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SYMPOSIUM ON HIGH FLUX MATERIALS TESTING REACTORS

Held in Brussels, Belgium,
September 21-26, 1959

UNITED STATES
ATOMIC ENERGY COMMISSION
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REACTORS—GENERAL

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CENTRE D'ETUDE DE L'ENERGIE NUCLEAIRE

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UNITED STATES
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CONTENTS

THE AEC TEST REACTOR PROGRAM

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ABSTRACT

The need for extensive materials and components testing as a part of the AEC reactor development program was recognized in the early days of the organization of the program. The first group of nuclear reactors proposed by the AEC in 1948 included the Materials Testing Reactor, known widely as the MTR, a high flux reactor intended to provide these irradiation services and to supplement the research reactors then in use at or planned for the AEC National Laboratories.

The small amount of experience available at that time in test reactor operation, the unknown hazards associated with the experiments and tests contemplated and the desire of the AEC to minimize property damage and avoid injury to surrounding populations in the case of an accident led to the location of the test reactor at an isolated site in Idaho. Safety was further emphasized by the procedures the operator was to follow in the hazards evaluation of the experiments and the manner of operation of the test reactor.

The AEC planned to use the MTR to perform irradiation services for contractors engaged in work for a number of its Divisions such as Biology and Medicine, Military Applications, Production, Reactor Development, and Research, as well as for other Government agencies. A policy board consisting of representatives from these Divisions was established in Headquarters to review and approve each experiment for conformance with the overall program objectives and to assign priorities. Approved experiments are then transmitted to the AEC Operations Office in Idaho and to the MTR operator where problems of scheduling, feasibility and safety are considered prior to the initiation of the irradiation.

Since 1952 when test reactor irradiations began, the number of experiments irradiated annually has increased, the experiments have become more complicated as experimenters learned how to perform more meaningful experiments and the need for improved and more accurate data has continued to increase.

Screening experiments involving small specimens of fuel or reactor materials loaded in simple capsules have developed into complicated and instrumented capsules. Engineering type tests demand larger volumes of high flux space and extensive loop systems in which a coolant is circu-

lated in a loop passing through the reactor core. Extensive auxiliary apparatus such as pumps, heat exchangers, control systems are required outside the reactor. The Engineering Test Reactor was constructed to provide these additional features and for the growing testing load.

To further supplement its requirements for additional test reactor space, the AEC uses space in the Canadian NRX and NRU reactors, purchases space from privately-owned reactors such as Battelle, General Electric Testing Reactor, and the Westinghouse Testing Reactor.

The presently available test reactors provide space for testing small specimens, prototype fuel elements, coolant compatibility studies and the radiation effects on structural materials. There is a need for reactors in which to test full size fuel elements in quantities large enough to provide a statistical evaluation of their performance. For such tests the AEC is planning or constructing reactors such as the Plutonium Recycle Test Reactor, the Heavy Water Components Test Reactor, and the Experimental Gas-Cooled Test Reactor.

Future requirements in the test reactor field indicate the need for further increases in neutron fluxes, more uniform neutron fluxes for the duration of the irradiation, reduced neutron flux gradients in the experimental space, better instrumentation for neutron flux determinations, and for temperature measurement and control.

* * * *

The need for extensive testing of fuels, materials and components for use in constructing and operating nuclear reactors in the presence of nuclear radiations was recognized early in the organization of the United States atomic energy program. Professor Eugene P. Wigner recognized in the latter part of 1942 that energetic neutrons and fission fragments, produced in the fission process, could displace atoms from their equilibrium positions and that the solids whose atoms were displaced could have their physical and mechanical properties affected.

Theoretical and experimental studies were begun and with them discussions were held and conceptual designs undertaken for test and research reactors which would be suitable sources of these energetic radiations in which to carry out such studies and tests. Such reactors must produce fast neutron fluxes of magnitudes at least comparable to those encountered in nuclear power reactors in order to test radiation damage effects in reactor structural materials, as well as in reactor fuels. They must also produce copious quantities of neutrons of appropriate lower energies to cause fission in the test reactor fuel and in fueled experiments.

During the period 1944 to 1948 there developed out of the discussions and research activities of the Manhattan District and its successor, the Atomic Energy Commission, additional uses for research and test reactors such as the production of radioactive materials to be used in tracer studies, the preparation of intense radioactive sources, studies of chemical radiation effects, and studies to determine breeding and conversion ratios of fertile materials.

These discussions were participated in by physicists and chemists, as well as engineers interested in the construction of future nuclear reactors for power sources. In October of 1948, a group of key reactor engineers and scientists met at the newly formed Argonne National Laboratory to review requirements and set objectives for the Commission's reactor program then being formulated. A major objective agreed upon was the need of a high neutron flux reactor capable of producing high thermal fluxes for isotope production, research interests and fission fuel studies and also fast neutron fluxes for radiation damage studies of materials. A conceptual design of a core for such a high flux reactor proposed by Professor Eugene P. Wigner and his associates in 1945, was selected and named the Materials Testing Reactor. This basically was a heterogeneous reactor, light water moderated and cooled, with core design optimized for maximum fast and thermal fluxes. It was provided with extensive experimental ports and facilities such as neutron beam holes, thermal column, shielding study facility.

