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UNITED STATES AIR FORCE

LOCKHEED NUCLEAR PRODUCTS
Lockheed Aircraft Corporation, Georgia Division

VOLUME 1 OF 6
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Third Semi-annual
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UNITED STATES AIR FORCE



LOCKHEED NUCLEAR PRODUCTS

LOCKHEED AIRCRAFT CORPORATION
GEORGIA DIVISION -- MARIETTA, GEORGIA

FOREWORD

The proceedings of the Third Semi-Annual ANP Radiation Effects Symposium, held at the Dinkler-Plaza Hotel in Atlanta, Georgia, October 28 through 30, 1958, are in six volumes. Each of the first five volumes presents the unclassified papers from one of the five sessions; the sixth volume presents classified papers from all five sessions.

Each volume contains a complete table of contents and an index of authors. Volume One contains a list of the names of all who attended the Symposium.

Typist.

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APPLICATION OF RADIATION EFFECTS DATA TO DESIGN AND DEVELOPMENT PROBLEMS

by

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The addition of a nuclear environment to the environments normally encountered by the aircraft designer has required that the suitability of materials and components be reassessed on a broad scale. Much of the activity of materials engineers and designers of the past few years has been concerned with this assessment. This report presents a summary of the concepts and procedures utilized in applying radiation effects data to the solution of design problems and comments on developments of the past year.

Conceptually, application of radiation effects data involves, simply, the comparison of observed effects on materials properties with the properties requirements of the anticipated application. The analytical processes necessary in implementing this comparison include consideration of the various radiation dosage units, determination of the relative effectiveness of different types of particles in producing damage, and the evaluation and extrapolation of data. Limiting factors in such an analysis include difficulties with dosimetry, lack of knowledge of radiation effects under combined environments, and limited knowledge of the materials properties necessary in specific engineering applications.

Despite the approximations necessary, this analytical approach serves a number of purposes to real advantage. Preliminary materials selection can be made with sufficient confidence that the major effort of design and development can proceed; time is thereby gained in which to define materials performance. Key problem areas have been defined, and the definition has led to studies to develop new and improved materials.

INTRODUCTION

Much of the effort on the ANP program to date has been concerned with evaluating the problem created by the radiation environment. Radiation has been found or is suspected of being analogous to temperature in the sense

that it affects materials performance in all other environments. This has meant that almost all of the materials and components of an aircraft have had to be examined to determine their suitability in this additional environment.

In completing this examination, a more or less common approach for applying the existing radiation data to design and development activities has been developed by the various contractors in the ANP program(1-5). This approach consists simply of a comparison of test data with the requirements of an application; this is not a new or novel concept other than in its implementation with respect to the radiation data. As is the case in any such activity, the comparison is most effectively accomplished when one possesses knowledge of the environment, conditions, and other important parameters of both the test and the application. In the radiation effects area, such knowledge generally has not existed to the extent that a final solution to an engineering problem could be obtained. The work to date, therefore, represents only a segment of the over-all radiation damage problem. Several useful purposes, nevertheless, are served by using this approach.

This paper summarizes the processes utilized in applying radiation data to design and development activities. Following a description of the over-all procedure, the bases, limitations, and recent developments in the steps are reviewed. The discussion is limited to the organic materials, since - with a few exceptions - these are the materials most susceptible to damage and will constitute the principal problem area; the same or a similar process is used in applying data on metals and inorganic solids.

GENERAL DESCRIPTION OF THE PROCEDURE

The approach generally used in applying radiation effects data to a particular materials problem consists simply of a comparison of the observed property changes to the properties requirements of the application. The initial information requirements are the accumulated radiation effects data and knowledge of the application including the radiation level, the lifetime desired for the component, and the composition of organic materials used. With this information at hand, the following steps are completed:

1. Conversion of radiation dosages from those of the experimental observations to those anticipated in the application.
2. Evaluation of the amount of property change permissible in the application.
3. Computation of the time period required in the application to attain a dosage corresponding to the permissible amount of property change.

Figure 1 is an example of the use of this general approach to the application of data as followed at GE-ANPD. In this example, use is made of "threshold dosage" and "equivalent dosage" in relating the experimental dosage to that of the application. Various investigators have defined a number of such "dosages" as a means of facilitating the dosage conversion computations; these will be briefly described in a subsequent section.

Although this approach is simple in concept, much effort in the ANP program has been devoted to accumulating the radiation data, to establishing the bases of procedure, and to facilitating the completing of this type of analysis. Several comprehensive summaries of the available radiation data have been prepared^(1,6,7) and the Radiation Effects Information Center at the Battelle Memorial Institute is engaged in keeping the information up to date.⁽⁸⁾ Also, reports are available in which dosage conversion factors are tabulated;^(1,4,5) these considerably simplify the dosage conversion procedure.

BASES AND LIMITATIONS IN APPLYING RADIATION DATA

One of the major problems in implementing an analytical approach for applying radiation data has been the establishment of the bases for the various steps. Factors that have required consideration have included, for dosage conversion, the effectiveness of different particles in producing damage and radiation dosimetry, and, for extrapolation, study of the radiation damage process.

Relative Damage Effectiveness of Different Types of Particles

The essential requirement in relating damage in organic materials of different radiation environments is a knowledge of the relative effectiveness of different types of particles in producing damage. This point has been of interest since the beginning of modern radiation effects work. However, because of the experimental difficulty involved in accurately determining the number and energy of the particles, the problem has not been completely resolved to date. It is generally agreed, however, that the damage produced in an organic material by any one type of particle is a function of the amount of energy deposited by that particle in the material, and that, further, if differences exist, they are more in amount than in kind.

Theoretically, at least two views of particle effectiveness can be advanced. The first holds that since the major portion of the energy deposited by any type of particle is dissipated through ionization processes, the damage should be proportional to the total energy irrespective of the type of particle, i.e. equal damage will result from the deposition of equal amounts of energy. The second view, from a detailed consideration of

Figure 1

TYPICAL CALCULATIONS IN APPLYING RADIATION EFFECTS DATA TO
AN ENGINEERING APPLICATION

Problem: A natural rubber hose is to be used in a radiation field consisting of:

4×10^9 thermal neutrons/cm²-sec,
 1×10^8 fast neutrons/cm²-sec, and
 5×10^{10} gamma photons/cm²-sec.

The probable life of the hose is estimated by the following procedure:

Thermal neutrons

Equivalent dosage: 2.4×10^{17} n/cm² = 2×10^6 rads (obtained from tabulated summaries of data)

$$4.0 \times 10^9 \text{ nt/cm}^2\text{-sec} \times \frac{2 \times 10^6 \text{ rads}}{2.4 \times 10^{17} \text{ nt/cm}^2} = 0.03 \text{ rad/sec}$$

Fast neutrons

Equivalent dosage: 5×10^{15} n_f/cm² = 2×10^6 rads

$$1 \times 10^8 \text{ n}_f\text{/cm}^2\text{-sec} \times \frac{2 \times 10^6 \text{ rads}}{5 \times 10^{14} \text{ n}_f\text{/cm}^2} = 0.4 \text{ rad/sec}$$

Intermediate energy neutrons

Approximately the same as that from fast neutrons = 0.4 rad/sec

Gamma photons

Equivalent dosage: 4×10^{15} γ/cm² = 2×10^6 rads

$$5 \times 10^{10} \text{ } \gamma\text{/cm}^2\text{-sec} \times \frac{2 \times 10^6 \text{ rads}}{4 \times 10^{15} \text{ } \gamma\text{/cm}^2} = 25.0 \text{ rads/sec}$$

Total rads/sec = $0.03 + 0.4 + 0.4 + 25.0 = 25.8 \text{ rads/sec}$

From graphs of property changes, it is estimated that the material should withstand a dosage of 10^8 rads. Then

$$\frac{1 \times 10^8 \text{ rads}}{25.8 \text{ rads/sec}} = 3.88 \times 10^6 \text{ seconds,}$$

or about 1000 hours.

the interaction process, holds that the density of the ionization along the track of the damaging particle affects the reactions that occur and therefore influences the extent of damage.

Depending upon the view one wishes to take, a case in support of either of these views can be built up from the experimental data. One type of comparison of particle effectiveness can be made from the large body of data on organic liquids reported in the unclassified literature. The usual approach in these studies consists of the measurement of "G" or similar "letter" values that correspond to the number of molecules produced or reacting per 100 volts of absorbed energy. There are a number of review articles in which such values are tabulated and discussed.⁽⁹⁻¹²⁾

The use of "G" values per se assumes that the process under study is a linear function of radiation dosage since neither the dosage nor the weight of material in which the energy is deposited is specified. Now, from the extensive measurements made in the AEC and ANP programs, gas evolution from organic liquids does appear to be an approximately linear function of dosage, at least in the early stages, so that comparison of "G" values for gas evolution is possibly a valid process. It is doubtful whether other processes, such as the destruction or disappearance of the original molecules, is a linear function of dosage; "G" values for these processes, therefore, can be questioned unless they are determined as a function of dosage. This is evident in that, where comparative data exist, there is much closer agreement between the "G" values for gas evolution than, for example, for disappearance of the original material.

In general, where comparative data exist, the "G" values for gas evolution in the various review articles indicate very good agreement on an energy absorbed basis for the saturated aliphatic hydrocarbons, and agreement within a factor of two for unsaturated aliphatic liquids, and for aromatic compounds. Data of this type do not appear sufficient to evaluate the particle effectiveness in alcohols, ethers, ketones, and esters. Finally, there are a number of reactions which definitely appear to differ for different types of particles. These include some in which there exists a possibility for a chain reaction as in an unsaturated compound such as ethylene (gas phase), vinyl compounds, and some halogenated compounds, particularly bromides and chlorides. Comparisons of this type are not possible for organic solids within the available data.

Particle effectiveness comparisons have also been made on the basis of property change data and the indications are that equal energy-equal damage is a useful approximation. The results of a number of pile, gamma, and electron irradiations of elastomers and organic liquids were compared on the basis of energy absorption in APEX 261 and the agreement appeared to be well within a factor of two. Figures 2, 3, and 4 are more or less typical of the extent of agreement obtained in these comparisons. Considering the state of pile dosimetry techniques, the lack of agreement in these particular results could be attributed either to a real difference or to uncertainties

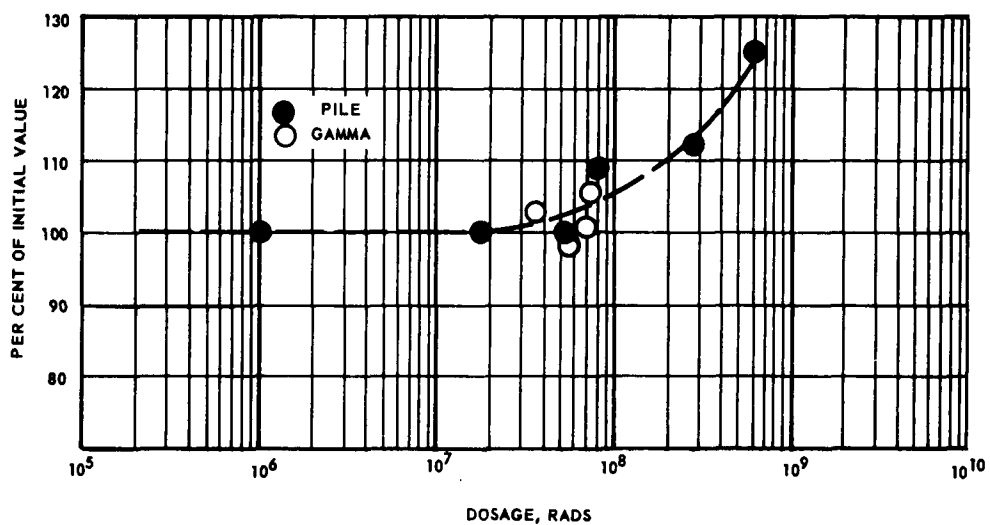


Fig. 2— Comparison of reactor and gamma radiation effects on the hardness of Neoprene W

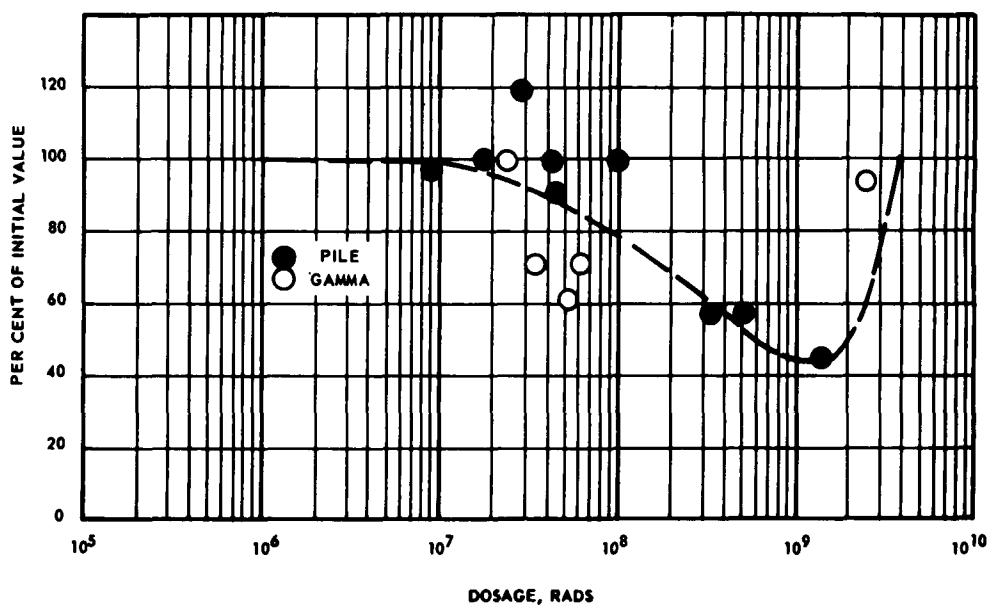


Fig. 3— Comparison of reactor and gamma radiation effects on the tensile strength of buna rubber (GR-S-50)

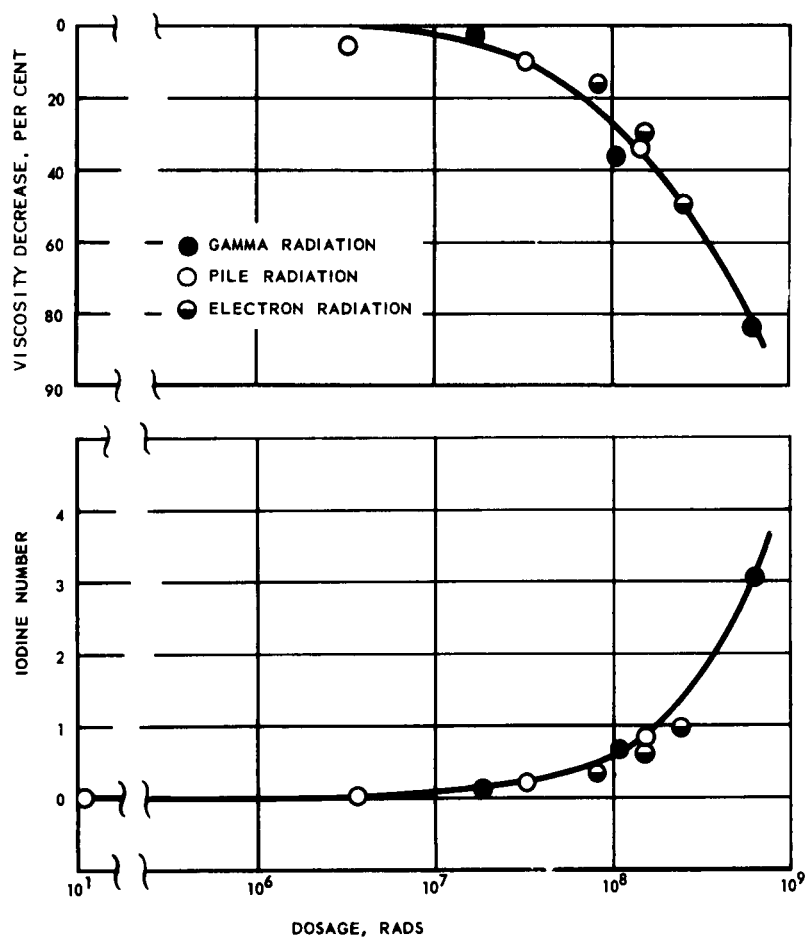


Fig. 4—Comparison of pile, gamma, and electron radiation damage to glycerine

in the calculations of energy absorption. Within the last year or two, workers at Stanford Research Institute have demonstrated quite good agreement between electron and gamma irradiations in both organic liquids and solids when results are compared on the basis of total energy absorption. (13,14)

Thus, although an accurate assessment of particle effectiveness is yet to be completed, the equal-energy-equal-damage approach is considered a good rule-of-thumb approximation that is accurate within a factor of two. Certainly the available data rather conclusively show that neutrons, or other heavy particles, are not 40 or even 50 times as effective in producing damage as has been found in some instances of biological damage. In certain practical applications, also, the problem of particle effectiveness is almost of academic interest only. For example, many applications involve the comparison of experimental data from gamma-only irradiations with an application in which the radiation field of neutrons and gammas has a high ratio of gamma particles to neutrons. In this instance, a possible error of a factor of two in assessing neutron damage will have a comparatively small effect on the final result.

Aside from the considerations of accuracy, there is the limitation to dosage conversion on the equal energy-equal damage basis that the comparison must be made for comparable environmental conditions. Obviously, equal energy deposition in a material at appreciably different temperatures or stress levels will not produce the same amount of damage since other factors enter into degradation of the material.

Radiation Dosimetry

In applying radiation effects data, radiation dosimetry is of importance in that it either simplifies the dosage conversion calculations or enables them to be completed with greater accuracy. The uncertainties and practical difficulties in completing this operation have undoubtedly hampered the progress of radiation effects studies. Progress in the area is being made, however; one evidence of this, the use of ergs per gram of carbon, will be apparent in this symposium.

The basic problem in dosimetry is that it is impractical or impossible to count the particles directly at radiation levels of practical interest; hence, the radiation field must be assessed by its effects or interactions with reference materials. Since the interactions are dependent on the elemental composition of the material and also on the types and energies of the particles, no one reference material can serve as an indicator of the interactions that will occur in all other materials.

These difficulties have resulted in the adoption of two philosophies of dosimetry. In one, the radiation field is described in terms of the amount of energy that it will deposit in a reference material. In the second, the actual amount of energy deposited in the material under study is determined - usually by computation from a description of the radiation field. The terms "exposure dose" and "absorbed dose" respectively, have been advocated for these two descriptions.

Exposure dose is the more convenient approach experimentally and is very useful where engineered materials or components are being examined. Where the radiation fields are of similar composition, the performance of the engineered components can be directly related to the exposure dosage since the same geometries and the same radiation are involved. However, when the radiation field is comprised of different proportions of the radiation particles, the performance of either materials or components must be assessed through the energy actually absorbed in the organic material. Computation of the energy absorption in components is not simple; complications include the odd geometries of the materials and the attenuation of the radiation field in traversing the housing or casing of components. For this reason, a description of a radiation field is the more usual and popular approach to dosimetry.

Since both exposure and absorbed dose may be reported in terms of energy absorption, it is important that the distinction between the two be kept clear. Exposure dose describes the radiation field in terms of the energy deposited in a reference material, while the absorbed dose is a description of the actual amount of energy deposited in the material under study. The recommended unit for absorbed dose is the erg per gram although the rad (100 ergs/gr), ev/gram, calories/gm, and other energy units have been used; Table I presents a brief summary of various exposure dose units.

Probably the most significant development of the past year is the use of ergs per gram of carbon to describe the radiation field. This unit was suggested by the ANP Advisory Committee for Nuclear Measurements and Standards as a means of obtaining uniformity in reporting work.⁽¹⁵⁾ It is measureable by means of a carbon ionization chamber and has the advantage - in contrast to some of the other units listed in Table I - of providing a direct measurement of the gamma energy. Even in a radiation field that deposits approximately equal amounts of energy from neutrons and from gammas, the carbon ionization chamber should give the gamma energy with an accuracy of better than 90%.

Widespread use of the erg/gm C in describing exposure dosage will considerably simplify the interconversion of radiation dosages. The energy absorption coefficient for gamma photons of hydrogen is approximately double that of carbon (and of oxygen and nitrogen as well) over a range of gamma energies from 0.2 to 3.0 Mev. Since these gamma energies cover the range of practical interest, this means that - to a good approximation - the energy deposition in organic materials will be higher than that in carbon by an amount corresponding to the percentage of hydrogen in the material. For example, a material with 6 per cent hydrogen will absorb 105 to 106 per cent of the energy absorption in carbon.

Equivalent Dosages

Equivalent dosages are the result of efforts to eliminate the necessity for completing detailed calculations of energy absorption for every

TABLE I
RADIATION UNITS USED TO DESCRIBE A RADIATION FIELD

UNIT	DEFINITION	REMARKS
Ergs per gram carbon or Carbon Dose	Amount of ionizing radiation (gammas, x-rays, and electrons) which imparts one erg per gram of carbon.	Ergs per gram Carbon, per the recommendation of the ANP Advisory Committee for Nuclear Measurements and Standards, is a means of describing a radiation field of gamma photons, x-rays, or electrons in terms of the amount of energy deposited in the reference material carbon. Because of the differences of various elements in absorbing energy from X or gamma rays, a radiation field of one erg/gm C will deposit different amounts of energy in different materials.
Roentgen	Amount of ionizing radiation (gamma and x-rays) which imparts 83 ergs per gram of air.	The roentgen, by definition, specifies both the type of radiation and the material; hence, it is an indirect means of describing a radiation field composed of x-rays and gamma photons. A radiation field of 1 roentgen will deposit from 83 ergs per gram (air, Teflon) to nearly 100 ergs per gram (polyethylene) in different materials.
Rep (roentgen equivalent, physical)	Amount of radiation of any type which imparts 93 ergs per gram of animal tissue.	The rep, by definition, specifies only the material; hence, it is an indirect means of describing a radiation field comprised of any type of nuclear particle including neutrons, alpha particles, and protons as well as x-rays and gamma photons. As with the roentgen, 1 rep of radiation will deposit different amounts of energy in different materials. For greatest utility, radiation fields described by rep should specify the fraction of total rep due to each type of particle.

TABLE 1 (CONTINUED)

UNIT	DEFINITION	REMARKS
Rem (roentgen equivalent, man)	The amount of radiation of any type which has the same biological effectiveness as 1 roentgen of x-rays or gamma radiation.	The rem involves the use of factors for the relative effectiveness of various particles such as protons, alphas, and neutrons as compared to x-rays or gamma rays in producing biological damage. Although the rem can be used as an indirect means of describing a radiation field, its use in material-damage work is limited.
Nv	The nv specifically refers to n neutrons per cubic centimeter moving with a velocity v centimeters per second in a given direction; hence, the product nv equals the number of neutrons of velocity v that traverse 1 square centimeter oriented in any direction.	Where v is not otherwise specified, nv or nvt is usually considered to apply to thermal neutrons. The nv is therefore a direct description of the number or flux of neutrons of near-thermal energies present per second in pile radiation. When nv is used, it is implied that the flux of fast neutrons and gammas present in the particular radiation field also pass through the unit area. Similar considerations apply to the nvt dosage unit.
Nvt	The nvt is the dosage unit corresponding to the nv flux unit and is equal to the product of nv and exposure time in seconds.	

materials application. Equivalent dosages are simply dosages of the same damage effectiveness for a given material expressed in various dosimetry units. Tables of these equivalent dosages reduce the calculations necessary in dosage conversion to the processes shown in Figure 1.

In setting up tables of equivalent dosages, workers usually selected a dosage corresponding to some specific amount of property change. In addition to simplifying dosage conversion, this permits comparison of the radiation stability of materials on a common basis. For example, at GE-ANPD, we have utilized terms such as Threshold of Damage dosage and Twenty-five per cent damage dosage, respectively, to refer to the dosage at which damage is first apparent and to the dosage at which at least one physical property of the material has changed twenty-five per cent from its value in the original material. Other organizations have used the same or similar terms with either the same or slightly different meanings. It is not the intention - where such terms are used - that these are necessarily the limiting dosages to which the material should be used. Obviously, a material of high strength may, after considerable irradiation, still be stronger than another material which has suffered less loss of strength percentage-wise but whose initial strength was low. Therefore, when the particular properties required of the material are known, the designer can make a more sound evaluation of materials by comparing the actual values of the particular property or properties in several materials. It is important in applying radiation effects data that the definition of threshold, 25 per cent, and similar "dosages" used in the different reports be recognized and that the specific properties of interest be considered whenever it is possible to do so.

In an approach toward obtaining a limiting dosage, Convair has proposed a Functional Threshold Dosage.⁽⁴⁾ Functional Threshold is defined as the minimum amount of absorbed energy required to change the properties of a material or functioning component to the extent that unsatisfactory operation of a particular system will result. In the absence of engineering test data, this dosage is derived from an analysis of the pertinent properties required of the material together with an evaluation of the permissible amount of change in these properties in the application as compared to the changes produced by radiation. It was envisioned that, as work progressed, experimentally observed Functional Thresholds would replace the computed values. This Functional Threshold Dosage has the advantages - in comparison to the various "dosages" described above - of being applicable to specific uses of the material and of eliminating the necessity for estimating performance from data on materials properties. Further, since it is related to the lifetime of the material in the specific application, the Functional Threshold Dosage is an important engineering quantity.

Dose Rate Versus Dosage

The principal means of extrapolating or extending the usefulness of radiation effects data has been based on the observation that damage is

a function of dosage irrespective of dose rate. This generalization has been supported by experimental observations by almost every organization engaged in radiation testing; a recent report by Harrington on elastomers and plastics contains probably the most extensive observations of this type.⁽¹⁶⁾ This has meant that where the dosage corresponding to a given amount of property change was known, the time required for this amount of change at any dose rate could easily be computed by simple arithmetic. Since experimental data have rarely been obtained at dose rates that correspond exactly to those of the application, this equal dosage-equal damage idea has been used extensively in assessing radiation damage problems.

At the same time, it has been recognized that extrapolations on this basis necessarily are limited. It is obvious and has been demonstrated in numerous instances that, where radiation is superposed on other environmental conditions, the time required for a given amount of change in a material cannot be based on the radiation data alone. It has been expected, also, that a dose rate effect would occur under combined environments, the basis being that two or more processes would be acting simultaneously to produce degradation of the material. Dose rate effects have also been observed in studies of chemical reactions⁽¹⁰⁻¹²⁾ and in some cases of practical applications such as the oil oxidation measurements at Shell Development Company⁽¹⁷⁾ and the observations of Goodman and Coleman⁽¹⁸⁾ that the dosage-to-failure of Kel-F as a dielectric is proportional to the square root of dose rate at temperatures of 30° and 90°C.

In view of these considerations, investigators have either allowed very large safety factors in estimating material lifetimes or have contented themselves with using the existing data as a means of selecting the more-radiation stable materials for testing under combined environmental conditions. The uncertainty of extrapolations into combined environmental conditions has been responsible for the increasingly loud hue and cry during the last year for more definitive test data.

Extension of the analysis of experiments recently completed at GE-ANPD indicates that the conditions under which dose rate effects will or will not occur can be delineated by considering the magnitudes of the sources of energy contributing to the degradation of the material. The experiments consisted of measurements of the time required for a given amount of oxidation to occur in an organic liquid over a range of dose rates and temperatures. The details of this work are to be discussed in another paper to be presented at this symposium, the conclusion, simply stated, is that the results can be described by an equation of the type

$$\ln t = \frac{C_1}{T} - \left(\frac{C_2}{T} + C_3 \right) \ln r + C_4$$

where t is the elapsed time required for a given amount of oxidation, T is the absolute temperature, r is radiation dose rate, and the C 's are experimentally derived constants. This equation is similar to the

familiar equation relating time and temperature for accelerated life tests, i.e.

$$\ln t = \frac{C_1}{RT} + C_2$$

where t is time, T is absolute temperature, R is the molar gas constant, and again C 's are constants, C_1 being commonly referred to as the experimental activation energy.

Computations from the experiments yield the over-all results shown in Figure 5. An important point is the indication - by computation at least - that there exists a dose rate of such magnitude that the thermally supplied energy is inconsequential. This is indicated by the horizontal line at the bottom of Figure 5. Now, if there exists a radiation rate that can completely over-ride or mask thermal effects in as rapid a reaction as oil oxidation, it seems a reasonable conjecture that at some dose rate other sources of energy-for-reaction will similarly be masked in other reactions and materials. Therefore, we may expect that, under a given set of conditions of stress, temperature, etc., there will be a dose rate X of such magnitude as to supply the predominant amount of energy that causes reaction. Since, at any dose rate equal to or higher than X , radiation energy is predominant, we would expect that equal reaction or damage would occur at the same radiation dosage. On the other hand, at dose rates less than X , the energy supplied for reaction by other conditions (stress, temperature, etc.) will be of the same order of magnitude as that supplied by radiation. Hence, at various dose rates less than X , the amount of reaction occurring will depend on both the radiation and the other conditions, and a dose rate effect will be noted. The value of X would be expected to vary for different materials and for different combinations of environments in a given material.

This view of the situation appears to explain the differences in data wherein a dose rate effect is observed in one instance and not in another. Harrington's data on Teflon, for example, were obtained by irradiation of samples without stress at room temperature. For this condition and in the absence of radiation, Teflon could be expected to degrade very little over a period of years, i.e. the condition supplies energy for reaction at a very low rate. A comparatively low radiation dose rate therefore, can supply energy-for-reaction of a magnitude such as to completely mask that supplied by this condition. As discussed above, where the radiation energy is much much greater than energy from other sources, damage should be dependent only on the radiation and therefore, should be dependent only on the dosage. The tests described by Goodman and Coleman in which a dose rate effect was apparent involved maintaining Kel-F samples under electrical stress during irradiation. For this condition, the dose rates employed (maximum of $\sim 2 \times 10^6$ ergs (gmC) $^{-1}$ hr $^{-1}$) apparently supplied energy of the same order of magnitude as that arising from the electrical stress and a dose rate effect was apparent.

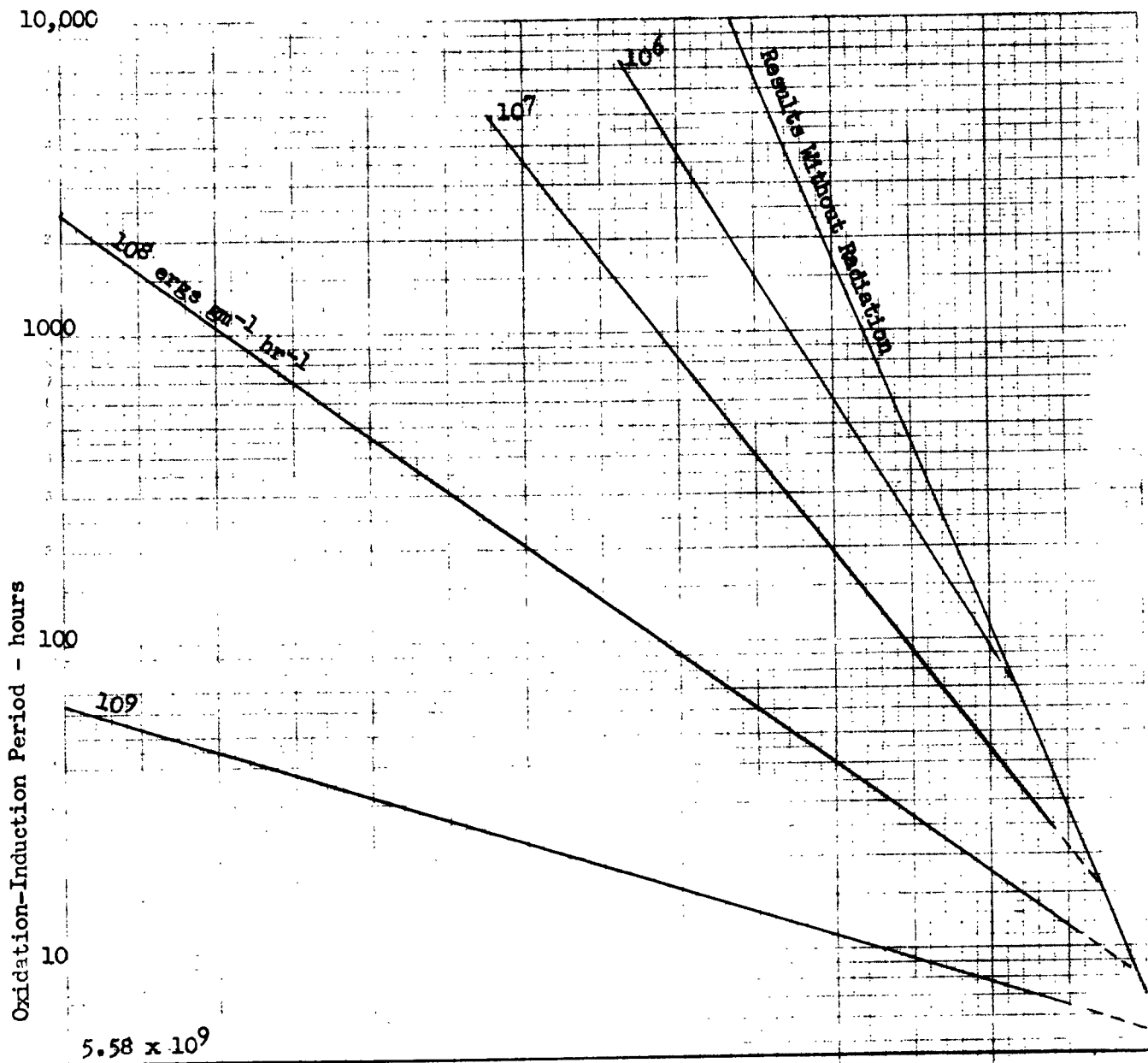
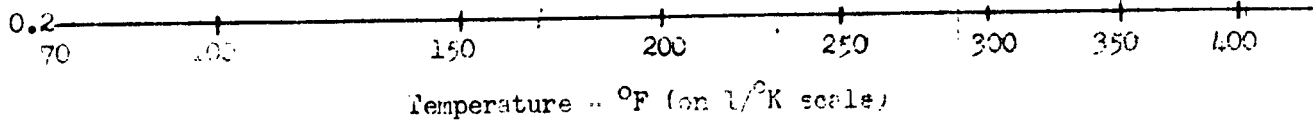


Figure 5

COMPUTED OXIDATION INDUCTION PERIOD OF INHIBITED DIESTER OIL
AT VARIOUS RADIATION DOSE RATES

1

Estimated for 5.58×10^{10} ergs gm⁻¹ hr⁻¹



VALUE AND USES OF THE GENERAL PROCEDURES

In view of the approximations and limitations of the procedure, the question, "What good is it?", naturally arises. Any attempt to answer the question can easily get into the realm of meaningless generalities; probably the greatest value of the work in this area has been to reduce the radiation effects problem to one that could be approached in a standard, logical manner. An objective in implementing the general procedure for applying data has been to eliminate the necessity for engineers to become specialists in still another area. Use of the procedure has led to the tabulation and presentation of data(1,2,4,5,19) in a form that at least partly eased the engineer's burden in this respect.

The principal use made of the procedure described has been in analysis of the over-all radiation damage problem. Armed with radiation effects data and tables of equivalent dosages, with charts of radiation levels and component locations about the power plant or airframe, and with information on the kinds and composition of organic materials used in the components, workers have been able to estimate lifetimes of materials and components. Despite their approximate nature, comparison of the estimates to the desired service life enabled designers to pinpoint the critical problems, to consider alternate component locations, and, in general, to establish the feasibility of the system from the standpoint of radiation damage. The completion of this job has been of value in permitting the major effort of the ANP program to be devoted to the main job of design and development of the power plant and airframe. Recognition of the critical problem areas permitted the Air Force and the contractors to begin work early in the program on the development of new or improved materials.

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RADIATION EFFECTS TESTING OF AIRCRAFT
SUBSYSTEMS AND COMPONENTS AT AIR FORCE PLANT
NO. 67 FOR THE ANP PROGRAM

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A few of the features of Air Force Plant No. 67 are briefly reviewed as they contribute to a discussion of the philosophy of radiation effects testing of aircraft subsystems and components.

Construction of Air Force Plant No. 67 in the southeastern part of Dawson County, Georgia, is nearing completion ahead of schedule. This facility will be operated for the Air Force by the Lockheed Nuclear Products Branch of the Georgia Division of Lockheed Aircraft Corporation.

Many of the features of this plant were discussed at the last Radiation Effects Symposium at Columbus, Ohio; details of the facility have been published in LAC Report No. 143, "GNAL Facility Capabilities." A few of the features will be briefly reviewed, however, as they contribute to a discussion of radiation effects testing of aircraft subsystems and components.

The major functions of this facility will be to irradiate, test, and evaluate aircraft components and complete subsystems under dynamic environmental conditions. To accomplish these functions, some rather specific requirements must be satisfied. The first is to provide for dynamic testing of aircraft subsystems and components in realistic radiation fields; that is, in fluxes comparable to, and in some cases in excess of, those expected to fall on the particular system at its most likely location in a nuclear-powered aircraft. The second requirement is the ability to handle large test articles rapidly. The third is for an adequate handling and testing capability. Large test articles must be moved to and from the reactor safely with means provided for evaluating them before, during, and after irradiation without exposing personnel to excessive radioactivity.

These requirements are met at Air Force Plant No. 67. Figure 1 shows the site plan with insets of the two major laboratory areas. Test articles are moved between these areas by the Hot Materials Transportation System, a standard gage railroad equipped with a manned locomotive and specially designed test cars, which are 10 feet wide and 15 feet long and which can support a 15-ton test article. It is in the Radiation Effects Laboratory shown in the lower inset that the test articles are mounted on the test cars and instrumented. This building also houses the hot-cells and warm laboratories, where post-irradiation testing and evaluations are conducted. The Reactor Building, shown in the upper inset, houses the reactor and provides space for test articles to be grouped around it, as shown in Figure 2. Electrical services at these car positions provide a total of 900 kilo-volt-ampere of a-c power and 50 kilowatts of 28 volt d-c power. The a-c power is provided in the following forms:

- 440-volt, 3-phase, 60 cycle
- 120 and 208-volt, 3-phase, 60-cycle
- 120 and 208-volt, 3-phase, 400-cycle

In addition to the power leads, the following instrumentation or control leads are available at each car position:

- 208 pairs of size AWG No. 20 conductors, twisted and shielded. (This is equivalent to 75 four-wire transducer channels, 50 two-wire thermocouple channels, and 8 pairs for spares.)
- 84 channels of low-voltage coaxial cables
- 10 channels of 5000-volt coaxial cables

Additional leads can be installed as needed. Mating boards installed in the test car pits connect all power and control leads to the test article through a corresponding member on the test car as it is positioned in the Reactor Building, shown in Figure 2. The leads from the car pits are connected through the pre-amplifier system, installed in the Reactor Building basement, and extended through tunnels to the control and recording instrumentation, as shown in Figure 3. This instrumentation includes both analog and digital recording equipment, as well as indicating and control panels for conducting measurements on subsystems and components.

The flow of a test system through Air Force Plant No. 67 is shown in Figure 4.

The ultimate objective of the Radiation Effects Program to be conducted at Air Force Plant No. 67 is the development and dynamic environmental proof-testing of aircraft subsystems essential to the mission requirements of ANP manned aircraft. In addition to those subsystems essential to functional and weapon system requirements, this program will develop and proof-test under dynamic conditions the additional flight test instrumentation and ground support equipment required for experimental nuclear flight. The work will incorporate the assistance of subcontractors whose interests and achievements contribute to the ANP effort.

The Radiation Effects Program consists of three phases. The development phase has as its final objective the qualification of equipments and components as required by their specific aircraft applications. The developmental testing of these items will be carried out under simulated operating conditions; that is, the conditions expected in the nuclear aircraft. The second phase is the proof-testing of subsystems consistent with the airplane requirements. For the experimental aircraft, the first phase will be completed prior to the 90% functional release date in order to give sufficient lead time for acquiring equipment prior to installation; the second phase will be completed before the first nuclear flight. Proceeding concurrently with these two phases will be a third phase intended to establish subsystems reliability consistent with the overall reliability requirements of the airplane. It is expected that this reliability program will be continuous.

DEVELOPMENTAL PROGRAM

The developmental program will be devoted principally to determining the components and equipments to be utilized later in subsystem proof-test programs. Components and equipments will be selected on the basis of design requirements and functional criteria specifications. The assistance of the subcontractor will be utilized in predicting expected lifetimes based on estimated damage due to irradiation, maintenance limitations due to activation, and service life under normal aircraft environmental conditions. Concomitant with the lifetime estimate, a random failure rate estimate will be made for nuclear environments when such information is available. The feasibility of accelerated rate testing will be determined from existent information in order to utilize fully the Air Force Plant No. 67 capabilities.

Following a lifetime and failure rate estimate, recommendations may be made for aircraft design changes such as relocation or local radiation shielding or for material substitution in the component or equipment to be effected by the subcontractor concerned.

PROOF-TEST PROGRAM

For proof-testing, the entire subsystem will be analyzed with respect to the functional requirements of the various components and equipments of which it is comprised. Of primary importance in this analysis will be the determination of all environmental factors, both nuclear and conventional, as they will exist for the individual item in the airplane. Of equal importance will be the determination of the acceptable functional performance of the item in the specific application. This concept implements the principle that the performance requirements for a given component are controlled by the application and that they vary with it.

On the basis of this analysis, a test program will be formulated. The first step will be the definition of the primary and secondary objectives expected to be achieved. Next will be the design of the experiments to fulfill these requirements. Test panels will be designed and fabricated to simulate the particular subsystem functional and environ-

mental conditions. Differences in nuclear environmental requirements for the various components will be achieved through the judicious employment of position on the test panel and the use of spot shielding. Conventional environments will be combined with radiation in accordance with those that have been found to be synergetic. According to a study conducted by the Cook Electric Company Inland Testing Laboratory Division under contract to WADC, these are as follows:

Temperature - Humidity - Radiation
Temperature - Ozone - Radiation
Humidity - Ozone - Radiation
Temperature - Humidity - Ozone - Radiation
Pressure - Radiation

Some results from test programs indicate that vibration in combination with some or all of the other parameters should be added to this list for subsystems and components evaluation. Figure 5 shows a section of polyethylene insulated thermocouple lead wire that was subjected to a gamma dose of 2.46×10^{10} ergs/gm (carbon) during one of the Lockheed Nuclear Products test programs. In the stressed area severe crazing occurred; and under certain conditions, vibration could dislodge sections of crazed insulation. For this and other reasons, predictions of performance characteristics of subsystems and components in nuclear-powered airborne vehicles will be more reliable if they are based on data obtained from irradiations of operating test systems. Dynamic irradiation may produce either more or less severe effects than static irradiation, depending on the particular materials and components in the system. For instance, test systems containing elastomers that are stressed during operation would be expected to exhibit more severe degradation during dynamic than during static testing. Bonds between molecules in a stressed elastomer have a tendency to break and to reattach themselves to other molecules, thus tending to relieve the stress. These bonds will, when subjected to irradiation, have an even greater tendency to break and not only attach to other molecules of the elastomer, but also to combine with extraneous ions and molecules, such as ozone and water vapor, which may be present in the surrounding atmosphere. On the other hand, a system in which hydraulic fluid circulates would be expected to sustain less change to the hydraulic fluid under dynamic than under static irradiation. Assuming a dose variation over the system, the fluid in circulation would receive a lower dose than would that part which under static conditions would be in a high dose area. Under irradiation, hydraulic fluids evolve gases. If allowed to form in the lines, gas pockets would contribute to erratic operation. Under dynamic operation, however, these gases will tend to be trapped in the fluid reservoirs.

Since the conventional environmental requirements will not in most cases be uniform for all components of a subsystem, attention will be focused on individual component requisites that will be satisfied within feasibility limits. This would, for example, take the form of incorporating into a test article such items as heaters or blowers for local temperature control.

As stated in the beginning, Air Force Plant No. 67 is not yet in operation; although major effort is directed toward its completion and activation, some work has been underway in the preparation of test articles for irradiation at the facility. One of these is shown in Figure 6. This test article was described in detail by R. N. Miller in "The Effect of Gamma Radiation on the Operating Characteristics of an Electro-Hydraulic Servo System." This system will be mounted on a railroad test car and controlled from a module in the operations building similar to the mock-up shown in Figure 7.

To summarize: The philosophy incorporated into the design of Air Force Plant No. 67 and into the concepts for its operation will, through the testing programs to be executed, provide the Air Force with the radiation effects information on aircraft subsystems required for the design and construction of a nuclear-powered airplane.

