their principal effect upon the gonads or the whole body and the lowering of others to control exposure of the gastro-intestinal tract to 15 rem per year.

The reductions in these values do not modify the basic approach of the Commission with respect to levels of radiation and concentrations of radioactive materials in unrestricted areas --- that is, areas which are outside the control of the employer. The Commission's regulations are designed to make it unlikely that individuals in unrestricted areas receive exposure in excess of 10 per cent of the limits established for radiation workers.

All AEC-licensed activities are subject to inspection by the Commission.
BIBLIOGRAPHY