While the major objective of the testing program then being formulated was materials testing for the newly formed nuclear reactor program, the MTR was called upon to serve numerous other AEC programs. The Commission's first high flux reactor was within sight, and all who had interests in high flux irradiations wanted a piece of the reactor. For example, Research programs desired high intensity neutron beams, the highest neutron flux positions for heavy isotope production; Biology & Medicine wanted the highest flux positions for high specific activity Co^{60} teletherapy sources; additional urgently desired requirements arose less frequently in other AEC programs and in other government agencies. The attempt to provide space and experimental ports to satisfy these many customers and the total lack of experience as to what constituted the best suited facilities for reactor materials testing lead to the incorporation into the MTR design of some experimental facilities which found little or no use in subsequent years. I shall refer to them later.

Ten years ago test and research reactor operating experience was nonexistent. Problems of reactor control existed, the effects of sudden additions of reactivity as might happen in an experiment failure were unknown, and the types of experiments and tests which reactor scientists and engineers would want to perform were only vaguely defined and the hazards involved were difficult to evaluate, hence the geographical location of the MTR and the operating policy and procedures to be applied received extensive study and attention by the Commission. The AEC was determined to avoid accidents or malfunctions if at all possible, and should an accident occur, to minimize property damage and avoid injury to surrounding populations. The emphasis on safe operation lead to the decision to locate the MTR at a site remote from populated areas and one whose potential use for other purposes was small. This resulted in locating the MTR at the National Reactor Test Station in Idaho.

Safety played a prominent role in determining the operating procedures, hazards evaluations or revelations, and safeguard measures which the operator established during the startup period and continues to follow. These procedures are discussed in considerable detail in other papers presented at this meeting.

The operating experience with the MTR and the safe and successful performance of a wide variety of experiments and tests therein during the past seven years, the successful operating policies and procedures developed by the operator in collaboration with the AEC, the reactor safety studies sponsored by the AEC, coupled with the experiences obtained by the National Laboratories in the operation of their research reactors, relieved to a considerable extent the earlier concerns which prompted the selection of a remote site for the test reactor. Consequently, large test reactors are now being located in less remote sites, in some cases near densely populated areas. Where the experiment creates risks or hazards which are not considered acceptable in a populated area, the Commission will insist that they be performed at the remote site.

The Division of Reactor Development was assigned the responsibility for constructing and operating the MTR as a service facility to all AEC Divisions. The Director, Division of Reactor Development is responsible for broad irradiations services planning and for determining policies governing the use of nuclear reactors to provide these irradiation services including approval of irradiations and establishment of priorities. A Test Reactor Irradiation Services Board was established to advise and assist the Director in the planning and administration of the irradiation services. This Board consists of a Chairman and a Secretary from the Division of Reactor Development, a representative from each of the Divisions of Research, Production, Biology and Medicine, Military Applications, and Licensing and Regulations. The Board members were selected from the various Divisions whose contractors are the sources of the requests for irradiations.

Requests for irradiation services are submitted on a standard form to the Secretary of the Irradiation Services Board. Each request is reviewed by the Board member of the Division whose contractor made the request for conformance with the Division's program objectives, priority evaluation and determination of budget activity against which irradiation charges are to be made.

Conflicts arising out of situations where two or more contractors desire the same space at the same time are considered by the Chairman, Secretary, and the Board members concerned. This group explores the factors involved, attempts to find other suitable space or solutions, or schedule changes which will allow the requestors to meet their desires. If the Board cannot arrive at an acceptable solution it can make a recommendation to the Director, Division of Reactor Development, who can render a decision which is binding if only Reactor Development contractors are involved. If other Divisions are involved and the Reactor Division Director's decision is questioned, the case may be presented to the General Manager of the Atomic Energy Commission for a final decision.

In these considerations by the Board use is made of the AEC's Operations Office in Idaho, which has the responsibility for detailed planning and scheduling of irradiation services in accordance with instructions from the Director, Division of Reactor Development, and the operating staffs of the test reactors. The operating staffs provide the technical information and evaluations required by the Board. The Board is not limited to these sources of technical information; they may call upon independent technical sources and upon consultants where considered necessary.

Board actions on irradiation requests are given final Headquarters approval or disapproval by the Director, Division of Reactor Development, and the requestor is notified of the action. Approved requests for irradiations are transmitted to the Idaho Operations Office for scheduling in an appropriate test reactor. The test reactor operator then determines the feasibility of performance of the requested irradiations, makes the hazards evaluations, sees to it that the experiment conforms with established safety and engineering standards, and finally schedules the irradiation. The satisfactory completion of these items by the operator is signified by his Project Manager's approval of the irradiation request. Final AEC approval is given by the Idaho Operations Office.