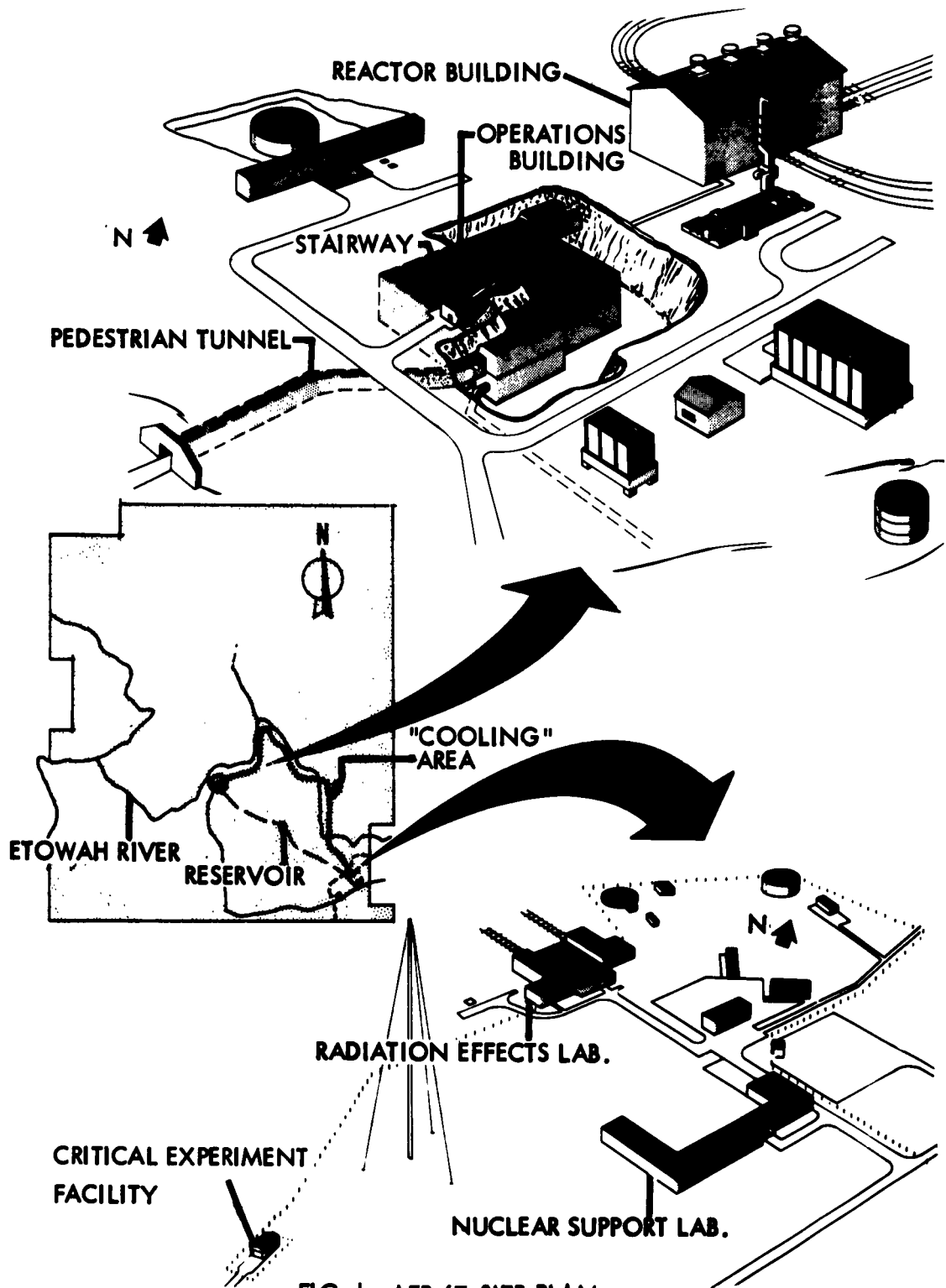


FIG. 1 AFP 67 SITE PLAN

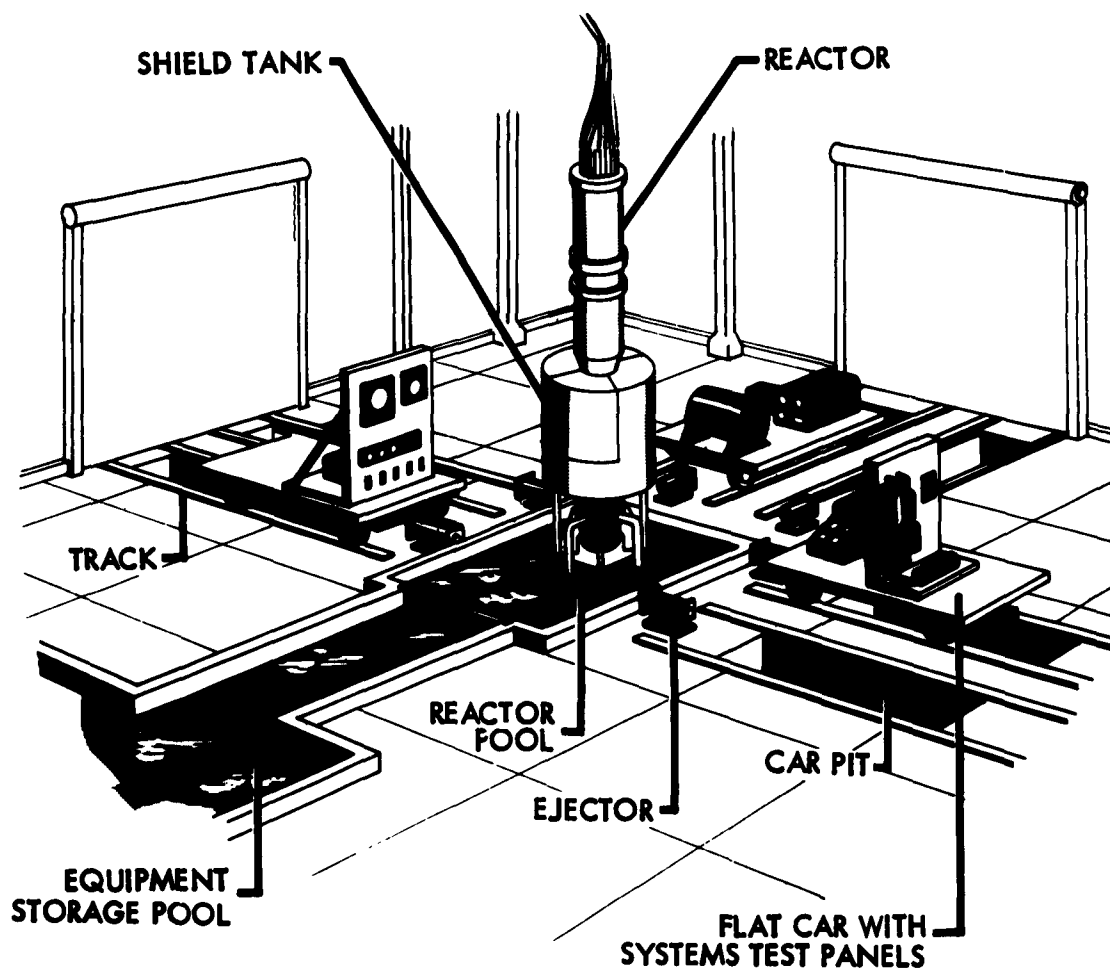


FIG. 2 RADIATION EFFECTS FACILITY, IRRADIATION TEST

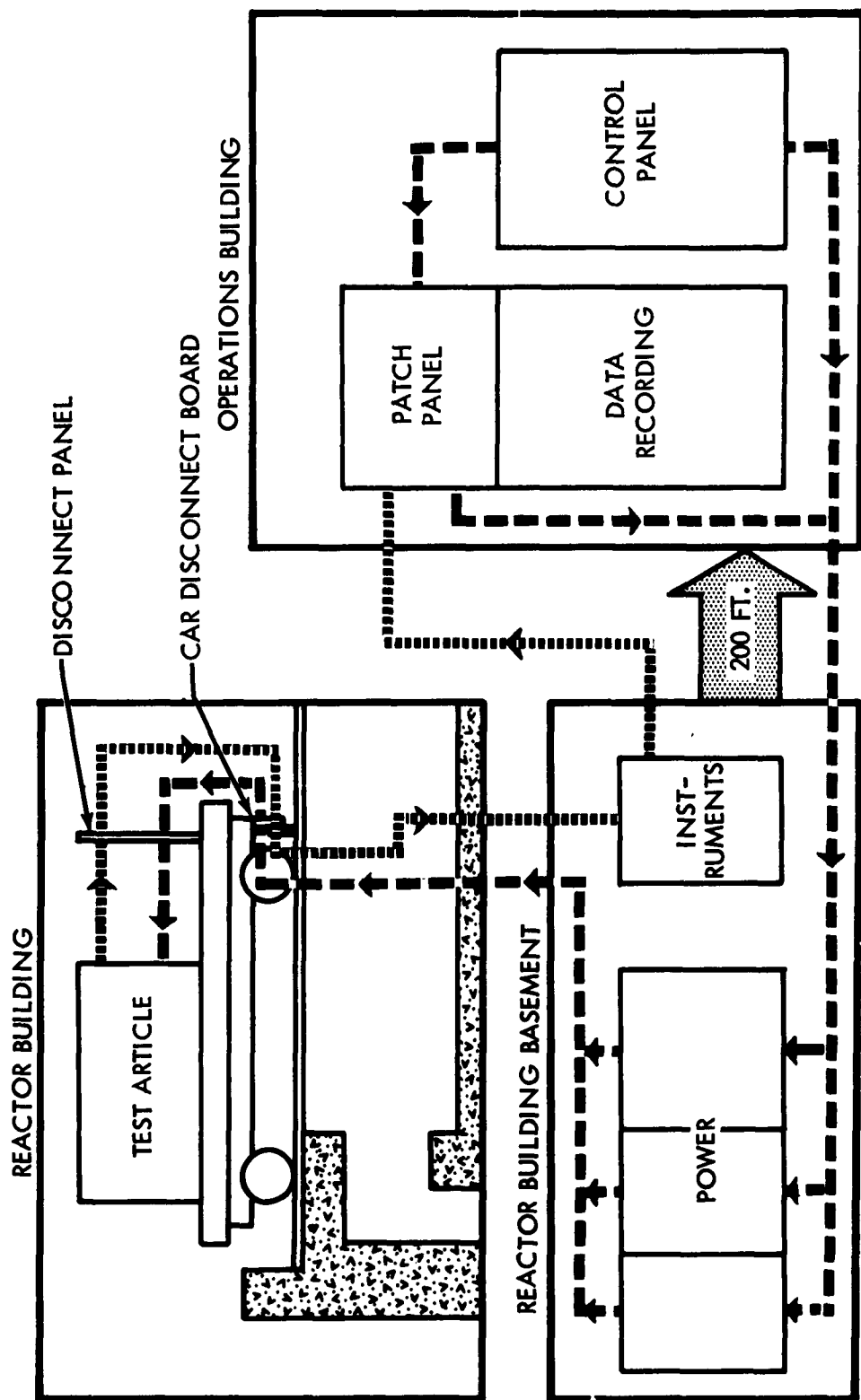


FIG. 3 REF CONTROL AND DATA FLOW

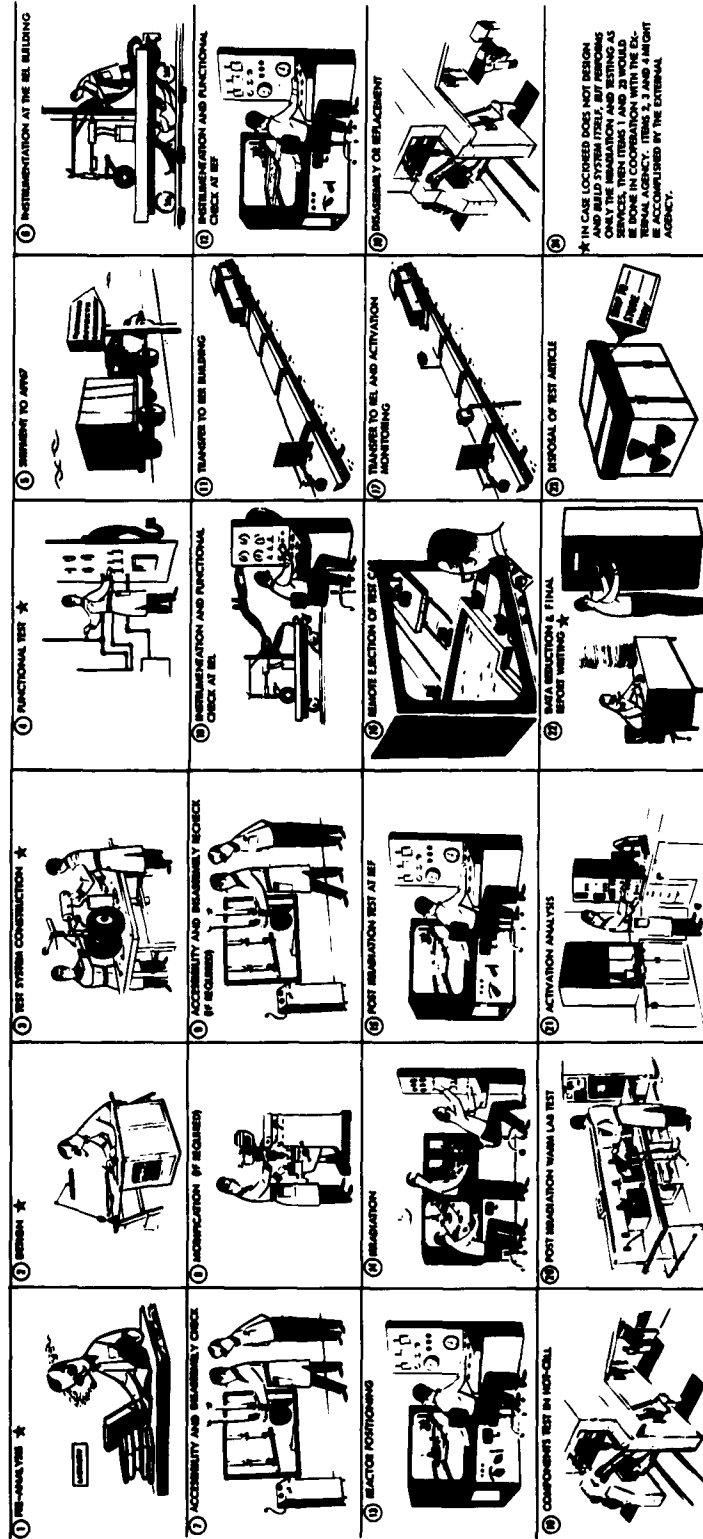


FIG. 4 TEST ARTICLE FLOW SHEET



FIG. 5 IRRADIATED POLYETHYLENE WIRE INSULATION

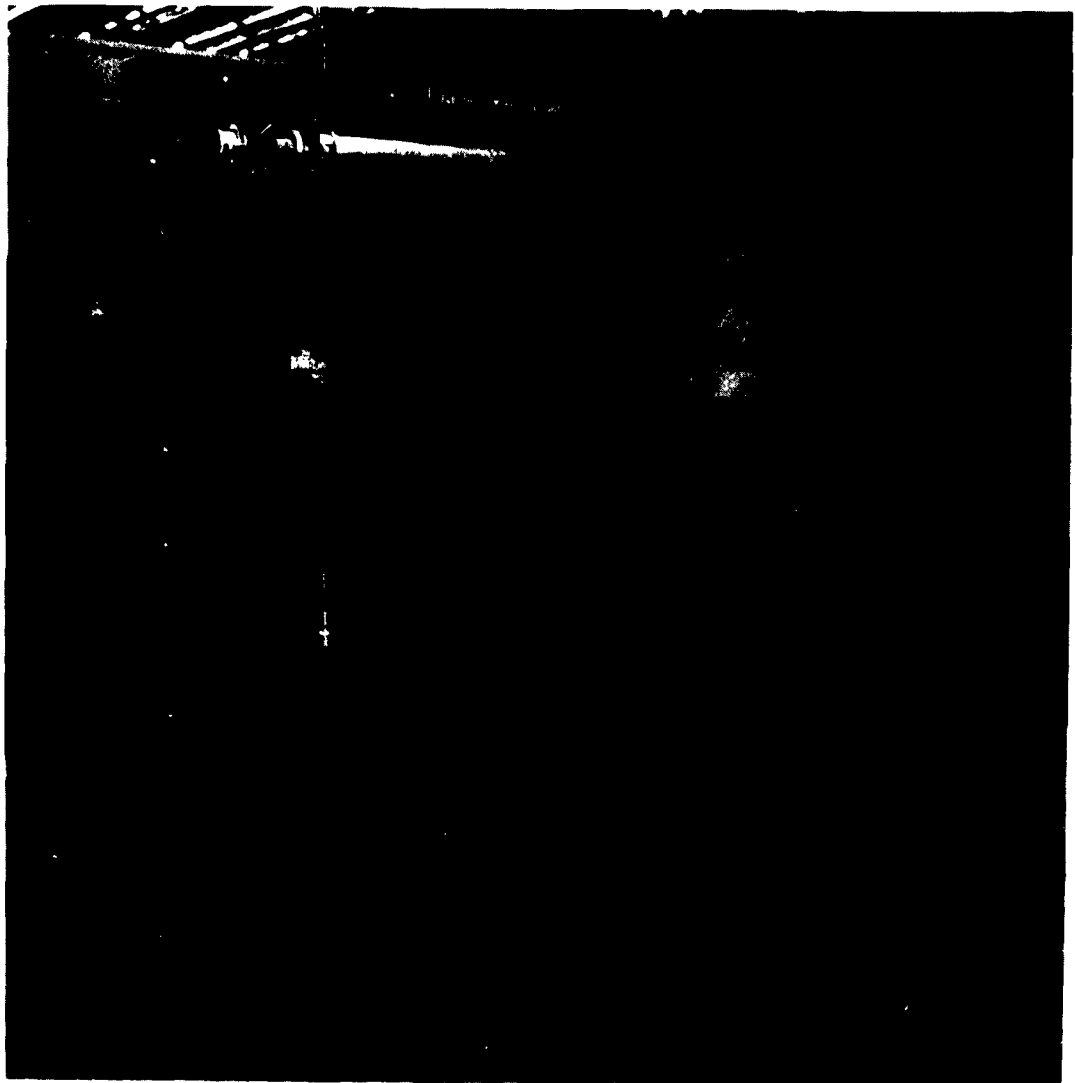


FIG. 6 YAW DAMPER TEST SYSTEM

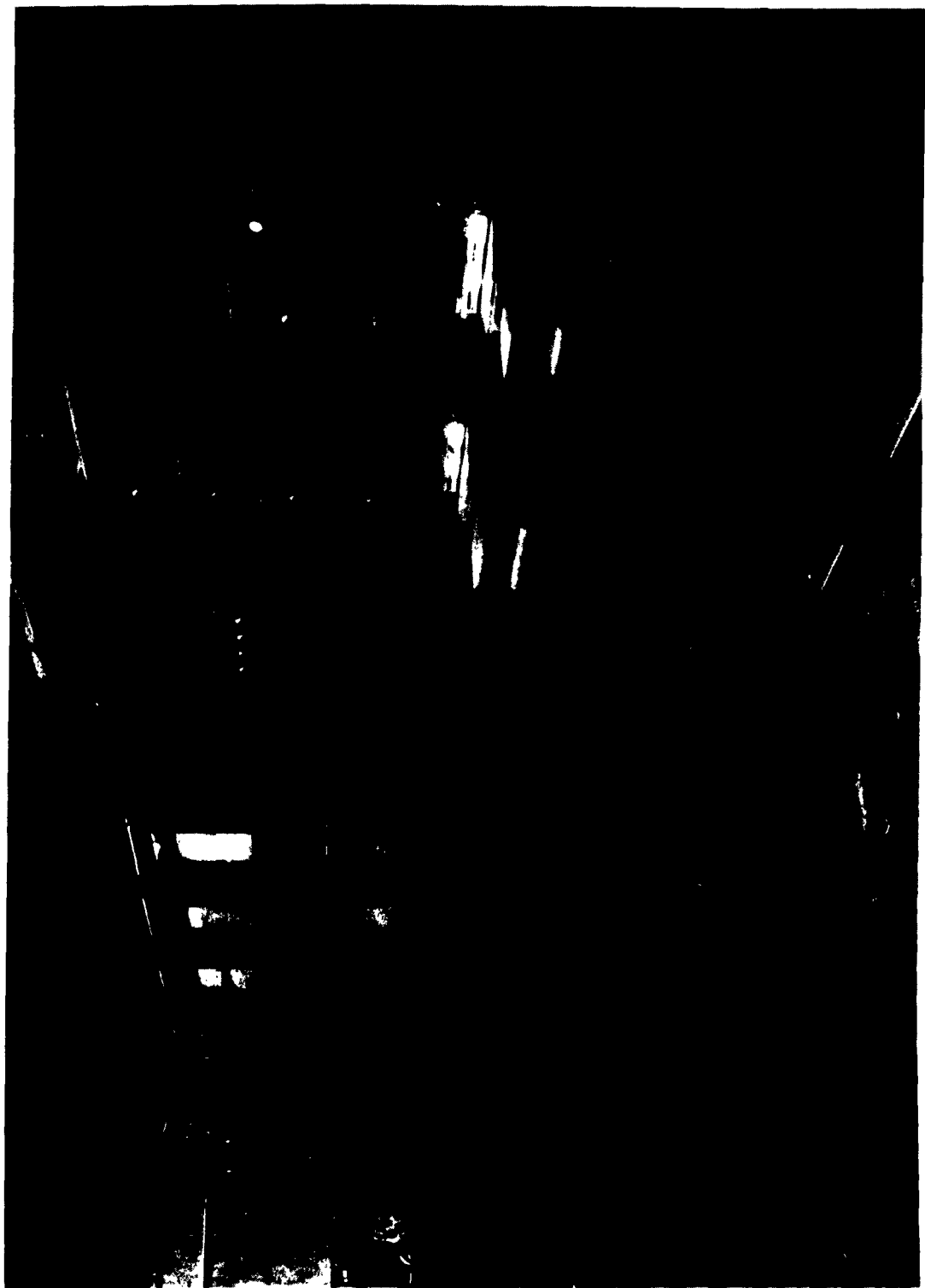


FIG. 7 INSTRUMENTATION MODULE MOCK-UP

**MISSION AND TRAFFIC CONTROL REQUIREMENTS
FOR THE WS/125A**

by

JAMES R. BURNETT

**Bendix Systems Division
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Ann Arbor, Michigan**

This paper is classified and is bound in Volume Six.

NEW RADIATION TEST FACILITIES IN THE
GENERAL ELECTRIC COMPANY

by

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This paper describes new radiation test facilities in operation or being installed by the General Electric Company in its General Engineering Laboratory, Schenectady, New York, and its Vallecitos Atomic Laboratory, Pleasanton, California.

The facilities in the General Engineering Laboratory include (1) a 15,000 curie gamma source, (2) a 1 mev, 10 kw electron accelerator, and (3) a 1 mev, 10 kw ion accelerator. In operation, the gamma source is placed, unshielded, in a room which is approximately 22 feet long by 12 feet wide by 10 feet high; thus large equipments such as motors, transformers, control equipment, and electronic systems can be irradiated under actual environmental and operating conditions for long periods of time and at dose rates up to about 500,000 roentgens per hour. The electron accelerator will provide a 1 million volt, 10 milliamperes beam current, giving a dose rate estimated at 10^{11} roentgens per hour. The ion accelerator will be used primarily as a source of fast neutrons. Using a deuteron beam and a beryllium target, a fast (greater than 1 mev) neutron flux of 10^{11} neutrons/cm²/sec with an accompanying thermal neutron flux of less than 10^{10} neutrons/cm²/sec is obtained over a volume of several hundred cubic inches.

Facilities at Vallecitos Atomic Laboratory for irradiation and testing of materials, components, and systems include, (1) the 30 MW General Electric Test Reactor, (2) the 30 MW Vallecitos Boiling Water Reactor, (3) a 30 KW Nuclear Test Reactor, (4) complete

hot cell services, and (5) versatile gamma irradiation facilities. The GETR combines the large fluxes of a high specific power tank-type reactor with the accessibility of a pool-type reactor. Fluxes above 10^{14} n/cm²/sec. of both fast and thermal neutrons are available. Space is provided for both capsule and loop experiments as well as for a wide variety of more unusual spatial requirements. The VBWR provides access for irradiation tests upon fuel elements and control rods within the reactor core together with other gamma-neutron irradiation spaces outside the pressure vessel. The NTR provides neutron fluxes up to 10^{12} n/cm²/sec. with the additional convenience of full control in adjusting the reactor performance to meet the specific needs of the experiment. A radioactive materials laboratory provides a full coverage of services ranging from preparation of assemblies for irradiation to post-irradiation measurements upon highly radioactive materials by remote operation. A variety of gamma irradiation services are provided using both cobalt-60 and fission product radiation sources to achieve dose rates as high as 10^7 r/hr.

INTRODUCTION

This paper describes new radiation test facilities either in operation or now being installed by the General Electric Company. These facilities have been financed entirely by General Electric Company funds and are located in its General Engineering Laboratory, Schenectady, New York, and its Vallecitos Atomic Laboratory, Pleasanton, California. These combined facilities will provide radiation sources and equipment for nearly all types of radiation work -- for studies on materials or machinery; for basic research or for qualification testing; and for studies using charged or uncharged, light or heavy particles.

FACILITIES IN THE
GENERAL ENGINEERING LABORATORY
SCHENECTADY, NEW YORK

The radiation sources located in the Company's General Engineering Laboratory in Schenectady include:

1. A 15,000 curie gamma source(a)
2. Two 1000 curie gamma sources
3. A 1,000,000 volt 10 kw ion accelerator(b)
4. A 200,000 volt, 200 watt ion accelerator
5. A 1,000,000 volt 10 kw electron accelerator(b)

The new facilities -- the two one million volt accelerators and the 15,000 curie gamma source -- are housed in a former overspeed test pit for generator rotors which has been rebuilt into a modern radiation laboratory. Because of the hazard associated with the high speed testing of large rotating machinery, the former facility was made up of a considerable mass of re-enforced concrete and the very substantial footings required for it. As a result this facility has been quite easily adapted to a radiation laboratory, although a considerable amount of concrete has been added for radiation protection.

A plan view of the radiation laboratory is shown in Figure 1. The original 46 ft. diameter test pit was practically covered over with an 8 ft. thick concrete cap. The gamma irradiation room, the electron accelerator and irradiation room, the ion accelerator, and the machine control room are all located on top of this cap. The walls of the test pit were composed of a three foot thick concrete wall, followed by 7 feet of sand and dirt, in turn followed by 4 more feet of concrete. The seven feet wide annular space has been excavated and enlarged to provide a maze and irradiation room for the ion accelerator. The test pit itself is reserved for a general experimental area.

The 15,000 Curie Gamma Source

The gamma facility has been designed to provide for the long-term irradiation of large equipments under operating conditions. Figure 2 shows the irradiation room, the source being inside the expanded aluminum screen near the center of the room. The room itself is approximately 22 feet by 12 feet wide by 10 feet high. The cobalt-60 is contained in stainless

-
- (a) Presently 6000 curies. To be 10,000 curies by the end of 1958, 15,000 curies by June 1959.
(b) Available by the end of 1958.

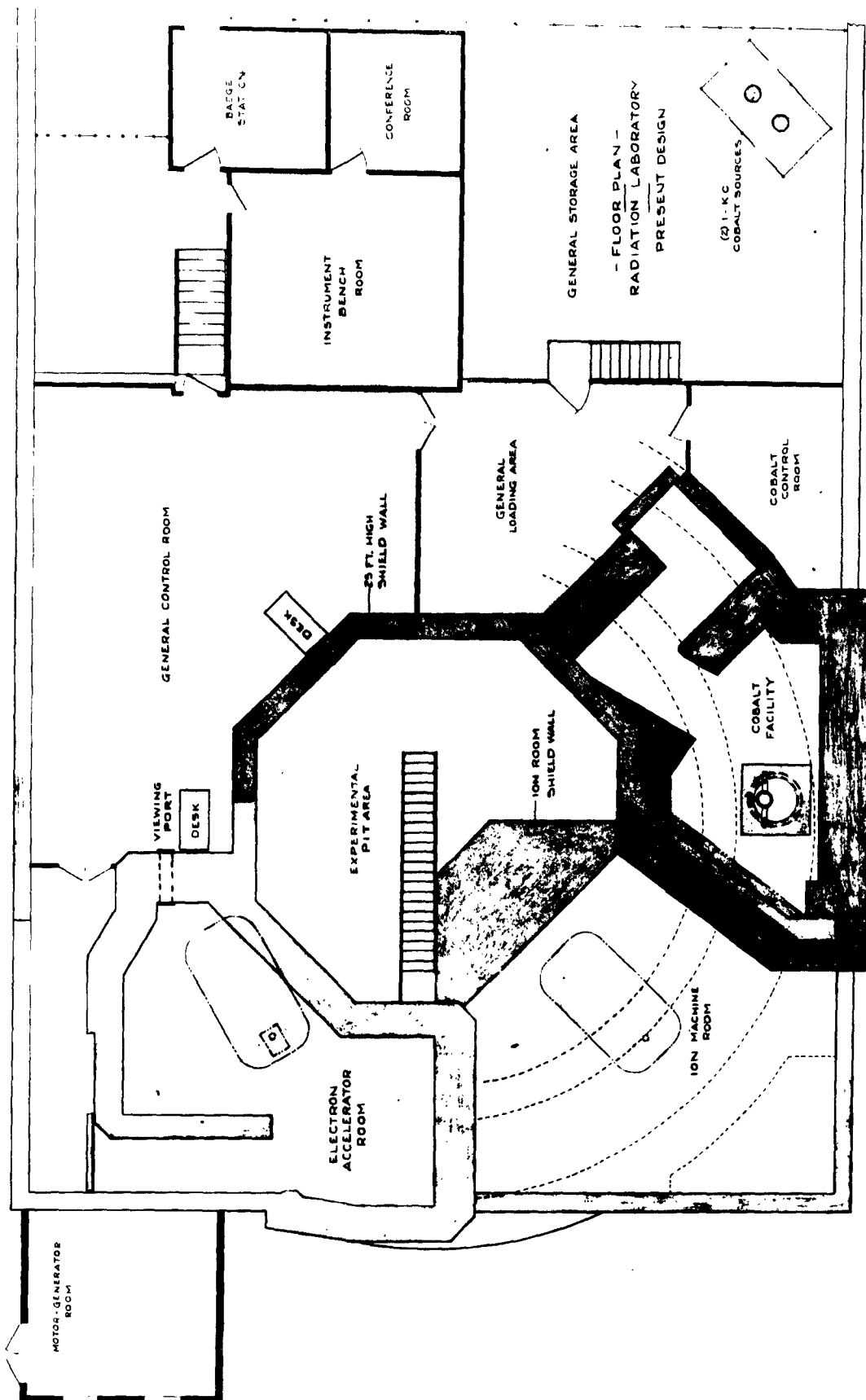


Figure 1

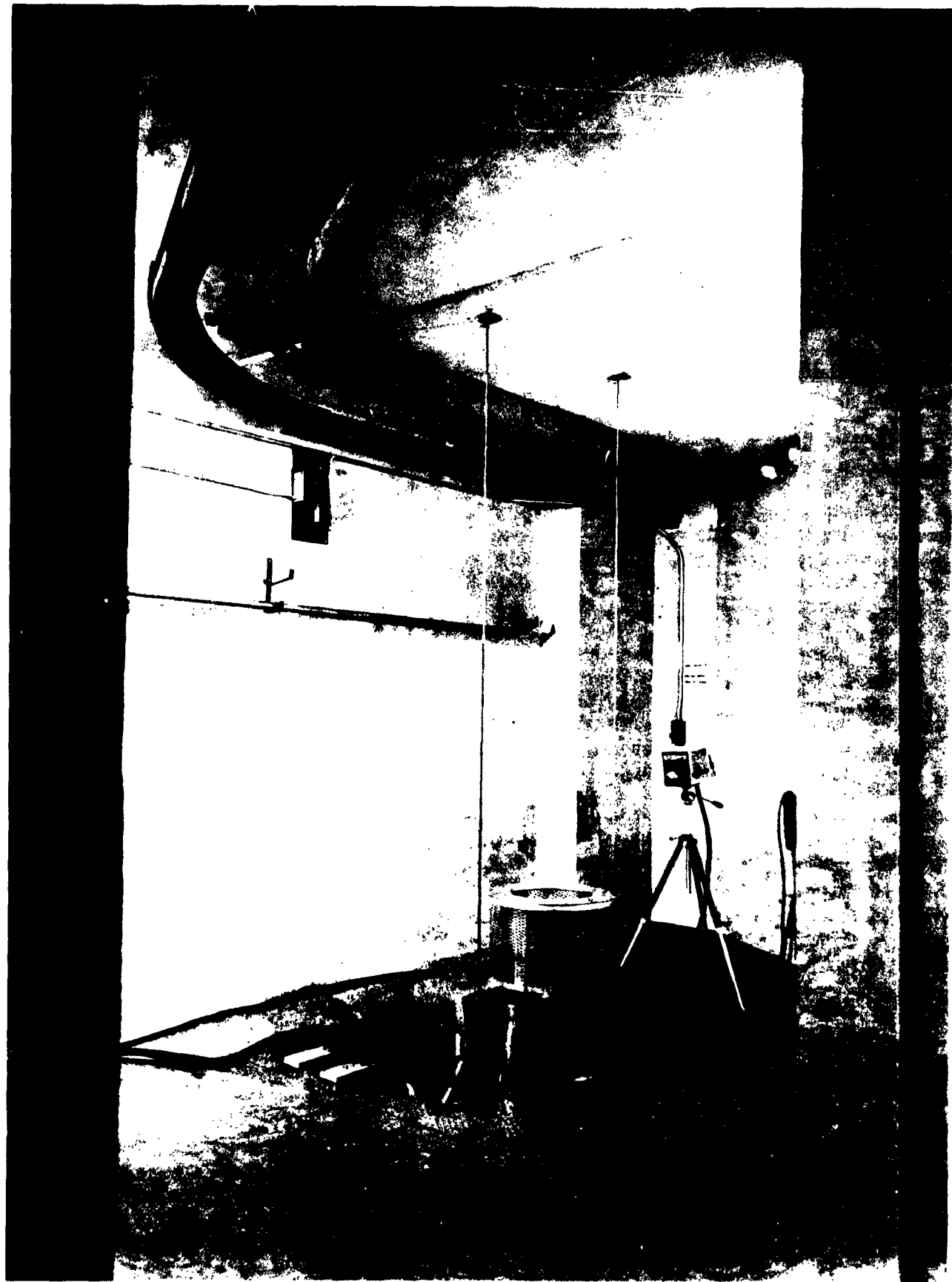


Fig. 2. Gamma irradiation room.

steel-clad plates, 14 inches long by 2 inches wide by 3/16 inch thick. Up to 20 of these plates may be placed around the periphery of the stainless steel "basket" shown in Figure 3. When the facility is not in operation, the basket is stored at the bottom of a 24 foot deep, water-filled well. After the equipment to be irradiated is set up in the room, the basket is raised into the room inside the expanded-aluminum screen. In this position the source is unshielded and any equipment in the room is irradiated at a dose rate that depends upon its distance from the source. Equipment located in the 15" diameter by 15" high expanded metal basket being held by the operator in Figure 4 will be irradiated at a dose rate of about 500,000 roentgens per hour; equipment located just outside the screen will receive a dose rate of about 90,000 roentgens per hour; while the dose rate in a far corner of the room will be about 1000 roentgens per hour. All these dose rates are for the 15,000 curie loading. In all instances the irradiations are carried out in the room and there is no need for encasing the samples in water-tight containers.

The gamma irradiation room is connected to its control room by 10 three-inch wiring ducts. This allows the equipment under irradiation to be connected to various instruments and sources of power, air, water, etc., located in the control room. Thus equipment can be tested and its performance monitored continuously during an irradiation. When leads must be kept short in order to make high frequency measurements, the instruments may be located in the irradiation room and read by means of a closed circuit television system. A water-filled window and system of mirrors supplement this system. Figure 5 shows the television system in operation.

Uses for the 15,000 Curie Source

There is at present little information on how equipment will operate under moderate or low radiation dose rates for long periods of time. Most of the data on radiation effects are derived from short time tests at high dose rates. The extent to which these data can be extrapolated to long time radiation effects at low dose rates under the combined conditions of radiation, time, temperature, humidity, mechanical stress, etc., is not well defined. This new gamma source represents a facility where large pieces of equipment such as motors, transformers, electronic circuits, and hydraulic systems can be tested under actual operating conditions for long periods of time -- months or, if need be, years. We feel that this facility will offer an excellent proving ground for more or less standard equipment which must operate at low dose rates, but for long periods of time.



Fig. 3. Stainless steel basket to hold CO-60 plates.

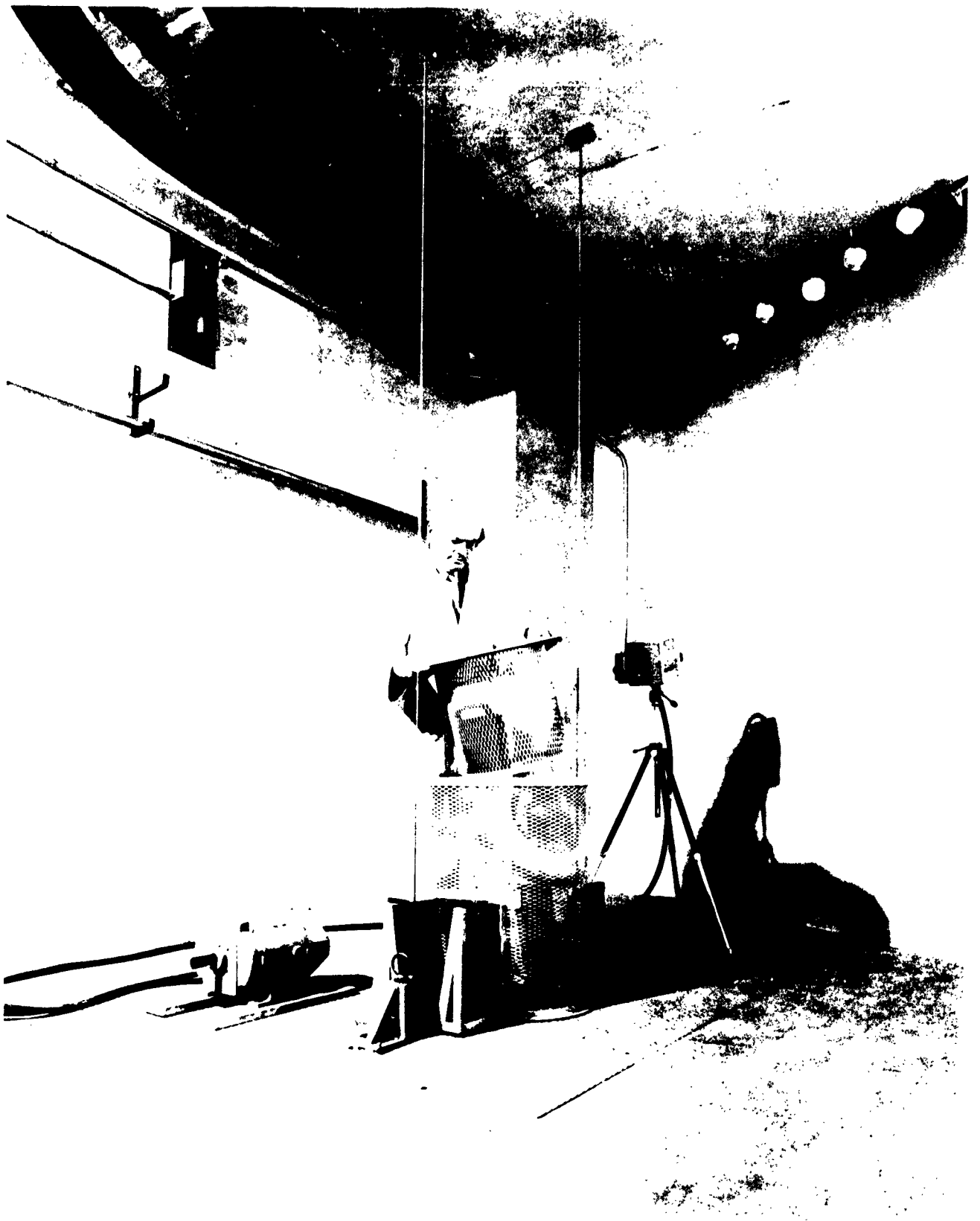


Fig. 4. Gamma source showing central irradiation basket.



Fig. 5. Closed circuit television in operation.

We do not wish to overemphasize the low dose rate aspects of this source, however. Equipment somewhat larger than a cubic foot in volume can be irradiated in the central basket at dose rates of about 600,000 roentgens per hour. Even higher dose rates may be obtained over smaller volumes by rearranging the cobalt plates.

The Electron Accelerator

Figure 6 shows the 1 million volt electron accelerator. This accelerator and the 1 million volt ion accelerator are of our own design and manufacture. They utilize a Cockcroft-Walton power supply and are designed to give a d-c beam current of 10 milliamperes at 1 Mev. The rather mammoth size relative to commercial machines results from the open construction. The open construction, rather than the pressurized construction of commercial machines, was used because of its low cost.

The primary feature of the electron accelerator is its high beam current and therefore its high dose rates -- we estimate a dose rate of about 10^{11} roentgens per hour a few inches from the window. Of course, utilizing this high dose rate will involve solving the heating problems that can arise from concentrating 10 kilowatts in a small volume, so that although the high dose rate cannot be used in many instances, there will be many experiments where it can be used.

Uses for the Electron Accelerator

Because of the low penetrating power of 1 Mev electrons (about 1/4 inch in a material the density of water) thin samples or liquids which can be stirred are required for use with this machine. We visualize that much of the work of this machine will be irradiating plastics or other organic materials, either for improving them with radiation or for radiation damage studies. A large number of samples can be irradiated in a short time by utilizing a system for scanning the beam over a width of several feet and a conveyor for moving the samples under the beam.

We emphasize the use of the machine on organic materials because, to a large extent, radiation damage in organic materials is a function only of the absorbed energy and not the irradiating particle. Thus the high dose rate of the machine makes it possible to conduct radiation damage studies in short times on these materials, where as much longer times would be required for reactor or gamma ray tests where dose rates could be several orders of magnitude lower.

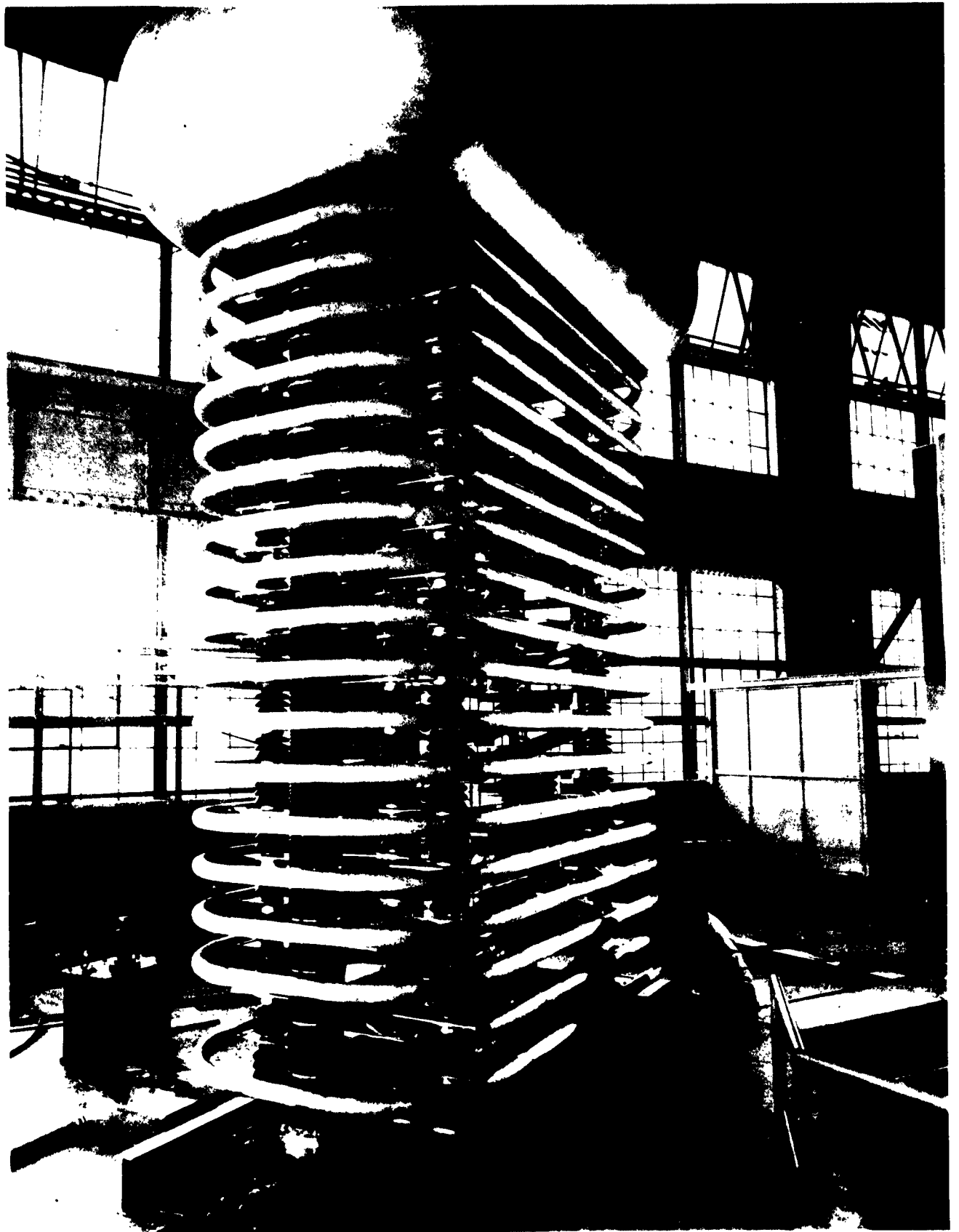


Fig. 6. 1 MEV electron accelerator.

The electron accelerator can also be used to a limited extent to study the effects of radiation on inorganic materials, since electrons, like neutrons, can create interstitial-vacancy pairs in crystal lattices. Although the two types of radiation are not entirely comparable, since an electron can effect only individual atoms, whereas neutron will affect large clusters of atoms, much work has been done studying the nature of radiation effects, especially in semiconductors, using electrons.

The Ion Accelerator

The ion accelerator is shown under construction in Figure 7. This machine is essentially identical to the electron accelerator except that it is of opposite polarity and contains an ion source instead of an electron source.

This accelerator will be used primarily as a source of high energy neutrons produced by bombarding a beryllium target with 1 Mev deuterons. This will yield a neutron intensity of about 10^{11} fast neutrons/cm² sec near the target. The neutrons will cover a spectrum of energies ranging from about 1 Mev to around 5 Mev. The thermal neutron flux on the other hand will be about 10^{10} neutrons/cm² sec if a water-filled moderator tank surrounds the target, and essentially negligible if no moderator tank is used. Thus the accelerator should provide a damage-producing fast neutron flux comparable with that of some reactors, with an accompanying thermal neutron flux that is two or more orders of magnitude below that found in a reactor. The fairly narrow range of energies of the fast flux will give simpler dosimetry problems than a reactor and at the same time the problems of induced radioactivity will be greatly reduced.

The ion accelerator picture isn't all rosy. While it has some important advantages over a reactor, it also has some important disadvantages. Since the neutrons are emitted more or less uniformly in all directions from the target, the intensity falls off inversely as the square of the distance from the target. Thus the high intensities are available only near the target and large samples would see a noticeable decrease in intensity at the outside of their volume. Thus the high dose rate figure applies only to samples a few cubic inches in volume.