This operating procedure, which applies to the MTR, ETR, and other test reactors, assures on-site surveillance of the test reactor operations and of the tests and experiments by the AEC representatives, it removes from the operator and the on-site AEC representatives the pressures which individuals or contractors might bring for favored attention to their irradiations over those of others, it requires Headquarters approval for each irradiation but does not burden Headquarters with the details of administration and management at a distance, this approval keeps Headquarters in closer contact with their programs, it permits rapid changes in programs by being able to quickly review the effects of such changes on other programs and determine whether these changes are acceptable before the field and operator become extensively involved.

The MTR began operation in 1952 by irradiating specimens in sealed capsules. By 1954 several high pressure water loops, one single pass gas-cooled loop and a liquid metal loop were in operation. During these years reactor programs received some assistance in performing irradiations from the Hanford production reactors and research reactors at Argonne, Oak Ridge and Brookhaven. During this period it became evident that some of the experimental facilities provided in the MTR were not suitable for the tests envisioned for the reactor development programs. For example, the beam holes were not acceptable for fuel or structural materials tests since the neutron flux along the axis of the beam hole varied too rapidly. Steep neutron flux gradients produced non-uniform heat generation in the fuel specimens and unsatisfactory test results. The reactor was designed to accommodate many capsule experiments, but it was soon discovered that uninstrumented capsules are not satisfactory for some kinds of experiments and instrumented capsules could not be used, for there was no provision for taking instrument leads out of the reactor tank. These uninstrumented capsules left the experimenter totally uninformed as to the temperature of the specimen; upon its removal from the reactor he frequently found the specimen had never reached or had exceeded the temperature he desired for the experiment. As a result of these failures considerable effort has gone into instrumented capsule design with provisions for continuous temperature measurements and external temperature control of the specimen being irradiated. This defect in the MTR was later remedied by the addition of a spool piece to the top of the reactor tank, the spool piece being provided with numerous penetrations through which instrument leads from the inside of the tank could pass to the outside.

The need for additional test space developed and the construction of the Engineering Test Reactor was undertaken. The ETR design profited by the earlier irradiation experience; in particular, attention was directed to providing more loops and larger test volumes within them. The test reactor core was lengthened to three feet and changes were made in the control system to achieve more uniform neutron fluxes.

The skills of the experimenter are usually ahead of the tools available to him. He can detect weaknesses and suggest improvements more quickly than the improvement can be accomplished. This is especially true in the reactor development field. The experimenter wants to perform his tests and experiments under the most carefully controlled conditions so that he can reproduce results and detect the factors which influence them. The small specimens of fuel materials undergoing basic property or materials screening tests must have their temperature carefully controlled and the temperature history well recorded if the results of one test are to be compared with another. This establishes a requirement for constant neutron fluxes throughout the entire exposure period, a condition not easily attained.

Larger specimens used in prototype fuel element development are called upon to generate high heat fluxes throughout the length of the specimen. To avoid specimen distortion due to thermal stresses the experimenter desires a uniform neutron flux over its entire length.

All of these tests are performed in either instrumented capsules or in in-pile loops. In-pile loop irradiation tests come closest to achieving the desired irradiation testing conditions.

A loop can provide the desired coolant conditions of flow rate and heat removal. The test specimen temperature can be controlled by varying the coolant flow. Constant and reproducible experimental conditions can be produced, a condition which is extremely important, for the performance of fuel materials is often very temperature-sentistive, and instrumented, controlled in-pile loop tests are needed to determine the performance limits of such fuel materials.

Thus there is a continued request for loops in which to perform these tests, the number of which keeps growing as new reactor projects appear and as increasing effort is devoted to fuel materials studies aimed at cost reductions in the fuel cycles, and as the effort to determine the factors causing fuel element failures increases.

Another advantage of loops is that they permit carrying tests to destruction of the materials being irradiated. If corrosion is the major interest of the test then wide variations of coolant condition and material preparation can be explored under conditions closely identical to those existing in the reactor for which the materials are being tested. Corrosion products being confined to the loop system do not contaminate the test reactor. Radioactive materials and fission products are likewise contained, and may be cleaned up without disturbing the experiment.

As the length of the power reactor fuel cycles increases and higher burnups take place the test reactors will be called upon to perform tests and irradiations extending over periods of several years, thus decreasing the number of experiments which can be completed in a year by a single facility.

The number of loops which can be advantageously used in a test reactor, particularly in the test reactor core where the most desirable testing conditions exist, are limited (a) by the space surrounding the reactor which is available for auxiliary equipment such as pumps, heat exchangers , control assemblies, instrumentation and piping, which are required for the operation of each loop; (b) by the number of access ports in the top, bottom and sides of the reactor containment vessel; (c) by the conflict for space for loop piping, control rods and drives and other items requiring access in the core region; (d) by the problem of scheduling a large number of loop experiments in such a way that the down-time required for the installation of experiments and by experiment failures do not cause an excessive amount of lost operating time; (e) by the interaction of one experiment with another which can result in excessive neutron flux shadowing, or the failure to maintain desired experimental conditions.

The Commission has recognized the need for more loops in which to perform reactor tests and experiments and for more test reactors in which to place these loops. The Commission is purchasing space and renting loops in privately-owned test reactors to supplement AEC-owned test reactors in supplying these needs. This action is a part of the deliberate attempt on the part of the AEC to encourage the establishment of privately-owned facilities in which to perform work for atomic energy programs.