A second disadvantage of the ion accelerator as a neutron source is that irradiations of more than a few hours or days become impracticable. We anticipate that continuous irradiations of more than a few days will not only be excessively expensive but will probably show an appreciable

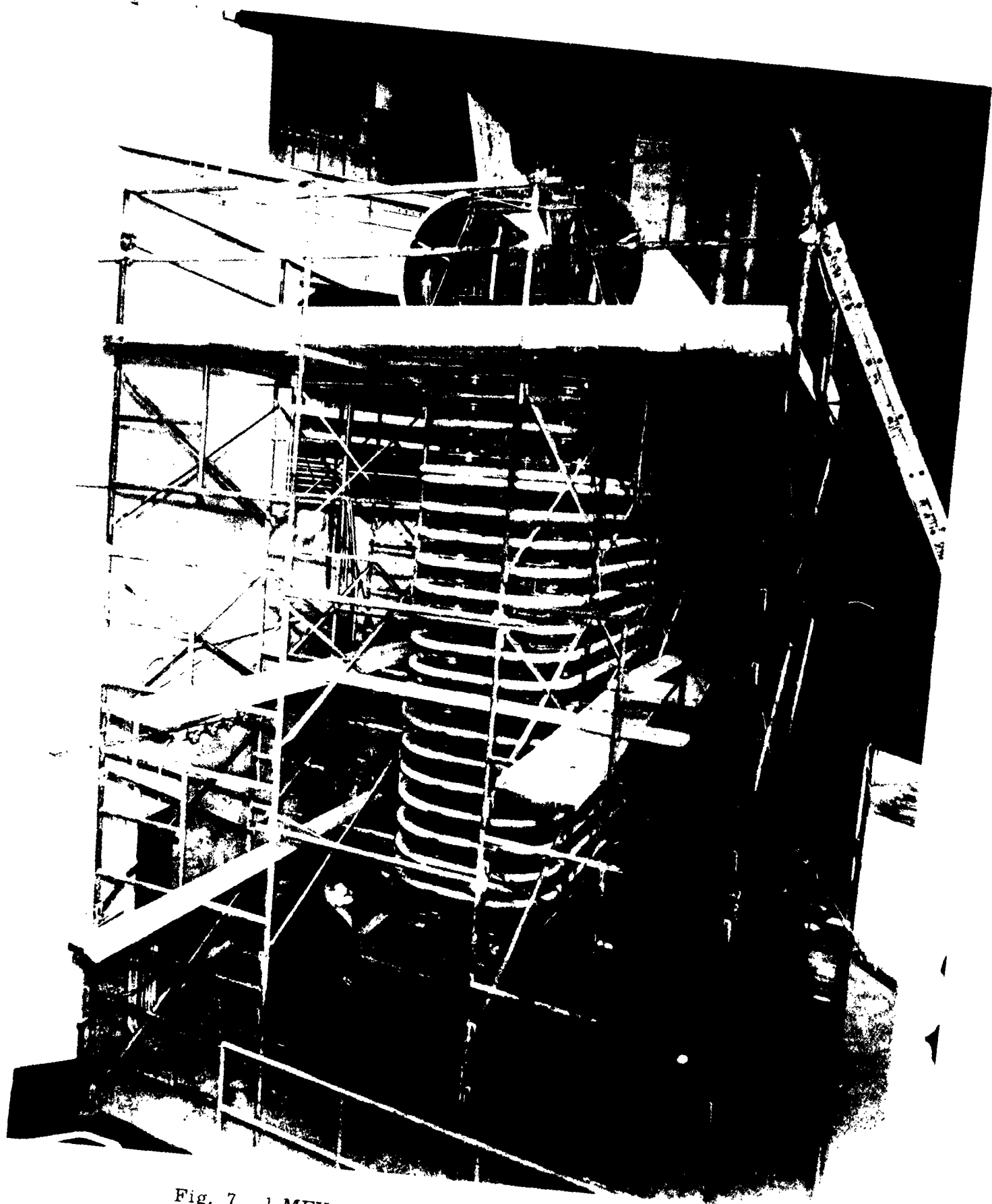


Fig. 7. 1 MEV ion accelerator during construction.

variation in available neutron dose rate because of gradual deterioration of the ion source.

Uses for the Ion Accelerator

As stated previously, we anticipate that the greatest use for this accelerator will be as a neutron source. As such it is well suited to rather basic research-type studies where reasonably mono-energetic neutrons are required and high total doses are not. The low residual radioactivity will allow relatively easy post-irradiation examination of samples. In addition to radiation effects studies, possible uses include using the neutrons to make radioactive isotopes for tracer studies, activation analysis, and inhomogeneity studies. Other possible uses include neutron cross section measurements, shielding studies, and studies of nuclear phenomena.

We have emphasized the use of the accelerator as a neutron source, but the possibilities of direct ion bombardment should not be neglected. Possible uses for direct ion bombardment include:

1. Simulating neutron damage studies in metals. Most metals are not affected appreciably by neutron bombardment except at very high doses -- doses that must be accumulated in months or years in a nuclear reactor. It is possible that this damage can be simulated in much shorter times by ion bombardment if the samples are thin enough that the ions can penetrate through the material. This greatly shortened bombardment time occurs for two reasons:

- a) There are more ions than neutrons, even in the highest flux reactors. A beam current of 10 milliamperes/cm² gives a particle density of 1.6×10^{15} ions/cm² sec compared with a figure that ranges from 10^{11} to less than 10^{15} neutrons/cm² sec for most reactors.
- b) The ion bombardment is essentially 100 percent efficient -- that is, in a sample of a thickness comparable to the range of the ions, each ion will interact at least once with the sample as it passes through. On the other hand, in a sample of reasonable thickness, probably less than one percent of the neutrons will interact with the material.

2. Surface treatment of material. Examples include:

- a) Making p-n junctions in a semiconductor by bombarding, say, n-type material with p-type ions.
- b) Introducing high concentrations of crystal defects on a surface to study certain types of surface phenomena.

The properties of these radiation sources are summarized in Figure 8.

FIGURE 8
COMPARISON OF NEW GENERAL ENGINEERING LABORATORY
RADIATION SOURCES

	<u>SOURCE</u>		
	<u>Gamma</u>	<u>Electron</u>	<u>Neutron</u>
Penetration	High (inches to feet)	Low (fraction of inch)	High (inches to feet)
Dose Rate	Low (500,000 R/hr)	High (10" R/hr)	High (10" fast n/cm ² sec 10 ¹⁰ thermal n/cm ² sec)
Sample Size	Large (cubic feet)	Thin Tapes or stirred liquids	Small (cubic inches)
Time of Irradiation	Long (Hours to years)	Short (Minutes to hours)	Short (Minutes to hours)
Sample Cooling	No Problem	Problem	No Problem
Induced Radioactivity	None	None	Low

FACILITIES LOCATED AT THE
VALLECITOS ATOMIC LABORATORY
PLEASANTON, CALIFORNIA

Vallecitos Atomic Laboratory (VAL) is a section of General Electric's Atomic Power Equipment Department. The Laboratory is this country's largest privately financed atomic research facility. It is located on a 1594-acre tract of land near Pleasanton, California. The Laboratory is dedicated to the peacetime applications of atomic energy and to reducing costs of nuclear power.

The purpose of this part of the presentation is to provide information about those facilities at VAL which are available for research and development activities in the radiation damage field. Some of these facilities have been completed during the past year. Others will begin operation within the next few months. These facilities, together with those at the Company's General Engineering Laboratory which have just been described, provide extensive radiation testing capabilities.

Facilities at the Vallecitos Atomic Laboratory for irradiation and testing of materials, components, and systems, include, (1) the 30 MW General Electric Test Reactor (GETR), (2) the 30 MW Vallecitos Boiling Water Reactor (VBWR), (3) a 30 KW Nuclear Test Reactor (NTR), (4) complete hot laboratory services, and (5) versatile gamma irradiation facilities. Each of these facilities is described briefly in the following pages.

General Electric Test Reactor

General Design Features - The most recent major addition to the facilities of the Vallecitos Atomic Laboratory is the 30 MW General Electric Test Reactor (GETR). This facility represents a new concept in test and research reactors. It permits a wide variety of simultaneous high-flux experiments to meet the needs of a diversified research program. The advantages of high specific power in a tank-type reactor and the large, easily accessible experimental space of a pool-type reactor are combined in the GETR.

The reactor core is contained in an aluminum pressure vessel which is submerged in a pool. The pool serves as a reflector and provides a large, flexible, irradiation zone. This arrangement permits freedom in the location, size, and configuration of loops and capsules. At the same time, high-flux investigations may be conducted in loops and capsule positions in the reactor core.

With this flexible arrangement, the GETR can be used to perform engineering tests on reactor fuel and components, general industrial testing, isotope production, and neutron and gamma research. It is anticipated that this facility will be in operation by the end of 1958.

Reactor Structure - The GETR is housed in a steel containment building. This building, together with the other buildings of the facility, are shown in Figure 9. This picture indicates the state of facility construction in August, 1958. Figure 10 shows a section through the center of the reactor containment building. The interior of the building is a four story structure, containing the reactor pool and associated canal and various areas for equipment connected with reactor operation and experiments. The reactor is located in the bottom part of the pool near the center of the building. An artist's conception of the GETR pool assembly is shown in Figure 11. The three foot long core is contained in a two foot diameter aluminum pressure vessel where cooling is maintained by a forced convection down flow system. The four large pipes into the pressure vessel are inlets and outlets for the coolant. The bottom mounted control drives allow the upper portion of the reactor to remain relatively unobstructed for access to experiments and fuel.

Advantage is taken of the high neutron leakage core by placing many experimental facilities in the pool surrounding the pressure vessel. Some examples of such pool facilities are shown in Figure 11. They are (1) hairpin loops, (2) capsule tubes, and (3) a beam port for a variety of physics experiments. In addition, Figure 11 shows three internal through loop facilities.

During operation, the reactor is cooled by circulation of high purity water through the pressure vessel. The water is pressurized at 140 psia and coolant inlet and outlet temperatures are 120°F and 140°F, respectively. The pool water surrounding the pressure vessel is circulated and cooled by a separate cooling system.

Figure 12 is a horizontal section through the pool and pressure vessel at the level of the reactor core. A variety of core and pool irradiation facilities are shown. The core provides three 3" by 3" vertical openings for through loops. Also, there are two types of core capsule positions having diameters of 1.5". Eight of these positions are located in beryllium filler pieces in the core and eight are in beryllium reflector pieces surrounding the core inside the pressure vessel. In addition, there are eight more core



Fig. 9. Site picture of GETR.

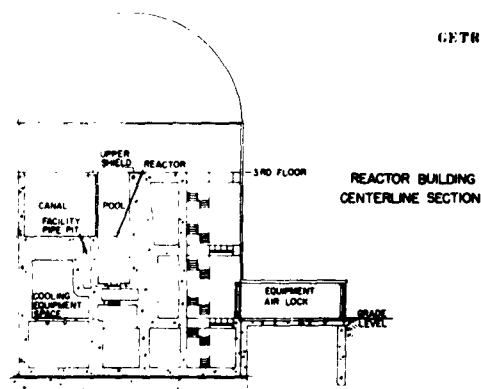


Fig. 10. GETR building centerline section.

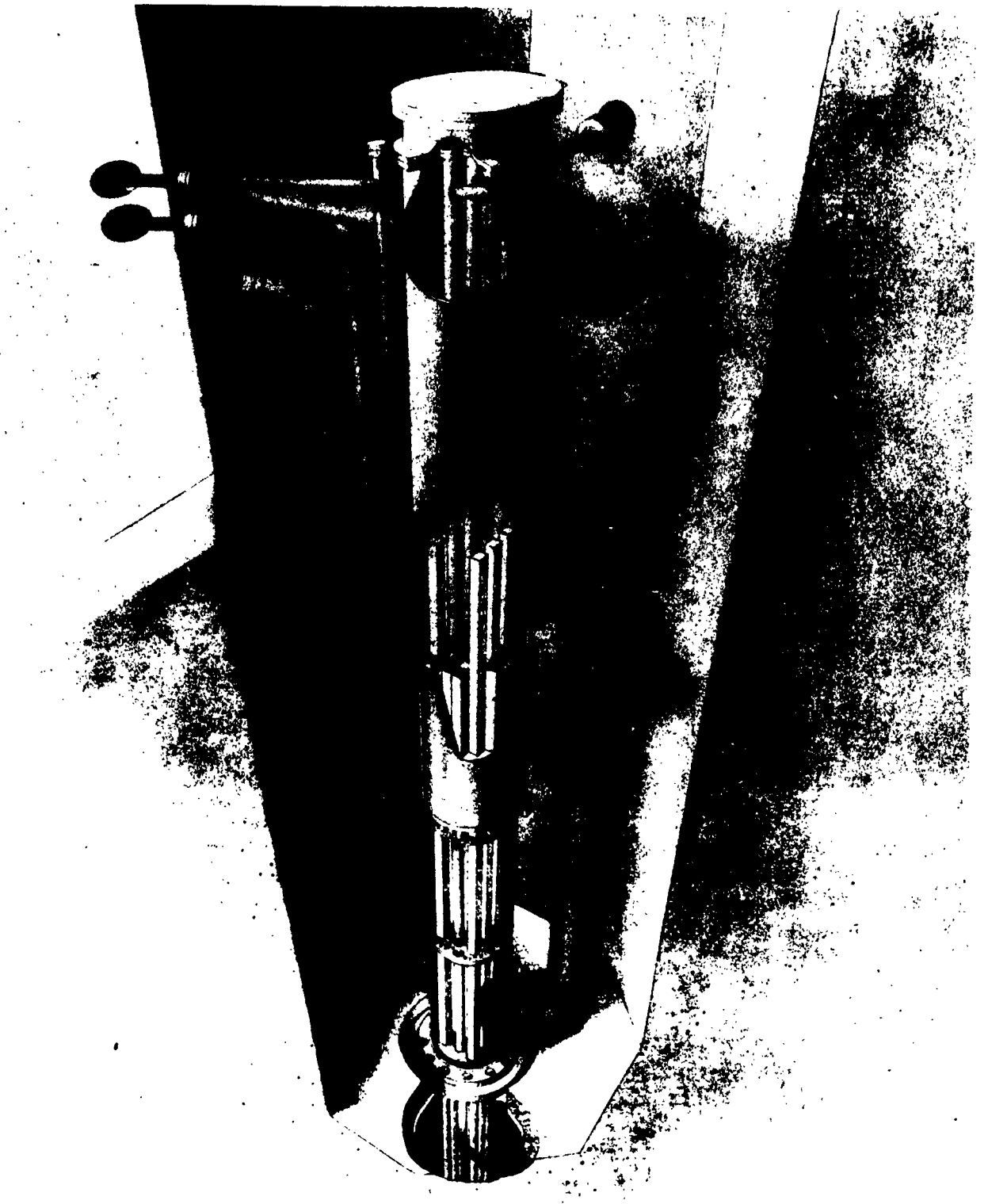


Fig. 11. GETR assembly.

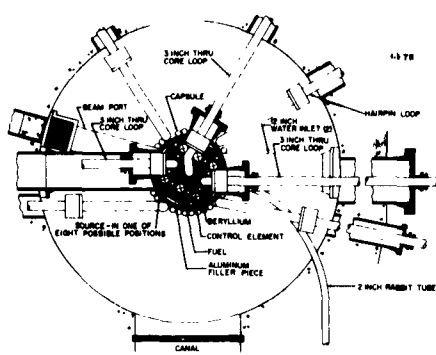


Fig. 12. Horizontal section through GETR pool and reactor core.

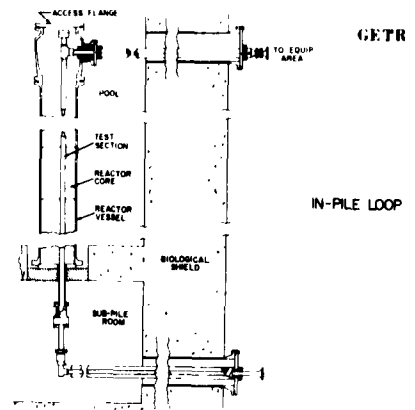


Fig. 13. GETR in-core loop.

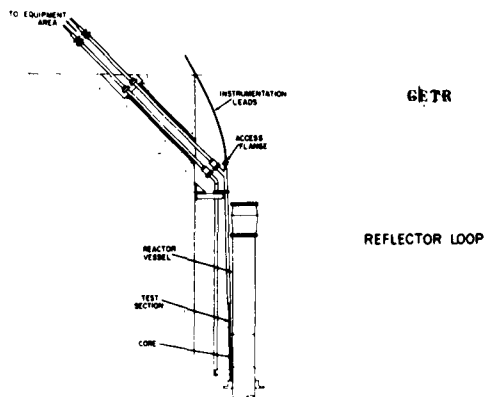


Fig. 14. GETR reflector loop.

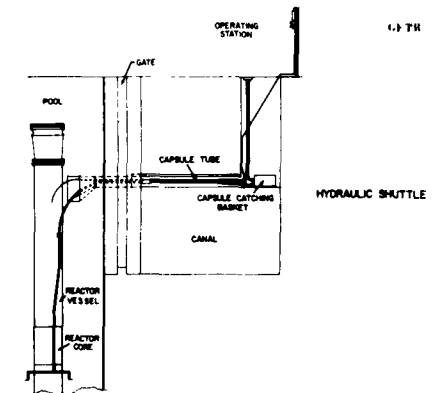


Fig. 15. GETR hydraulic shuttle.

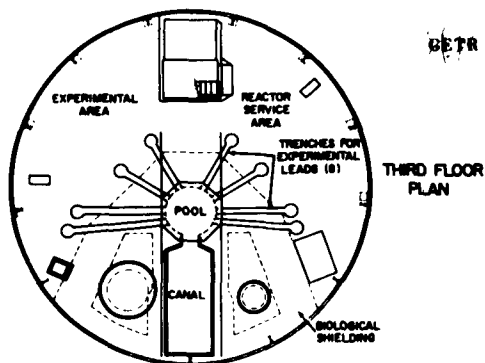


Fig. 16. GETR third floor plan.

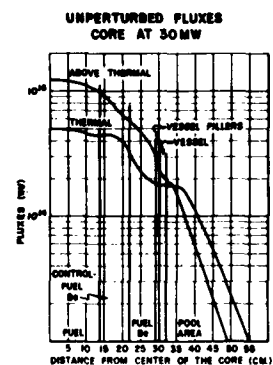


Fig. 17. GETR unperturbed neutron fluxes.

capsule positions in the beryllium reflector with a diameter of 0.5" (not shown in Figure 12).

Just outside the pressure vessel and opposite the core, there are 31 more capsule irradiation positions. Twenty-nine of these positions have a diameter of 1.5" and two have a diameter of 2.875". Space is also provided for four hairpin loops of the general design shown in Figure 11. Other pool facilities include a 1.85" diameter hydraulic shuttle tube and an 8" diameter beam port. In addition to the specific irradiation positions indicated, a variety of other spatial arrangements are available in the pool to meet the requirements of the experimenter.

Figures 13, 14, and 15 are vertical sections through the pool showing how various types of irradiation facilities are installed. The inlet and outlet connections for an in-core loop go horizontally through the pool wall near the top and bottom of the pressure vessel (Figure 13). Access to the pool may also be obtained through 18" diameter openings which emerge at an angle of 45° through the third floor at the top of the pool (Figure 14). Figure 15 shows the structure of the hydraulic shuttle tube which provides short term irradiations for capsules just outside the pressure vessel wall. A 4 foot minimum radius of curvature in the tube provides passage for capsules approximately 5" long.

Arrangements at the top of the pool are shown in the third floor plan in Figure 16. Eight trenches, 12" wide and 6" deep, emerge from the edge of the pool and contain experimental leads connected with irradiation activities. The service canal is separated from the pool by a removable steel panel. A steel missile shield covers the pool during reactor operation.

Radiation Intensities - Because of the high power density of the GEIR, very intense neutron and gamma radiation fields will be encountered. Information about the calculated unperturbed fast and thermal neutron fluxes is given in Figure 17. The plot shows the variation of flux with radial distance from the center of the core. The design of the reactor is such that substantial fluxes will be found well out into the pool area. High fast-to-thermal neutron flux ratios are also observed. The absolute flux values in Figure 17 are actually higher than those which will be found in the operating reactor due to the build-up of poisons in the core, presence of neutron absorbing experimental equipment, etc. However, the fast neutron flux will be considerably less affected than the thermal flux, resulting in an even higher operating fast-to-thermal ratio.

More realistic estimates for the thermal neutron fluxes which will be encountered during reactor operation are given in Figure 18. This illustration gives the flux pattern in a horizontal cross section of the core. Perturbed thermal flux values have been given for many of the experimental locations previously described. The maximum thermal flux is about 2×10^{14} per cm^2 per sec. and a flux of about 1×10^{14} is found at the pool capsule locations. The corresponding maximum perturbed fast neutron flux is about 9×10^{14} per cm^2 per sec. The perturbed fast and thermal fluxes are about equal at the pool capsule locations. The availability of such high neutron fluxes makes possible accelerated irradiation tests, often at substantially higher fluxes than those encountered in practical applications.

Gamma heating in the reactor core will be as high as 20 watts per gram in the high flux positions. This is reduced to about three watts per gram in the pool capsule locations.

In summary, the General Electric Test Reactor has been designed to be a very flexible irradiation facility, combining a high degree of accessibility to large volumes with intense radiation fields and a high ratio of fast-to-thermal neutrons.

Vallecitos Boiling Water Reactor

The VBWR was constructed to serve the following purposes:

1. A test facility for certain critical components and parameters of the reactor GE is building for the Commonwealth Edison Co., at Dresden, Illinois.
2. A tool for the development, demonstration and testing of other boiling water reactor designs.
3. A training aid in the operation of a boiling water reactor.
4. A research tool to provide reliable information concerning stability, load response characteristics, safety and fuel economy of boiling water reactors.

The reactor has been in operation at high powers for about a year. It is fulfilling well its intended purposes. The reactor facility is shown in Figure 19 during a typical period of full power operation.

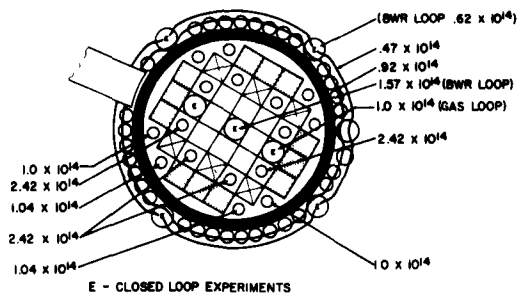


Fig. 18. GETR perturbed thermal neutron fluxes.



Fig. 19. Site picture of VBWR.

The VBWR is being brought to the attention of this meeting because of its high degree of usefulness in carrying out certain radiation tests. The types of tests referred to are (1) in-core irradiations of fuel and control element materials, (2) irradiation corrosion tests within the pressure vessel outside the core, and (3) radiation tests conducted in instrument tubes just outside the pressure vessel.

The details concerning the design and operation characteristics of the VBWR have already been described elsewhere⁽¹⁾ and will not be considered here. However, Figure 20, which presents a section view of the reactor, will be helpful in connection with radiation testing considerations. The reactor normally operates with high purity water as coolant, at a pressure of 1000 psig and a temperature of 545°F. Feedwater enters the pressure vessel above the core and flows down through baffles to the plenum below the core, then up through the fuel elements. Water and steam then leave the core, water to flow down around the outside of the core to re-enter at the bottom, the steam to exit through the baffle and pipe at the top of the vessel. For forced circulation, the core baffles in lowered position (as shown in Figure 20) prevent downward flow of water. Instead, the water is pumped through an extra loop and re-enters the vessel as feedwater.

The average and maximum thermal neutron fluxes in the core are 2×10^{13} and 5×10^{13} per cm^2 per sec., respectively. Lower fluxes are available in the instrument tubes (2 - 3 tubes are available). Fuel and control materials and corrosion test specimens can be mounted within the pressure vessel by conforming to requirements for supporting the specimens. Also, specimen materials must be compatible with the general environment inside the pressure vessel. The instrument tubes are straight and will accommodate samples with outside diameters up to 3". Service connections into the tubes to the irradiated materials are readily made.

Nuclear Test Reactor

The NTR is a fully-enriched, water-cooled and moderated, graphite reflected thermal research reactor with a full power rating of 30 kilowatts (heat). This facility is a valuable physics laboratory tool enabling one to perform a wide range of basic nuclear experiments. However, it is considered in the present discussion because of its utility as a flexible source of high intensity reactor radiations. An interior view of the NTR cell is pictured in Figure 21.

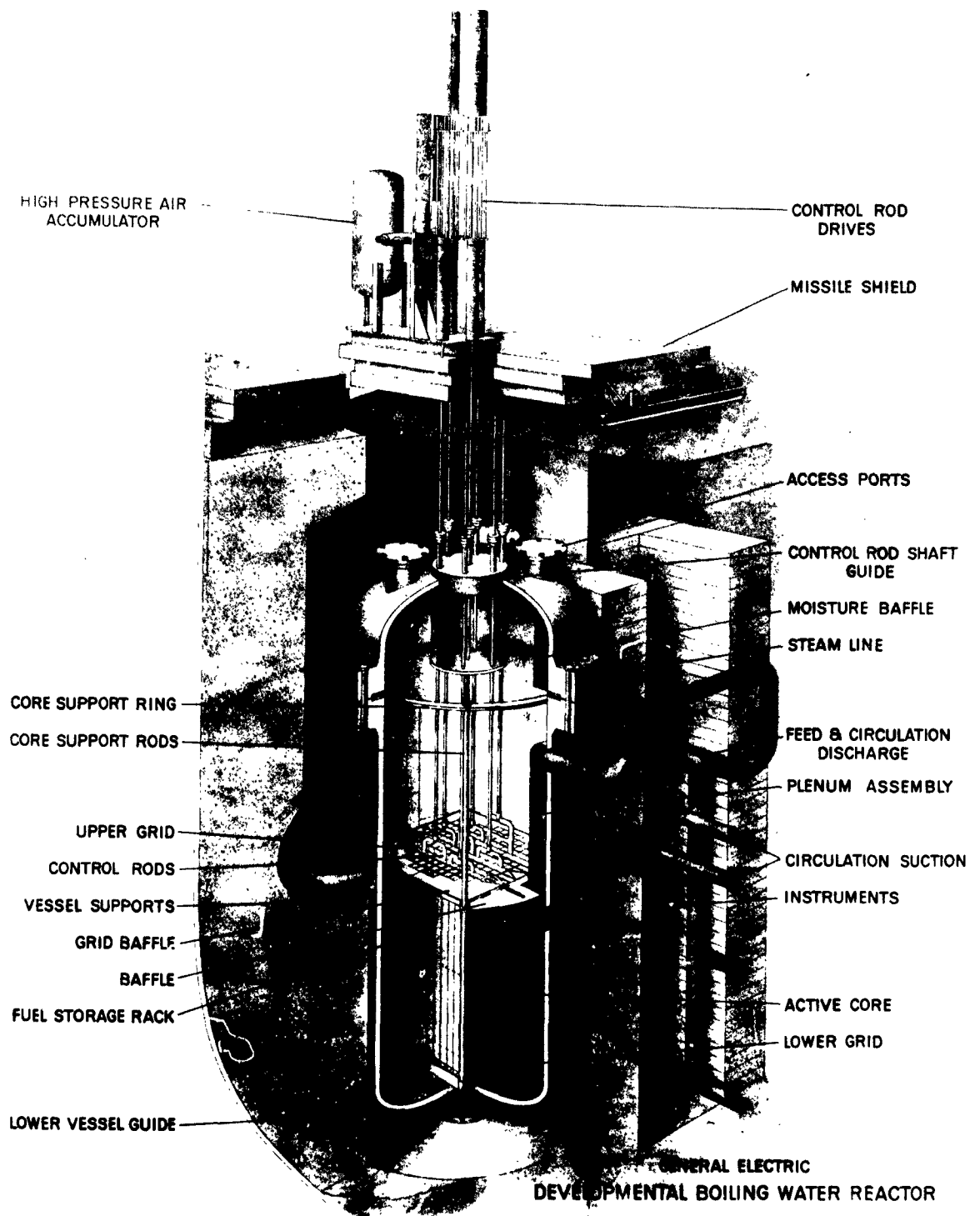


Fig. 20. Section view through VBWR pressure vessel.

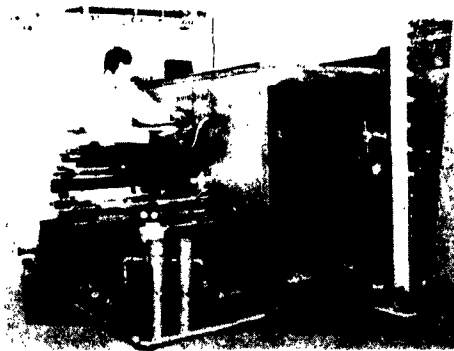


Fig. 21. Interior view of NTR cell.

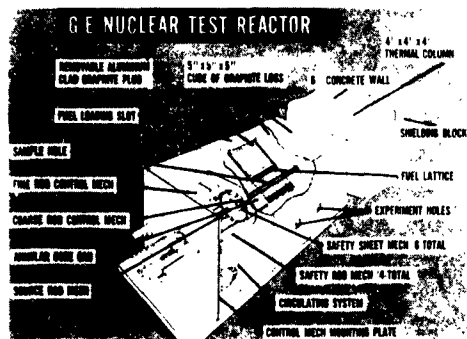


Fig. 22. Section view of NTR structure.

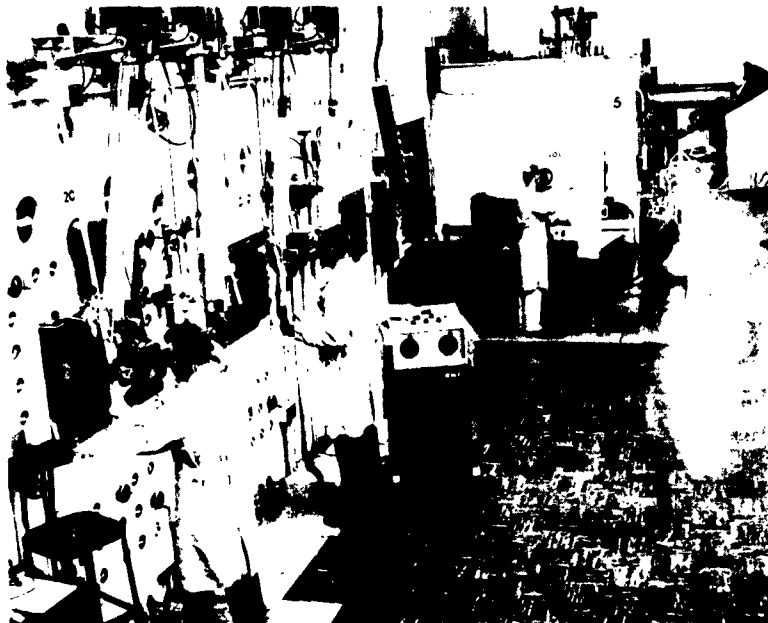


Fig. 23. Floor activities at the RML hot cells.

The structure of the NTR is shown in Figure 22. This reactor essentially consists of a core, a horizontal annular aluminum cylinder containing moderator and fuel, centered in a reflector, a 5' cube of graphite. The active fuel lattice of the reactor is contained in a horizontal annular aluminum can 19" long, 18" in diameter, and having an inner annular diameter of 12". Sixteen fuel elements are placed into the 3" annulus thus formed. Each fuel element consists of a number of U-Al alloy fuel discs on an aluminum support rod forming a skewered type assembly. The can containing this fuel assembly is completely filled with high purity, demineralized light water which serves as both a moderator and a coolant. The central hole of the annular aluminum can is completely filled with a 12" (diameter) cylinder of reactor-grade graphite. The graphite cylinder encloses the active length (19") of the central sample tube. This is an aluminum tube with an inside diameter of 3" which extends along the central axis of the assembly for the entire length of the reactor.

The horizontal central sample tube is the principal irradiation facility of the NTR. Samples are positioned in the tube by means of graphite plugs which are so coupled that they fill completely the unused length of the tube. The flux along the 19" active length of the sample tube during full power operation is about 10^{12} neutrons per cm^2 per sec. The tube emerges from the center of the reactor through an external graphite thermal column. The NTR is also provided with a second irradiation location at lower flux. This is a vertical thimble in the form of a 4" x 4" x 5' aluminum can which extends through the main graphite pack of the reactor and is approximately tangent to the annular fuel container.

In carrying out irradiations with the NTR, the reactor's flexibility is of considerable importance. With this facility, one has the convenience of full control in adjusting reactor operation to meet the specific needs of the experiment.

Hot Laboratory Services

The preparation of assemblies for irradiation and the subsequent examination of the irradiated systems are important parts of almost any radiation testing program. Post-irradiation examinations of reactor irradiated materials are often made quite difficult because of the radioactivities generated from neutron capture. In view of the need for such services, a fully equipped Radioactive Materials Laboratory (RML) has been put into operation at the Vallecitos Atomic Laboratory.

The heart of the RML is the hot cell area. Four high-level hot cells have been provided, each with a shielding capacity for more than a million curies of mixed fission products. Each cell has an operating area 17' long by 6-1/2' wide and is 14' high inside. There are three operating stations in each cell and each station is equipped with a lead glass window. In addition, the stations are equipped with master-slave manipulators. The general cell area is also equipped with remotely controlled manipulators and overhead cranes. There are many access holes in the cell walls which allow the introduction of various electrical, hydraulic and/or mechanical services into the cells. Figure 23 shows a view of activities in one fourth of the hot cell area.

Certain stations in the hot cells are often set up to perform specific operations. Figure 24 gives a view of activities at a hot cell lathe station. In another case, an entire separate low-level cell has been constructed for handling metallographic specimens (shown in background of Figure 23). Specific tools are provided for many other types of examinations of radioactive material. In addition, the RML is equipped with a large water pool, dry storage vaults, and numerous other facilities connected with high level radiation work.

With these facilities, the RML is able to provide a wide range of irradiation services. Systems to be radiation tested can be designed, built and given pre- and post-irradiation examinations. The following operations are representative of the large number of specific remote operations which can be performed in the RML:

1. Uncanning of specimens from capsules.
2. General machining, cutting and disassembly operations.
3. Dissolution and radiochemistry operations.
4. Visual and stereomicroscopic examinations and photographs
5. Weight, density, and dimension measurements
6. Metallographic preparation and examination
7. Hardness, tensile, impact, bend testing, et al.
8. Thermal expansion, thermal conductivity measurements et al.
9. Performance tests on mechanical and electronic equipment
10. Welding and mass spectrographic leak detection



Fig. 24. An operation at a RML hot cell lathe station.



Fig. 25. Irradiation operations at the cobalt-60 facility.

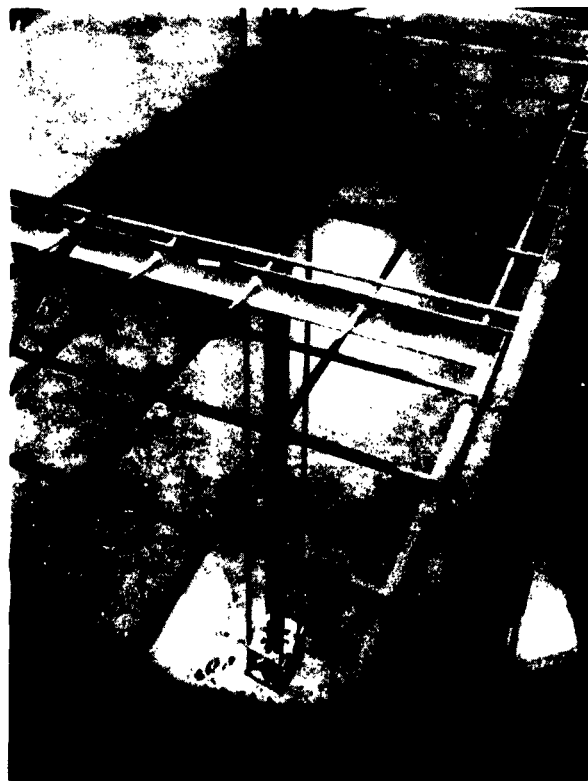


Fig. 26. Structure of the cobalt-60 facility.

Gamma Irradiation Facilities

Two types of water-shielded gamma irradiation facilities are now in operation at the Vallecitos Atomic Laboratory. One type employs cobalt-60 as the radiation source while the other uses irradiated fuel elements from the Vallecitos Boiling Water Reactor. The present installations provide dose rates as high as 10^6 roentgens per hour. It is anticipated that both types of facilities will be expanded with the operation of the GETR to provide larger irradiation spaces at dose rates up to 10^7 roentgens per hour.

Irradiation operations at the cobalt-60 irradiation facility are shown in Figure 25. A sample is about to be lowered into a watertight pipe which extends down into the radiation source. The source itself, which provides dose rates up to 1.5×10^6 roentgens per hour, is shown in Figure 26. It consists of two cylindrical capsules, each 1.2" in diameter and 9" long, containing the radioactive cobalt and mounted in a rack which is shielded by eight feet of water. This unit is in the process of being expanded to a total strength of 10,000 curies mounted in ten such capsules. Later provisions are to include a perforated horizontal source base plate so that the ten capsules can be arranged like pegs in a punch board to achieve a more flexible facility.

The fission product radiation source is in a separate pool near the VBWR where irradiated fuel elements can be used conveniently close to the reactor. A similar fuel element irradiation facility providing dose rates as high as 10^7 roentgens per hour will be located in the canal at the GETR when irradiated fuel is available from the latter reactor.

In addition to the use of cobalt-60 gamma sources locally, plans are underway to produce this material for sale as high intensity radiation sources. The GETR will provide the large fluxes which are required to achieve high specific activities. The facilities of the RML will be employed to fabricate the radioactivity into sources of various kinds. It is expected that cobalt-60 will be produced at the rate of about 100,000 curies per year at specific activities of about 50 curies per gram of metallic cobalt.

Summary

A brief description has been given of five facilities at the Vallecitos Atomic Laboratory which are available in

connection with irradiation tests upon materials, components, and systems. The facilities are (1) the 30 MW General Electric Test Reactor, (2) the 30 MW Vallecitos Boiling Water Reactor, (3) a 30 KW Nuclear Test Reactor, (4) complete hot laboratory services, and (5) versatile gamma irradiation facilities. These units, together with those at the Company's General Engineering Laboratory which have been described earlier, provide extensive radiation testing capabilities.

REFERENCES

- (1) Nucleonics Reactor File No. 4 - Reactors on the Line -
VBWR Nucleonics - February 1958

THE CONVAIR RADIATION EFFECTS TESTING SYSTEM

by

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This paper and an accompanying movie describe the radiation effects testing system at Convair-Fort Worth, including the shuttle system to transport specimens in special environmental chambers. The hardware and controls necessary to meet the environmental criteria and the versatility of the system are described.

INTRODUCTION

Experience gained at Convair through several years of radiation effects experimentation has led to the design and installation of a versatile transport and testing system. Some of the outstanding features of this shuttle system, as it is called, include:

1. Ability to irradiate samples ranging in size from foils used in activation research and dosimetry mapping, through materials and electronic components, to complex subsystem assemblies taken from aircraft systems;
2. Ability to position, irradiate, and remove samples from the reactor area rapidly and safely, resulting in an appreciable increase in reactor utilization; and
3. Flexibility of design to accommodate additions to the basic concept, such as the environmental system discussed later in this paper.

Rather than attempt to describe in detail the design and operating characteristics of the system, I would like to present a short motion picture which describes the system and shows its operation during our most recent material and component irradiation experiments in September of this year.

((MOVIE))

ENVIRONMENTAL SYSTEM

There is considerable experimental data to support the conclusion that nonnuclear environments such as temperature and humidity affect the degree of radiation damage to materials and components. It is important to make a thorough study of these interactions to assure the correct selection of particular materials and components operating under specific environmental conditions to be encountered in a nuclear aircraft. It is also necessary to obtain this information on broad classes of material and components to allow the establishment of qualification specifications on various aircraft systems, as it is virtually impossible to mock up the identical nuclear and nonnuclear parameters expected.

To provide the experimental facilities and hardware to accomplish this task, the design criteria were based on the necessity that specimens must be irradiated and tested, as well as transferred and stored, while a predetermined set of environmental conditions is maintained. The three elements of the environment to be controlled include the temperature, the humidity, and the atmosphere (Fig. 1).

Temperature is to be controlled between -80°F and $+450^{\circ}\text{F}$, with a maximum variation of $\pm 5^{\circ}\text{F}$. This range of temperature control will be available in the irradiation, storage, and test chambers, and during transport.

The relative humidity will be controlled from 25 to 95 percent (± 4 percent) over a temperature range of 50 to 130°F in the irradiation, storage and test chambers.

The composition of the atmosphere will be controlled by circulating either air or inert gas through the system. In addition, a monitoring system will be provided to record the ozone, oxygen, and water vapor content of the air circulating through the system.

ENVIRONMENTAL SYSTEM CRITERIA

- TEMPERATURE
-80°F TO +450°F
- HUMIDITY
25% TO 95% R.H.
@ 50°F TO 130°F
- ATMOSPHERE
AIR OR INERT GAS
COMPOSITION MONITORING

At the Irradiation Position

As can be seen in Figure 2 air is circulated through a closed loop by means of an axial flow fan. The air is forced over refrigeration coils for low-temperature control. Brine for these coils is furnished from a CO₂ refrigerator located out of the radiation area. The air also flows over resistance heaters which provide the high and ambient temperature control.

When an experiment requires humidity control, air is passed through a water-bath humidifier loop to add the proper water content. For very high relative humidities, the refrigeration coil is flooded by a water spray which saturates the air. This saturated air is then reheated to the temperature which will maintain the proper water vapor content. In addition, dry air may be introduced into the system for circulation when very low humidities are required.

The instrumentation for the temperature control consists of sensing elements located in the ducts adjacent to the irradiation positions, controllers which vary either the amount of brine through the refrigeration coils or the current to the heaters, and recorders located in the control room which provide a continuous record of operations.

The instrumentation which provides humidity control consists of sensing elements located in the ducts, controllers which determine the amount of air through the humidifier loop and the heater temperature necessary to maintain the proper relative humidity, and recorders which provide a continuous record of operations.

Figure 3 is a conceptual drawing of the hardware installation in the reactor area. The major portion of the control equipment is located in the handling area; in fact, the only equipment located in the pool consists of a fan and the low-temperature refrigeration coils. The equipment is thus protected from high radiation levels which would, in a short time, impair the reliability of the equipment; routine maintenance during regular shutdown periods is also facilitated.

FLOW CHART TEMP - HUMIDITY ENVIRONMENT

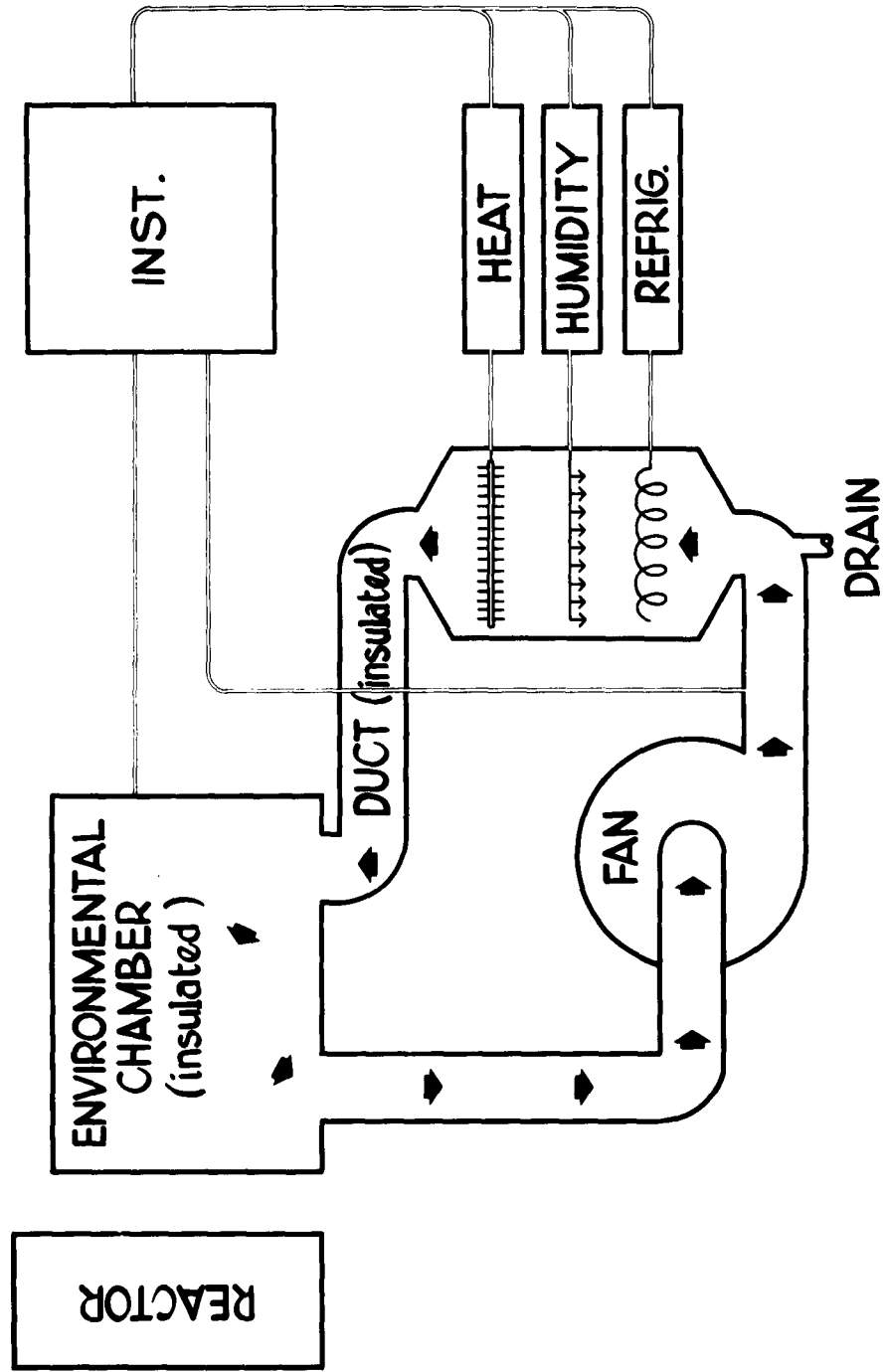


FIGURE 2

NPC 7424

GENERAL ARRANGEMENT

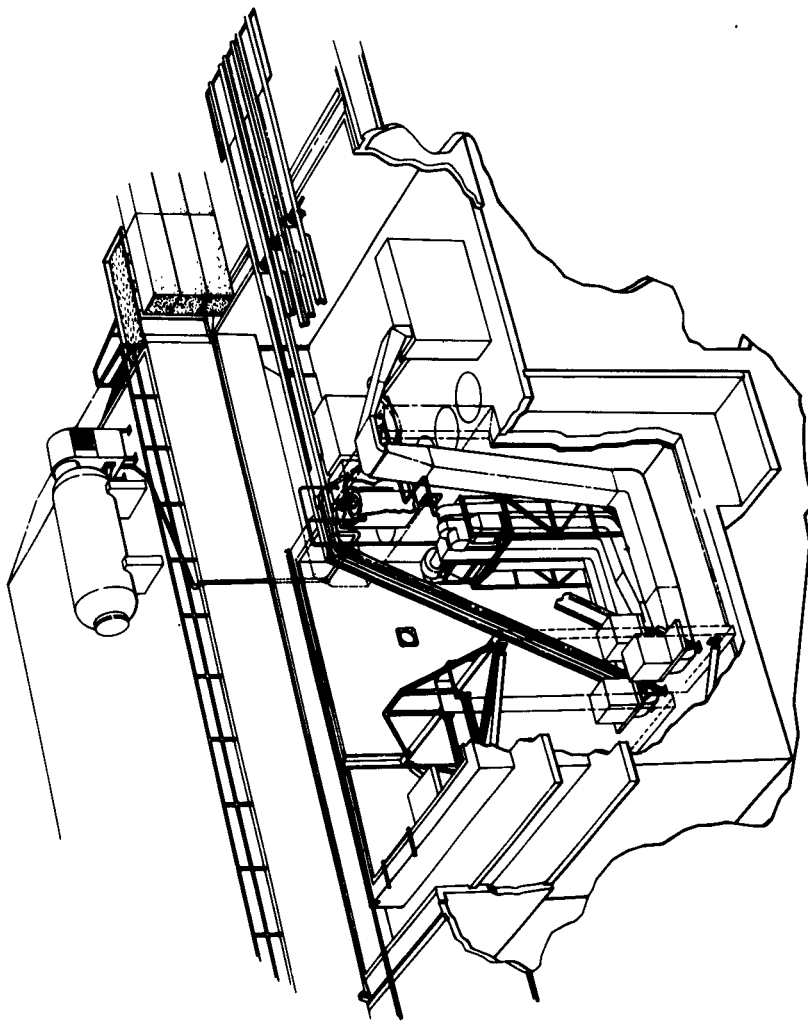


FIGURE 3

NPC 7425

Figure 4 is an exploded view of the portable irradiation chamber. The outside and inside walls of the chamber are made up of very thin aluminum sheeting. The void between the walls is filled with a high silicon-content insulating material. The bottom of the chamber is removable, and it is this lower section that is used as a base plate to mount the sample racks and trays. The louvers are normally closed so that the box is a sealed container. When the box is moved into the irradiation position the louvers open automatically which, in effect, makes the chamber an integral part of the environmental duct system. This method of introducing conditioned air into the box provides control during the irradiation period as well as during the transfer and subsequent operations with the chamber.