The present MTR loadings consist of 10-12 loops, and the ETR has 12 installations scheduled. It is not expected that all the ETR installed loops will be operated at one time; operating experience is required before the number can be determined and present installation schedules indicate about mid-1960 all the installations will be completed. Current estimates indicate the need for 21 additional loops, exclusive of those loops which require higher neutron fluxes than existing test reactors can provide.

The presently available AEC test reactors, together with assistance from privately-owned test reactors, should be able to supply the number of loops the various AEC projects estimate will be needed during the next three to five years with the following exceptions, namely, (a) they cannot supply the high neutron fluxes anticipated, (b) they cannot supply the constant, uniform neutron fluxes in the test regions the experimenters desire, (c) they do not offer the mechanical features forming a part of the loop system which permit rapid sample changing, sample removal without interfering with refueling operations, dry regions in which to make instrumented lead connections, and (d) space adequate in size and volume in which full size fuel elements may be tested, either singly or in large numbers for statistically significant performance data.

The AEC and privately-owned test reactors have upper limits to the neutron fluxes they can provide of about 5×10^{14} n/cm^2/sec thermal and 1×10^{15} n/cm^2/sec epithermal and fast. AEC power reactors now in operation have maximum thermal fluxes of 2.5×10^{14} n/cm^2/sec and 5.3×10^{14} n/cm^2/sec epithermal and fast. The AEC National Laboratories are designing and developing high flux research and isotope producing reactors to operate in the neutron flux range of $3\text{-}5 \times 10^{15}$ n/cm^2/sec.

Higher operating neutron fluxes are expected in power reactors as the fuel element development and technology improves, reactor core design becomes more exact, core flux flattening techniques are proven, hot spots in fuel elements are reduced and heat transfer rates increase. Likewise, these developments are applicable to research reactors and can aid in achieving higher flux research reactors to serve as tools for the research scientist in his studies.

The trend towards higher neutron fluxes by both power and research reactors requires test reactors capable of producing higher flux test space in which to perform engineering tests. The AEC is at the present time preparing conceptual designs for an advanced test reactor capable of producing thermal neutron fluxes of at least 2×10^{15} n/cm^2/sec and epithermal and fast fluxes of 2×10^{15} n/cm^2/sec in the test loops forming a part of this reactor. This Advanced Test Reactor will also include features which will provide the experimenter with a neutron environment of greater constancy in both time and space than in existing test reactors. The design objectives of this advanced test reactor include such goals as (a) unperturbed neutron flux variations at any point in the test reactor core not to exceed $\pm$ 10% with time throughout an operating cycle, with variations over a number of successive operating cycles not to exceed $\pm$ 20%, (b) flux gradients along the length of a test specimen not to exceed $\pm$ 20% in 2 feet, (c) loops designed integrally with the test reactor and built in at the time the test reactor is constructed, (d) improved mechanical features which will allow easy and rapid removal and reinsertion of test specimens in the loop, (e) improved neutron flux monitoring facilities so that flux values and any changes therefrom may be obtained at any desired time throughout the test reactor operating cycle.

Conceptual designs of such an Advanced Test Reactor will be discussed in another paper at this meeting.

The functions of the test reactors as developed over the years are essentially the creation of a neutron environment in which to irradiate reactor fuels and materials for basic fuel and materials studies, for fuel and materials screenings, for prototype fuel element development, and determinations of corrosion, dimensional changes, diffusion of fission fragments, effects of long exposures, burnup rates, high burnups, for proof tests of prototype fuel elements and for studies of various control materials.

These test reactors are, however, not suitable for testing full size fuel elements. Except for low temperature irradiations of fuel elements as performed in the MTR, for example, where considerable experience in the use of the Al-clad highly enriched U-Al alloy fuel is obtained and lesser experience with other materials such as a 20% enriched U-Al alloy fuel and Pu-Al alloy fuel, no test reactors are specifically used for full size fuel element tests.

Test reactors are currently in the planning stages or under construction in which to prove out power producing reactor systems such as the Heavy Water Components Test Reactor at Savannah River, the Plutonium Recycle Test Reactor at Hanford, and the Experimental Gas-Cooled Test Reactor at Oak Ridge. These reactors will be provided with isolated test channels in which to perform tests on full size individual fuel elements. Beyond this, tests of full size fuel elements are performed in the core of the reactor for which they are designed.

I hope that this review of the experiences with our test reactor program, mistakes as well as fruitful approaches, and our plans for the future in this area will be of some assistance to all of you who are engaged in or just entering the field of atomic power production, and in doing so, plan to include some form of testing of reactor fuels, materials and components.

USES OF HIGH FLUX REACTORS

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ABSTRACT

Advances in reactor technology and in basic research have revealed serious deficiencies in our knowledge of the interaction of neutrons with matter. In this paper, we discuss the use of high flux reactors to remove a number of these deficiencies.