Transporting the Specimens

Samples are transferred from the reactor area to the Irradiated Material Laboratory by means of a transport cart (Fig. 5). The cart contains an insulated duct system with provisions for temperature control similar to the loops located in the irradiation positions. Air is circulated through the irradiation chamber (located on top of the transport cart) and duct loop by means of a fan. A drawer holding dry ice provides the low temperature control, and high temperatures are maintained by means of resistance heaters located in the duct. No humidity control is provided in the transport cart because the insulated irradiation chamber can maintain the humidity within design limits during the short time required for transport.

Storage and Testing

The pre-irradiation and post-irradiation testing of samples is performed in the Irradiated Materials Laboratory (IML). Equipment in this laboratory consists of storage facilities for changing samples from the irradiation or storage chambers to the testing chambers, and testing machines.

ENVIRONMENTAL CHAMBER

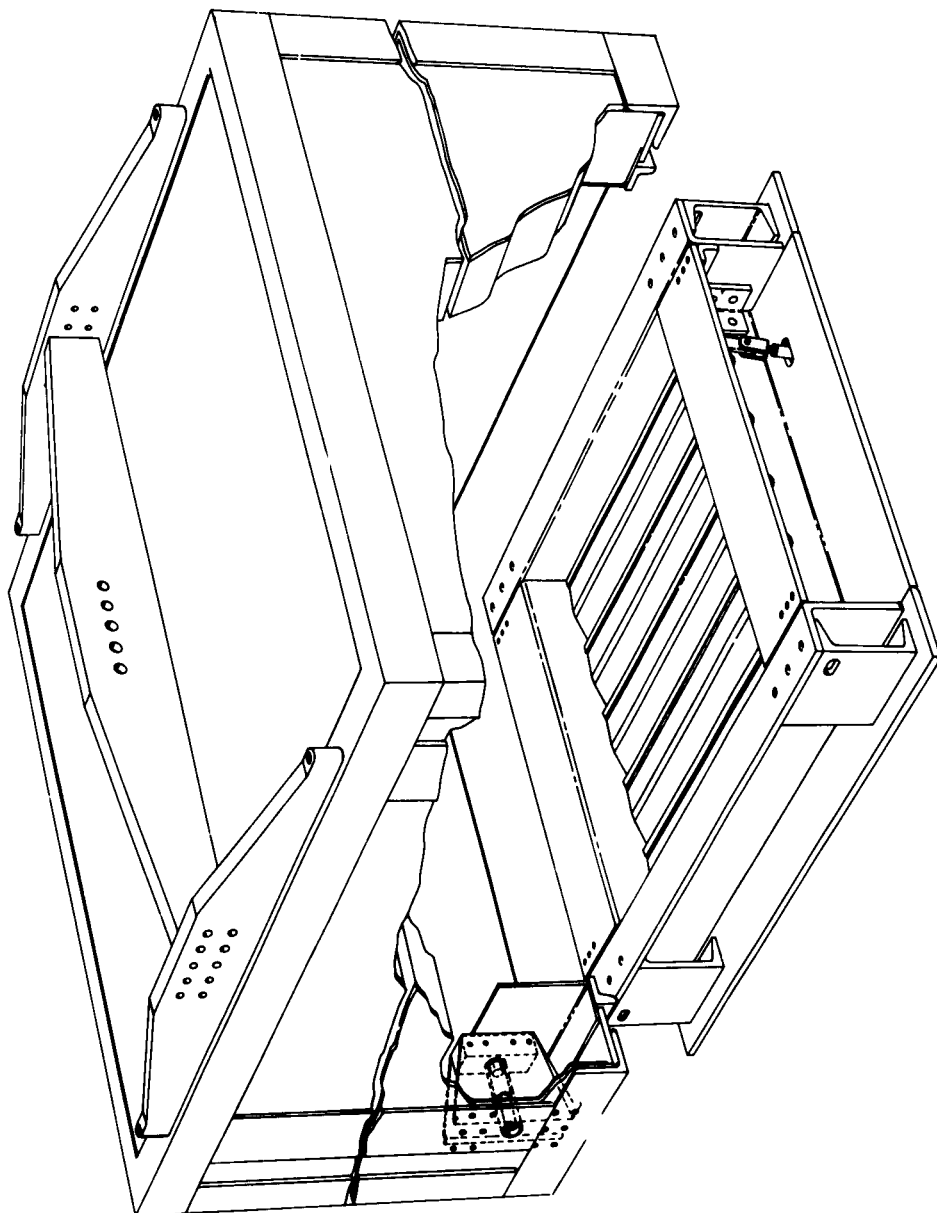


FIGURE 4

NPC 7426

TRANSPORT EQUIPMENT

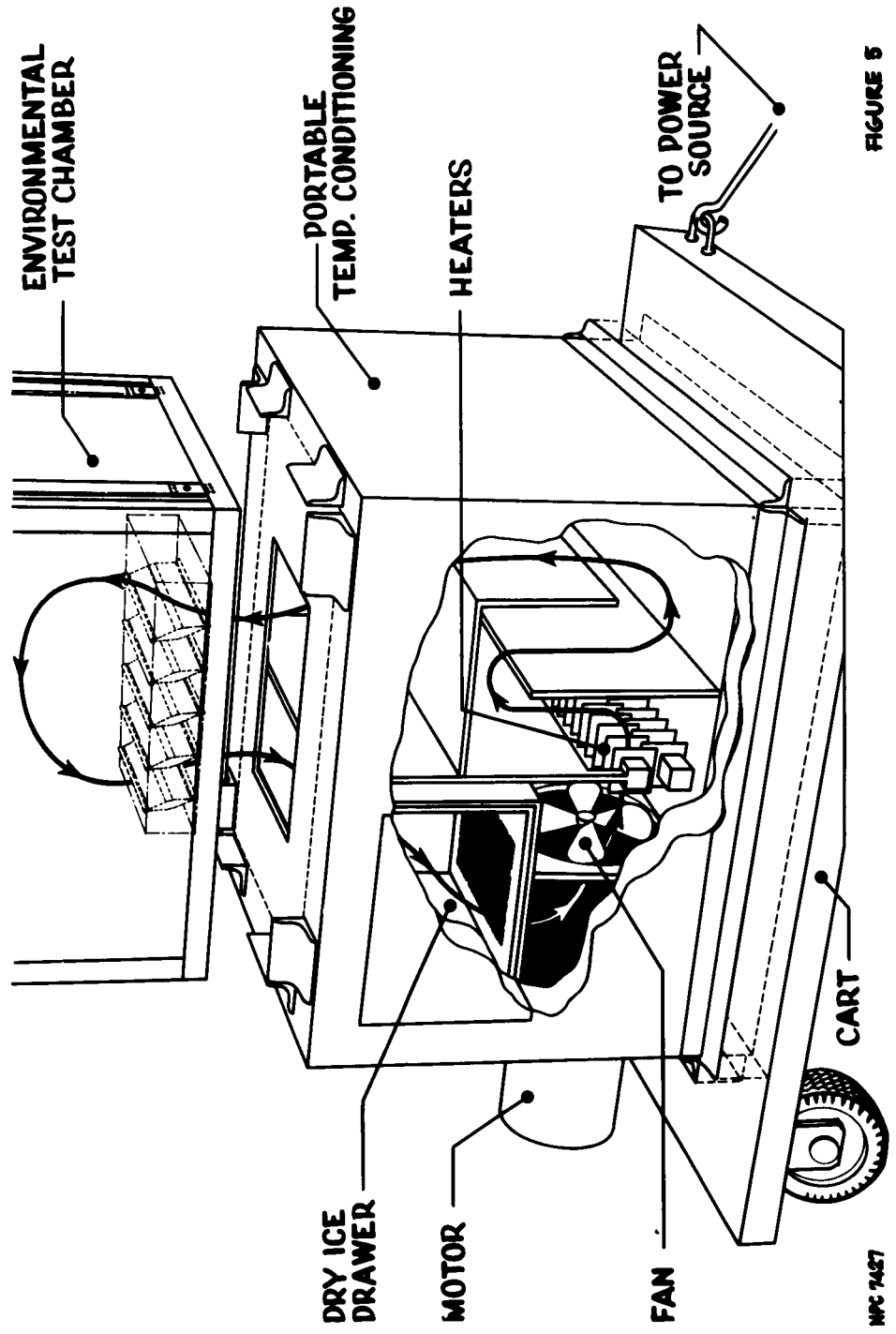


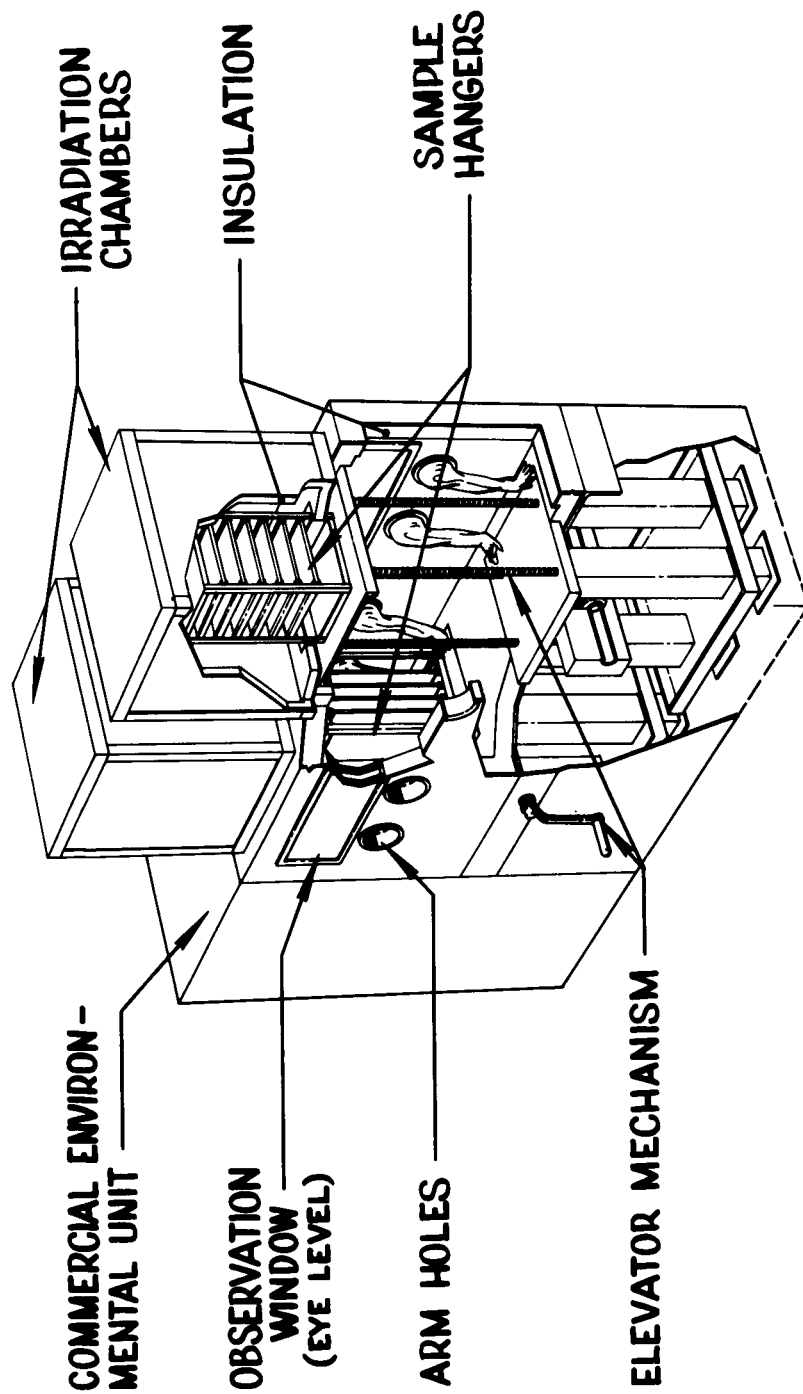
FIGURE 5

The IML storage facilities consist of a series of tables with refrigeration, heating, and humidity equipment installed underneath. There are provisions for storing eight chambers under varying environmental conditions at any one time. Figure 6 shows a method of transferring from the irradiation chambers to a testing or storage chamber. The irradiation chamber is set on top of the handling box. The lower section of the box is disconnected from the insulated cover and lowered into the box by means of a mechanical table. A testing chamber is connected to the end of the handling box. Samples are either removed by hand from the irradiation cage to the test chamber or consolidated into another chamber for storage. The testing chamber is then removed from the handling box and moved to the test machine for testing.

SUMMARY

The shuttle portion of the system has been installed and is in operation. The environmental portion of the system, being fabricated at the present time, should be operating early next spring. With this system, materials, components, aircraft subsystems, and other assemblies can be irradiated in various imposed environments. These assemblies can be operated and data obtained while being irradiated.

ENVIRONMENTAL UNIT



NPC 7426

FIGURE 6

RADIATION EFFECTS REACTOR

by

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The Radiation Effects Reactor is a 10-megawatt, pressurized, heterogeneous, light water cooled and moderated reactor, using fully enriched uranium in ETR type elements. It is mounted on a hydraulic lift, which raises it from the pool to the height of the systems mounted on railroad cars around the pool for irradiation. Coolant flows through swivel pipes at 3000 gpm. Control and instrumentation cables are routed through an overhead conveyor system to lessen radiation damage to them.

A new core will contain 11.6% Δk excess reactivity and have a lifetime of 7000 megawatt hours at rated power. Calculated flux values at the pressure vessel surface are thermal neutron current - 3×10^{11} nv, above thermal neutron current - 8×10^{11} nv, and gamma flux - 6×10^{13} mev/cm²-sec.

The Lockheed Radiation Effects Reactor, which is generally known as the RER, was developed by the General Electric Company from a Lockheed design concept. This reactor was designed to yield a maximum neutron and gamma leakage current compatible with the power level and safety requirements. It is a heterogeneous type, light water cooled and moderated, and designed for 10-megawatt operation with ETR type elements. The reactor is equipped with practically no shielding.

The RER is housed as shown in Figure 1 in a building constructed of steel beams and aluminum siding. It is mounted on a hydraulic lift with a 30-foot travel, and it can be positioned vertically in such a manner that the center-line of the core is at any point within this range up to 10 feet above the floor of the building. Control and instrumentation cables are attached to the top of the reactor and run up through an overhead conveyor system and then down through a tunnel to the Operations Building, where all reactor controls and instrumentation are located. (Located in the Operations Building are the console, the graphic panel, the recorder rack, and the amplifier cabinet.) Coolant is supplied to the reactor through swivel pipes, which are equipped with Barco joints to allow movement. The pumps, heat exchangers,

and other items of coolant equipment are located in a covered pit adjacent to the Reactor Building. The stainless steel pressure vessel, which is illustrated in Figure 2, consists of one cylinder 3-1/2 feet long and 3 feet in diameter and another cylinder 12-1/2 feet long and 2-1/2 feet in diameter, joined by a conical transition section. The vessel is 1/2 inch thick throughout, and it is capped at the bottom with a 1/2-inch ellipsoidal head. The upper closure is a flat, circular, forged plate, 3 feet 9 inches in diameter and 3 inches thick. The pressure vessel bears the ASME Boiler and Pressure Vessel Code stamp.

Coolant enters the pressure vessel from the bottom through two 8-inch nozzles and passes up above the core through two chutes, down through the core, up between the inner tank and pressure vessel, then down two chutes and out two 8-inch nozzles. This flow path, shown in Figure 3, provides a downward flow through the core and prevents the core from being drained in case of a break in the coolant line.

The upper closure separates the pressure vessel from the equipment tank, which contains the rod drive mechanisms, the fission counter drive and pre-amplifier, a venting system for the pressure vessel, and a sump pump. Holes in the upper closure accommodate the control rods, the regulating rods, and the fission chamber.

Standpipes above the equipment tank for control and instrumentation cables provide a disconnect point above water for use when the RER is in the pool.

The core support structure consists of a grid plate, a support plate, and a control rod shock damper. This structure provides support and alignment of fuel element, the source, the control rod lower guide and damper assembly, and the inner tank. The grid plate, which positions the various core components, consists of stainless steel members 1/4 inch thick and 3-3/4 inches deep, assembled to form an egg-crate type lattice. The aluminum support plate provides support for the fuel elements and source; and it houses rollers, which provide alignment for the control rods. The grid and damper support includes the main cone, shock absorber tubes, and structural members, which combined form a unit with the lower grid plate support. The entire structure is supported by a ring, bolted to the vessel wall.

The inner tank, a right aluminum cylinder 1/4 inch thick and 8 feet high, is aligned at the bottom by the grid and damper assembly and supported and aligned at the top by a ring welded to the pressure vessel. The purpose of this tank is to channel the coolant flow through the core and to support the hold-down plate. The hold-down plate, which covers the entire core, is made up of three hinged sections to facilitate handling and refueling. It provides roller support for the control rods, restrains horizontal movement of core components, and prevents vertical movement of the fuel elements. Each of the 32 fuel elements comprising a core loading is made up of 18 fuel plates, roll-swaged into two side plates. (The arrangement of these fuel elements is shown in Figure 4.) Each of the fuel-poison control rods contains 14 plates. The fuel used is fully enriched U-235,

176 grams per fuel element and 111 grams per control element. Dummy elements fill the grid spaces not used for fuel elements. Reflector elements may be placed in the space between the core and the inner tank.

The cadmium poison section of the control rods is permanently attached to the fuel section. Each of the four control elements has an average value of $4.5\% \Delta k/k$. One regulating rod of 30-70 cadmium-silver alloy has a value of 0.3%.

The RER uses an antimony-beryllium source, which will have an activity of 330 curies during normal operation because of the regeneration that takes place when the antimony is exposed to the core flux.

The control rods are of the electro-mechanical type, with mechanical fingers actuated by a magnet to grapple a knob at the upper end of the control rod. Control rods move at a speed of 4.5 inches per minute. Scram forces come from a spring, which gives it an acceleration of 5 g. The regulating rod, which is bolted to the drive, moves at a speed of 45 inches per minute.

The shield tank surrounding the reactor provides an attenuation of 10^3 for fast neutrons and 10^5 for thermal neutrons. This tank, which contains an outer liner of Boral, is divided vertically into three sections. The center section is further divided radially into quadrants, and each quadrant is sub-divided horizontally into two components. The quadrant facing away from the test car positions is removable; this arrangement will facilitate possible future replacement by a component test cell. The Boral shielding is removable from one quadrant facing a test car position. The sections of the tank can be filled and drained remotely from the graphic panel.

The RER is equipped with a start-up channel, a fission counter; a log N channel, a compensated ion chamber; three safety channels, ion chambers; two power level channels, and the gamma monitor channel. (These channels are illustrated in Figure 5.)

There are 15 scram circuits that can automatically cause a scram when a predetermined condition exists. These include period, flux level, temperature, and pressure circuits. Both relay and electronic scram circuits are employed.

Normal coolant flow is maintained by one of the 3000-gpm pumps illustrated in Figure 6, with the other on stand-by. Pressure and coolant level is maintained by an automatic pressurizer and make-up system. The pressure is supplied by bottled nitrogen.

Two heat exchangers operating in series remove the heat from the primary system, dissipating it in the cooling tower. Inlet temperature is 120°F and outlet temperature is 142°F . About 30 gpm of the primary coolant is by-passed through a demineralizer, thus maintaining purity. Make-up and emergency coolant

is supplied from a 100,000-gallon demineralized water storage tank.

Two air-actuated valves in parallel automatically open upon loss of normal coolant flow and allow the water in the storage tank to flow by means of gravity through the reactor to the 150,000-gallon drain-and-hold tank. Initial flow is over 1000 gpm, and a remotely operated valve can be used to throttle the flow. The flow rate and the volume are sufficient to prevent meltdown of the core.

To obtain a core lifetime of 7000 megawatt hours, the following amounts of excess reactivity are required:

Xenon	5.9%
Fission Products (except Xe)	1.5%
Fuel Burn-up	2.9%
Temperature	0.8%
Control	0.5%
Total	11.6%

The calculated average thermal neutron leakage current is 3×10^{11} nv; the average above thermal neutron leakage current, 8×10^{11} nv; and the average gamma leakage flux, 6×10^{13} mev/cm²-sec.

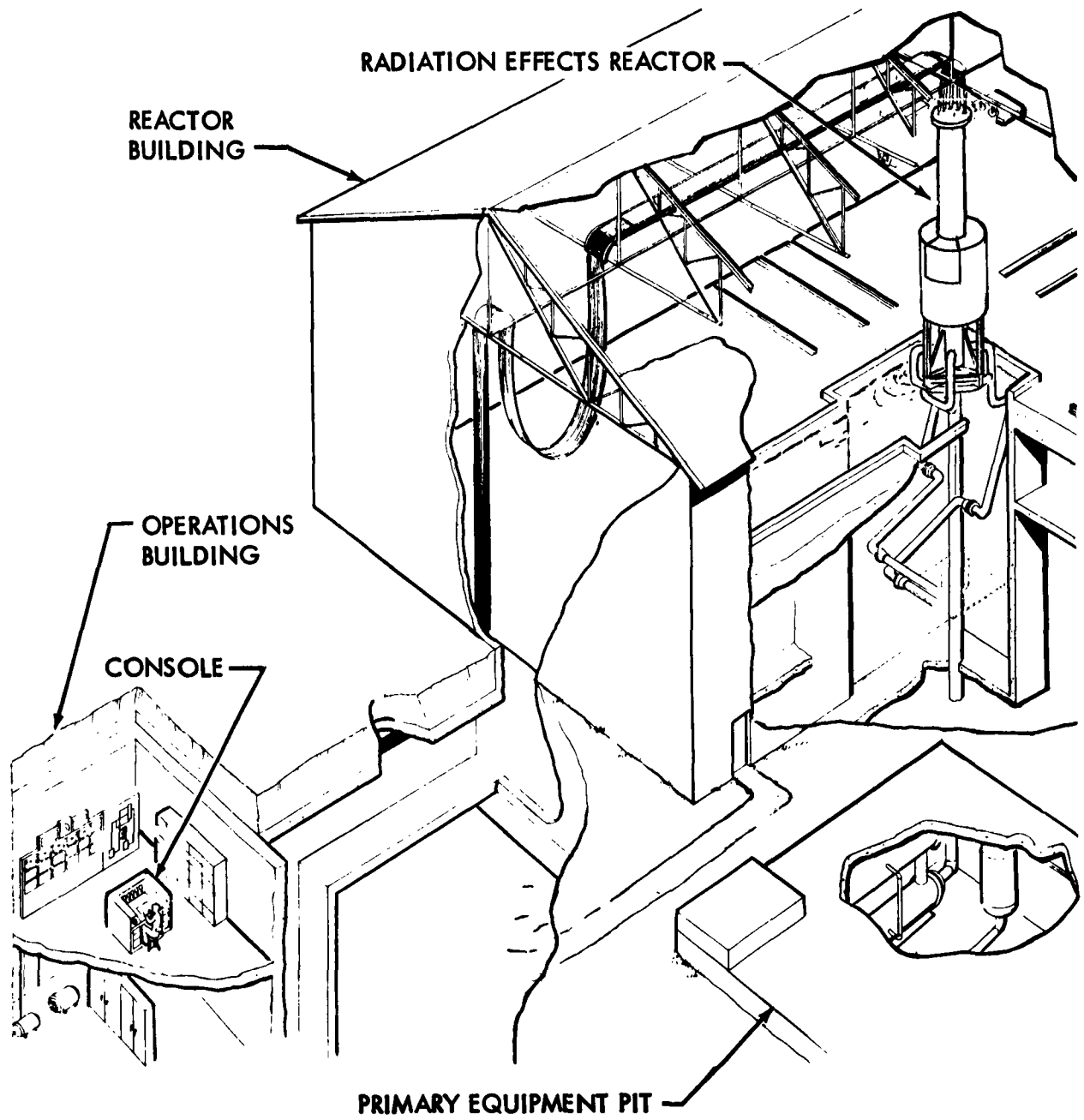


FIG. 1 REF REACTOR BUILDING AND RER

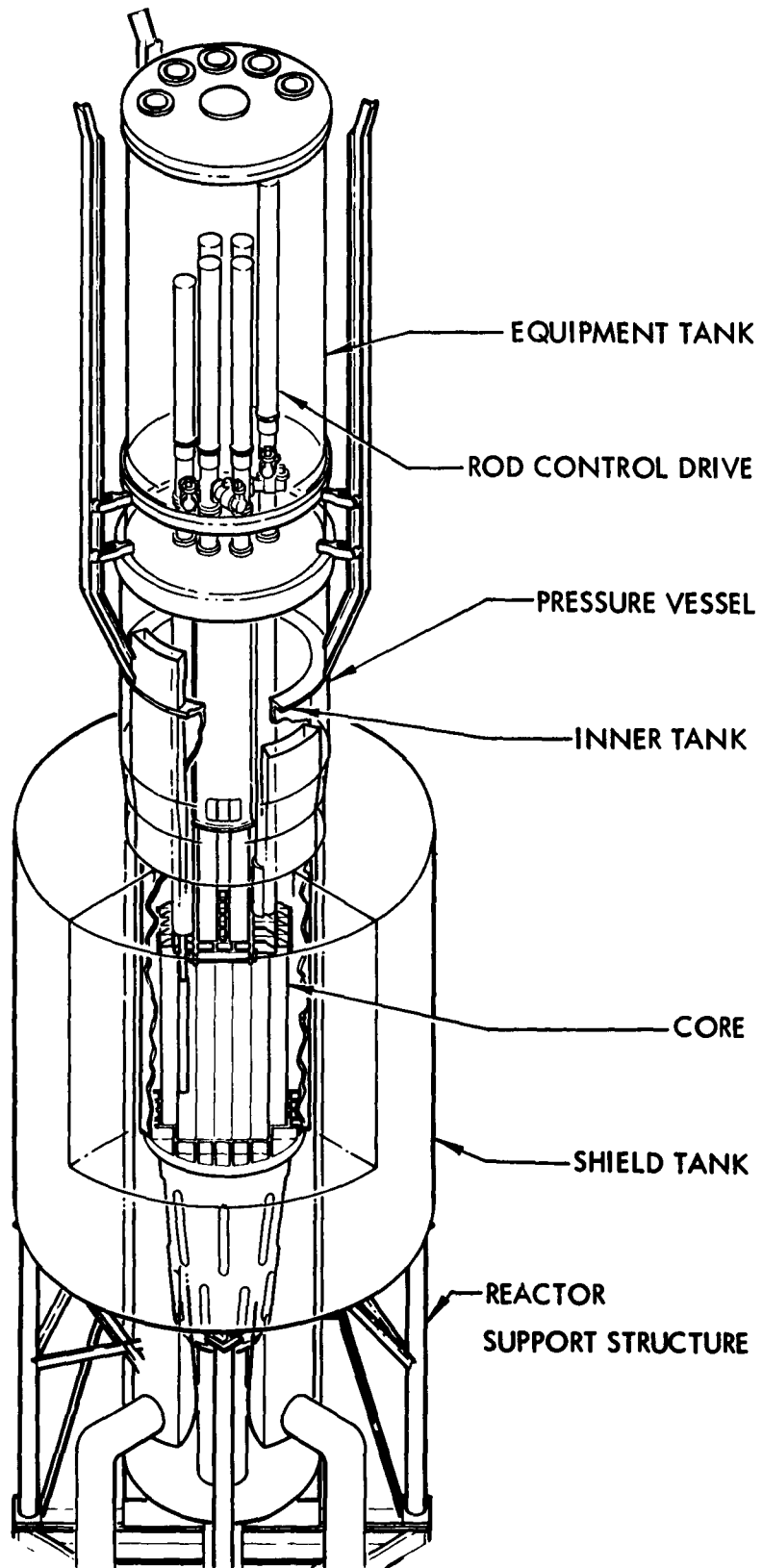


FIG. 2 RADIATION EFFECTS REACTOR

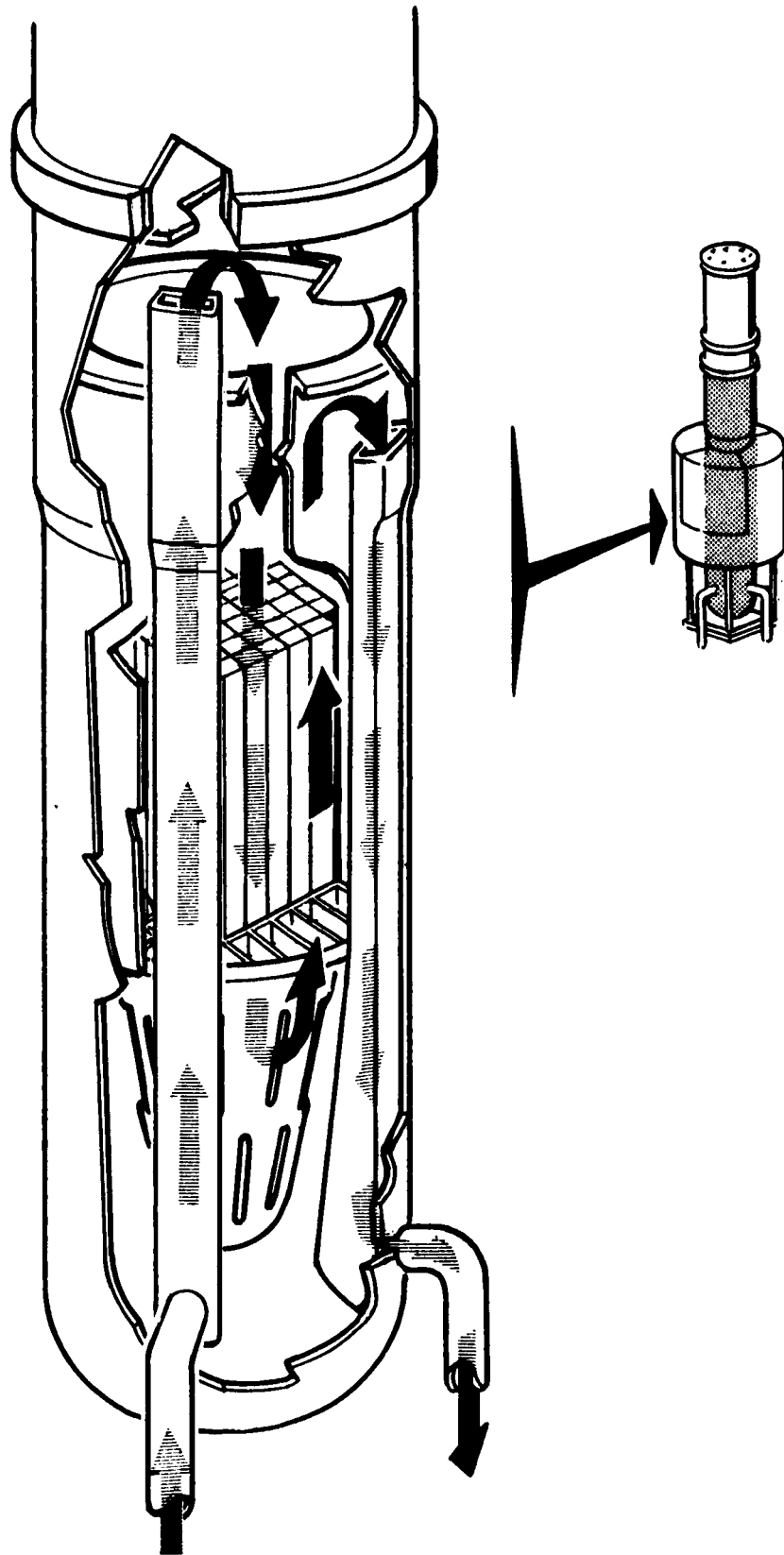


FIG. 3 RER WATER CIRCULATION

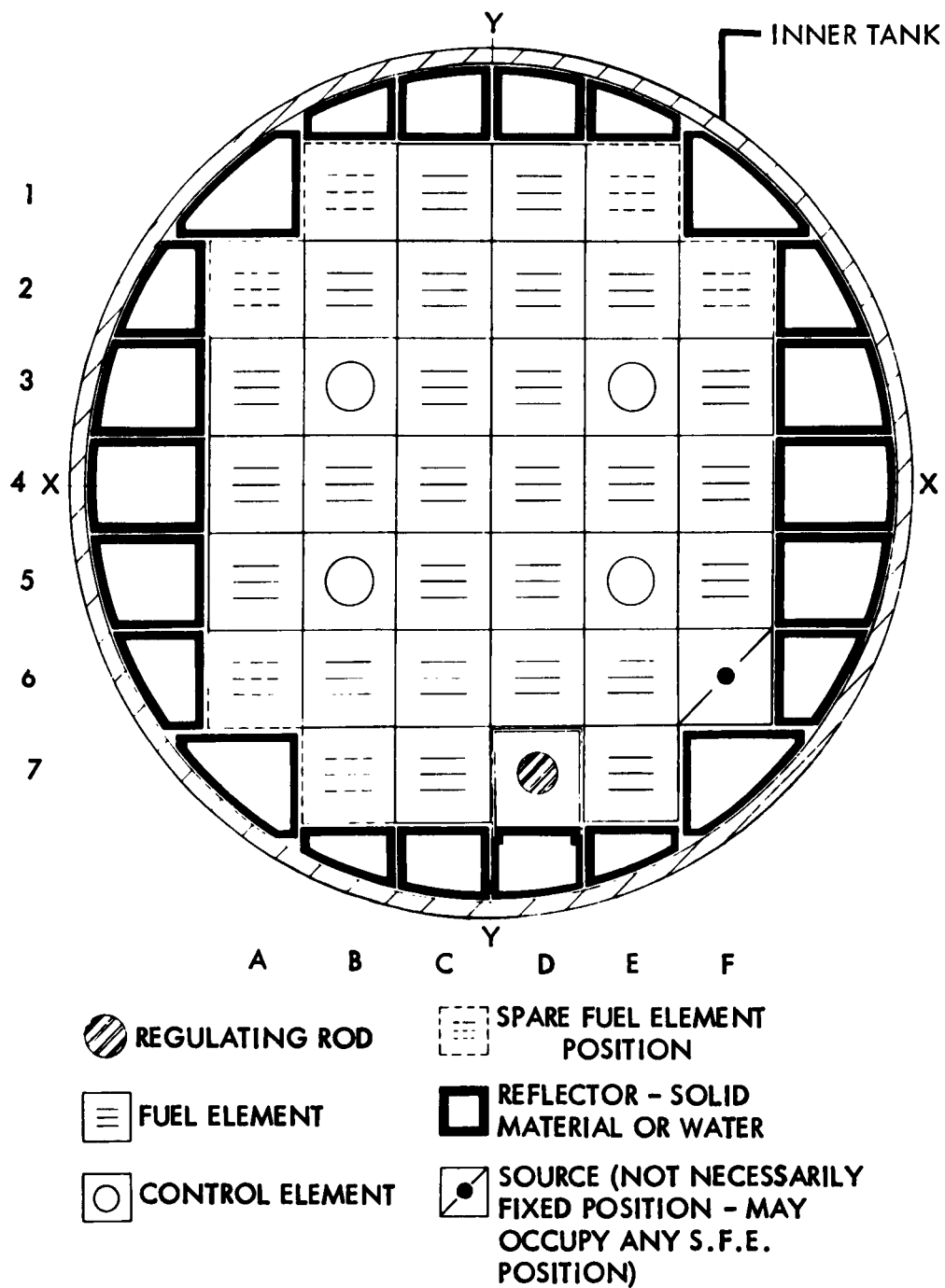


FIG. 4 CORE ARRANGEMENT

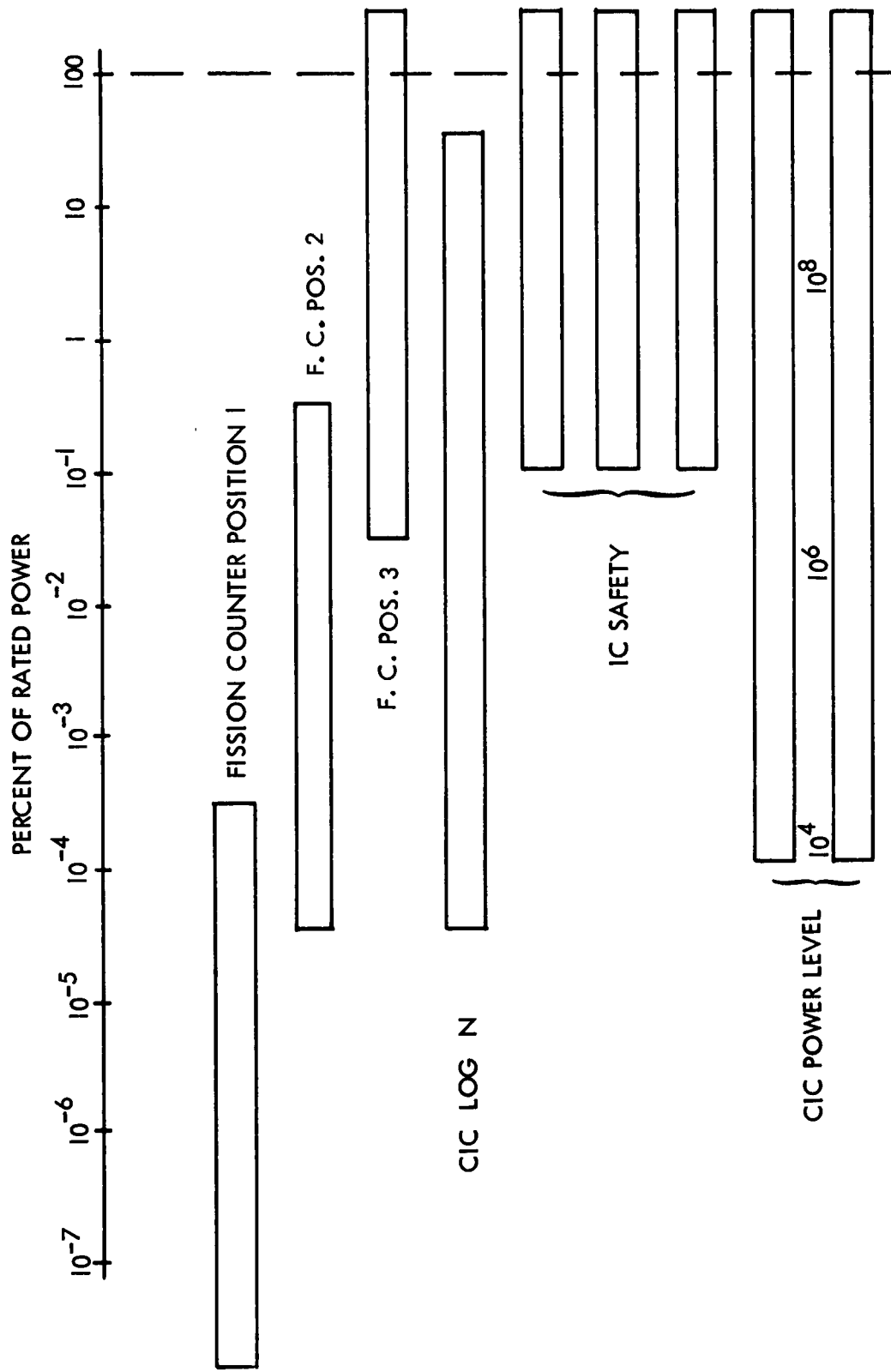


FIG. 5 NUCLEAR INSTRUMENTATION CHANNEL RANGES

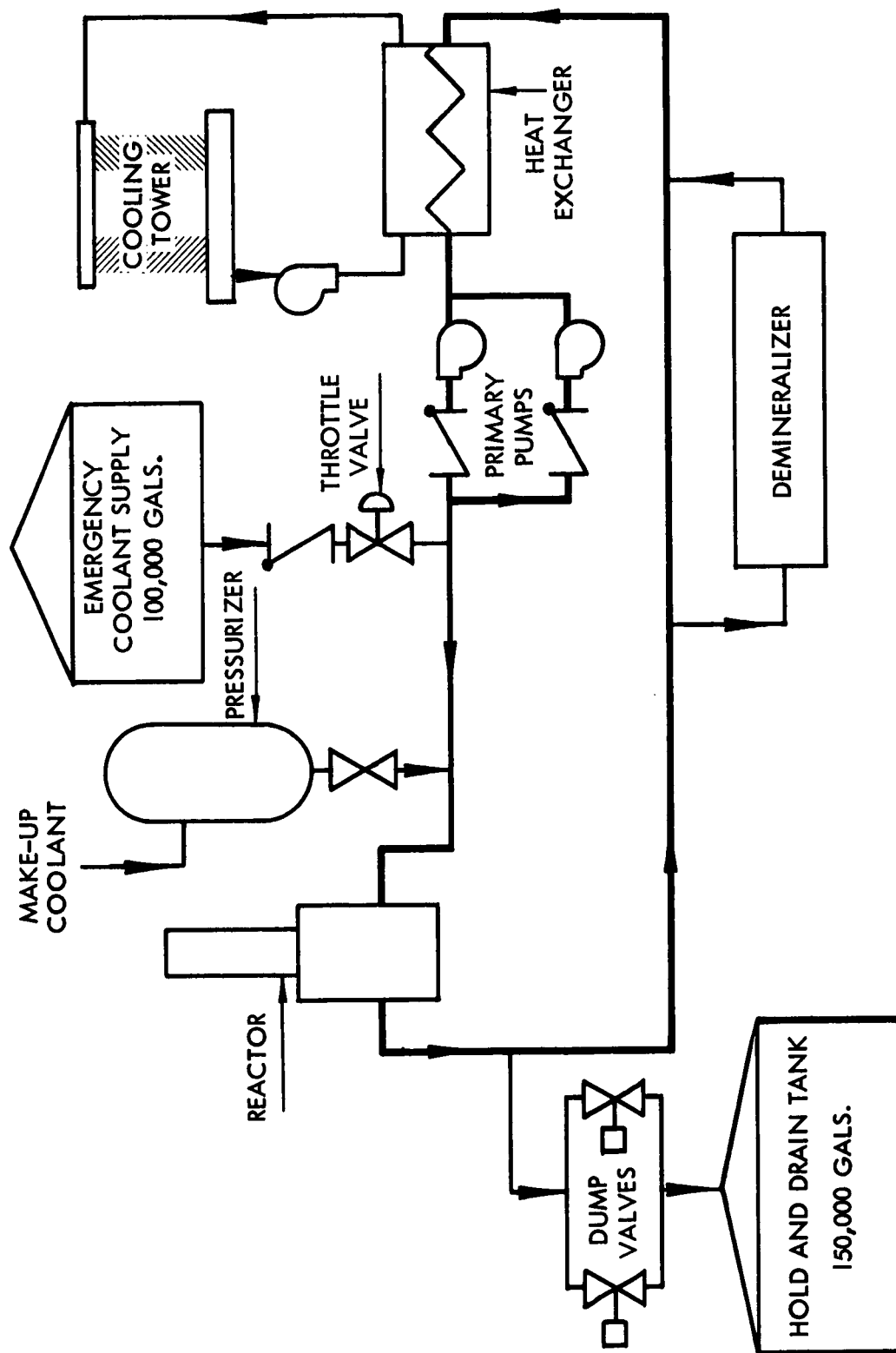


FIG. 6 PRIMARY COOLANT SYSTEM

A DESCRIPTION OF A MULTI-KILOCURIE IRRADIATION FACILITY AND THE ASSOCIATED RADIATION DOSIMETRY

by

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M. E. Krasnow, E. R. Rathbun, and
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Inland Testing Laboratories and Cook Research
Morton Grove, Illinois

Inland Testing Laboratories, a division of Cook Electric Company, has designed and is operating a large multi-kilocurie cobalt-60 irradiation facility. With this source of gamma radiation, a dose rate of 10^6 roentgens per hour is possible from approximately 62 kilocuries of activity.

The design of the source configuration and cave facility makes possible the testing, both static and dynamic, of a large variety of materials within a high radiation flux environment.

This paper deals primarily with the developmental research regarding source configuration design, albedo characteristics of the cave, assay and calibration of the source by graphite ionization chamber dosimetry, and assembly and isodose plot of the completed source.

The present source configuration yields optimum conditions for irradiation of a large number of materials at a high radiation dosage.

As early as 1948 the Nuclear Science Group of the Cook Electric Company of Chicago, Illinois, initiated radiation effects studies with the nuclear reactor of Pennsylvania State University. In 1953 the Inland Testing Laboratories was established as a separate division of the company, and subsequently Air Force contracts prompted expansion of the new division in all fields of environmental testing. As a participant in the ANP program, Inland Testing Laboratories was awarded the contract for the construction and operation of a multi-kilocurie cobalt-60 irradiation facility for testing the effects of radiation on aircraft fuels, hydraulic fluids, and lubricants under dynamic conditions. Nucleodyne Corporation, another division of the Company, had

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Inland Testing Laboratories, a division of Cook Electric Company, has designed, constructed, and is operating a large multi-kilocurie Cobalt 60 irradiation facility. The Cobalt 60 source at this facility is approximately 50,000 curies and produces a dose rate of 10^6 roentgens per hour.

The design of the source configuration and the large cave facility makes possible the testing, both static and dynamic, of a large variety of materials within a high radiation flux environment. This facility also provides sufficient space to perform many tests simultaneously at relatively uniform high dose rates.

Specifically, this paper is concerned with the developmental research regarding the design of the source configuration, the albedo characteristics of the cave, the assay and total activity of the source as measured by graphite ionization chamber, and the assembly and isodose plot of the completed source.

INTRODUCTION

In 1950 the Inland Testing Laboratories were established as a separate division of the Cook Electric Company, and subsequently Air Force contracts prompted expansion of the new division in all fields of environmental testing. The Nuclear Science Group of Inland Testing Laboratories initiated radiation effects studies with the nuclear reactor of Pennsylvania State University in 1956. Also, as a participant in the ANP program, Inland Testing Laboratories were

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awarded the contract for the design, construction, and operation of a multi-kilocurie Cobalt 60 irradiation facility to determine the effects of radiation on aircraft fuels, hydraulic fluids, and lubricants under dynamic conditions.

Prior to the completion of the irradiation facility and the assembly of the source, it was essential to evaluate the various hazards associated with the handling and use of approximately 50,000 curies.¹ Investigations such as shielding calculations, ozone activation, and thermal radiation problems connected with the exposed source, in addition to confirmation of design of the source configuration and reliable methods of radiation dosimetry were some of the factors studied for an understanding of these hazards.²

Simultaneously with these investigations, the chemistry laboratory was obtaining base data on the fuels, oil, and lubricants submitted for testing by the Air Force.¹ Such data were necessary for comparison purposes in the proper evaluation of materials in a radiation environment under dynamic conditions.

With this brief history and introduction, the remainder of this paper will discuss in more detail the characteristics of this gamma facility which will include the source configuration and radiation dosimetry.²

THE INLAND TESTING LABORATORIES GAMMA IRRADIATION FACILITY

The Cobalt 60 facility is actually two structures in one. The inner structure, built of thick concrete, houses the hot cell; and the outer structure, of ordinary construction material, contains the laboratory and service areas. As shown in Figure 1, the hot cell and the control room adjoin the chemistry laboratory. The hot cell, or cave, shown in Figure 2, has two types of shielded walls; the front wall is heavy density magnetite concrete and is 48 inches thick while the other three walls are of ordinary concrete and are 68 inches thick. To maintain the proper temperature and a slightly negative pressure, the cell has a separate air-conditioning system.