In the field of chemistry the most obvious applications of high flux reactors are those which involve multiple order reactions such as the production of transuranic isotopes. Other uses include: study of thermal fission cross-section of isotopes with low cross-sections such as Np^{237}; measurements of ν for isotopes of $Cm^{245,247}$ and the Cf isotopes; radiation chemistry studies to determine the non-linear behavior with integrated flux of energy storage and the dependence of energy storage effects on the rate of radiation.

While chemistry experiments usually involve irradiations of samples, physics experiments are typified by beam hole facilities, such as the Argonne Fast Chopper. A detailed examination of the limits of detectability of such a tool when used with a flux of 5×10^{15} n/cm^2-sec shows that: total cross-sections can be determined with high precision up to 10 kev, and in the range of 10 kev-50 kev the accuracy would compare with present day Van de Graaff techniques; all resonances of 90% of all stable nuclides can be well resolved to 100 ev and detected to 500 ev and so on. Similar improvements can be made in other physics investigations, and in this paper we discuss the quantitative improvements possible in a number of beam type experiments.

Neutron diffraction studies, the principle tool of many solid-state and metallurgical investigations, require high resolution and hence, high source strength. One can also speculate on what might be possible with ultra high fluxes (10^{16}-10^{18}) and these include studies of the neutron-neutron interaction, simulated low order nuclear explosions, and studies of materials problems likely to be encountered in a thermo-nuclear reactor.

Advances of nuclear technology prompt the consideration of densities in the range 1-10 Mw/l. and this requires basic heat transfer research and fuel element testing at high flux levels. Examples of progress in research and test reactor design leading to advances in power reactor design are plentiful: to name a few, MTR design studies played an important role in the development

of boiling and pressurized water reactors; similar roles have been played by CP-5, HYPO, and X-10. Current possibilities for ultra high flux ($>10^{16}$) reactors are described, and their special problems presented.

INTRODUCTION

It is a commonplace that as soon as a research facility is built demand arises for another which is both bigger and better; this is the price we pay for rapid advances in basic research and technology. It is noteworthy therefore, that it is only within the past two to three years that there has been much discussion of testing and research reactors with thermal neutron fluxes in the 10^{15}-10^{16} range. The reason is that the advent of CP-5, MTR, and their counterparts allowed an unparalleled expansion of research and development involving neutrons. So little was known in these fields that it took the researchers some time to catch up with the builders. Now that has happened, and the needs of basic research and reactor technology are distressingly evident. In this paper we will discuss the uses to which high flux reactors can profitably be put; we will make an attempt to include all fields of endeavor but our emphasis will be on research in the physical sciences.

To this end it is legitimate to inquire as to the general consequences of raising the neutron flux level. Most obvious is an increase in precision of most experiments; there are few experiments of interest in the physical sciences which require low fluxes (but there are important exceptions in the biological sciences) so that increasing the flux at the very least decreases the statistical error in the information gathered. Of far greater importance, however, is the possibility of converting an experiment which is largely qualitative in nature into one which is quantitative. Perhaps the most striking examples here are the experiments on asymmetries in beta-decay: the parity non-conservation experiments. There are many others of a similar nature. To complete the chain, there are experiments of a new type which will become feasible for the first time at these flux levels; since most of these can be at best qualitative, completion of these experiments will inevitably give rise to demands for yet a higher flux level.

CHEMISTRY EXPERIMENTS

It is in the field of chemistry that the most pressing need for higher flux levels occurs. The chemistry of the transuranic elements and the production of new elements by heavy ion bombardment has progressed to the point where microgram quantities of Cf^{252} are insufficient to permit progress at a rapid rate. Cf^{252} is not the only heavy isotope desirable, but it is certainly the one most wanted. Using Pu^{239} as a feed material, the production rate of Cf^{252} is proportional to the 9th or 10th power of the flux; Cm^{244} is produced at a rate proportional to the 5th power of the flux and is a possible feed material in the high flux reactors now being designed. In particular, plentiful supplies of Cm^{244} would enable one to produce Bk^{247} by cyclotron bombardment. This very long lived ($\sim$10,000 yrs by α-decay) isotope would be ideal for heavy element chemical studies. At present, however, the only available isotope with a reasonably long half life is Bk^{249} (β^-, $t_{1/2}$ = 1 yr) and this has been produced only in microgram quantities by plutonium irradiation. The heavy isotopes are useful tools for further investigation and interesting objects of study in themselves. The chemistry of heavy isotopes is complicated by their activity. To study the chemistry of Cf, therefore, it has been suggested that Cm^{244} (or

Pu^{242}) be used as a feed material for the production of Cf^{252} and Bk^{249}. The Cf^{252} is useful as a fission source but far too active for chemical studies. The Bk^{249} besides being itself a subject of study, can also be allowed to decay to Cf^{249} which, having a half life of about 500 years, is a good isotope for the study of Cf chemistry. After Bk^{249} is extracted, the Cm^{244} can be returned for recycling. Figure 1 is a graph of isotope production in an average flux of 3×10^{15} starting with Cm^{244} as feed material. We will not belabor this point further except to point out that the advent of Pu recycle reactors and of fast breeder reactors with high power densities and Pu fuel loadings require a sound background in heavy element chemistry before adequate fuel processing can take place.

Further examples of experiments not feasible at lower fluxes are attempts to produce nuclides such as Hg^{206}, Zr^{98}, Nd^{152}, etc.

Experiments which can be shifted from qualitative to a quantitative basis or whose precision can be greatly improved are those which study the systematics of neutron interaction with fissionable materials and the fission process itself. Both Cm^{244} and Cf^{252} are of interest in this regard since the primary mode of decay of these isotopes is spontaneous fission.

Another sort of chemistry experiments that bear directly on reactor design are experiments on energy storage. Two general classes of effects are expected to appear at higher flux levels: one, the non-linear effects of high integrated doses; two, possible effects proportional to the rate of irradiation.

PHYSICS EXPERIMENTS

While the chemistry experiments just discussed are irradiation experiments which, typically, involve multiple capture processes, the usual physics experiment is a beam experiment. The number of beam experiments either performed or proposed is so large that it will serve our purpose to discuss just two of the most important cases in detail. For the sake of concreteness, a source strength of 5×10^{15} n/cm^2-sec is compared with the CP-5 flux of 3×10^{13}.

Argonne Fast Chopper

Figure 2 is a picture of the Argonne Fast Chopper; we wish to consider the measurement on neutron cross sections with this chopper. Qualitatively, the discussion is also applicable to measurements with a crystal spectrometer.

Total Cross Section

The total neutron cross section is usually determined by a measurement of the neutron transmission of a sample of the material being studied. For measurements of this kind an increase in the flux to 5×10^{15} would permit a great improvement in the resolution of the time-of-flight system and a reduction in the minimum weight of the sample being studied. Let us consider these two effects in more detail.

The narrowest resolution width now in use in neutron transmission measurements is that of the Argonne fast chopper at CP-5, for which the resolution width is 12 mμsec/m. With existing detectors and electronic equipment the resolution width could be reduced to 6 mμsec/m, with the same counting rate