The source assembly, illustrated in Figure 3, is located near the center of the cell and is closer to the front wall containing the lead glass window. This permits efficient operation of the Argonne remote manipulators. The source is raised and lowered from its storage well of water by a remote chain-drive mechanism, provided with a fail-safe release device. The final source container, illustrated in Figure 4, is a stainless steel cylinder 24-1/4 inches outside diameter and 13-1/2 inches high, containing within its walls slugs of Cobalt 60 three layers thick.

The engineering design of the source was confirmed by use of source mockups. Figure 5A illustrates a source mockup configuration and the procedure of preparing the configuration using slugs of very low activity Cobalt 60. The P. R. Bell-type single-channel pulse height analyzer shown in Figure 5B was used to make the intensity measurements on the mockup sources. Figure 6 shows a typical example of the gamma spectrum from such source mockups. The results of the mockup experiments were in good agreement with the predicted theoretical values as illustrated in Figure 7. The configuration of wall thickness three slugs and height 18 inches was arbitrarily selected as the reference source to which all other configurations were compared. The calculated ratios agree favorably with the measured intensity ratios, for the estimated accuracy of the data is of the order of 10 percent.

ALBEDO CHARACTERISTICS OF THE IRRADIATION CELL

Prior to the assembly of the Cobalt 60 operational source, it was necessary to evaluate the albedo characteristics of the irradiation cell. The albedo characteristics of the irradiation cell were of interest, since such "reflected" radiation could be a significant contribution to the radiation dose.

The P. R. Bell single-channel pulse analyzer was utilized in this study; and a 12 millicurie Cobalt 60 source was positioned directly over the scintillation crystal and photo tube assembly. As shown in Figure 8, a lead cone was placed directly on top the crystal as shielding against the direct rays of the source. A comparison of the resultant spectra, shown in graphic form in Figures 9 and 10, with the unshielded spectrum, shown in Figure 11, gives some indication of the effectiveness of the shielding. The scans indicated in Figures 9 and 10 compare favorably with a plot of energy spectra of emergent Cobalt 60 photons by the Monte Carlo calculations made by J. R. Perkins³ for Albedo of concrete, which are illustrated in Figure 12.

ASSAY OF COBALT 60 AMPULES AND ASSEMBLY OF THE IRRADIATION SOURCE

The Atomic Energy of Canada, Ltd., supplied Inland Testing Laboratories with Cobalt 60 as produced by the reactors at Chalk River. The cobalt slugs were encapsulated in aluminum cylinders having over-all dimensions of 3/8 inch diameter by 1-1/8 inches long.

The assay of the Cobalt 60 slugs was conducted with a CO₂ graphite ionization chamber in conjunction with a Kiethley Model 410 micromicroammeter. The CO₂ graphite ionization chamber used for the Cobalt 60 dosimetry measurements is a modification of the chamber described by Ballweg and Meem⁴ for use

at the Oak Ridge National Laboratories for gamma flux measurements in the bulk shielding facilities reactor. A cross section of this chamber is shown in Figure 13. The graphite wall thickness was reduced from 0.777 cm to 0.239 cm in order to meet the requirements of the Bragg-Gray principles specifically for Cobalt 60 gammas.⁵ Calibration of this chamber against the standard kilocurie Cobalt 60 source of the Argonne Cancer Research Clinic at the University of Chicago showed agreement with the theoretical value within 5.6 percent. The experimental chamber constant was in good agreement with the theoretical constant, being within 3.1 percent.

In order that proper correlation between source activity and dose rate could be established, it was essential that representative samples of the 7000 slugs comprising the source be assayed to determine the total activity of the final assembled irradiation source. By statistical calculations, it was determined that some 100 slugs representatively selected from the total batch of 7000 slugs would be sufficient to determine the total activity of the source within an error of 1 percent. The method of assay was based upon the principles used by the Materiel Laboratory, Wright Air Development Center, Dayton, Ohio.^{6,7} The assay of the slugs and the assembly of the source was carried out as a joint operation utilizing the Argonne remote manipulators.

The ion chamber was positioned in a lead shield chamber shown in Figure 14. The center of the ion chamber was located exactly 1 foot from the center of the assay tube, and the entire assembly was located at a convenient position in front of the cave window within easy reach of the Argonne manipulators, as shown in Figure 15. All cables and gas lines to the chamber were run through the conduits in the cave wall and connected to the Kiethley 410 electrometer and power supply outside the cave.

In order to correct for the albedo and backscatter contribution to the ion current as obtained within the lead chamber, it was necessary to assay a selected slug in open air, where such contributing effects would be negligible.⁶ For this purpose, the 30 foot wooden derrick shown in Figure 16 was constructed to raise the slug by remote cables to a sufficient height to obtain an open air reading unaffected by surrounding materials. Measurements were taken at various known distances to obtain a best average value of activity for the reference slug. As shown in Figure 17, the activity of the reference slug corrected for decay and effects of temperature and pressure on the chamber current averaged 8.43 ± 0.24 curies in open air. A correction factor, based on the ratio of activity of the reference slug in "open air" and in the lead chamber, was then applied to all lead chamber readings to convert them to "open air" values. The slug values were also

similarly corrected for decay, ion chamber temperature, and pressure. The average slug value was determined to be 7.45 curies and thus, the total assembled source was calculated to be 50,000 curies with an error of not more than ± 5.5 percent.

ISODOSE MAPPING OF THE INLAND TESTING LABORATORIES' OPERATIONAL COBALT 60 SOURCE

The Cobalt 60 slugs were assembled by means of the Argonne manipulators into the cylindrical container (24-1/2 inches outside diameter and 13-1/2 inches high) shown in Figure 18 by end-to-end stacking. With the completion of the loading of the container, it was necessary to determine the radiation dose obtainable at various distances around the source.

Distances from the source were accurately measured at 1 foot intervals horizontally and vertically from the center of the source. Dosage measurements were made at these various distances by use of the graphite ionization chamber and Keithley electrometer.

In order to obtain an accurate isodose plot around the source, two separate plots of the dose rate data were prepared. The first plot, Figure 19, was obtained by considering the dose rate throughout the cell at constant height, and the second plot, Figure 20, was obtained by measuring dose rate at various vertical heights at constant distances from the source.

From such graphs it was then possible to prepare an accurate isodose plot at distances surrounding the source, as shown in Figure 21. By use of such isodose charts, materials to be irradiated at specified dose rates can be located at predetermined positions. Exposure times can then be calculated to obtain any desired radiation dose. In addition to these isodose plots, supplementary methods of dosimetry such as cobalt glass, ferric-ferrous and ceric-cerous are used to more accurately determine the dose rate for a particular irradiation experiment. For large equipment, integration of dose is necessary for greater accuracy.

It will be noted in Figure 21 that the highest dose rate of 10^6 roentgens per hour is obtainable at points 2 inches from the inside wall of the cylinder along the mid-plane. In order to take advantage of this maximum dose rate, the stage-insert shown in Figure 22 was constructed to suspend test materials within the volume of the cylinder. This in no way interfered with the access of test equipment located around the outside perimeter of the source. By such manipulations, maximum utilization of the source was obtained.

SUMMARY OF THE VARIOUS TYPES OF IRRADIATION TESTING BEING CONDUCTED WITH THE ITL HIGH-GAMMA FLUX SOURCE

The ITL Cobalt 60 high-gamma flux source offers a wide variety of radiation testing possibilities. The design of the source and support equipment makes possible dynamic testing of operational equipment where data of this type are required, as for example in a nuclear-powered aircraft.

As an example of the types of environmental testing being carried out with the ITL multi-kilocurie gamma source the following are cited:

1. Under Air Force ANP contract, dynamic testing of lubricants under radiation environmental conditions utilizing:
 - a. Erdco universal tester to evaluate the performance of fluid lubricants in gears or bearings
 - b. Panel coker to evaluate coking tendency of fluid lubricants under radiation conditions
 - c. Fuel coker to evaluate deposit forming tendency of fuels
 - d. Grease tester to evaluate the performance of a grease in a bearing at high speed and high temperature.
2. Under subcontract with Lockheed Aircraft Corporation:
 - a. Aircraft hydraulic systems under operational conditions
 - b. Landing gear systems under operational conditions
3. Contract with the Army Quartermaster Corps: food preservation studies.
4. Various commercial considerations: pasteurization studies and glass colorization.

These examples serve to illustrate briefly the variety of uses that a source of this type can offer.

SUMMARY

The irradiation facilities and assembly of the Inland Testing Laboratories multi-kilocurie Cobalt 60 source have been described. Some details as to the

albedo characteristics of the hot cell and methods of slug assay and isodose mapping of the assembled source have been discussed.

The assembled source of 50,000 curies $\pm$ 5.5 percent of Cobalt 60 will produce a dose rate of 10^6 roentgens per hour at a point 2 inches from the inside wall of the source cylinder along the mid-plane.

The ITL irradiation source and facility are designed for maximum utility of both static and dynamic testing of a wide variety of materials under an environment of high gamma radiation flux.

REFERENCES

1. M. Z. Fainman, et al. ; "Evaluation of Fuels, Hydraulic Fluids, and Lubricants under Dynamic Conditions in the Presence of Gamma Radiation," First ITL Progress Report, Contract AF 33(616)3865.
2. M. Z. Fainman, R. E. Simpson, N. F. Offenbergl, "Evaluation of Fuels, Hydraulic Fluids, and Lubricants under Dynamic Conditions in the Presence of Gamma Radiation," Second ITL Progress Report Contract AF 33(616)3865.
3. J. R. Perkins, "Monte Carlo Calculations of Gamma-Ray Albedos of Concrete and Aluminum, "Journal of Applied Physics 26, 655 June 1955.
4. L. H. Ballweg and J. L. Meem, "A Standard Gamma Ray Ionization Chamber for Shielding Measurements, " ORNL-1028 (1951).
5. G. J. Hine and G. L. Brownell, Radiation Dosimetry, Academic Press, Inc., New York, N. Y. (1956).
6. M. C. Atkins, "Design and Use of a 23,000 Curie Cobalt 60 Source for Material Testing," Appendix A, W.C.R. T. TM 56-159, Materials Laboratory, Wright Air Development Center.
7. W. R. Burrus, "Standard Instrumentation Techniques for Nuclear Environmental Testing," Technical Note 57-207 (1957).

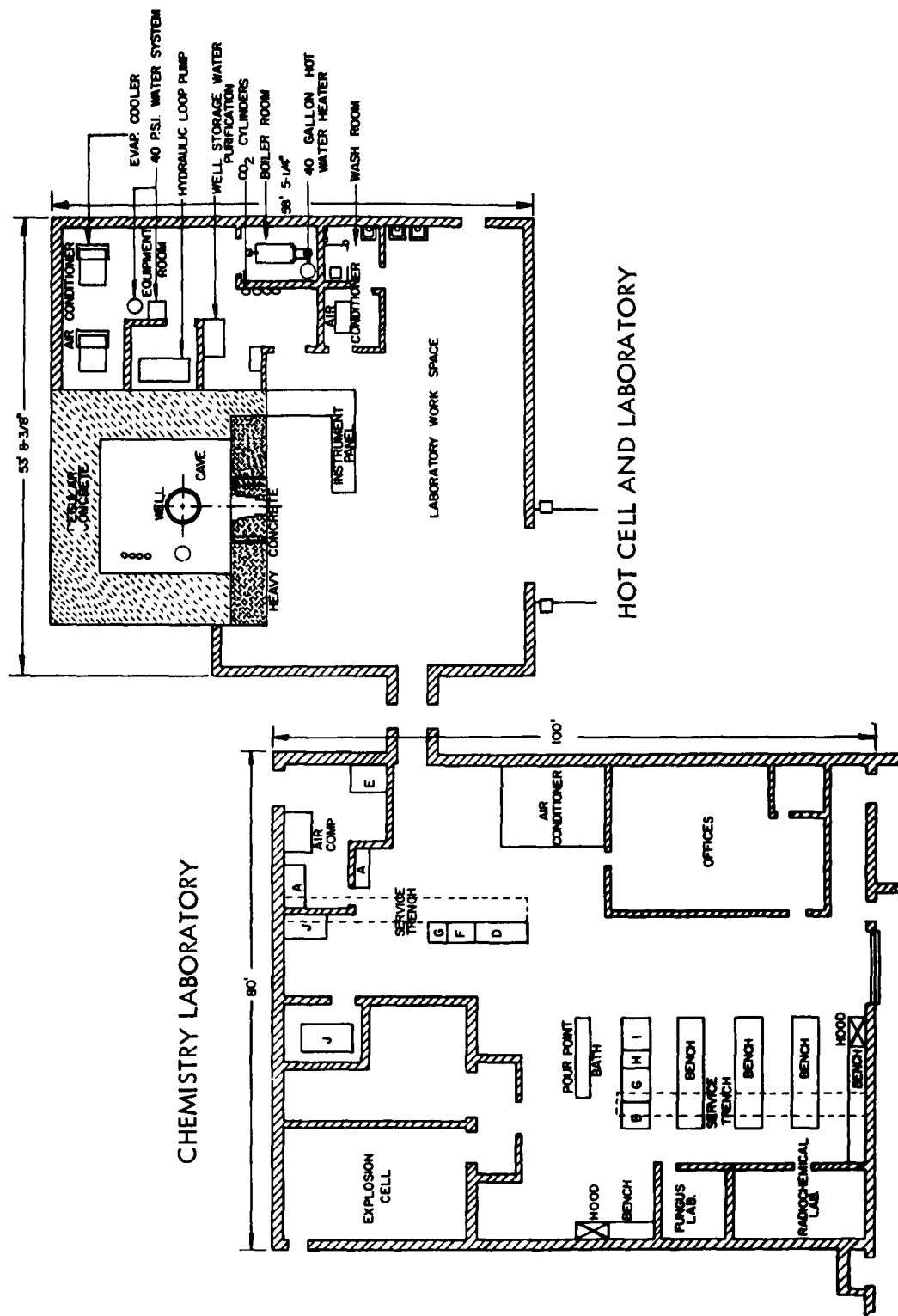


FIG. 1 FLOOR PLAN: CHEMISTRY LABORATORY AND HOT CELL

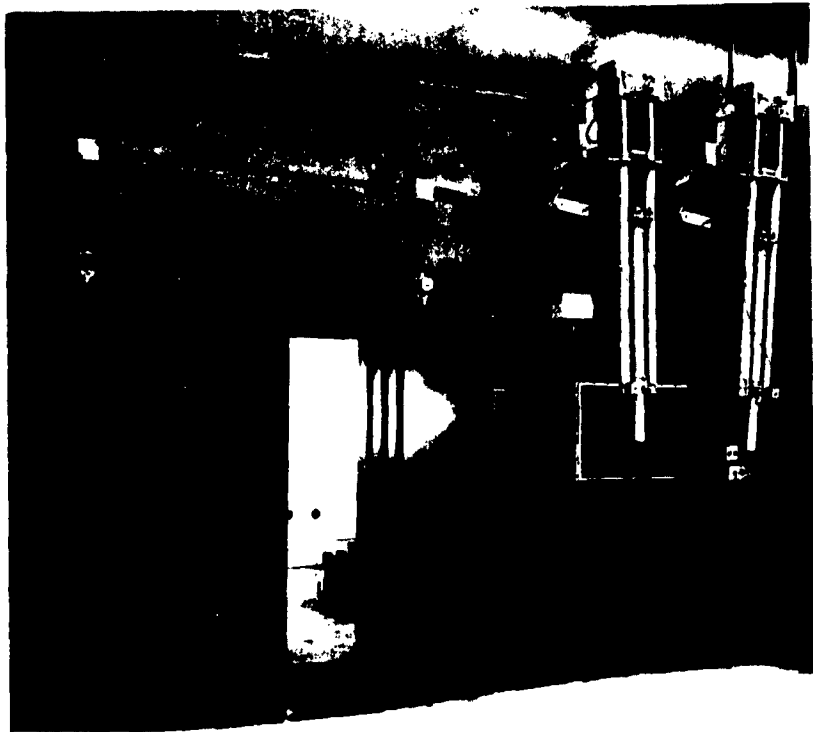
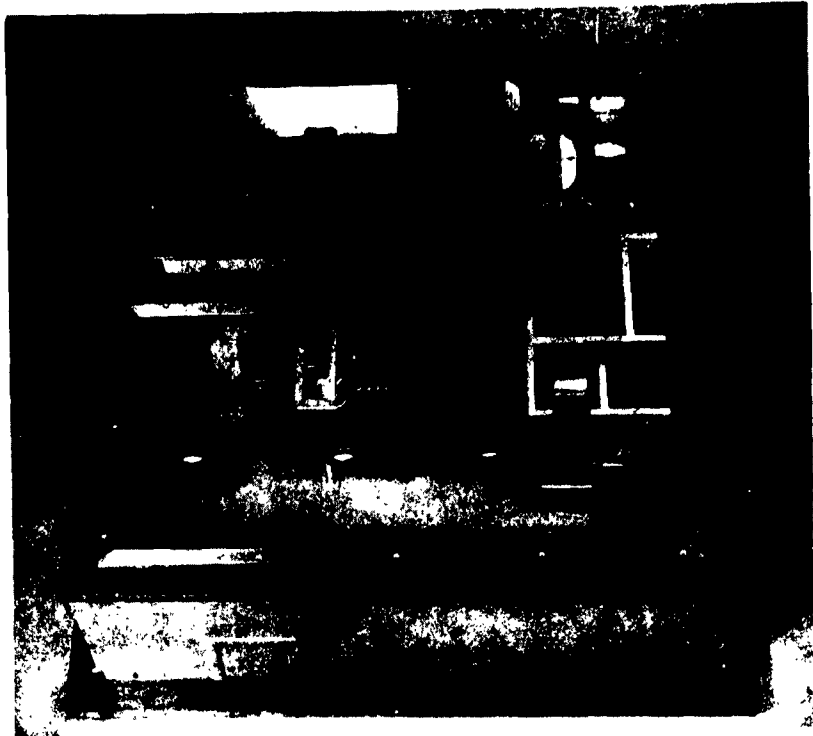


FIG. 2 COBALT-60 CELL

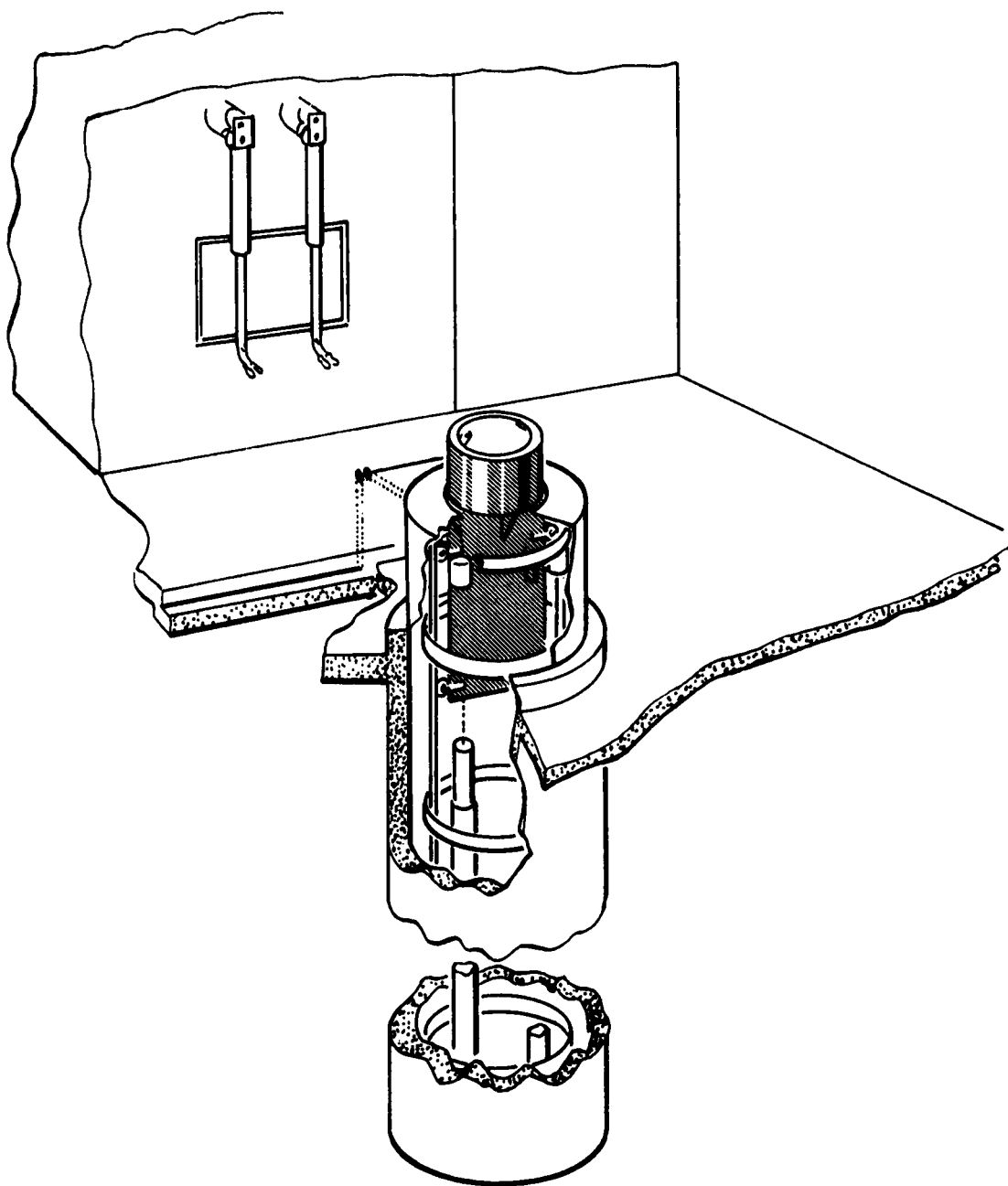


FIG. 3 SOURCE ELEVATOR

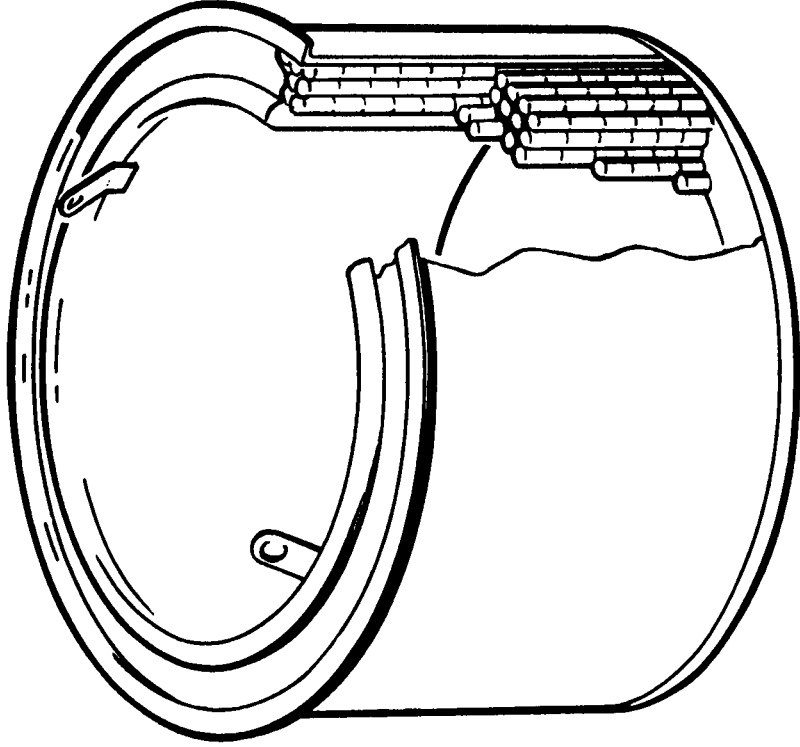


FIG. 4 COBALT-60 SOURCE



FIG. 5A ASSEMBLY OF Fe^{59} CONFIGURATION

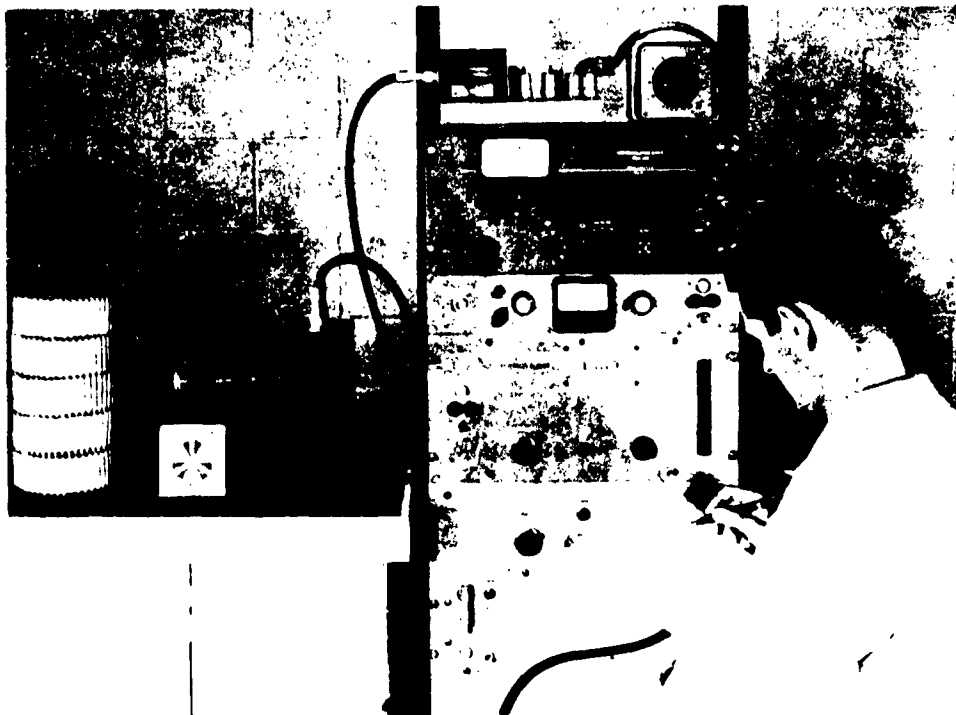


FIG. 5B INTENSITY MEASUREMENT OF Fe^{59} CONFIGURATION

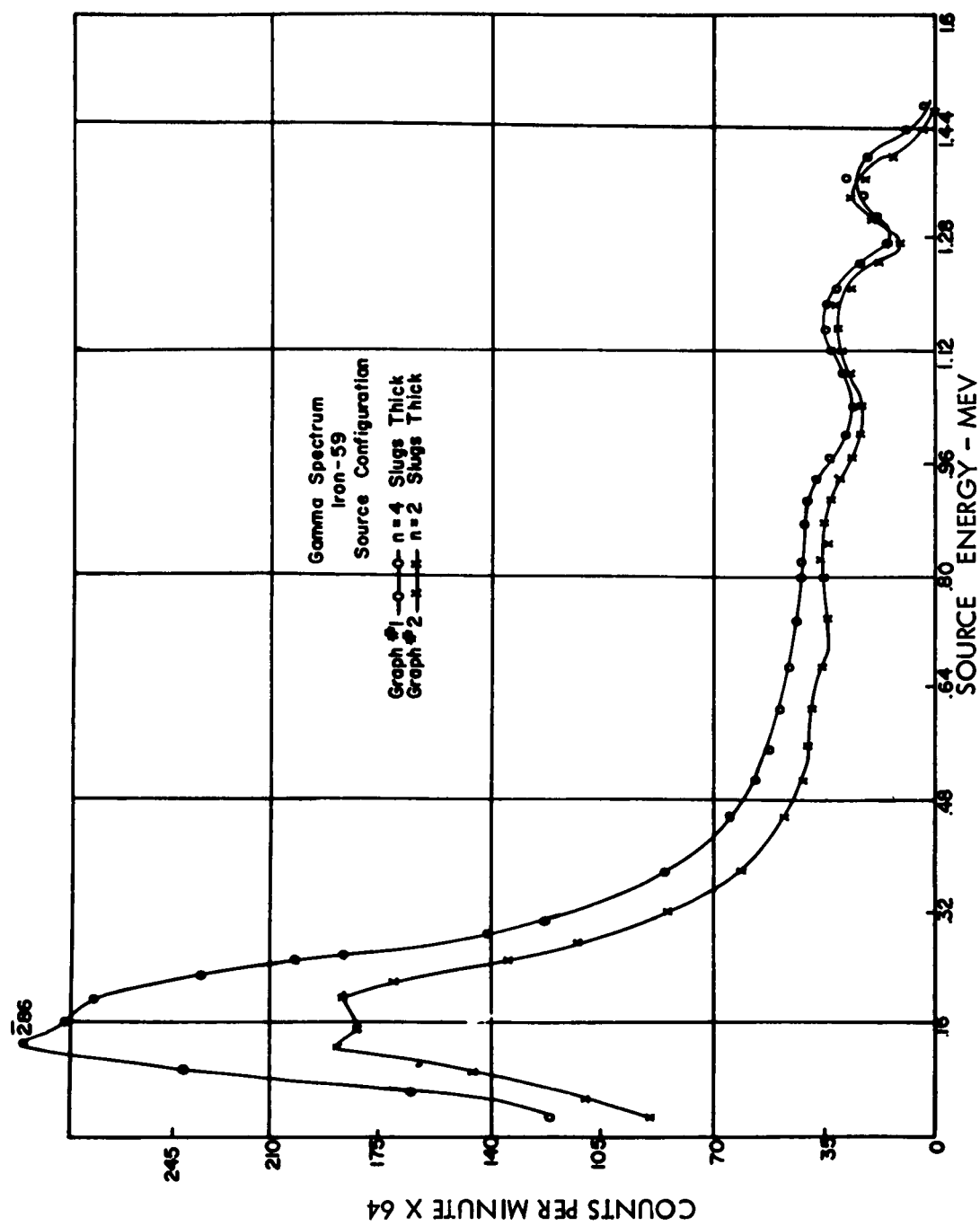


FIG. 6 GAMMA SPECTRUM Fe⁵⁹ SOURCE CONFIGURATION

TABLE I

N	H (IN.)	MEASURED R_D	K_{NH}	$\frac{R_D(NH)}{R_D(REF)}$	$\frac{K_{NH}}{K_{REF}}$
		(10^{-4} MEV/MIN)	(IN.)		
2	11.5	5.98	0.478	0.75	0.82
2	16.0	7.30	0.539	0.91	0.92
3	13.5	7.78	0.565	0.97	0.97
3	18.0	8.03	0.584	1.00	1.00
3	24.8	6.48	0.538	0.81	0.92
4	11.5	7.93	0.557	0.99	0.96
4	16.0	8.60	0.579	1.07	0.99
4	18.0	6.99	0.565	0.87	0.97

FIG. 7 EFFECT OF SOURCE WALL THICKNESS ON RADIATION DOSAGE

LOCATION OF ALBEDO MEASUREMENT FOR AIR



LOCATION OF ALBEDO
MEASUREMENT OVER SOURCE WELL



LOCATION OF ALBEDO
MEASUREMENT FOR CELL WALL

FIG. 8

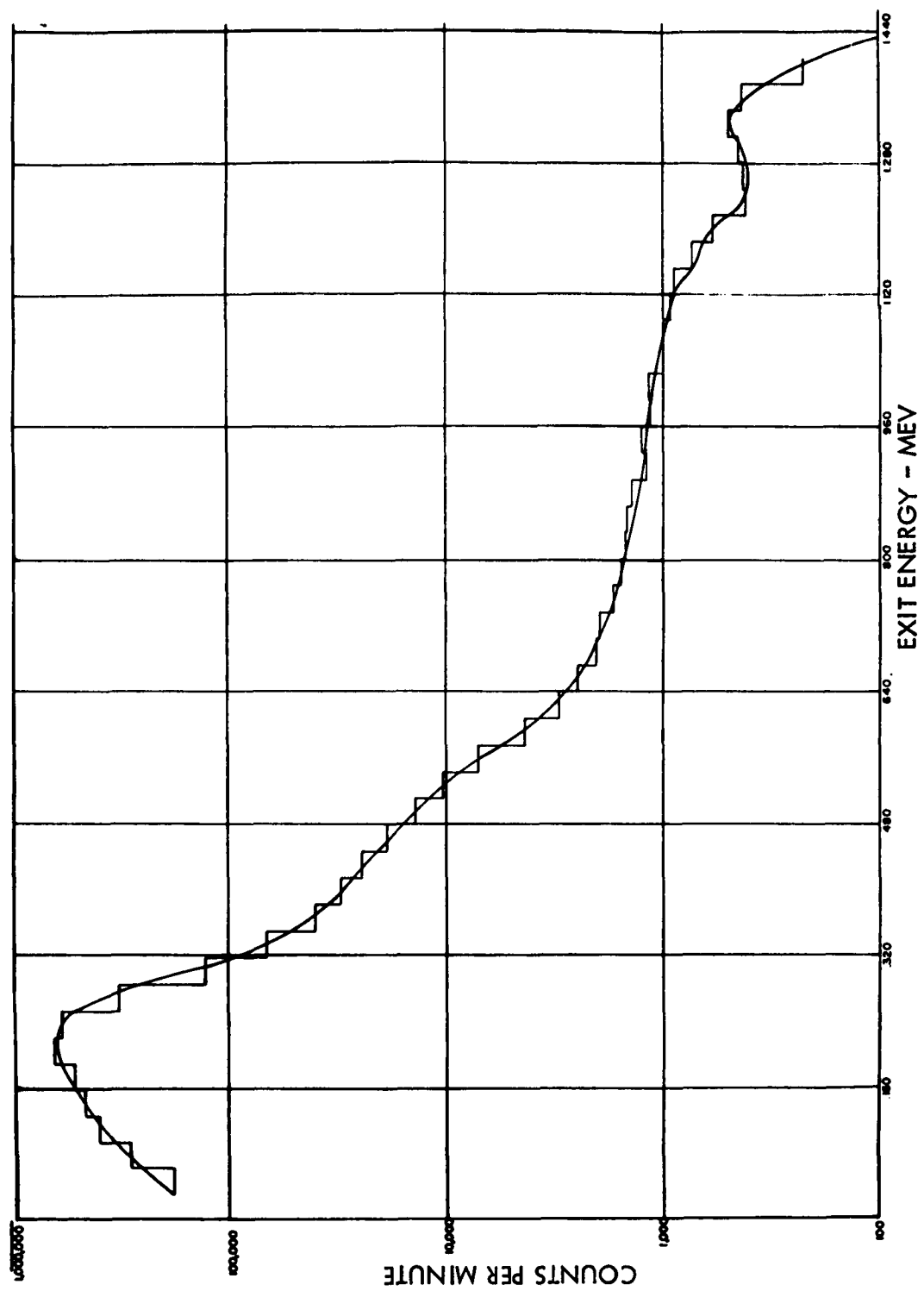


FIG. 9 ALBEDO OF MAGNETITE CONCRETE WALL OF SOURCE CELL

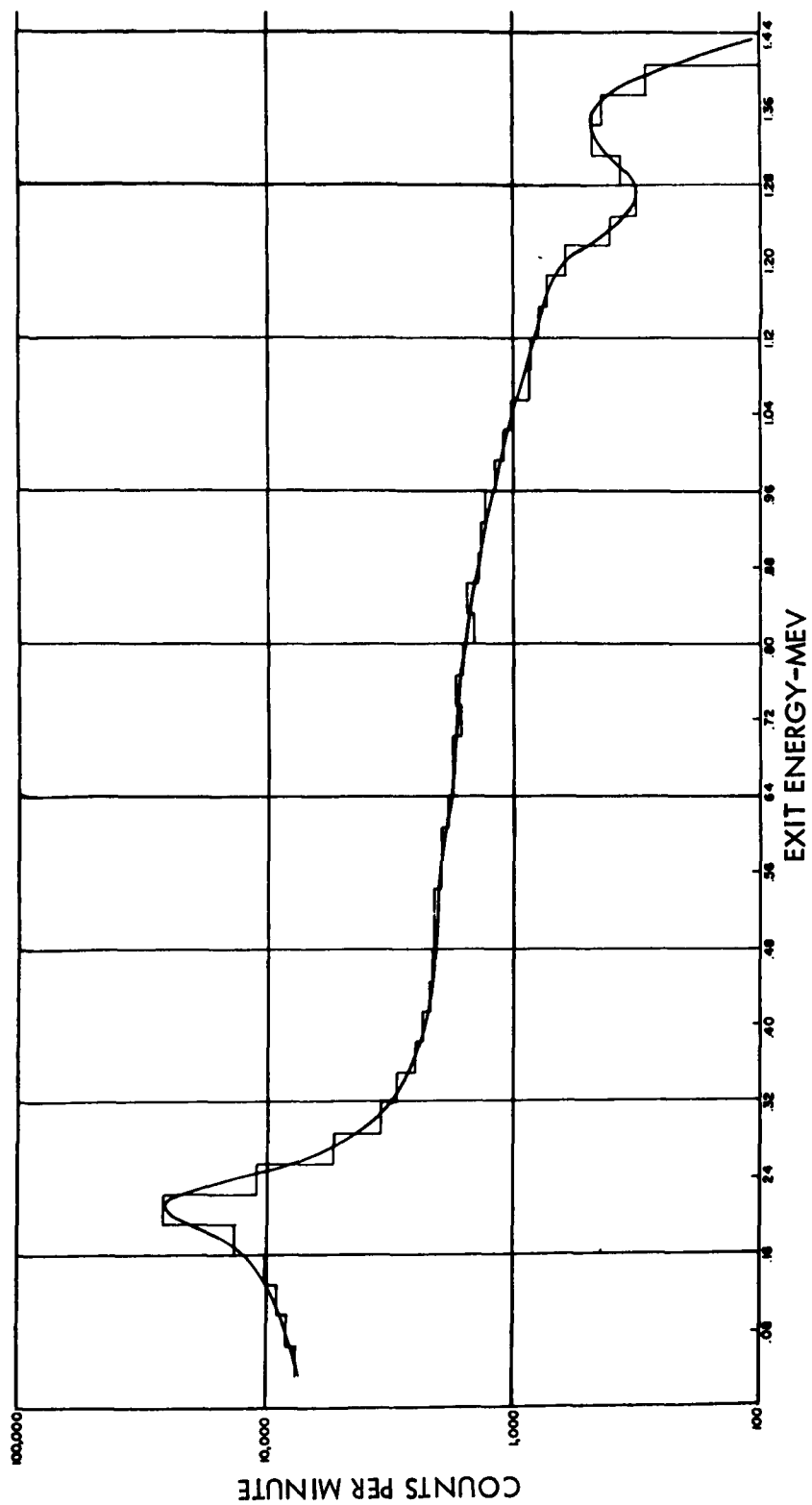


FIG. 10 ALBEDO FOR AIR

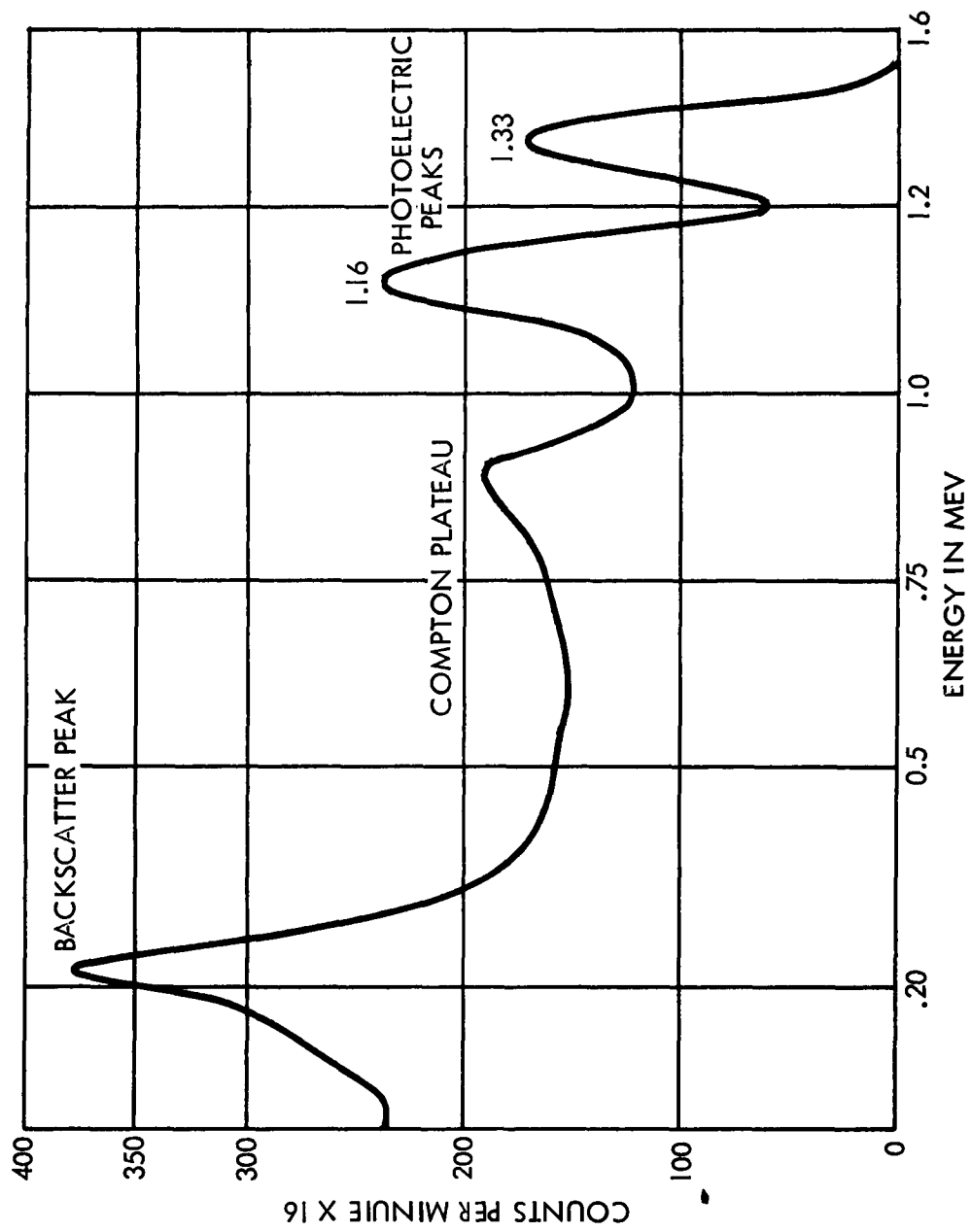


FIG. 11 SPECTRUM OF COBALT-60

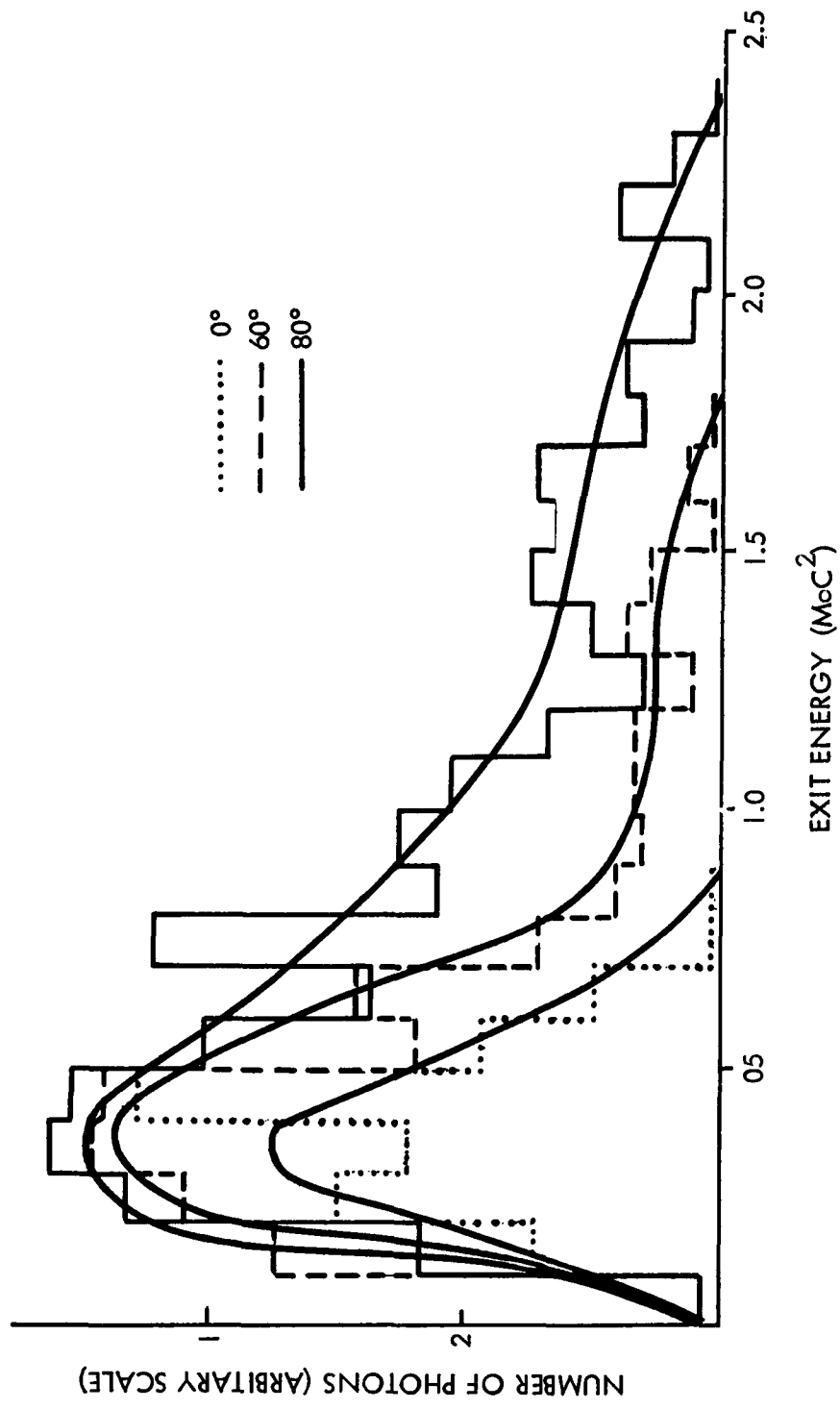


FIG. 12 ENERGY SPECTRA OF EMERGENT PHOTONS FOR Co^{60} GAMMA RAYS INCIDENT AT $\theta_0=0^\circ$, 60° , AND 80°

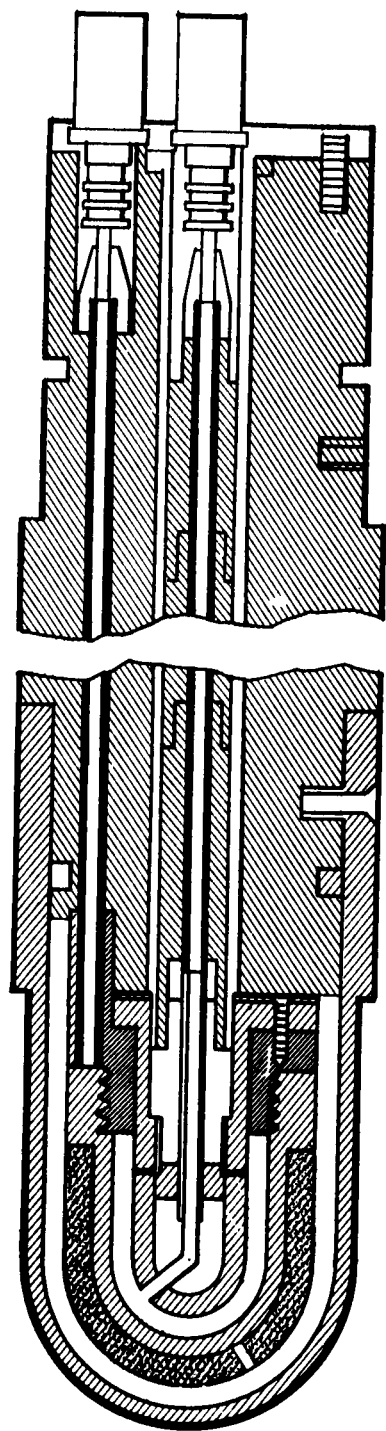


FIG. 13 GRAPHITE IONIZATION CHAMBER

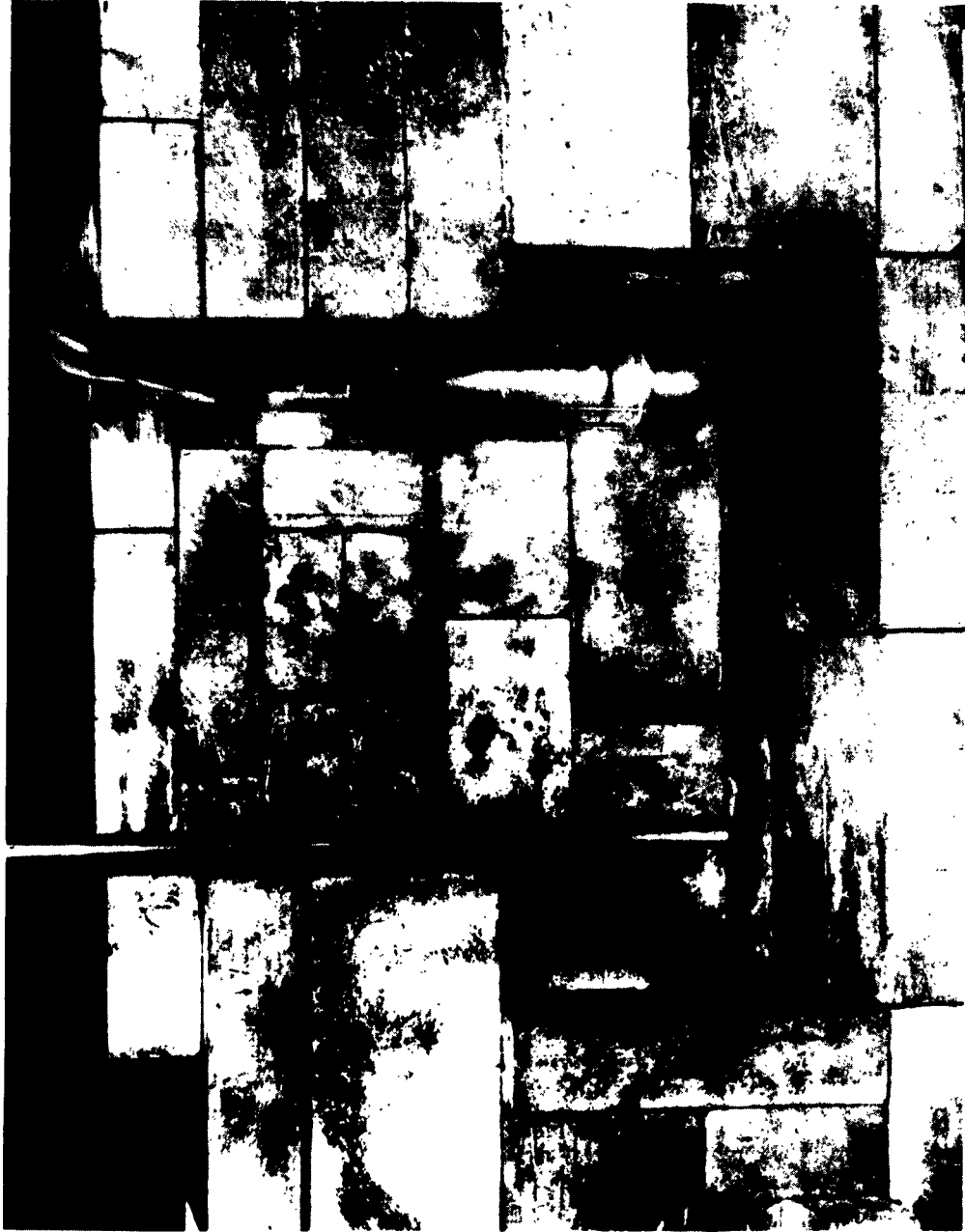


FIG. 14 SHIELDED ASSEMBLY FOR ASSAY OF COBALT-60 SLUGS

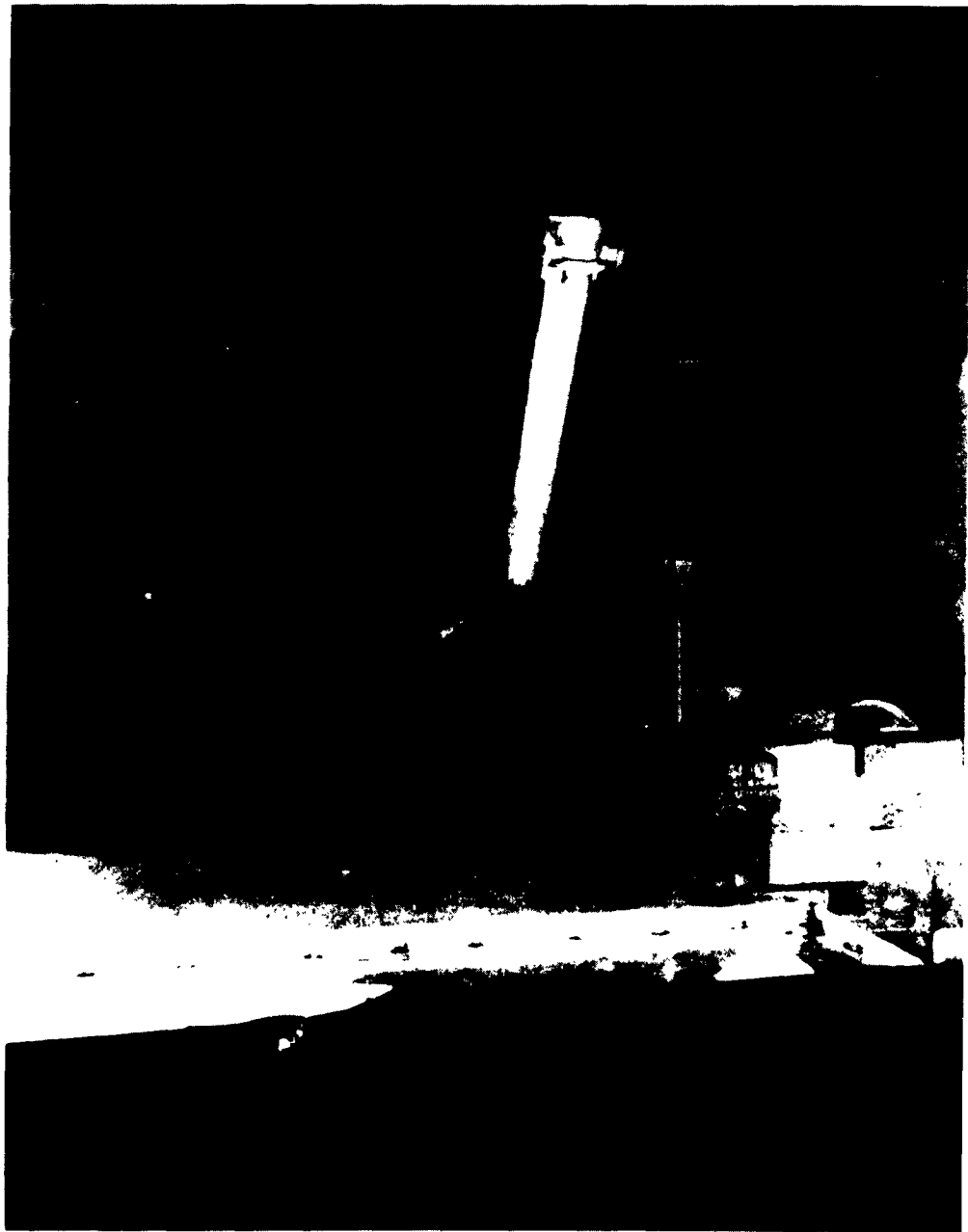


FIG. 15 LOCATION OF COBALT-60 SLUG ASSAY ASSEMBLY



FIG. 16 OPEN AIR ASSAY ASSEMBLY FOR COBALT - 60 REFERENCE SLUG

TABLE 11

DISTANCE (FT)	CURRENT (AMPS X 10 ¹¹)	DOSE RATE (R/hr)	DOSE RATE ¹ (R/hr) _C	CURIES	(CURIES) _D ² (DECAY CORRECTED)	DEVIATION (FROM MEAN)
0.5	24.7	482	495	8.78	8.94	0.02
1.0	6.0	117	120	8.50	8.65	0.27
2.0	1.63	31.8	33.7	9.26	9.43	0.51
4.0	0.375	7.32	7.52	8.50	8.65	0.27

Ave. 8.92 ± 0.27

FIG. 17 THE CURIE VALUE OF REFERENCE SLUG IN OPEN AIR

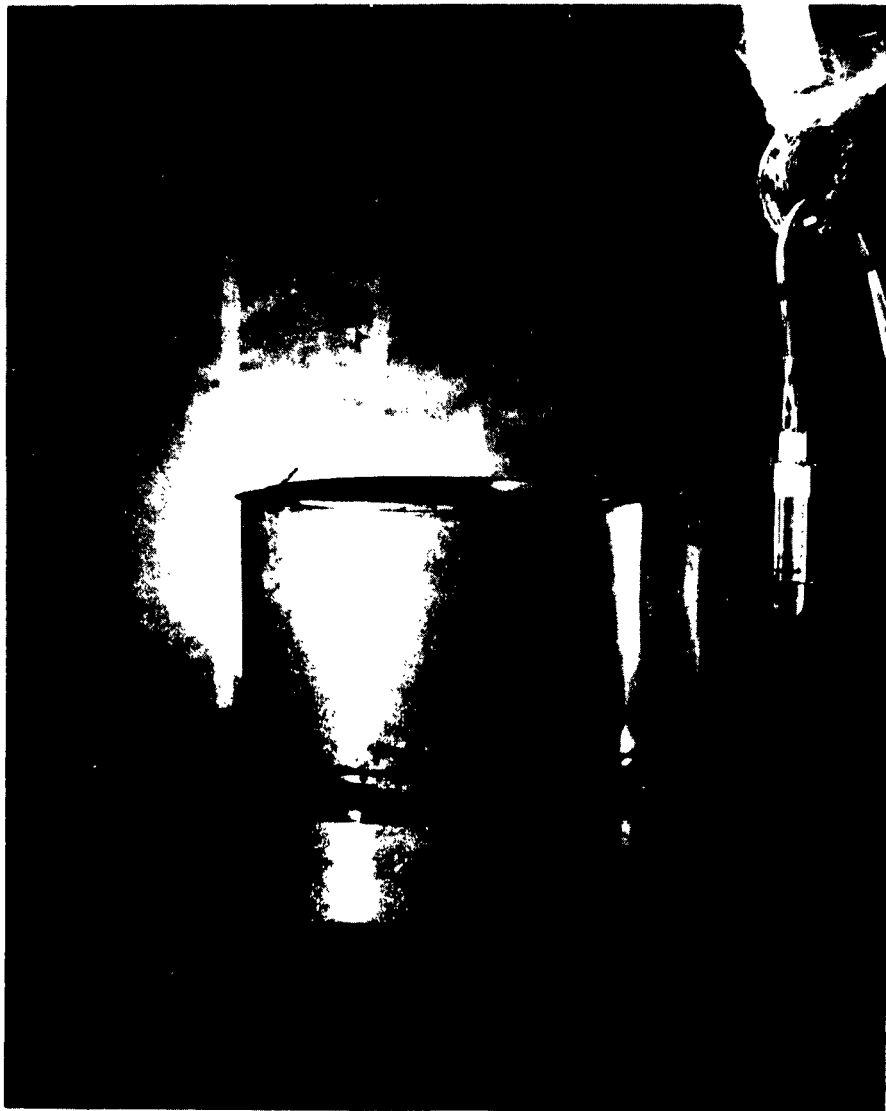


FIG. 18 ASSEMBLED COBALT - 60 IRRADIATION SOURCE

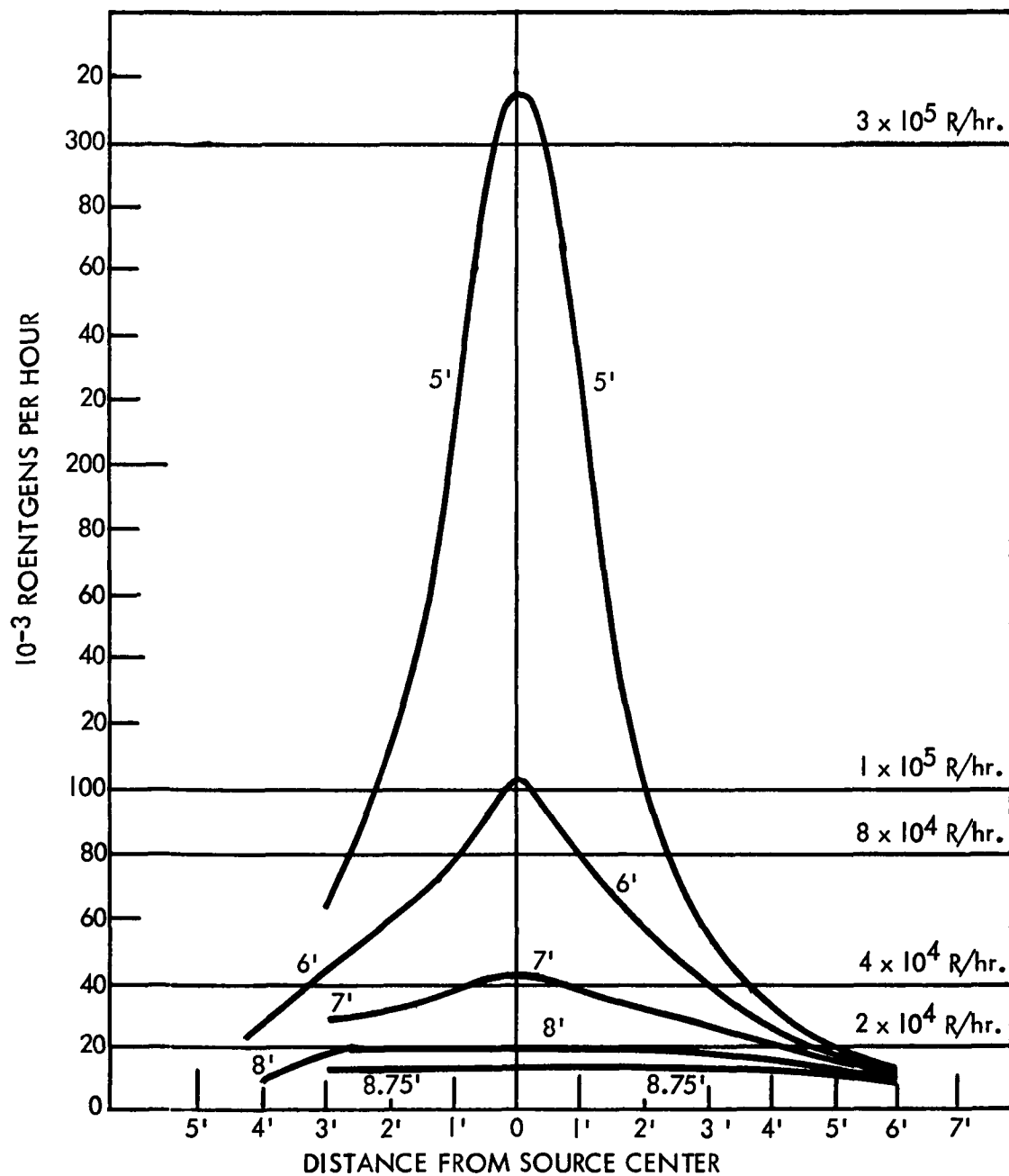


FIG. 19 LINES OF CONSTANT HEIGHT COBALT - 60 SOURCE

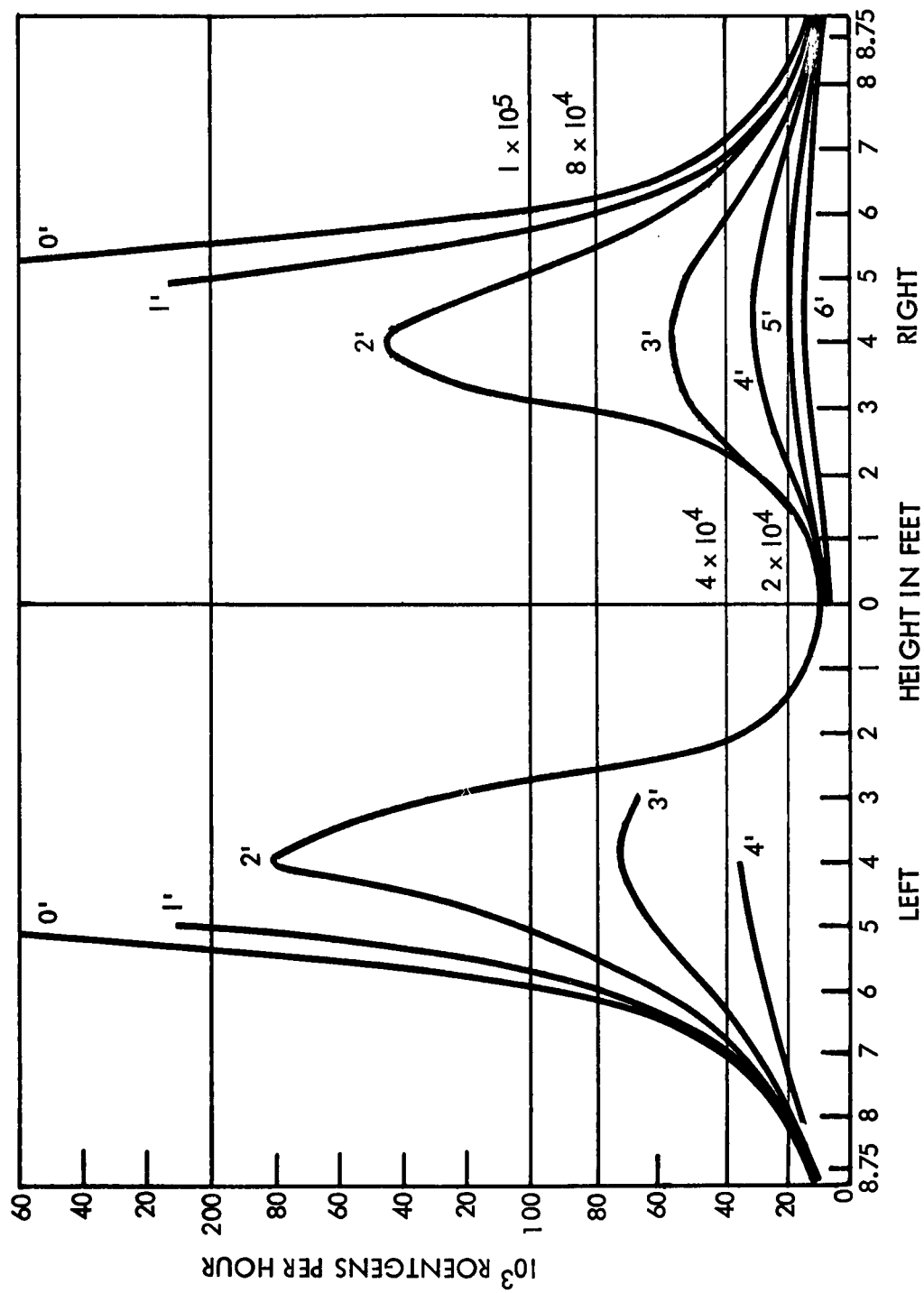


FIG. 20 LINES OF CONSTANT DISTANCE FROM COBALT-60 SOURCE

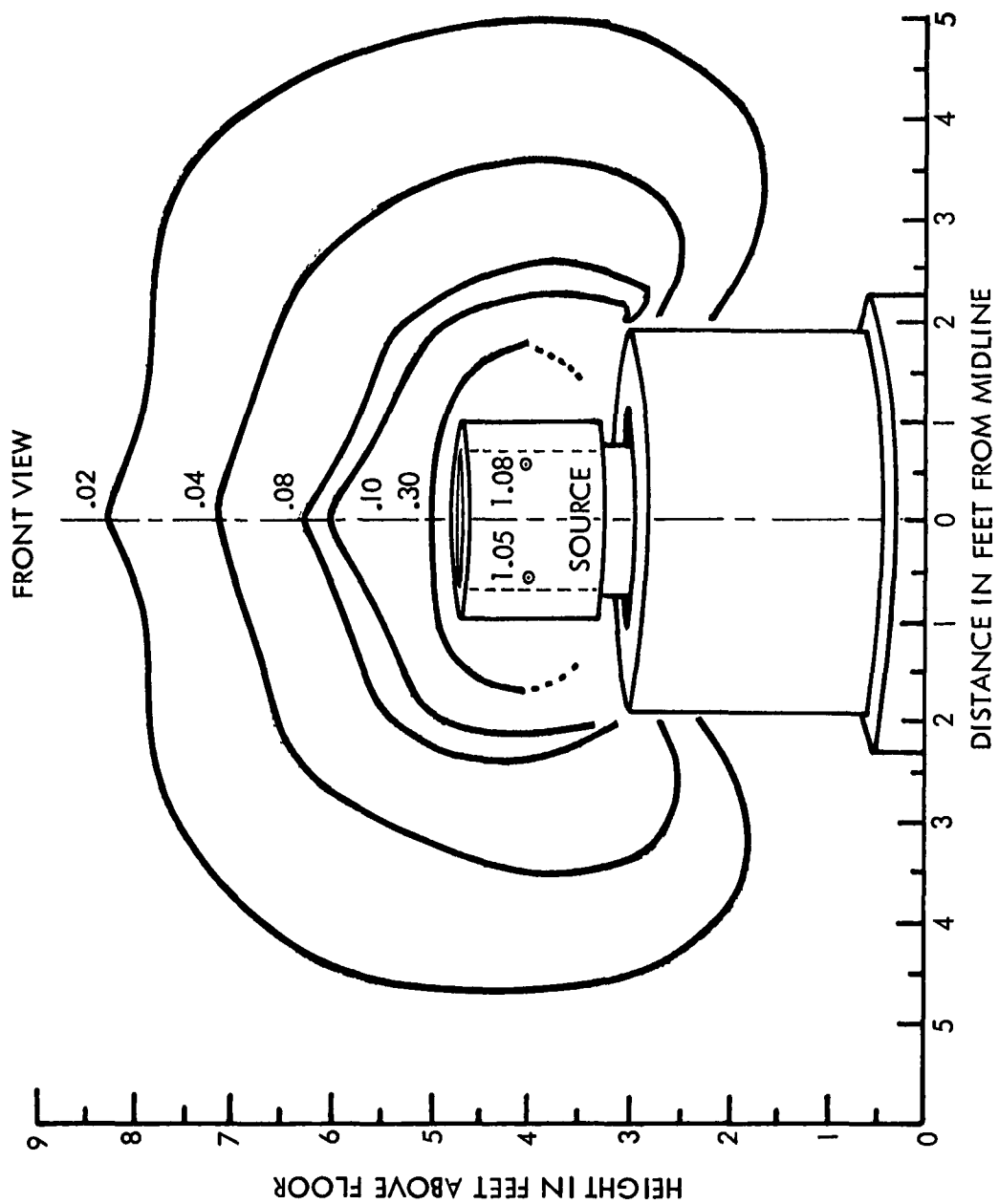


FIG. 21 ISODOSE PLOT OF COBALT-60 SOURCE IN MEGARENTGENS PER HOUR AT THE VERTICAL MIDPLANE

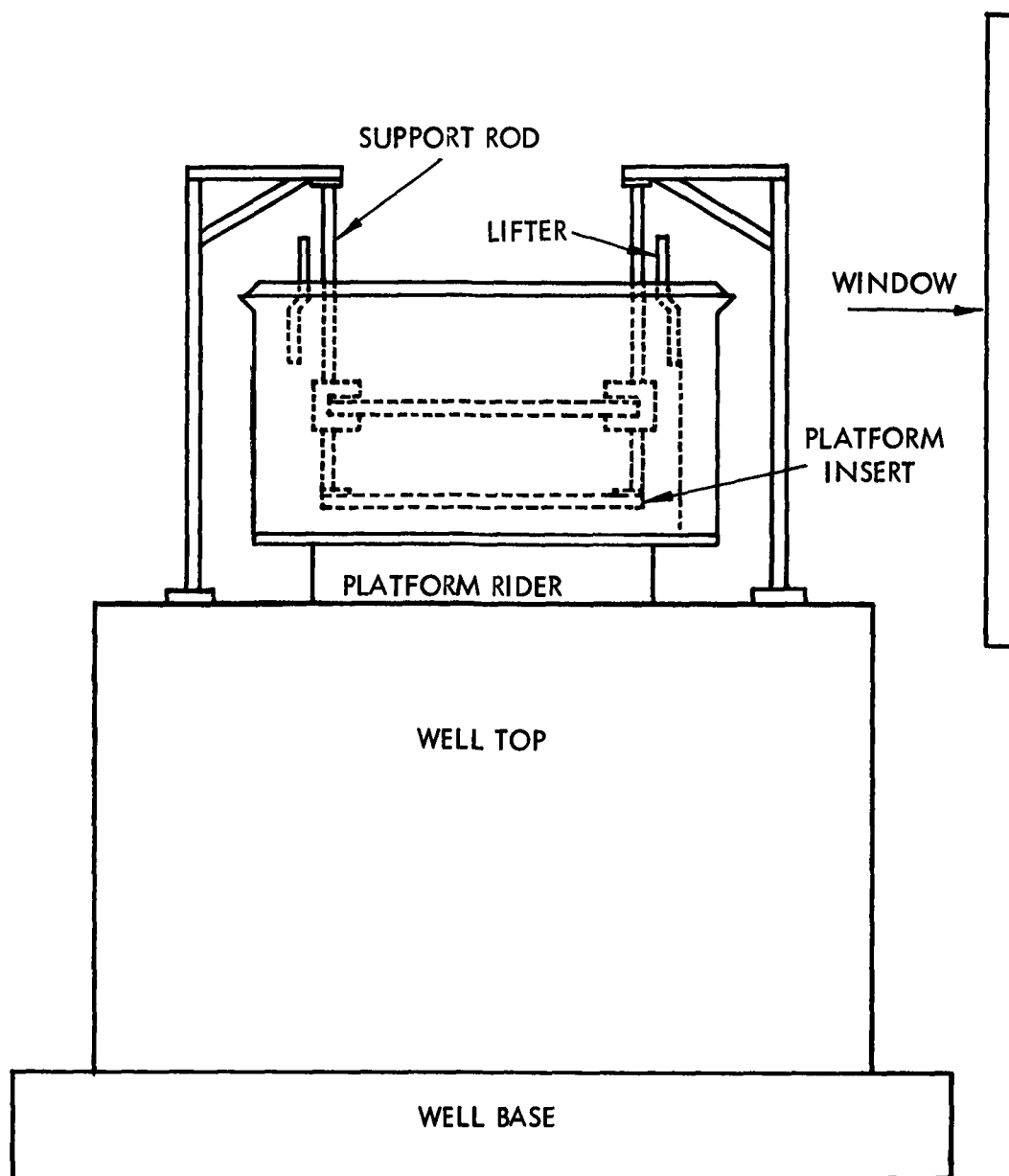


FIG. 22 INLAND TESTING LABORATORIES'
COBALT-60 SOURCE AND SUPPORTING EQUIPMENT

START-UP OF THE CRITICAL EXPERIMENT REACTOR

by

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The Critical Experiment Reactor operated at Air Force Plant No. 67 by Lockheed Nuclear Products Branch is used to establish core loadings for the Radiation Effects Reactor and to determine all pertinent nuclear characteristics of each RER core.

The CER was designed, manufactured, and installed by the Atomic Power Equipment Department of General Electric Company. Facility design was by Lockheed. Facility construction was begun December 27, 1957; and installation of the CER was begun April 9, 1958.

Criticality was achieved June 9, 1958, with a rectangular 20-element array containing approximately 3.260 kg U-235 and having approximately 0.3% excess reactivity. The four fuel-poison control rods were determined to have a shutdown worth of approximately $4\frac{1}{2}\% \Delta k/k$ each. The silver-cadmium regulating rod is worth approximately $0.3\% \Delta k/k$ negative reactivity. The RER operational loading, containing 11.6% reactivity, was achieved in the CER with a 32-element array, approximately 5.372 kg U-235. For this core the temperature coefficient is negative, approximately $-7.01 \times 10^{-5} \Delta k/k$ per °C at 75° F. The void coefficient is approximately $-4.35 \times 10^{-4} \Delta k/k$ percent void. The critical and operational core loadings, rod worth, and coefficient values were in good agreement with previously calculated values.

Operation has proved the CER to be stable and simple to control; and, since the initial "bugs" in the instrumentation channels and in the mechanical systems have been eliminated, most of the systems have been trouble-free.

The Lockheed Critical Experiment Reactor, illustrated in Figure 1, is housed in a prefabricated metal building 28 feet x 48 feet x 20 feet high, divided into two major sections, as shown in Figure 2 - reactor area and the control area. The reactor itself

is submerged in a pool of demineralized water 10 feet x 17-1/2 feet x 20 feet deep and controlled from the console illustrated in Figure 3. Since measured radiation levels at the pool surface are below AEC recommended levels for a 40-hour week exposure during the essentially zero power reactor operation, the control area has no shielding except that afforded by the water in the pool and by a portion of the concrete structure of the pool wall. The facility houses, in addition to the reactor, the reactor control system, auxiliary nuclear instrumentation, and very limited shop and office space. The entire building is insulated and heated; and, in addition, the control area is air-conditioned to provide a uniform environment for the reactor instrumentation. Auxiliary equipment includes a one-ton A-frame crane, spanning the pool, and a continuous circulation water treatment unit comprised of a pump, a filter, and a mixed bed ion-exchange resin demineralizer.

The facility housing the CER was designed by Lockheed. The construction contract was awarded to the Flagler Company and construction was started December 27, 1957. The facility design organization provided engineering coverage and liaison, and construction was approximately complete by May 16, 1958.

The Critical Experiment Reactor was designed and manufactured by the Atomic Power Equipment Department of General Electric Company. Installation started April 9, 1958, and was approximately complete by May 23, 1958. The Kaminer Construction Corporation performed the reactor installation under General Electric subcontract, with engineering liaison provided by General Electric Company and Lockheed.

The Critical Experiment Reactor, commonly known as the CER, is a zero-power reactivity measurement device used for establishing operational core loadings for each Radiation Effects Reactor core and for determining pertinent nuclear characteristics for each such core prior to transfer of the core to the RER. Although the purpose of the reactor is not that of performing basic nuclear mock-ups of new cores, the accessibility of the core and reflector locations, the central location of the control rods, and the nuclear instrumentation present - all make the CER well suited for nuclear mock-up work. Possible applications include investigating core-reflector geometries, testing various type fuel elements, and determining the effects of various reflector materials.

The CER is a swimming pool type reactor, the internal components of which are identical to those of the RER. Fuel elements are of the flat-plate type, containing approximately 176 grams of highly enriched U-235 per element. Four fuel-poison control rods within the core structure and a poison regulating rod on the periphery of the core provide the conventional regulation, safety, and shim functions. The grid, which is arranged as shown in Figure 4, provides space for inserting 32 fuel elements, 4 control rods, the regulating rod, and the 10-curie polonium-beryllium neutron source. In addition, various spare openings around the periphery of the core may be used for insertion of various reflector materials.

The grid and scram damper assembly, which supports the core of the reactor, is located 10 feet 4-1/2 inches below the normal pool surface. This assembly is itself supported by a stand of four stainless steel pipes mounted on a pedestal on the pool bottom. The CER grid and scram damper assembly is an exact duplicate of the corresponding RER assembly. In fact, this assembly and a number of other components of the CER have been designed to serve as spares for the RER and to make the CER as nearly identical to the RER as possible, so that the nuclear mock-up of the RER will be more realistic. These components (namely, the grid and scram damper assembly, the fuel elements, the control rods and drives, the regulating rod and drive, and the fission counter and drives) are described in a companion report on the RER.*

Significant features of the CER include the open core support structure, which provides accessibility for variation of core geometry and conduct of critical experiments, and the traversable top head, which was designed to make the center of the core easily accessible for control rod placement. Rod drive positions are indicated to the nearest hundredth of an inch, permitting detection of reactivity changes as small as $10^{-4} \Delta k/k$ with the regulating rod.

Nuclear detectors for the CER are arranged around the periphery of the core in water-proof cans. The start-up channel and low-level period channel receive their intelligence from the fission counter attached to the top head. This counter has limited vertical travel (31 inches). The rest of the detectors, three CIC's and two B-10's, are fixed; but they may be adjusted manually to a slight degree.

The instrumentation system, a block diagram of which is shown in Figure 5, provides multiple coverage from source range to operating power. The log N - period channel and the two flux level safety channel are operated off the three CIC's. The B-10's are data channels especially useful during start-up, but they are not tied into the reactor safety system. The reactor will scram electronically in the event of a short period from either the low-level period channel or the log N - period channel on a high flux indication from either safety channel and in the event of instrument trouble. The period recorder can be switched to record either low-level period or log N - period, and the flux level recorder will record either safety channel. Both recorders are set to scram the reactor at levels more conservative than the electronic scram of the channels being recorded. Additional instrumentation channels available for installation in the pool but not a part of the reactor safety system include four B F₃ channels, two Hornyak button channels, two F N D channels, and two gamma dosimeter channels. Power ranges are shown in Figure 6.

Neither the facility nor the reactor presented any difficult or unusual problems during installation, and both facility construction and reactor installation proceeded approximately on schedule. Checkout of the reactor mechanical systems and the reactor control and instrumentation systems prior to and during start-up revealed a number of minor, but sometimes exasperating inconsistencies, all of which were corrected by the General Electric Company. Typical of these were the control rod

"seated" switches, magnetically actuated by magnets in the control rods, which were found to be mutually interactive. This condition was resolved by relocation of the switches.

Initial operation of the CER, which was carried out by the General Electric Atomic Power Equipment Department, involved a thorough system checkout of all reactor systems, followed by a critical experiment, regulating rod calibration, build-up of the core to the operational RER loading, determination of the reactivity content of the operational core by distributed poison, determination of the temperature coefficient of the poisoned operational core, determination of the void coefficient, and determination of certain other coefficient data. Calculations of these data had been made by the Atomic Power Equipment Department, and they will be compared with the experimental results.

Safety is the keynote of all operation of the Critical Experiment Reactor; and, while the actual operating characteristics of the CER were as yet unexplored, the initial experiments were conducted under highly conservative conditions. The initial critical experiment presented a slight operational difficulty. The eight central fuel elements are made captive by the four control rods; therefore, to load the first eight elements requires access from directly above, precluding operation of the shutdown system. An operational safety philosophy in the CER, however, requires that any addition of reactivity be made with part of the available shutdown held in reserve. Since fuel elements of the type used in this reactor cannot achieve criticality without a water moderator, the paradox was resolved by lowering the pool water level below the core, loading the eight elements, withdrawing two of the four control rods approximately half way, and refilling the pool.

The approach to critical was based on the extrapolation of the inverse count rate data from the two B-10 channels; and, as the experiment progressed, the data indicated that criticality would be achieved with 19.7 elements, including the fuel sections of the four control elements. The plot of the inverse count rate data, shown in Figure 7, was remarkably conventional, with each incremental fuel addition confirming the validity of the extrapolation. The 20-element 4×5 core was critical with approximately $0.3\% \Delta k/k$, which, considering the simplifying assumptions needed to make the mathematical representations capable of solution, is a figure in reasonable agreement with the calculated critical loading of 17.9 plus or minus 1.0 elements. The critical loading arrangement is illustrated in Figure 8.

The first experiment following criticality was calibration of the regulating rod. Its worth was established by measuring the rising period induced by incremental removal of the rod. The curve on this method of calibration is given in Figure 9. For the 20-element core, the total worth of the regulating rod was approximately 0.3%, as compared with a predicted 0.3 - 0.5%. The worth of the regulating rod in the operational core was the same as in the critical core.

Once the regulating rod had been calibrated, a number of minor reactivity effects were investigated. Elements with varying fuel contents (plus or minus one gram of U-235 based upon manufacturing quality control records) were cycled through a given core position and were found to produce definite, though inconsistent, reactivity changes. Rotating an element 180 degrees about its longitudinal axis and adjusting its position very slightly also gave detectable reactivity changes. In each case the critical rod configuration was established before and after the change being investigated and the reactivity change was calculated from the change in regulating rod position. It should be noted that the thermocouples used for determining core temperature were initially faulty; hence a number of the rather sensitive reactivity measurements could have been indecisive as a result of water temperature variations. During the Lockheed re-run of the CER Test Program this factor will be considered carefully to permit refinement of the previously determined data where possible.

The control rod calibration was conducted coincident with the build-up to the operational core loading. As each element was added beyond the critical core, the differential worths of the various control rods were determined at a few points by the rising period method. Connecting these points gave a differential curve for each rod, which on integration gave the total rod worth. It should be recognized that calibration of a rod by this method required addition of several elements, and addition of each element had an appreciable effect on rod worth. Hence, considerable uncertainty exists in the experimentally determined rod worths. The worth of one rod is in the vicinity of $4\frac{1}{2}\%$, and the overall worth of four rods is between 18 and 20%. The calculated worth was roughly 6% per rod, or a total of approximately 24% shutdown worth. The operational core loading, as determined by this method, contained 31 elements and had an estimated 11.5% excess reactivity.

To recheck the excess reactivity, poison in the form of cobalt wires was added to the fuel elements throughout the operational core in a quantity calculated to permit criticality with all rods withdrawn. This experiment, which is considered more precise than the experimental method described in the previous paragraph, indicated 11.6% excess reactivity with 32 elements in the core and all rods withdrawn. The loading configuration is shown in Figure 10. The previously calculated operational loading contained 30 elements.

Determination of the temperature coefficient of the fully poisoned operational core was accomplished by measuring the incremental regulating rod motion required to maintain criticality while the pool was being heated. In practice, the reactor was placed on a very long rising period with the regulating rod. The power level was permitted to rise along with water temperature until the reactor temperature coefficient exceeded the reactivity insertion, and the power level would start to drop. The reactor would again be put on a positive period with the regulating rod and the process repeated. Meanwhile, core temperature as a function of time, was determined with thermocouples.

The pool was heated by a steam generator, which pumped steam through coils placed in the pool. The rate of temperature change was approximately 1.5° F per hour and the experiment was run for about six hours. The experimentally determined temperature coefficient at approximately 75° F was $-7.01 \times 10^{-5} \Delta k/k$ per ° C. The calculated temperature coefficient was $-10.6 \times 10^{-5} \Delta k/k$ per ° C.

Due to the physical interference between the distributed poison wire and the void containers, it was not possible to determine the void coefficient of the fully poisoned operational core. Instead, the operational loading with no poison was used for the void coefficient experiment. Plastic strips with measured air voids were inserted in each element, and the critical rod configuration was determined. The experiment was repeated with plastic strips containing the same volume of plastic but having no void. The difference between the two critical configurations was measured in terms of reactivity; and this figure, together with the value of the known void volume, permitted determination of the void coefficient. The experimentally determined value was $-4.35 \times 10^{-3} \Delta k/k$ percent void in water. The previously calculated value $-2.96 \times 10^{-3} \Delta k/k$ percent void in water had been calculated for a fully poisoned operational core just critical with all rods withdrawn.

Operation of the CER since completion of the start-up program has been reasonably trouble-free. In the first two months following completion of the start-up program considerable attention was given to improvement of the operating characteristics of the various instrumentation and control channels. There were no unplanned scrams during operation in this period, and normal maintenance during scheduled down periods was usually adequate for conduct of repairs as required.

The reactor control system has proved to be entirely adequate. Response to control rod position changes is conventional, and the reactor operates stably on its selected power level. Because of the low power levels encountered, transients during operation are practically nonexistent, and very little power level regulation is required.

In view of its simplicity of operation, it is expected that operation of the CER in support of the RER will be entirely satisfactory.

Due to its convenient construction, use of the CER for investigation of basic nuclear phenomena should also prove feasible.