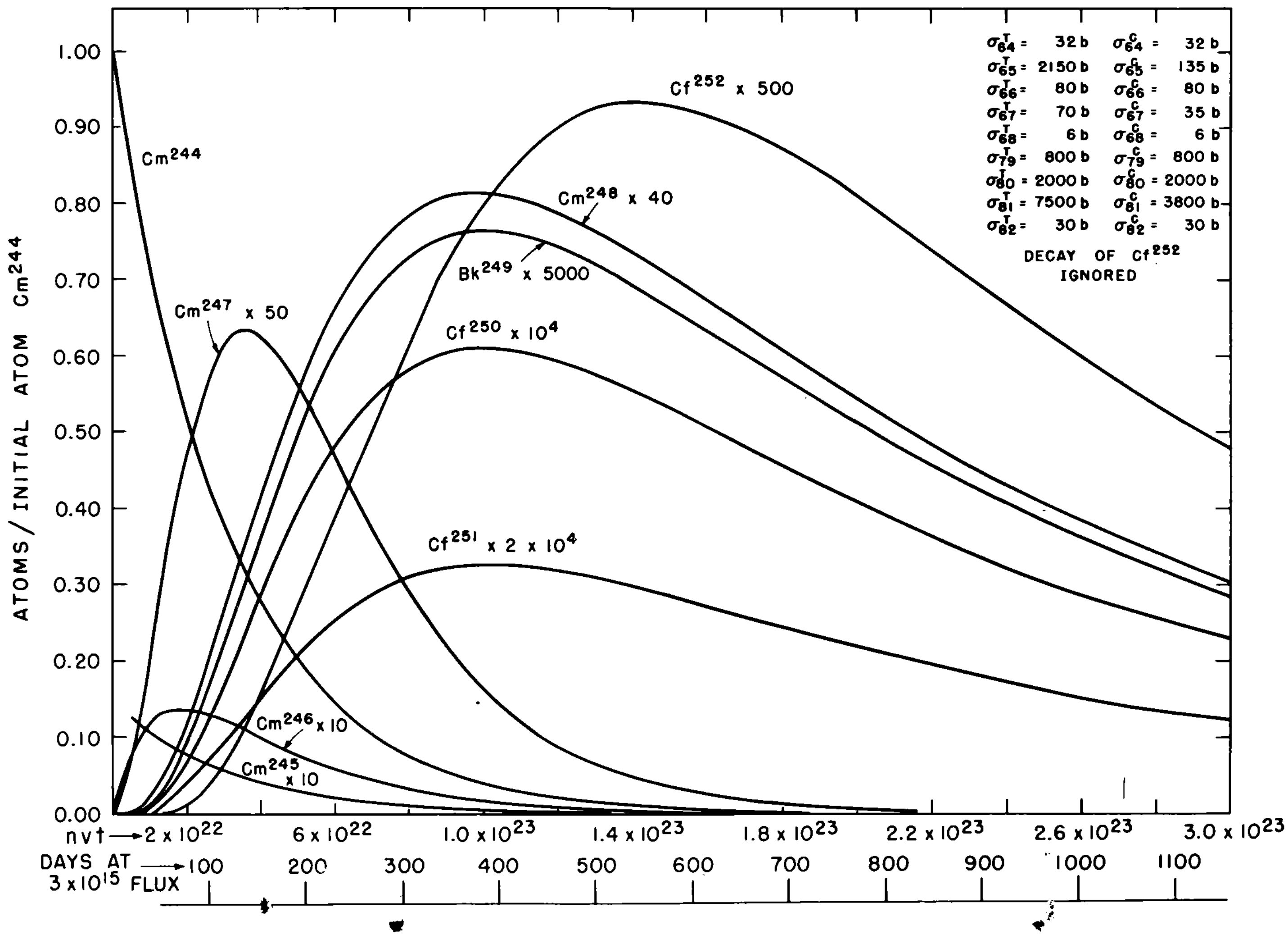

Fig. 1—Build-up of Cf^{252} and intermediate isotopes from Cm^{244}.

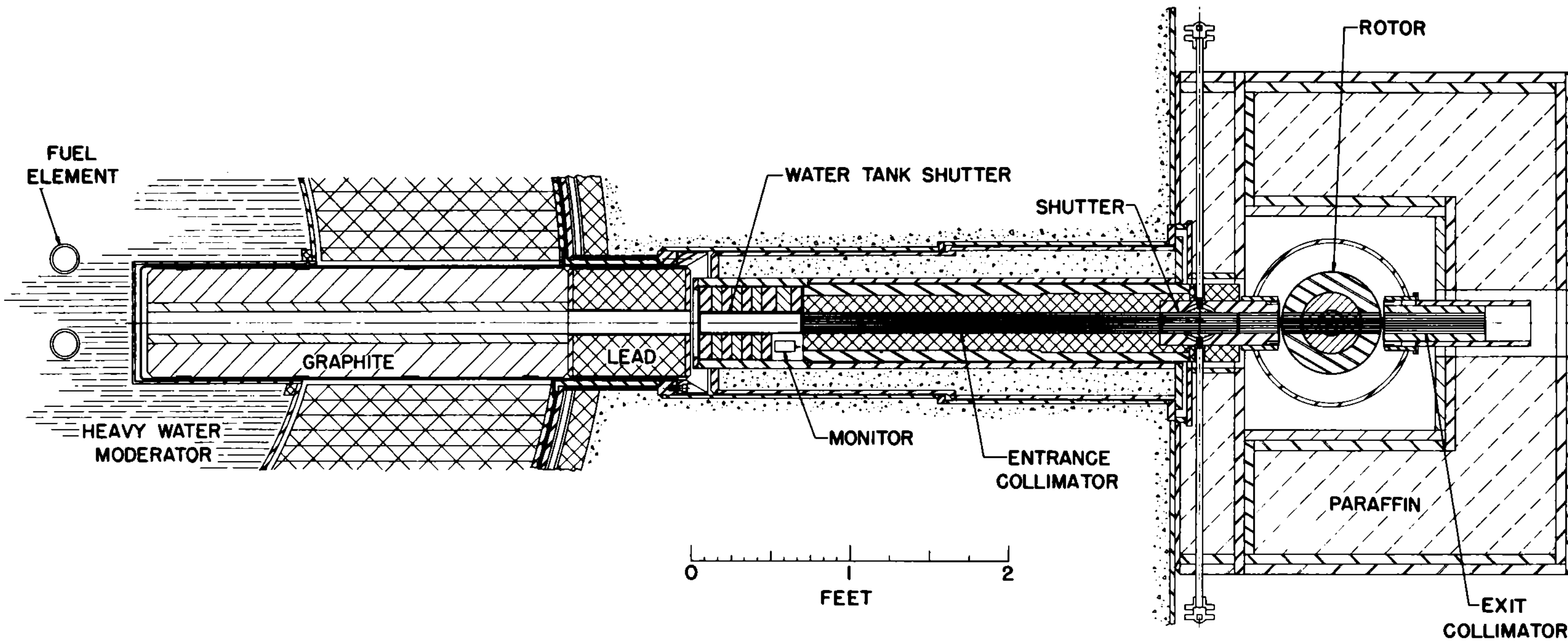

Fig. 2—ANL fast chopper.

per channel, if the neutron source strength were increased by a factor of about 12. If the flux were increased to 5 x 10^{15} by major modifications of the chopper and its associated equipment (but using well established techniques), a resolution of 1.5 mμsec/m could be obtained. The relative resolution as a function of energy of a system having this resolution is given in Table I.