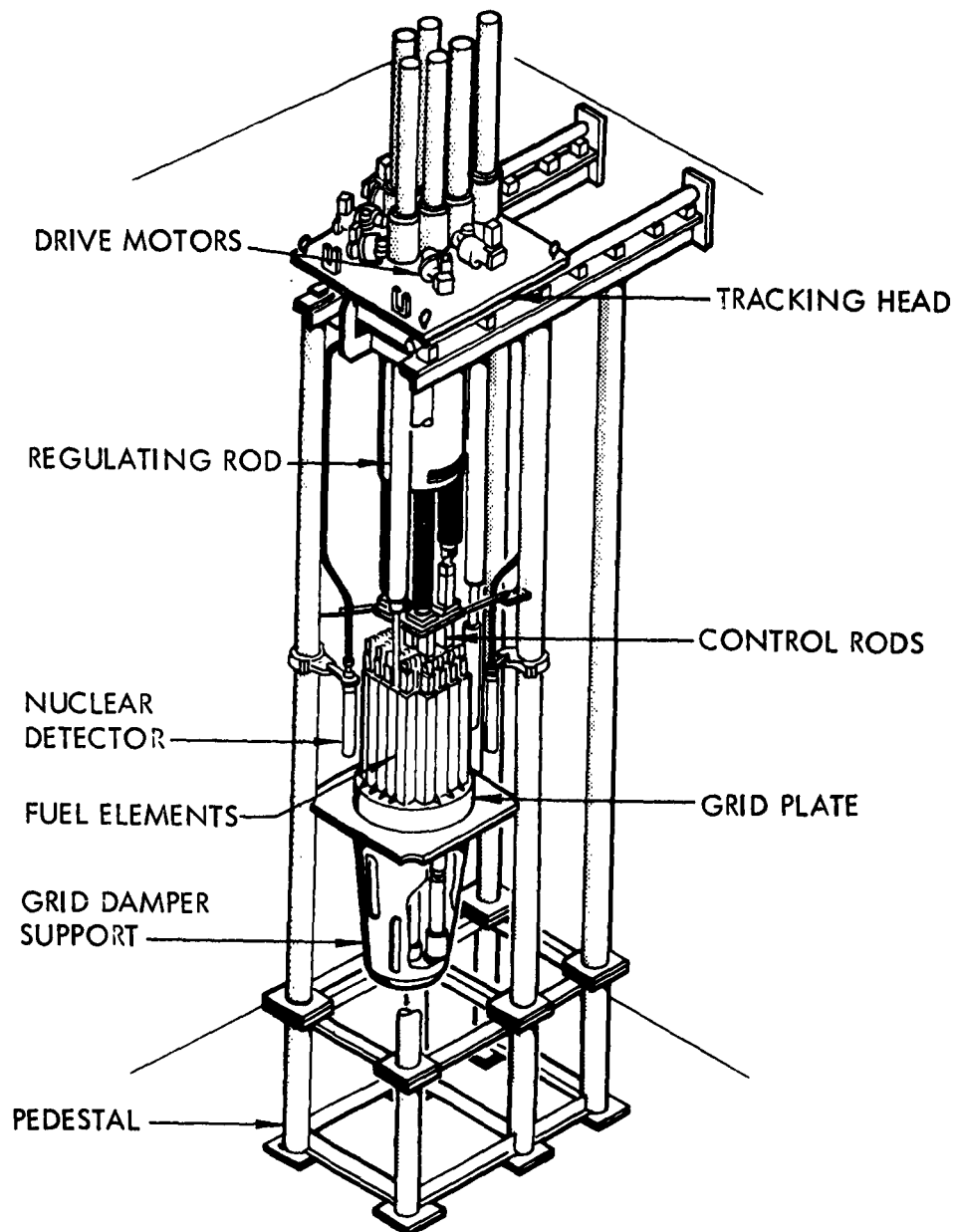


FIG. 1 CRITICAL EXPERIMENT REACTOR

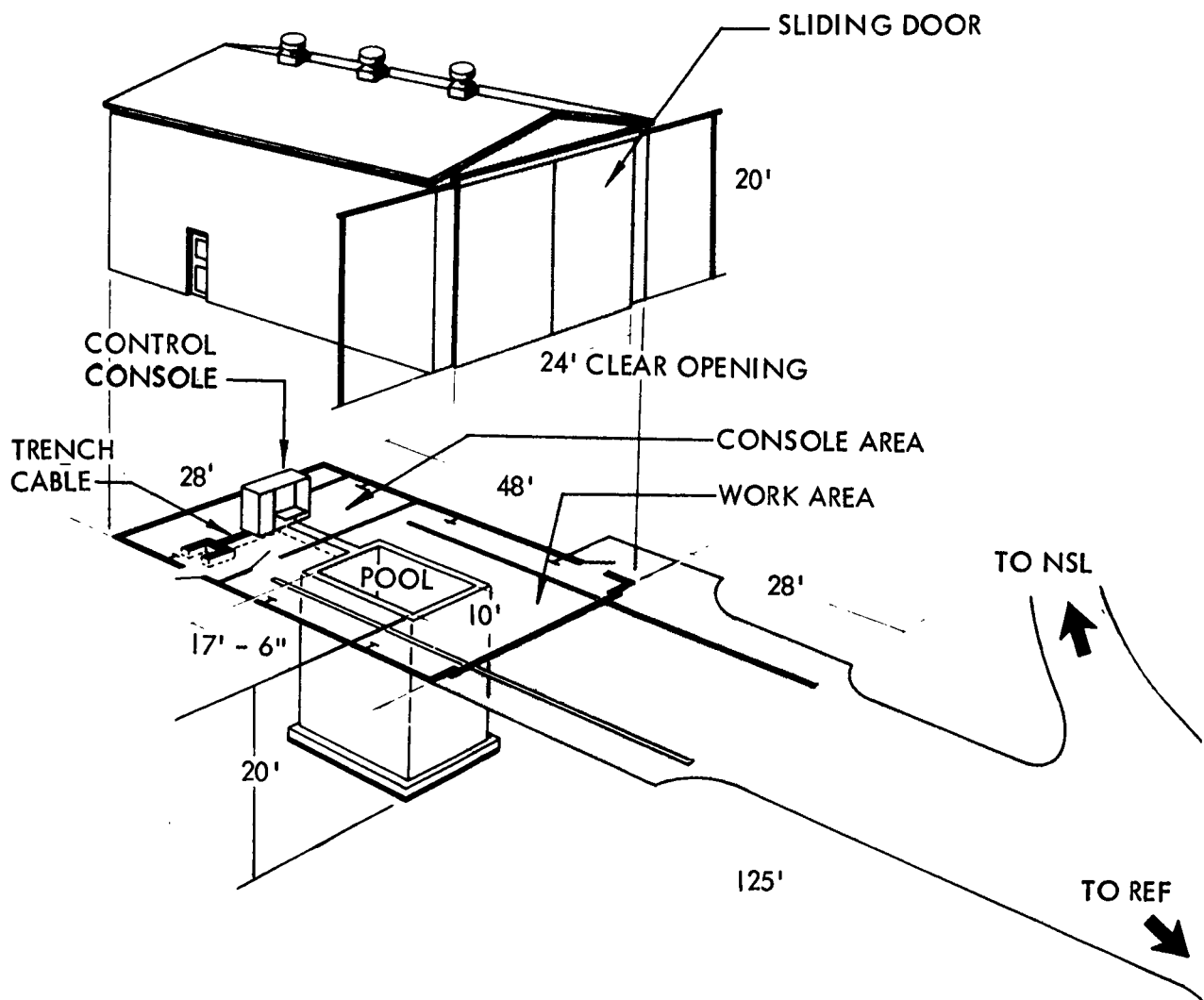


FIG. 2 CRITICAL EXPERIMENT FACILITY

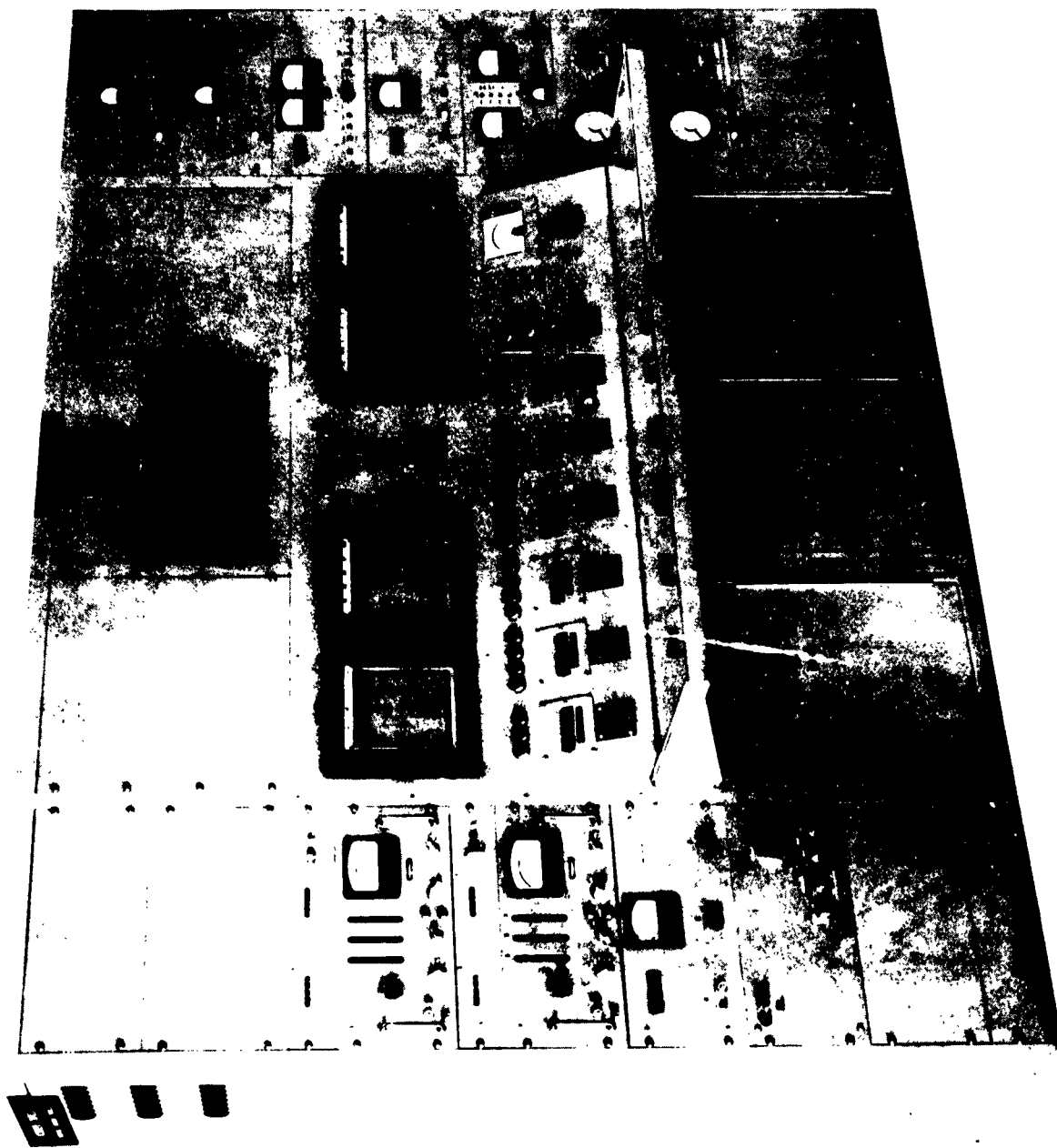


FIG. 3 CER CONTROL CONSOLE

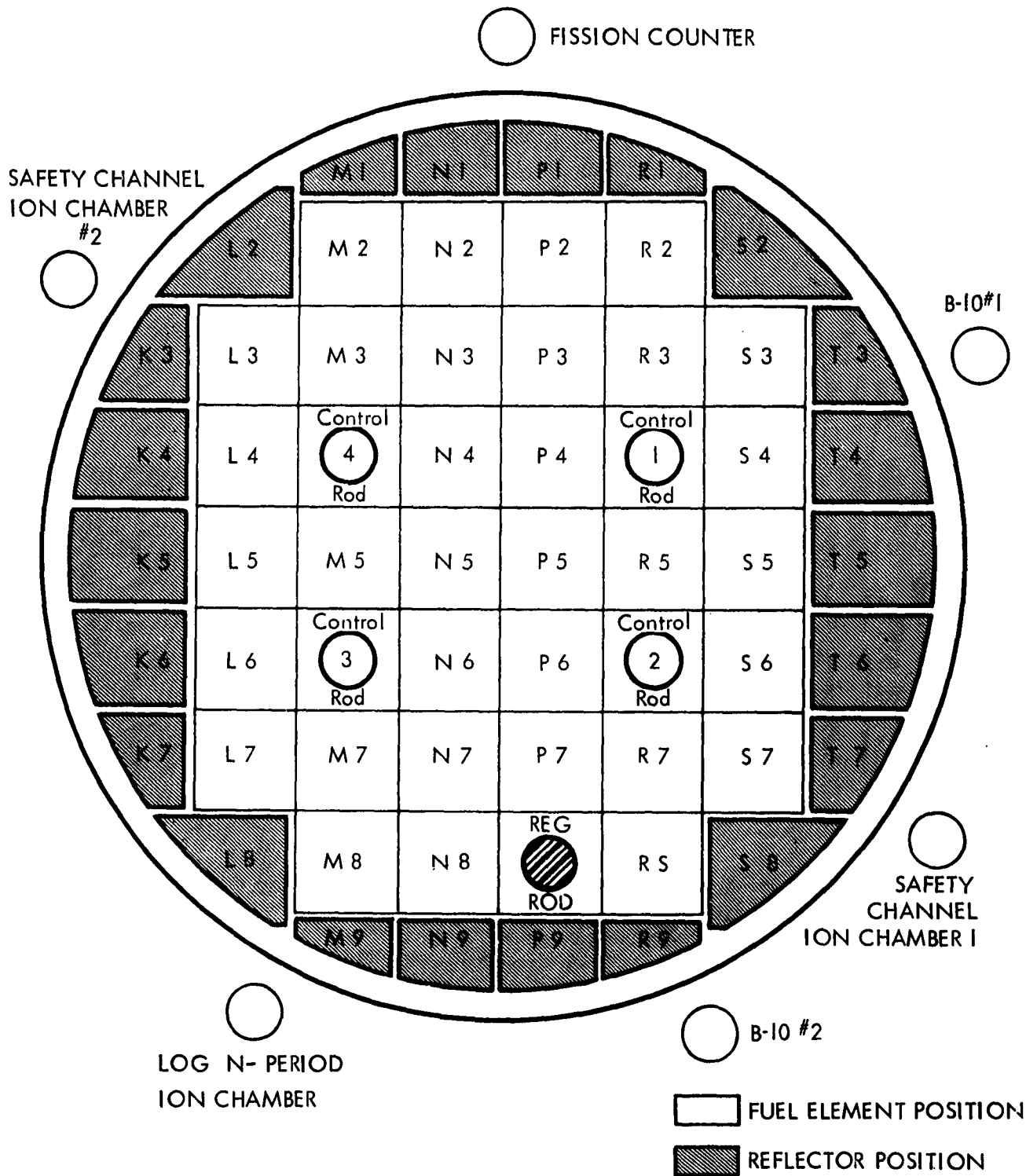


FIG. 4 CER GRID ARRANGEMENT

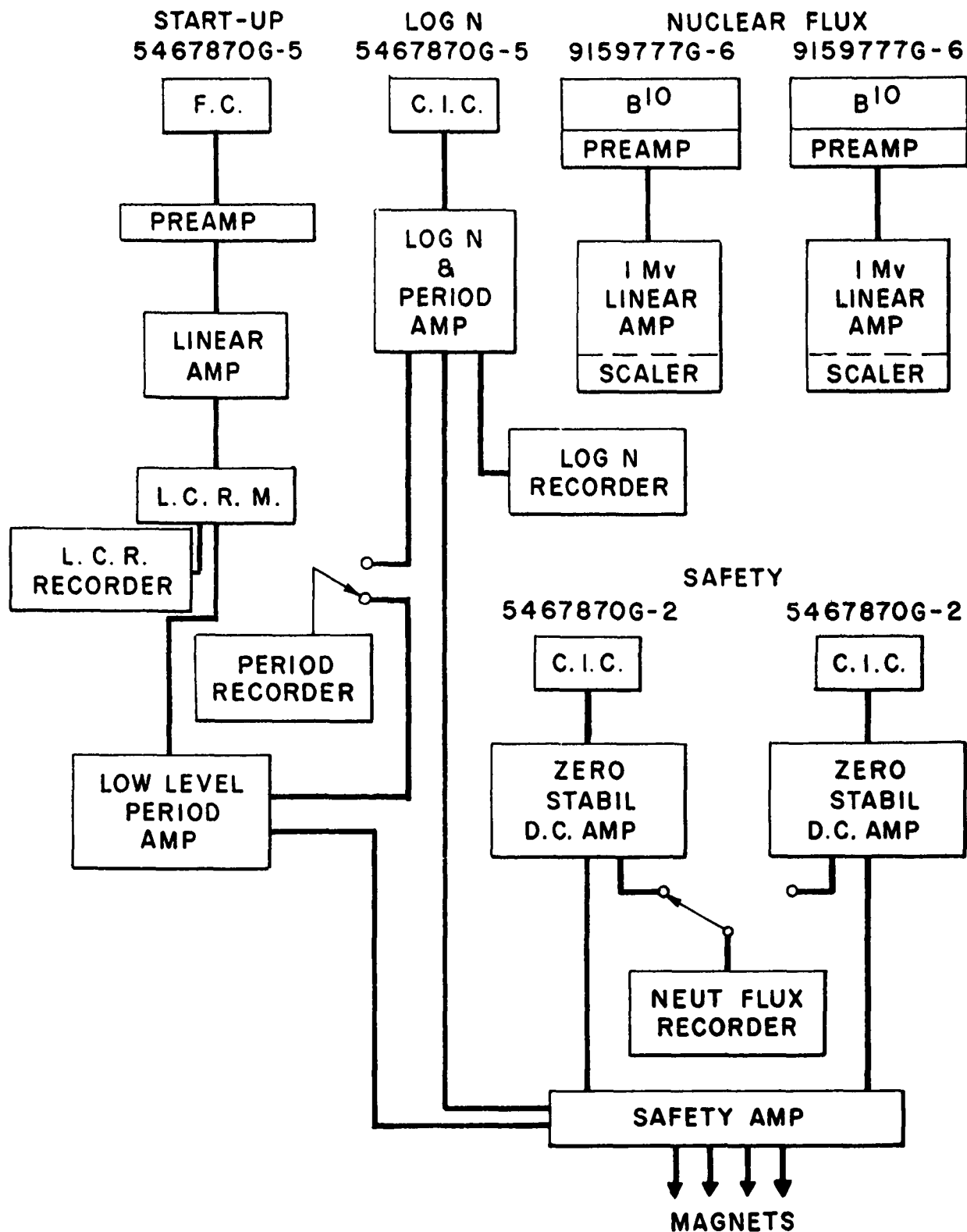


FIG. 5 NUCLEAR INSTRUMENTATION BLOCK DIAGRAM

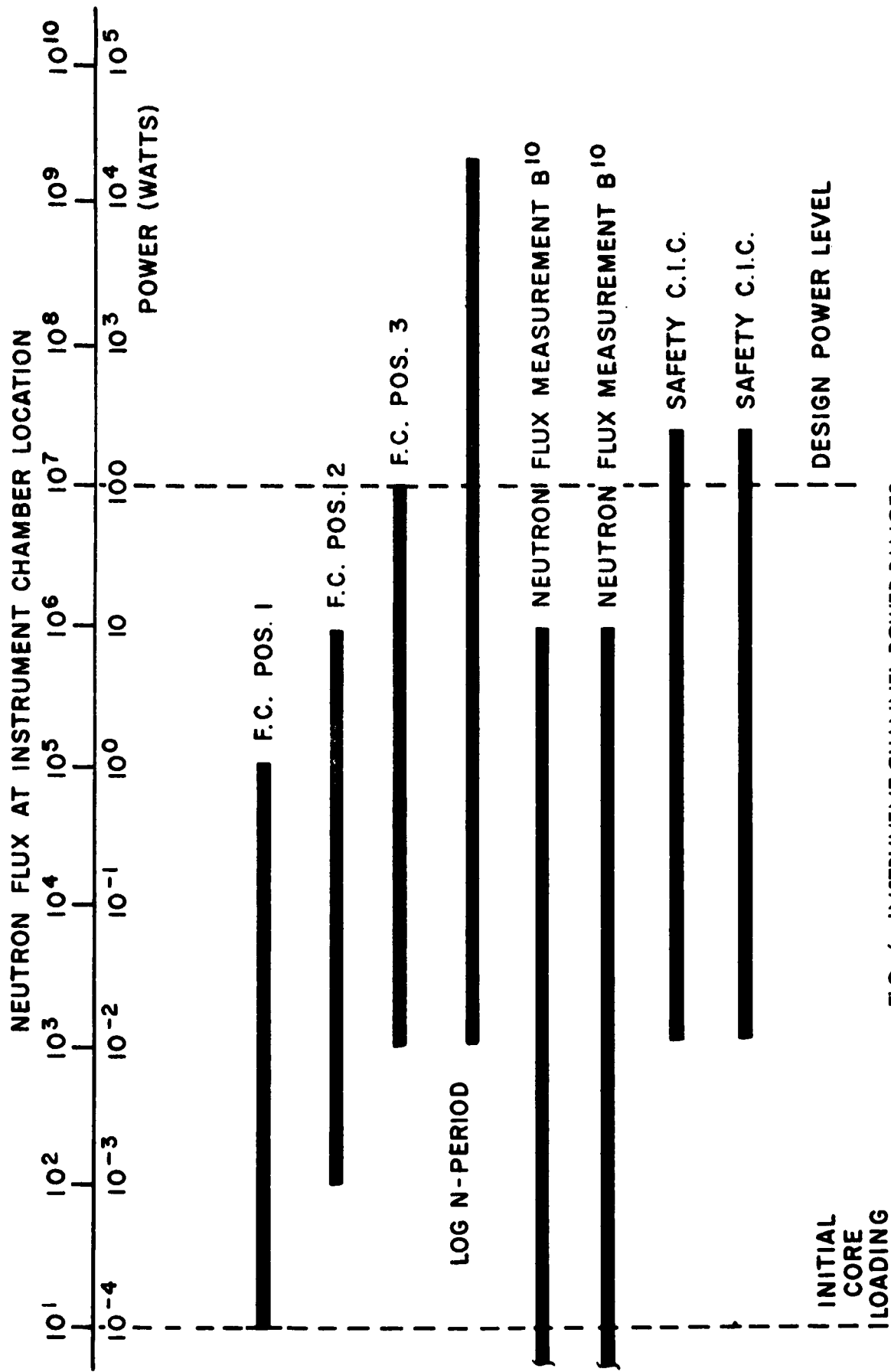


FIG. 6 INSTRUMENT CHANNEL POWER RANGES

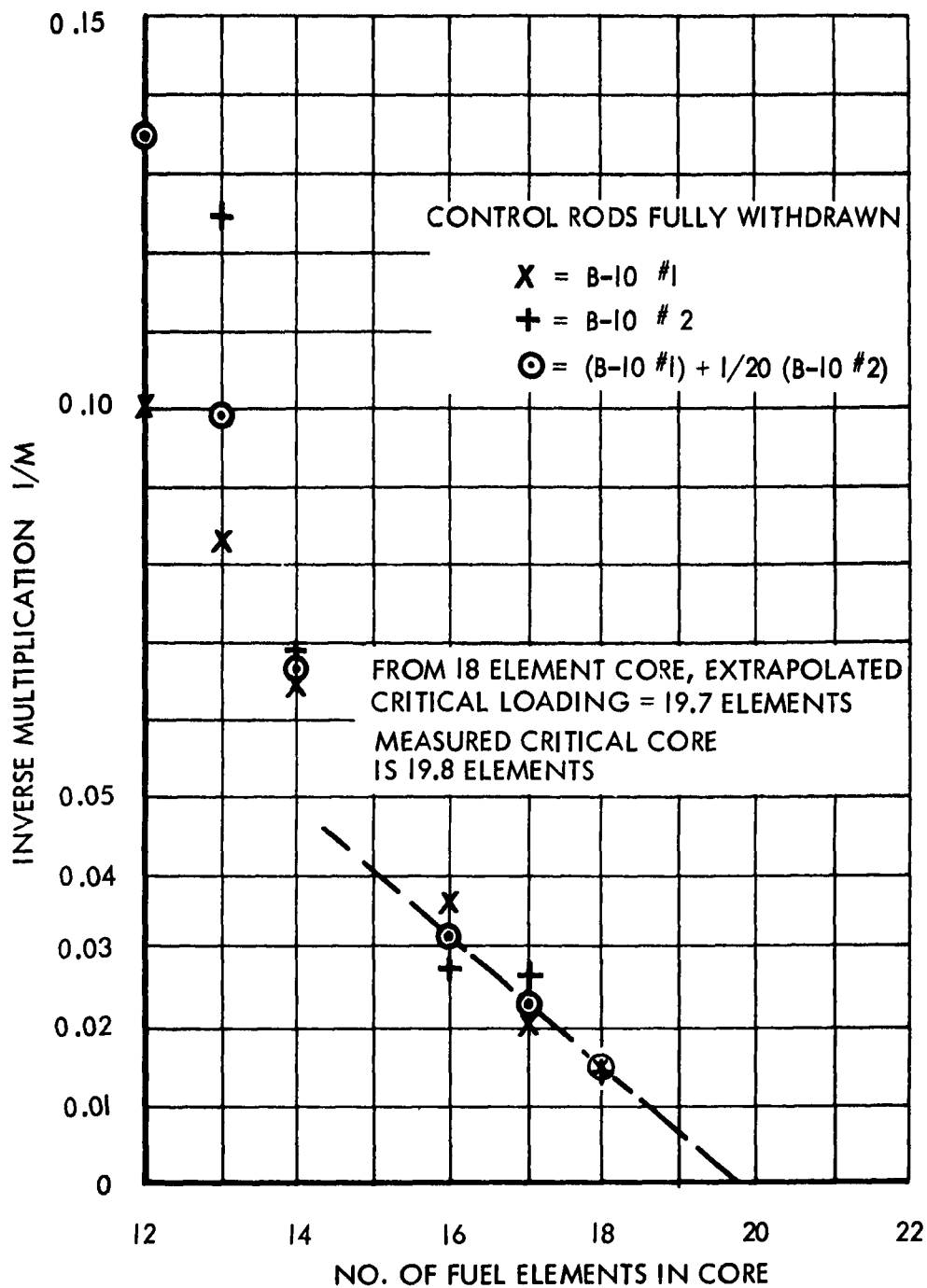


FIG. 7 INITIAL LOADING TO CRITICAL INVERSE MULTIPLICATION CURVES

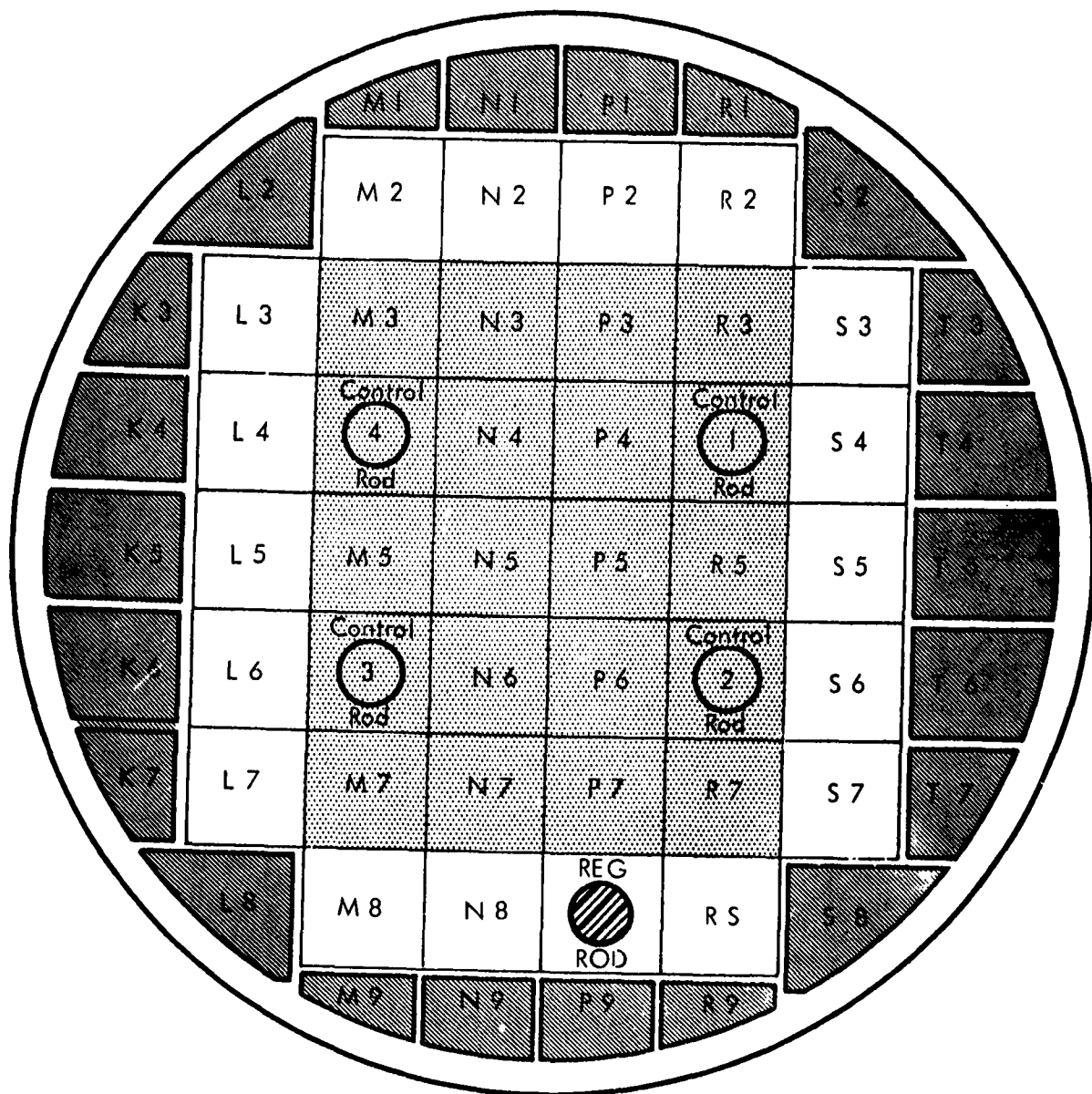


FIG. 8 CRITICAL LOADING

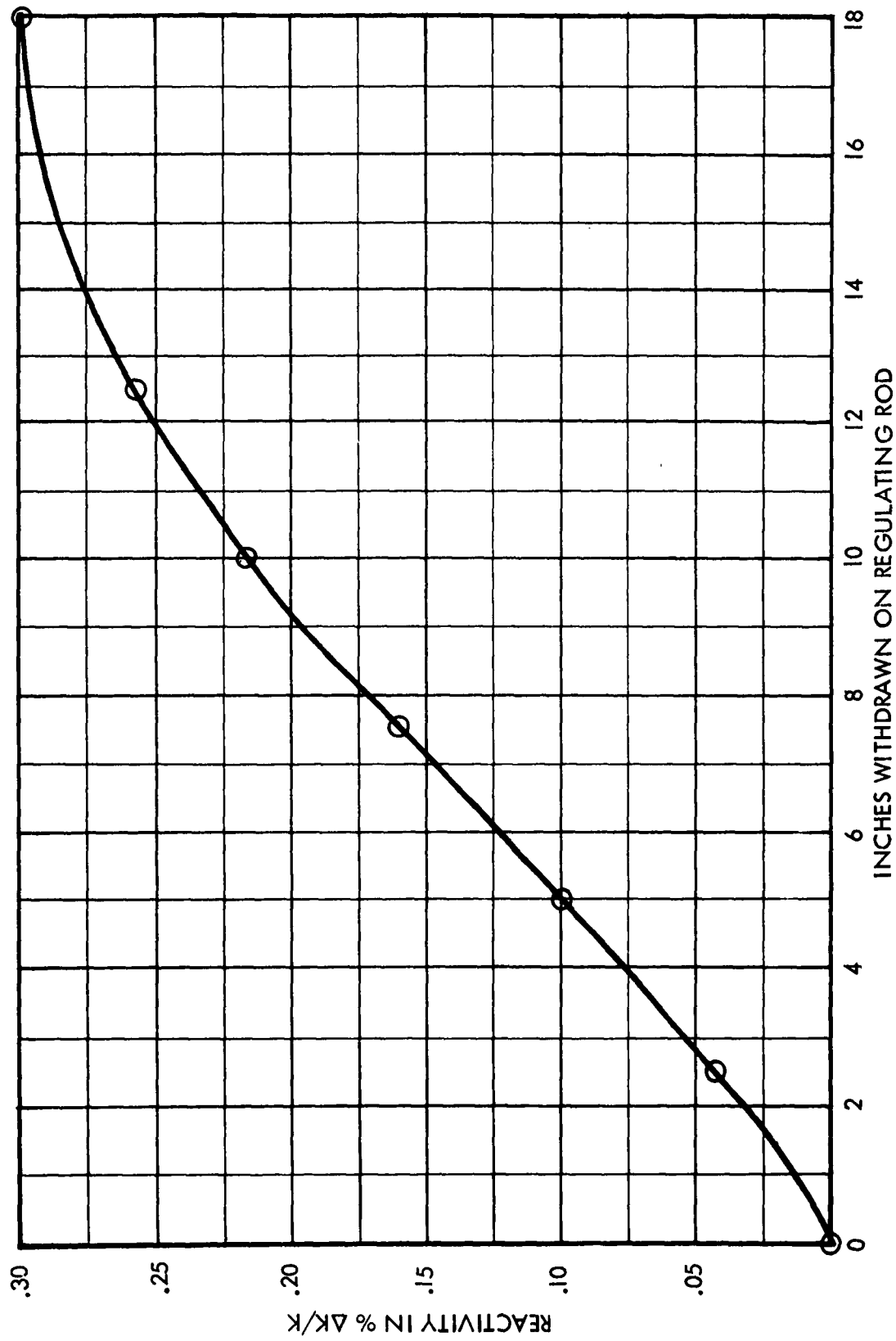
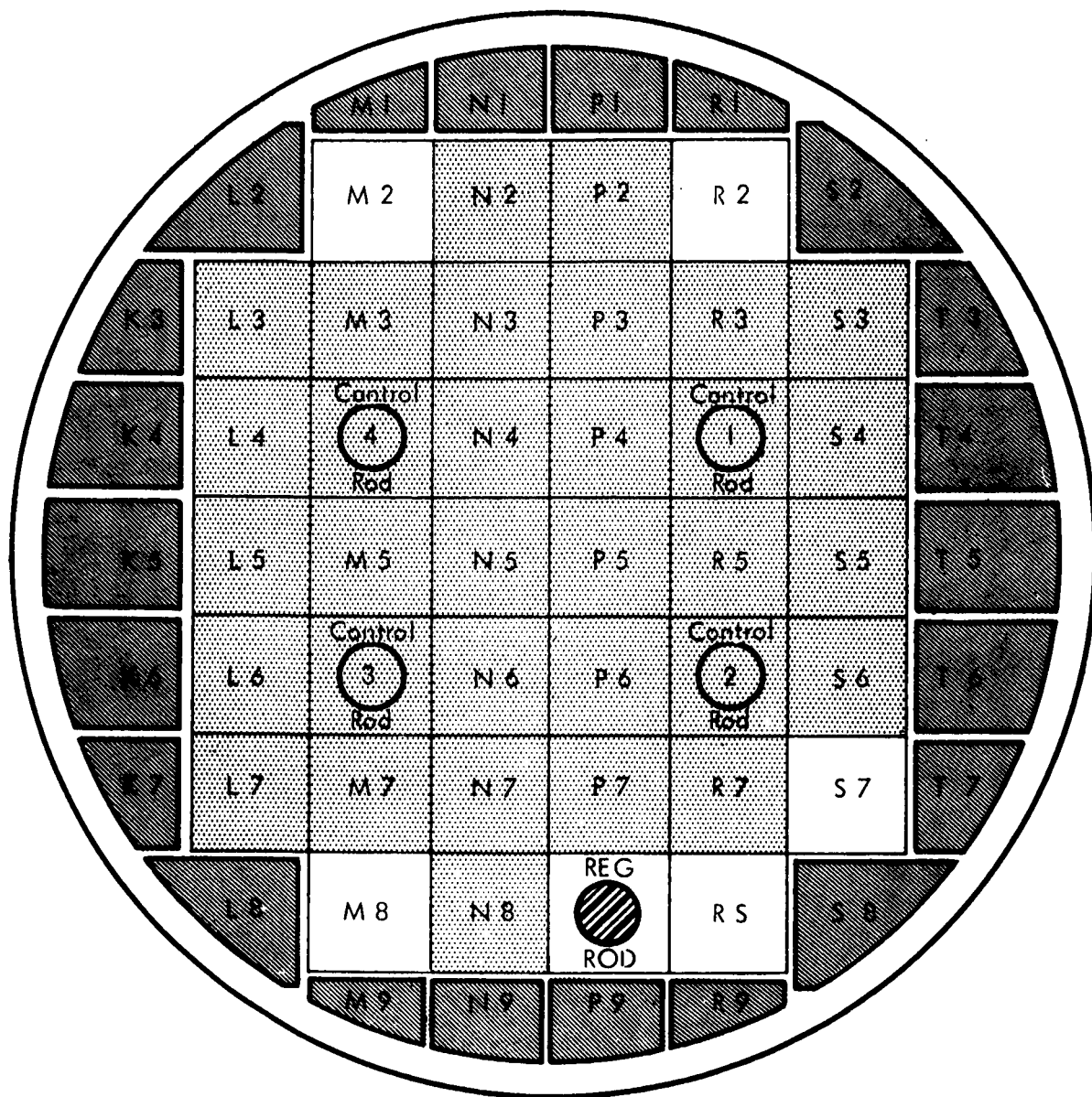


FIG. 9 CALIBRATION OF REGULATING ROD BY RISING PERIOD METHOD IN
32 ELEMENT CORE WITH DISTRIBUTED POISON - ALL CONTROL RODS OUT



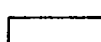
-  OPERATIONAL FUEL CONFIGURATION
-  FUEL ELEMENT POSITION
-  REFLECTOR POSITION

FIG. 10 - OPERATIONAL LOADING

REMOTE AREA MONITORING SYSTEM AT AIR FORCE PLANT NO. 67

by

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To conform to Air Force Plant No. 67 facility requirements for the capability of rapidly irradiating large numbers of test articles, including aircraft subsystems in operation, the Radiation Effects Reactor operates above ground with essentially no shielding. This condition necessitates the monitoring of neutron and gamma flux levels at selected stations of the site and at the perimeter during reactor operation. A remotely operated radiological monitor system has been developed for this purpose.

The functions of this system are as follows:

1. Supply primary power to the remote detectors and instrumentation.
2. Sequentially select the type of radiation to be monitored.
3. Provide a means of conducting the radiation analog currents back to the central station for recording and for identifying the radiation type and originating station.

To accomplish these functions some special concepts for power transmission and for signal selection and transmission have been developed. Individual detectors and special circuitry for the system are described in companion papers.*

POWER TRANSMISSION

In this system, a block diagram of which is shown in Figure 1, a single twin-lead of 75-ohm #14 wire, Amphenol type 14-023, is run between each remote station and the central recording station. Then auto transformer T_1 is adjusted until 110 volts appears across the primary of remote power transformer T_3 . To make this adjustment, two-way radio

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*"Area Monitoring for Radioactivity," R. L. Shipp

"Logarithmic Circuits for Radiation Dosimetry," L. A. Turner

communication is necessary; and, once this voltage is set, no further adjustment of T_1 should be needed. Approximately 300 milliamperes of 60 cps current is required in the line.

By use of the principle of superposition, the network between source and load may be reduced to that shown in Figure 2. As can be seen, the line is tuned to near series resonance; but since the overall "Q" is less than 1, normal temperature variations in reactive elements should cause negligible changes in load voltage.

CHANNEL SELECTION

The keyed oscillator is designed to complete one keying cycle every three minutes. During the first minute of the cycle, the oscillator causes the line to be energized at a frequency of 10 kc; during the second minute interval, there is no output from the oscillator; and for the third minute interval, the excitation frequency is 20 kc. The keying frequency is fed to all remote stations simultaneously so that at all times the same type of radiation is being monitored at every point.

There are two "Bridged T" filters in the remote station block marked "Channel Selector"; filter #1 is designed to suppress 20 kc. Filter #2 suppresses 10 kc. Each filter output is rectified and by means of a control transistor is used to close a relay. Thus, during the first period of the keying cycle when the oscillator frequency is 10 kc, the relay associated with filter #1 is energized; during the second interval with no oscillator output, both relays are de-energized; and for the duration of the third, or 20-kc interval, relay #2 is energized. By suitable connection of the relay contacts, the three detectors are sequentially connected to the d-c input, point P. The d-c analog current path is through a-c blocking filter F_2 , then through the path $T_3 - L_2 - L_1 - T_2$, through the filter F_1 , and into the recorder. Not shown in the diagram are d-c and 60-cycle blocking filters employed at the oscillator output and at each channel selector input.

DATA RECORDING

Two 12-point recorders are employed in the system. One is for the stations at the 3600-foot fence, and the other is for those along the perimeter fence. The recorder balance speed and the numbers of stations assigned are such that during the first minute of the keying cycle, the recorders will print the argon level at all of their associated stations. The originating station of each point of the chart is identified by a printed number. The type of radiation is identified by the position of left-hand and right-hand chart marking pens, which are controlled by the keying oscillator. Thus, in a three-minute interval the system will record and identify the argon, gamma, and fast neutron activity at all stations.

ELECTRONIC HARDWARE

Figure 3 is a schematic diagram of the keyed oscillator, which contains a 10-kc and

a 20-kc oscillator, each of which is in continuous operation. The output at J_2 is 10-kc, zero, then 20-kc, as previously stated, depending on the condition of relays $Ry-A$ and $Ry-B$. These relays are controlled in proper sequence by a motor-driven cam. Jacks J_1 and J_3 provide outlets for utilizing the 10-kc and 20-kc unkeyed signals for test and alignment purposes. A selector switch is provided so that either manual or automatic channel selection is possible.

There is a separate control panel of the type shown in Figure 4 for each remote station. The power cathode follower V_1 is driven by the keyed oscillator output and in turn drives the twin lead to the station. T_1 is used to set the voltage level at the remote station. Meter M_1 provides a visual indication of the activity at its associated station, and J_3 supplies signal to the 12-point recorder.

The circuitry required at each remote installation for channel selections and also the filters for preventing interaction of the various current components are shown schematically in Figure 5. All remote circuitry is located underground in 9-inch cylinders, an arrangement designed to minimize temperature variation.

In stringing the twin lead cable, use was made of all existing supports, such as trees, fences, and power line poles. In the few cleared areas where poles were required, inexpensive fence posts were used.

Although the system design is at this time unproved, it is expected to offer considerable advantage over conventional telemetering methods, both as to initial cost and as to reliability.

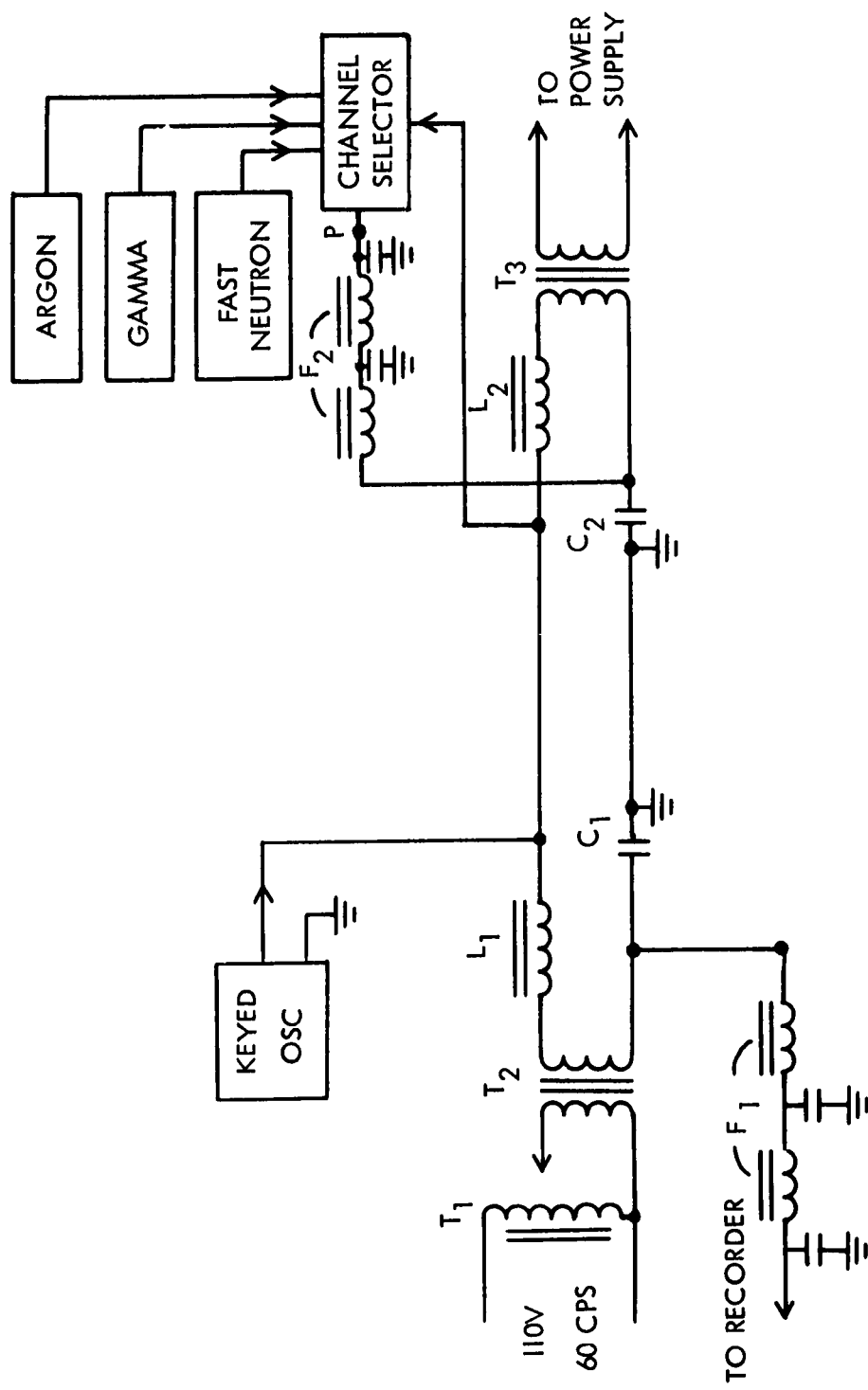


FIG. 1 SYSTEM BLOCK DIAGRAM

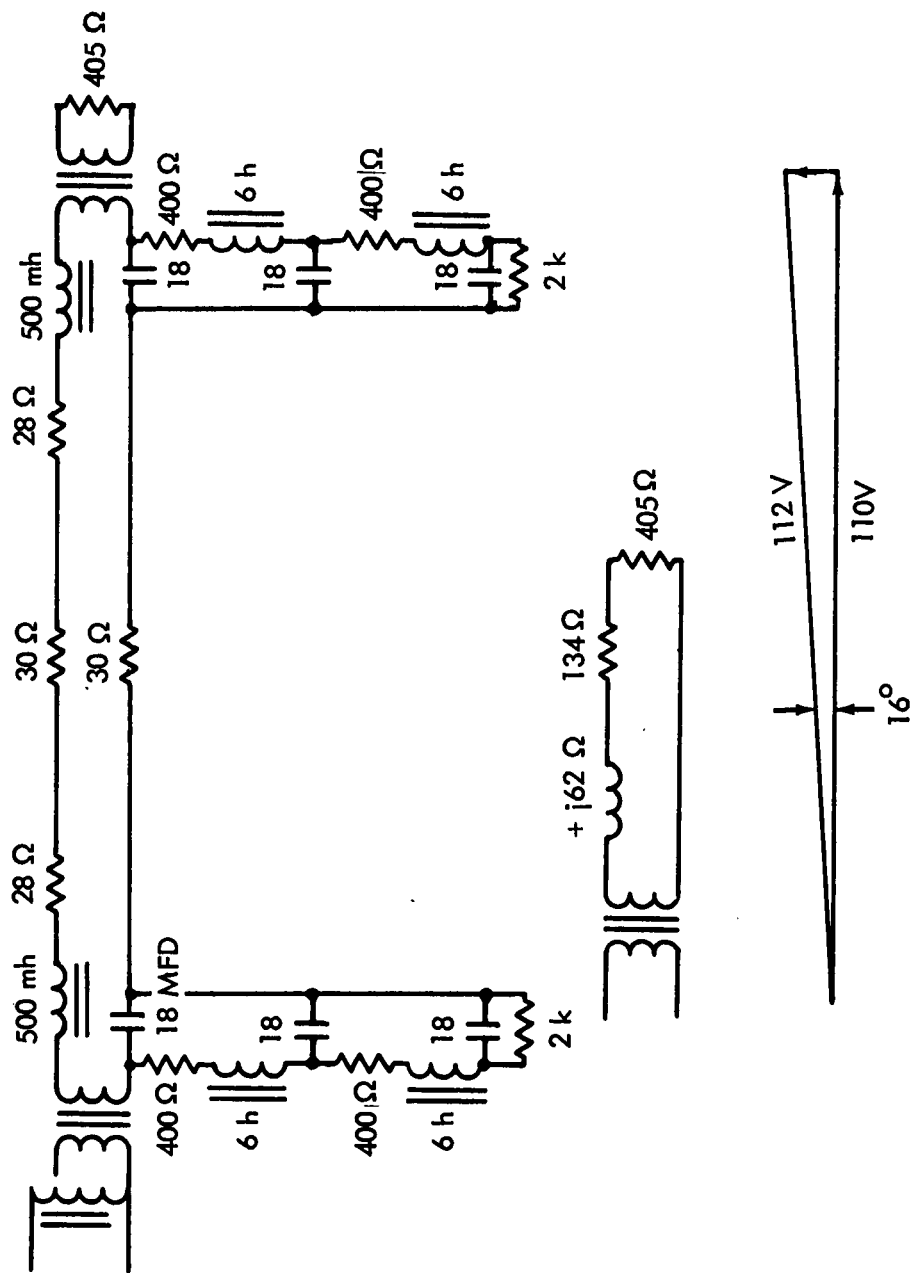


FIG. 2 NETWORK REDUCTION AT 60 CPS

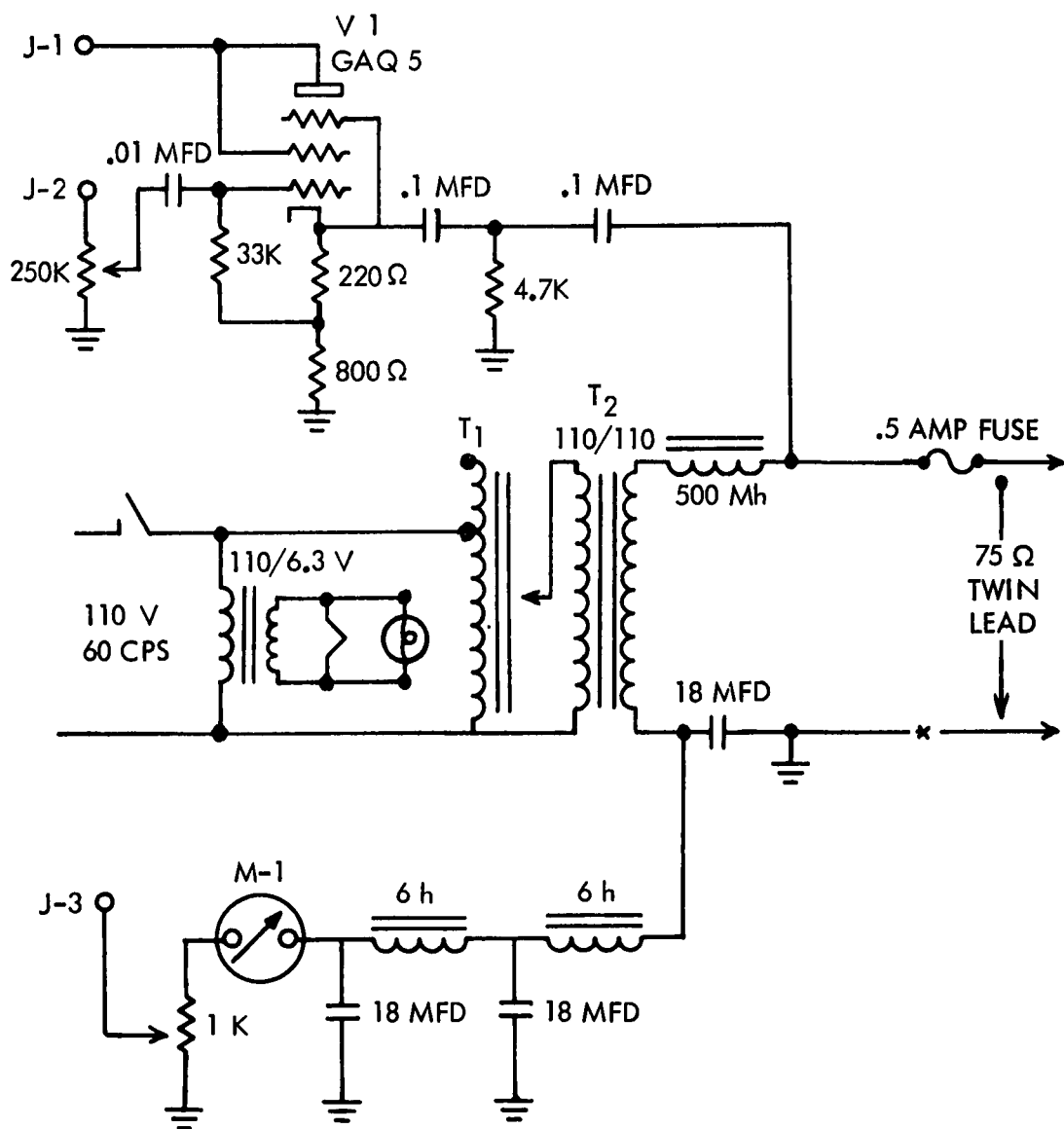
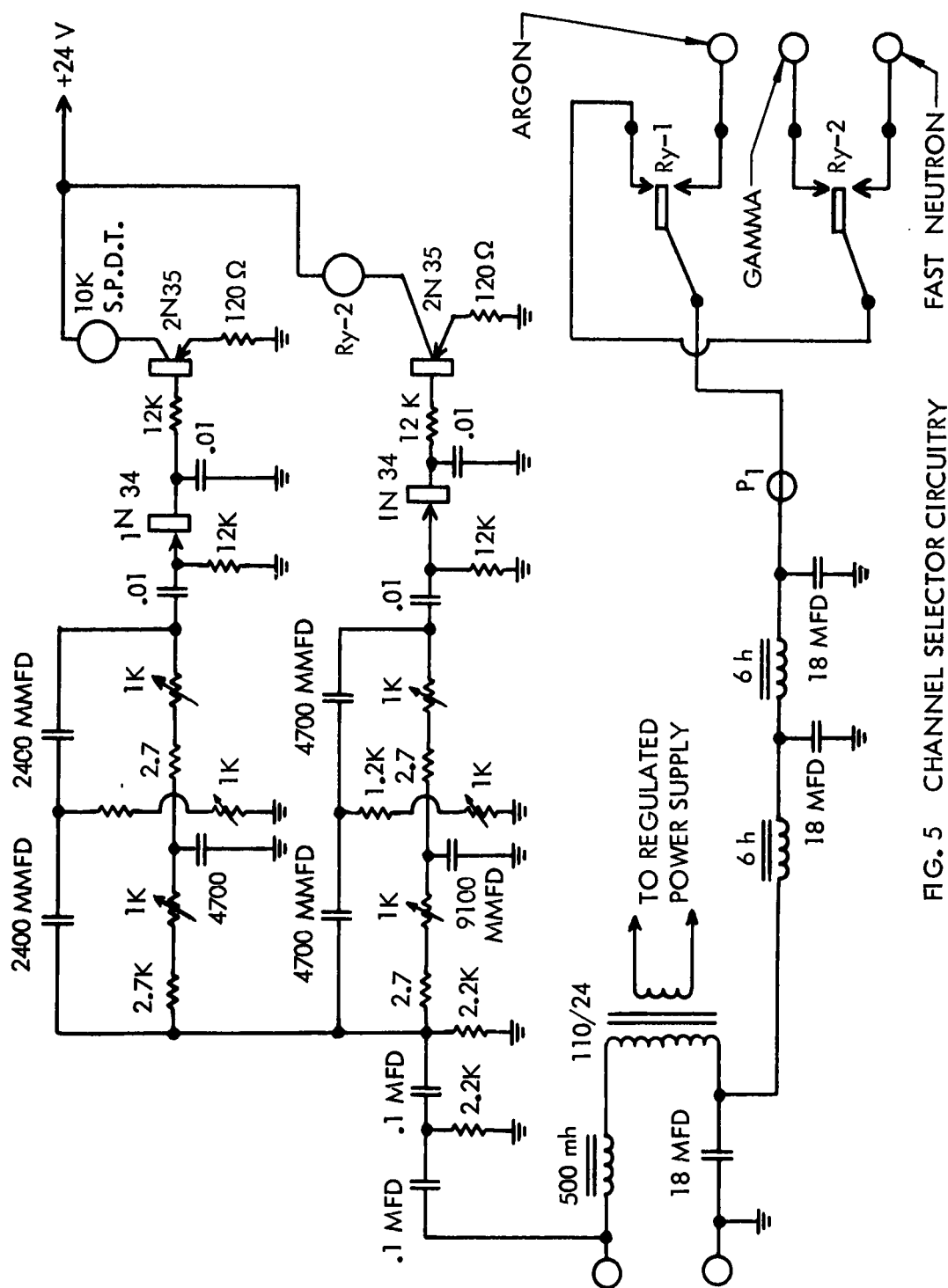


FIG. 4 RECORDER CONTROL PANEL



AREA MONITORING FOR RADIOACTIVITY

by

Roy Shipp

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The radiation effects reactor is designed to provide sufficient gamma and neutron leakage flux to irradiate large aircraft subsystems and components. This design requirement necessitates operation of the reactor above ground with a minimum amount of shielding, with the result that above-tolerance fluxes extend beyond the bounds of the reactor building. The predicted gamma flux, for instance, is one roentgen per hour at the 3600-foot radius exclusion fence.

To ensure the safety of people in the general vicinity of the site, monitor stations are situated around the exclusion fence to provide readings for use in controlling radiological hazards. Figure 1 is an area map showing the layout of the area monitoring system. Also, additional stations are located at the peripheral fence to measure radiological hazards at the site boundary. The radiological factors monitored at the station include concentration of argon-41, as well as intensities of fast neutron and gamma radiation. Data from the remote monitor stations are transmitted to central recording and alarm stations. Data primarily concerned with hazards associated with the operation of the Radiation Effects Laboratory and the Critical Experiment Reactor are transmitted to the Nuclear Support Laboratory.

The development of the area monitoring system was a result of the coordinated effort of three groups. Coverage in this paper, however, is restricted to the development of detectors and certain aspects of the argon-41 problem peculiar to this facility.

Design considerations for the entire system were remote operation, reliability, no protection from weather, and minimum power utilization.

GAMMA DETECTION

A 6-decade logarithmic rate meter circuit* was developed in the Nuclear Instrumentation Laboratory to cover the range requirements of the gamma monitoring system of background to 10 roentgens per hour. The detector, illustrated in Figure 2, is a 5-gallon ionization chamber, operated at atmospheric pressure. A series of concentric electrodes is used inside the chamber to reduce the value of the saturation voltage for a 10-roentgen per hour operation.

FAST NEUTRON DETECTION

For fast neutron detection a recoil proton type proportional chamber is used. The chamber is in reality a cluster of nine 4-compartment units, as shown in Figure 3. The horizontal disks are constructed of lucite, and the vertical dividers are constructed of polyethylene. Nine half-mill center wires are employed in the arrangement. The central stack of four units is brought out to one connector, and the eight stacks in the outer ring are connected to a second connector, with the result that all of the units are connected for low-intensity measurements and only the inner chamber is required for high-intensity measurements. The filling gas is argon diluted with 10% methane, a compound which permits operation at a lower potential than ethylene or methane alone would permit.

ARGON DETECTION

Argon-41, which is produced by neutron activation of naturally occurring argon-40, is a beta-gamma emitter with beta energy of 1.18 mev and gamma energy 1.37 mev. Half-life of argon-41 is 110 minutes. Argon-41 is an external rather than an inhalation or ingestion hazard. The maximum permissible concentration is based upon tolerance beta and gamma radiation received by immersion in a semi-infinite sea of air containing argon-41. The maximum permissible concentration of argon-41 is 5×10^{-7} microcuries per milliliter, or approximately one disintegration per minute per milliliter. Under normal meteorological conditions, the argon-41 produced in the air around the Radiation Effects Reactor will be disbursed before tolerance concentration accumulates. Under adverse meteorological conditions, however, it may be possible for above-tolerance concentration of argon-41 to accumulate and drift over the boundaries of the site, thereby creating a radiation hazard to the populace in the vicinity of the site. Monitors at the 3600-foot exclusion fence provide a warning to reduce the possibility of such an occurrence. Detection of argon at the exclusion fence constitutes a rather severe problem. The gamma radiation background resulting from normal operation of the reactor is about one roentgen per hour, and the increase in radiation level due to tolerance concentration of argon-41 is about 5 milliroentgens per hour. This increase in argon concentration would be lost in the normal statistical variations of the one roentgen per hour field if total gamma radiation intensity were used as a means of determining argon concentration. The use of a single channel scintillation spectrometer set to count gamma rays of the characteristic energy of argon-41 constitutes an effective method of measuring the gaseous activity. The effectiveness of this system is shown in Figure 4. The detector is placed in the open,

-2-

*"Logarithmic Circuits for Radiation Dosimetry," L. A. Turner

so as to be immersed in a sea of argon contaminated air. Such a system cannot be used at the 3600-foot fence, however, because of the overloading of the spectrometer by the high-intensity radiation from the reactor. A possible modification in this arrangement would involve placing the scintillation detector at the center of a 50-gallon drum, buried to provide shielding from the reactor radiation. This scheme was discarded, however, because of the complexity of the spectrometer circuitry involved and the associated lack of reliability.

The solution finally adopted involved the use of a buried proportional counter, as shown in Figure 5. The detector consists of a 7-liter central sample, well separated from the annular proportional counter by a large, thin beta window, an arrangement which provides a high geometry for counting beta radiation from argon-41 within the chamber. To remove particulate contamination, air is drawn through an absolute filter and allowed to flow through the central well of the chamber. This arrangement does not differentiate argon-41 from the other radioactive gases; but since argon-41 is the major radioactive component to be expected from activation of the constituent gases of the air, the arrangement is effective.

As mentioned earlier, argon-41 presents a problem only under adverse meteorological conditions. Therefore, the reactor will not be operated at full power when weather station data indicate the imminence of such a condition.

Because of the lack of experimental data relating to the reactor and the site, the personnel making calculations pertaining to the meteorological conditions warranting shutdown of the reactor compensate their calculations by selecting conditions having the large margin of safety. As a result, the present weather permissive conditions may impose excessive restrictions on the reactor operations. Experiments are now being conducted to secure argon-41 data under adverse weather conditions from samples collected near the reactor during preliminary low-power operations. The evacuated sample bottles are opened to the air by remotely operated valves. Then after the reactor is shut down, the sample bottles are rushed to the laboratory for argon-41 counting. These data will be used for recalculation of the argon-41 problem.

REF SYSTEM MONITORS

- — γ n_f A⁴¹
- — γ A⁴¹
- △ — γ
- ▲ — A⁴¹

NSL SYSTEM MONITORS

- — γ n_f A⁴¹
- — γ n_f
- + — γ

- || — LOCKED, LIMITED ACCESS
- .|| — GUARD CONTROLLED

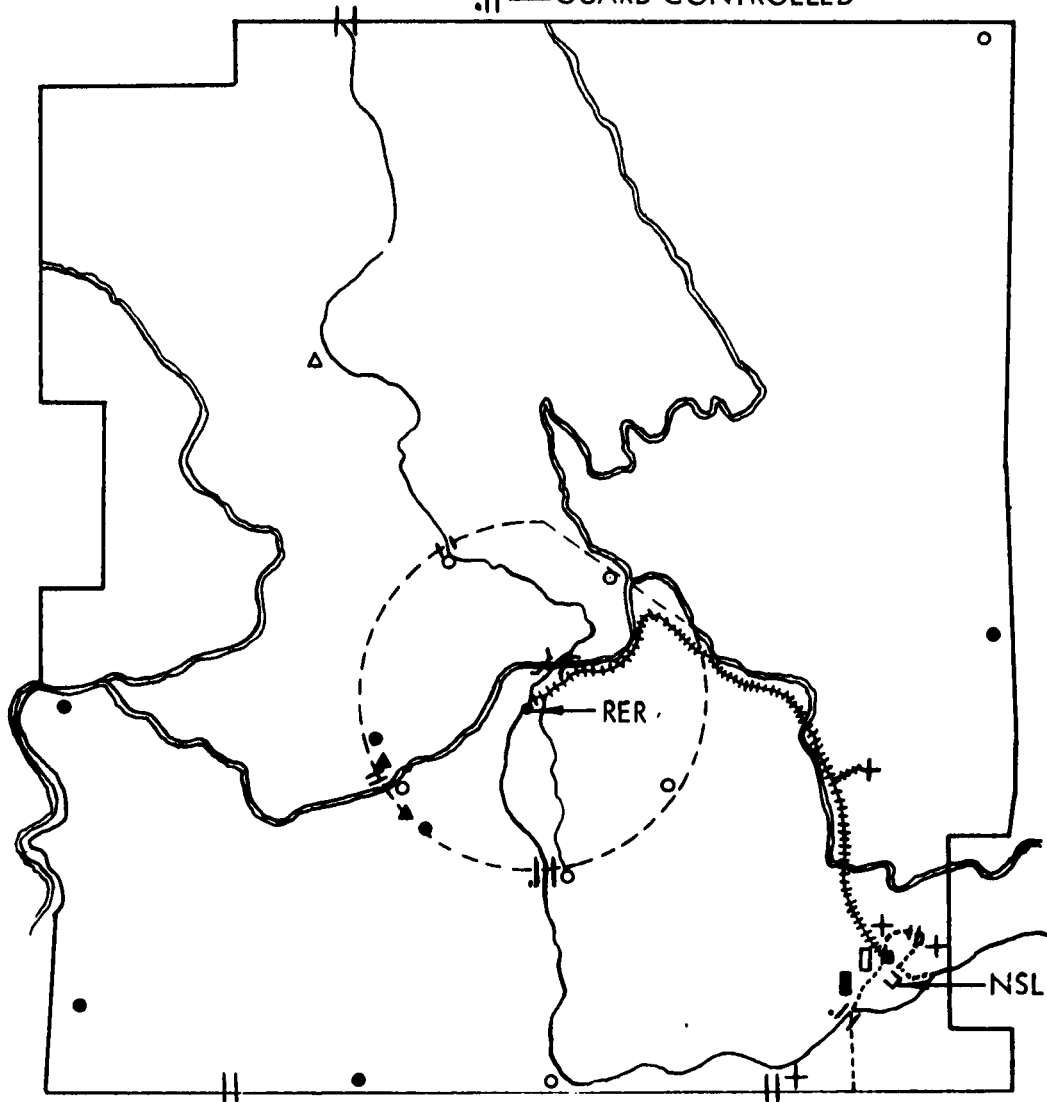
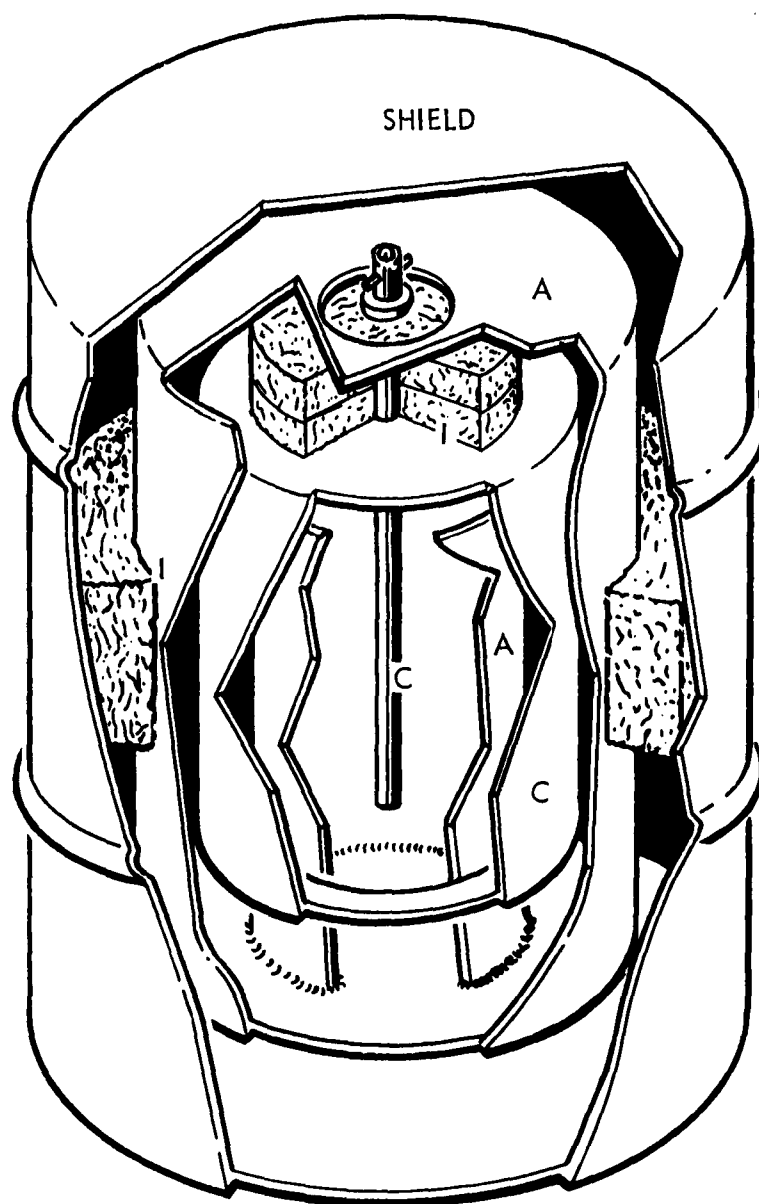


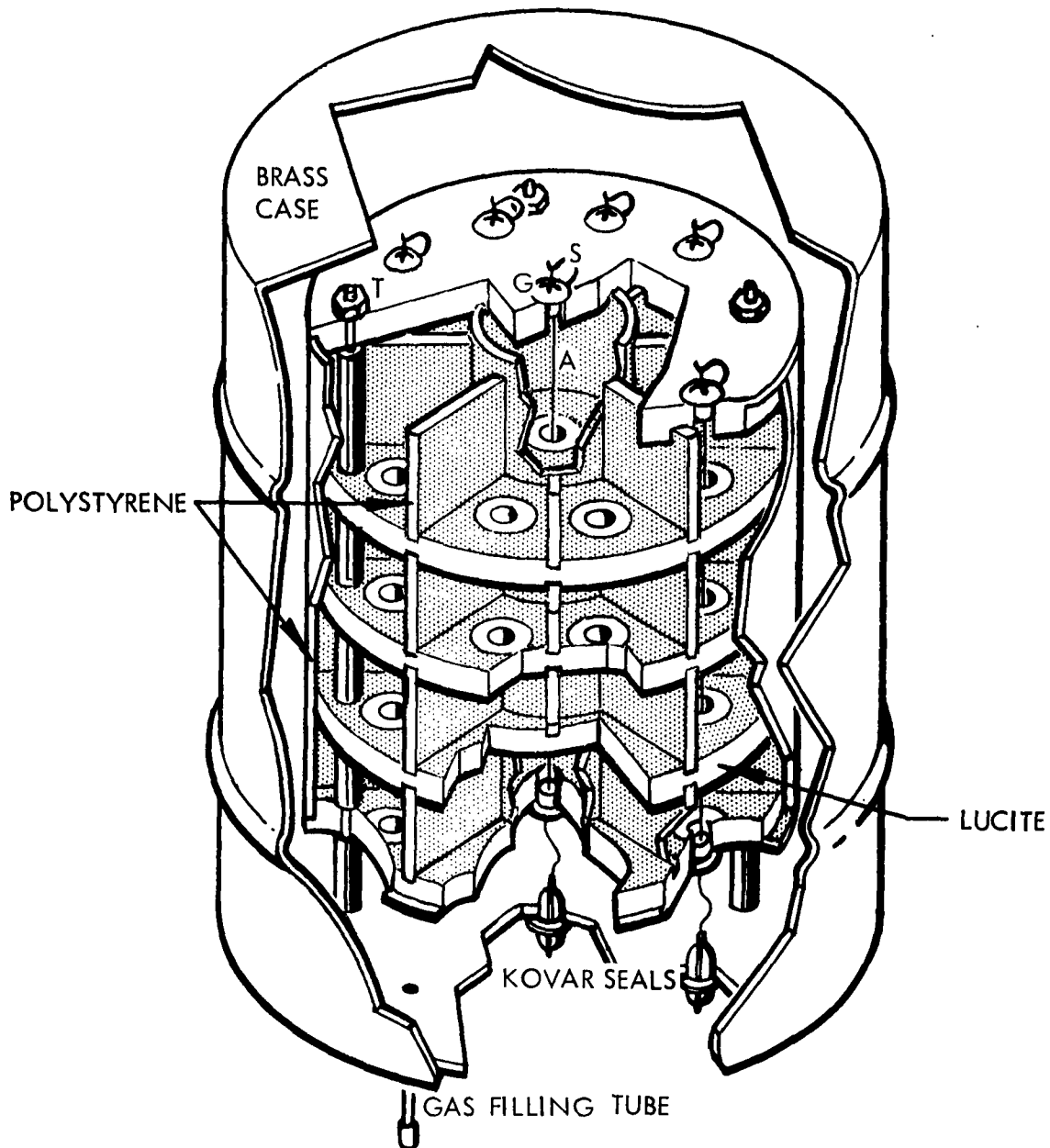
FIG. 1 AREA MONITORING STATIONS



A = ANODE
C = COLLECTOR ELECTRODE
I = INSULATOR

FIG. 2 Y IONIZATION CHAMBER

AQUADAG SURFACE



- A = 0.0005" TUNGSTEN CENTER WIRES
- S = TENSION SPRING FOR CENTER WIRE
- T = THROUGH BOLT
- G = GUIDE FOR CENTER WIRE

FIG. 3 FAST NEUTRON DETECTOR

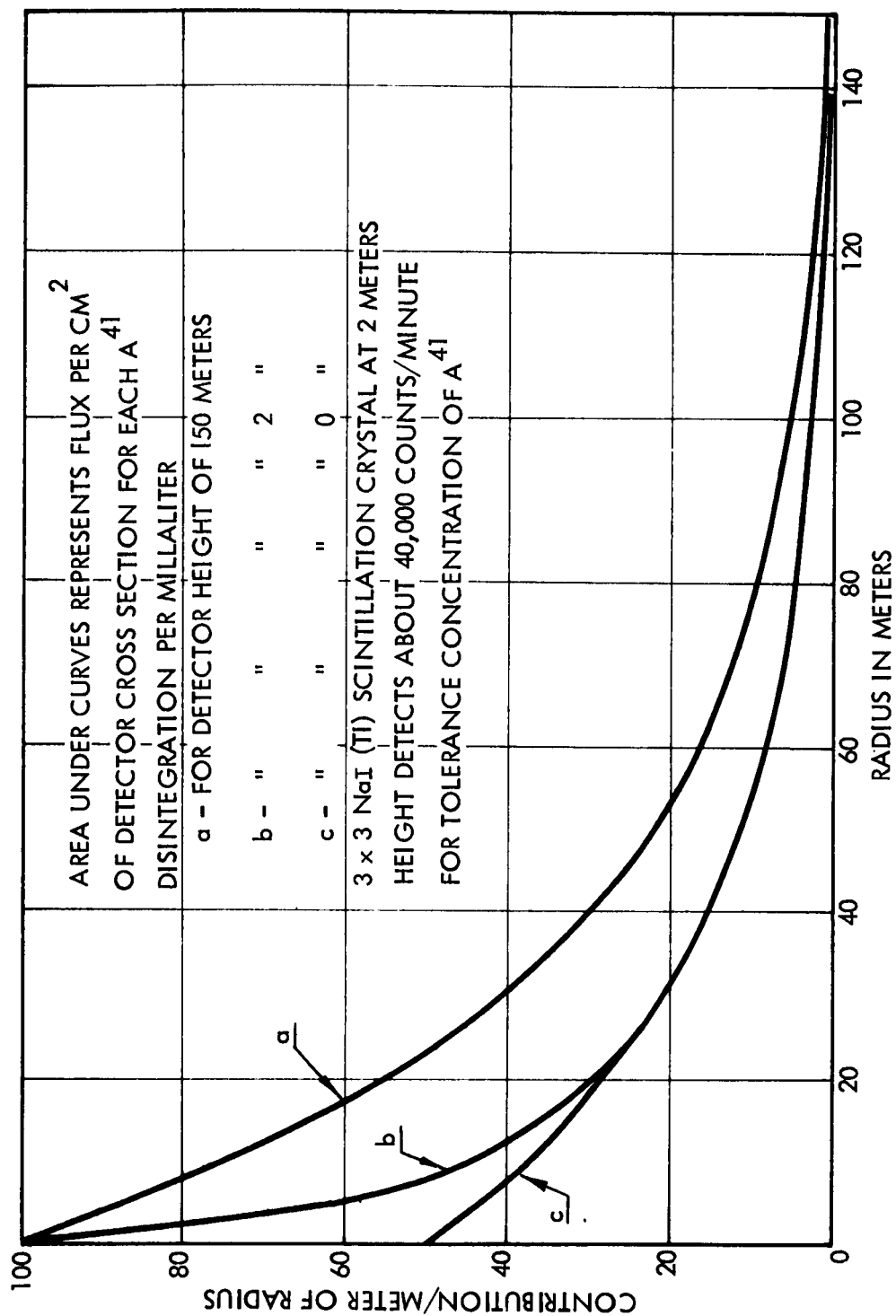


FIG. 4 GAMMA RADIATION FROM IMMERSION IN SEMI-INFINITE SEA OF ARGON-41

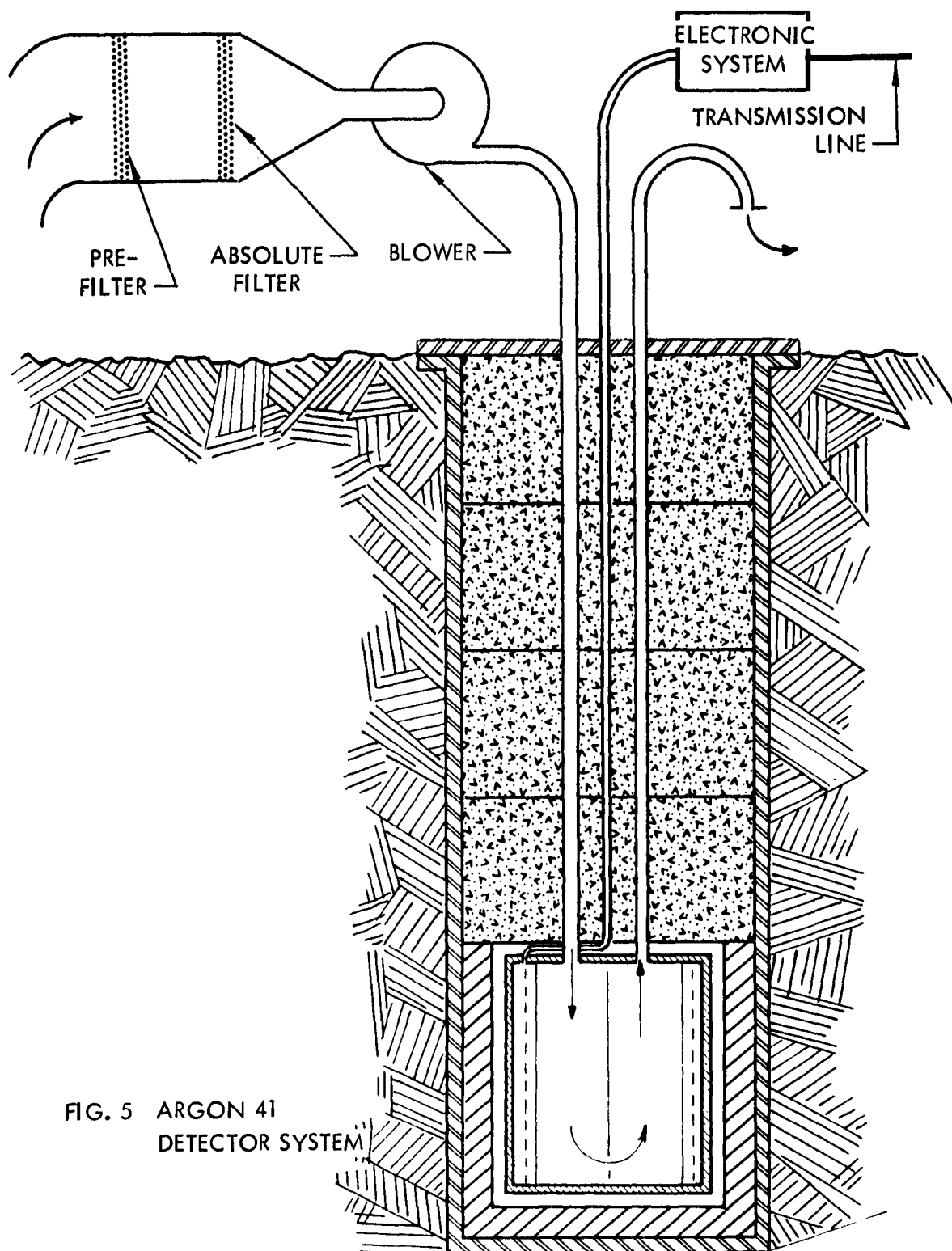


FIG. 5 ARGON 41
DETECTOR SYSTEM

LOGARITHMIC CIRCUITS FOR RADIATION DOSIMETRY

by

L. A. Turner

Lockheed Nuclear Products
Lockheed Aircraft Corporation
Georgia Division, Marietta, Georgia

At Air Force Plant No. 67 simple circuits are available for the logarithmic measurement of a-c and d-c voltages and frequencies or pulse repetition rates. Ranges of over 100 db are easily obtained.

In flux mapping at Air Force Plant No. 67 and in subsequent subsystems testing, the use of logarithmic circuits may be desirable as a general convenience or to avoid the necessity for range switching, which, in the case of a wide-range inaccessible detector, might not be practical. For most applications 100 db of range is sufficient, and this can be easily obtained if accuracies of ± 1 db are acceptable. Greater range and better accuracy can be obtained with more complex circuitry and with a careful choice of components.

Alternating current and a-c voltage can be measured by taking advantage of the overload characteristics of several cascaded amplifier stages. Applications would include such devices as logarithmic, rather than linear, amplifiers in pulse circuits where the extreme range would prevent overload or facilitate more rapid pulse height analysis.

Direct current and d-c voltage are most easily measured by making use of the logarithmic characteristic of a high-vacuum thermionic diode when operated under conditions in which it has a negative static resistance. A logarithmic micro-microammeter, when used with an ionization chamber, allows operation over a very wide range of field intensity without circuit adjustments. Applications include scanning devices and setups in which the approximate radiation fields are not known. A circuit of this type is used with a large ionization chamber to measure gamma dose in the remote area monitor systems.

Frequency can be measured logarithmically by adding the outputs of a number of circuits of different time constants or by the use of a modified univibrator with a logarithmic output. Logarithmic frequency or count rate meters can be used with

any type of radiation detector that gives a pulse output, such as FND and GM tubes, and allow scanning over a wide range of field intensity without the necessity of range switching. A logarithmic count rate meter is used with proportional counters for fast neutron and argon-41 measurements in the remote area monitor.

A-C VOLTAGE AND CURRENT

By adding the outputs of several cascaded amplifier stages, a-c voltage may be converted to its logarithm. All stages must have the same gain and overload characteristics for this description to apply.

In Figure 1 is shown the block diagram of a unit having a range of 100 db with ± 1 db linearity. This unit contains eight stages of 14 db gain each. Since the outputs of the odd and even stages are 180° out of phase, it is necessary that they be added separately and then combined after the output of the odd stages is inverted in phase. It is necessary that the amplifiers have very low phase shift over the operating frequency range in order that the stages add properly and that the output be in phase with the input signal. Figure 2 shows how the outputs combine to produce a log curve.

The range of this unit was 10 microvolts to 1 volt. Its range could easily be extended to high voltages by the addition of more stages of proper design to handle high signal levels, but extension of the range to lower inputs would require extreme care because of noise and microphonics. Transistorized circuitry should make it practical to operate to about one microvolt input with a relatively narrow bandwidth.

Alternating current can be measured as the voltage drops across a resistance in series with the circuit.

D-C VOLTAGE AND CURRENT

In a thermionic, high-vacuum diode, electrons are emitted from the cathode with sufficient energy to reach the anode, even though it may be at a slightly negative potential relative to the cathode. The number of electrons reaching the anode decreases logarithmically as the anode is made more negative. The current normally will become negligible in the range of -1 to -10 volts.

If this diode is used as a series resistance in a low-current circuit and an electrometer is used to measure its voltage drop, a logarithmic current circuit will be obtained. A lower limit on the current that can be measured will be set up by such factors as insulation resistance, ion currents, and photoelectric emission. An upper limit will be set by the current at which the static resistance of the diode becomes positive.

Since the voltage across the diode may be excessive, it is usually desirable to incorporate it in a feedback loop so that the effective input impedance will be

approximately zero ohms. A block diagram of such a system is shown in Figure 3.

D-C voltage may be measured as the current through a fixed resistance.

FREQUENCY

Frequency can be measured in several ways. In one, "flip-flop" drives a number of rectifiers within inputs limited by series capacitors and outputs limited by series resistances. Usually the logarithmic response is obtained by using a different value of series capacitance for each rectifier. Approximately one rectifier per decade is required for most applications. This system, when used for pulse counting, will show a short constant dead time at all count rates.

A schematic diagram of one rectifier and a block diagram of the system are shown in Figure 4, and Figure 5 shows how the outputs add to produce a logarithmic response.



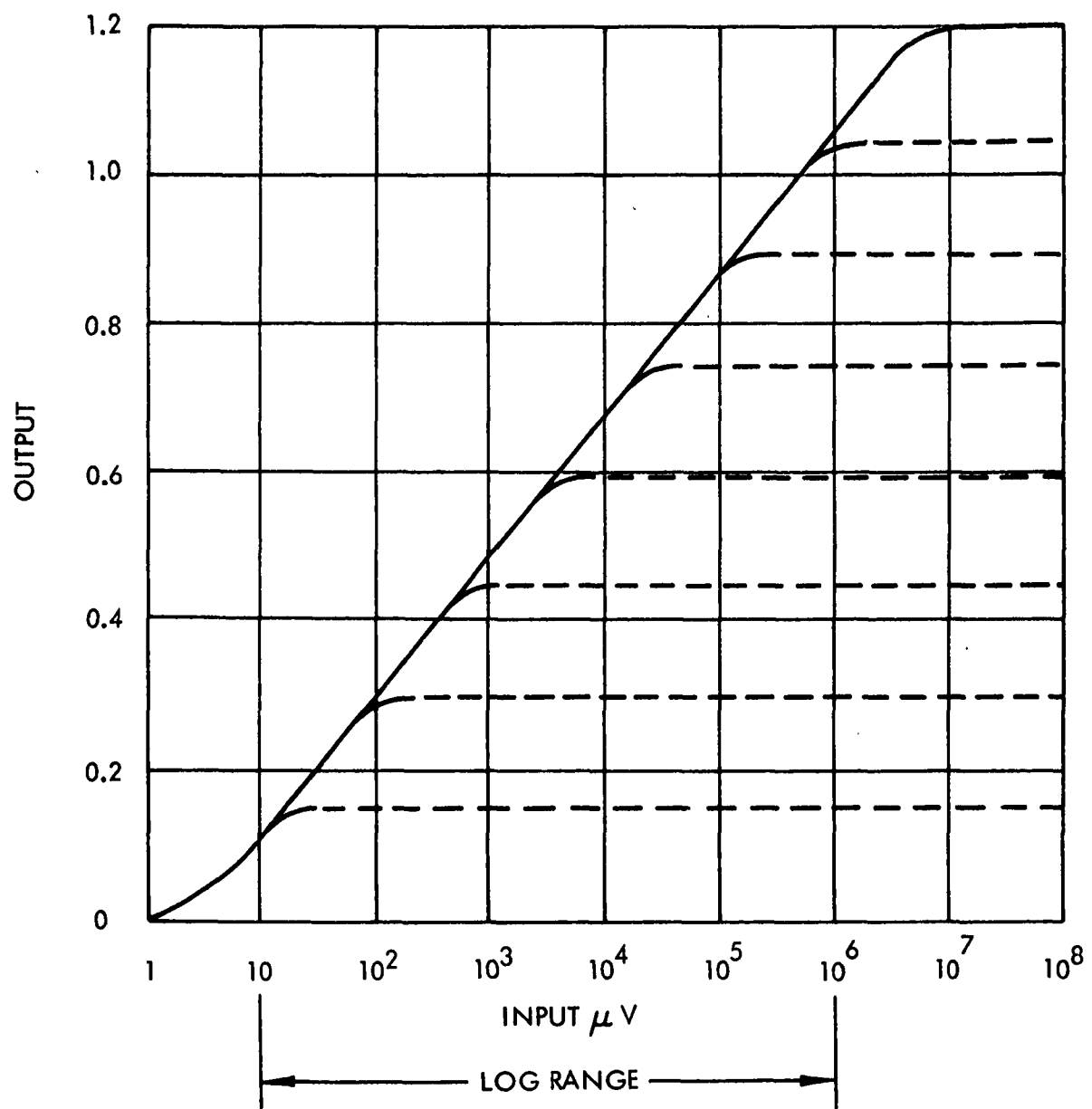


FIG. 2 LOG RESPONSE FROM VOLTAGE ADDITION

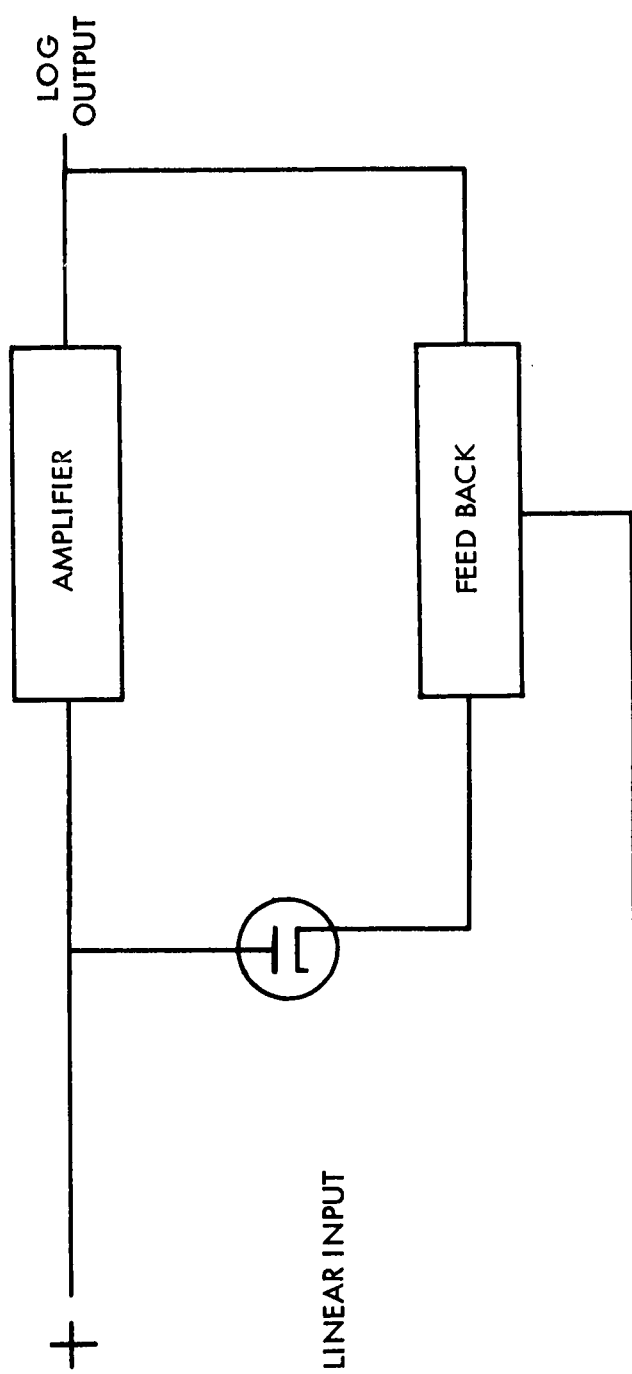


FIG. 3 DC LINEAR TO LOG CONVERTER

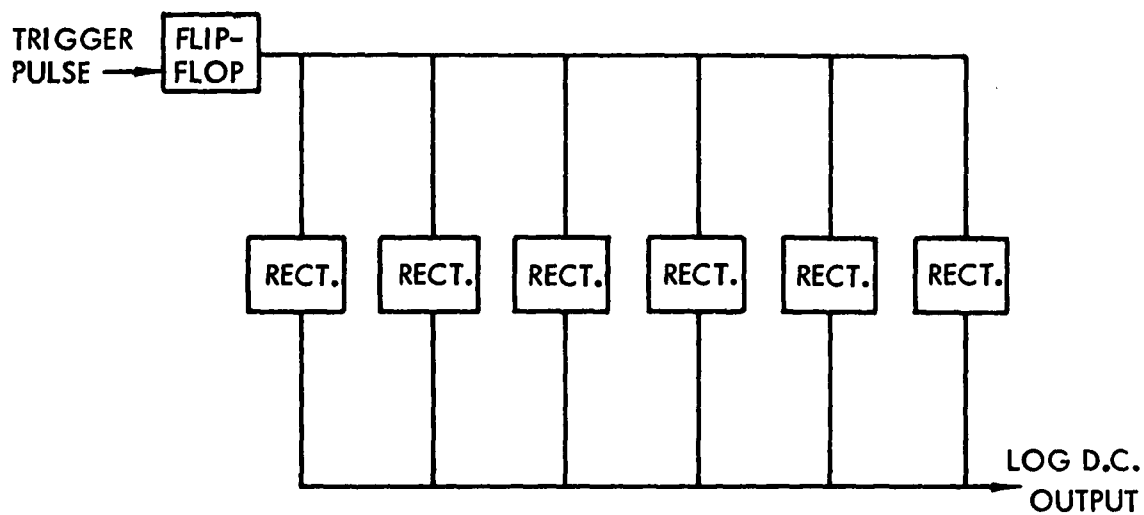
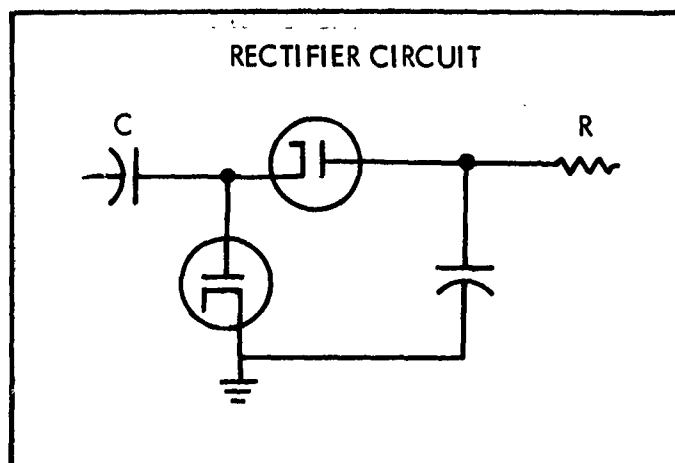
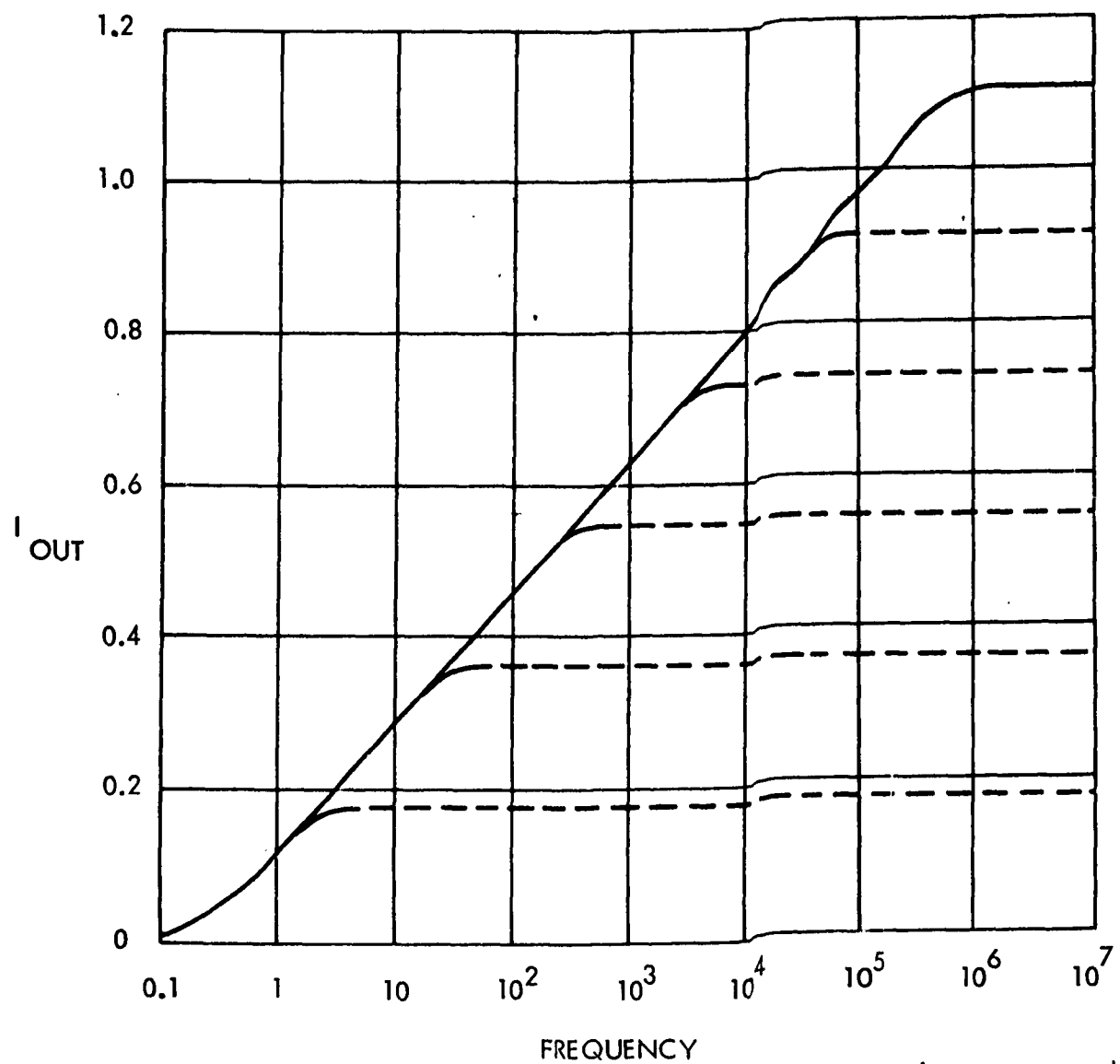


FIG. 4 LOG COUNT-RATE METER



$$I_{OUT} \approx \frac{I}{\sqrt{1 + \left(\frac{1}{F}\right)^2}} + \frac{I}{\sqrt{1 + \left(\frac{10}{F}\right)^2}} + \frac{I}{\sqrt{1 + \left(\frac{10^2}{F}\right)^2}} + \frac{I}{\sqrt{1 + \left(\frac{10^3}{F}\right)^2}} + \frac{I}{\sqrt{1 + \left(\frac{10^4}{F}\right)^2}} + \frac{I}{\sqrt{1 + \left(\frac{10^5}{F}\right)^2}}$$

FIG. 5 LOG FREQUENCY RESPONSE FROM CURRENT ADDITION