TABLE I

Relative resolution vs energy for a time-of-flight spectrometer having a resolution of 1.5 m sec/mμ

E (ev)	ΔE/E	ΔE (ev)
20	0.0002	0.004
10^2	0.00042	0.042
10^3	0.0013	1.3
10^4	0.0042	42
10^5	0.013	1,300
10^6	0.042	42,000
10^7	0.13	1,300,000

An examination of Table I indicates several ways in which an extremely high resolution spectrometer could extend the range of measurements of slow neutron spectroscopy:

a. The chopper would be competitive with all other techniques of measuring total cross sections up to an energy of 50 kev, and would be far superior to the Van de Graaff generator below 10 kev.

b. Useful measurements of average cross sections could be made up to an energy of 10 Mev. At the present time very few cross sections of this kind have been measured in the energy range from about 3 to 10 Mev.

c. All resonances of 90% of the stable nuclides would be well resolved up to an energy of about 100 ev.

d. Almost all resonances of 90% of all nuclides would be detected up to an energy of 500 ev.

An alternative use of a great increase in neutron source strength would be to study extremely small samples. At the present time it is necessary to have a sample having a cross sectional area of at least 2 mm^2 to obtain a resolution of about 0.1μsec/m, the poorest resolution which should be used if resonance structure is to be studied. For heavy nuclides an area of 2 mm^2 corresponds to a weight of about 10 mg of material, if only resonant structure is to be studied, and a weight of about 200 mg if the off-resonant cross sections are also to be measured. At the neutron flux considered it seems feasible, with minor modifications of existing equipment, to obtain a usable counting rate and a resolution of 0.1μsec/meter for a sample having a cross section of only 0.1 mm^2; major modifications in the design of the chopper might make it feasible to use an even smaller sample. An area of 0.1 mm^2 corresponds to a weight of 0.5 and 10 mg of a heavy nuclide for studies of the resonances and the off-resonance cross sections, respectively. These weights are small enough that it should be possible to obtain satisfactory samples for the study of all of the stable nuclides, many of the rare nuclides which are alpha unstable and many of the lighter nuclides which are beta unstable. Especially the latter category of nuclides constitute an almost untouched field for neutron cross section measurements. In addition to making it feasible to procure a wider range of samples, the use of a small sample makes it much easier to handle very radioactive materials.

Radiative Capture Cross Sections

The measurement of radiative capture cross sections is of the highest priority to meet the needs of the U.S. reactor program.[a] At the present time, the only known technique for measuring this cross section by the time-of-flight method, i.e., in the range of energy from 10 to 10,000 ev, is to observe the prompt gamma rays from capture with a detector that is so large that it is almost certain to detect some of the binding energy released in the form of gamma rays. Until now it has not been possible to use this technique in the region of energy accessible to the chopper because of the high background rate (from natural causes, such as cosmic rays, etc.) in the large detectors required for the measurement. However, an increase in the neutron source strength to 5×10^{15} would cause the rate from capture to be large enough to be measurable for most nuclides over the full range from thermal energy to about 10 kev.

TABLE II

Estimates of the minimum detectable capture cross section σ_γ for U^{238}, as measured with the present Argonne fast chopper at a source having a flux of 5×10^{15} neutrons/cm^2/sec. In the calculation it is assumed that a rate of capture of 10 neutrons/sec could be detected. Also, the sample is thin enough that the effect of self-shielding of the resonances would be unimportant.

E	$\frac{\Delta E}{E}$	Minimum detectable σ_γ (barns)	Approximate value of σ_γ (barns)	$\sigma_\gamma/\sigma_\gamma$ (min)
10^2	0.10	0.180	25	140
10^3	0.10	0.030	4.3	140
10^4	0.23	0.0018	0.56	310

Perhaps the most important (for reactor applications) and difficult measurement would be that of an average capture cross section in a range of energy where individual resonances could not be resolved. As an example of results which might be obtained in such measurement, estimates of the minimum detectable cross section for the typical material U^{238} are compared in Table II with the actual cross section for that material. It is seen that over a wide range of energy the actual cross section is several orders of magnitude greater than the minimum detectable cross section when a flux of 5×10^{15}

[a] See Nuclear Cross Sections Advisory Group, Compilation of Requests for Nuclear Cross-Section Measurements, September, 1957, BNL-463, T-103.

neutrons $cm^{-2}sec^{-1}$ is available. However, it is also clear, since these minimum detectable cross sections would have to be increased by a factor of 200 for a measurement at CP-5, that useful measurements cannot be made with the present source of neutrons. Thus the availability of a much higher neutron source strength opens up an entirely new area for cross section investigations.

Scattering Cross Sections

The measurement of the scattering cross section for slow neutrons is important for nuclear theory because it is the most straightforward and only proven way of determining the total angular momenta of the neutron resonances. With the present source strength only the most favorable resonances can be studied. However, with a flux of 5×10^{15} neutrons/cm^2 sec, the scattering cross section could be measured with accuracy for essentially all of the resonances which can be resolved with a resolution of about 0.1μsec/m, including the levels in the fissionable nuclides. The determination of the total angular momenta of the resonances in the fissionable nuclides is of particular interest because of the bearing of these results on the theory of fission.

Fission Cross Sections

A substantial increase in flux would have much the same effect on the measurement of fission cross sections as has already been discussed for the total cross section, i.e., it would allow measurements to be made with higher resolution and with smaller quantities of material than is now possible. For a material such as U^{235} which can be used in large quantities, the resolution of a fission cross section measurement can come within a factor of 2 of that obtainable in a transmission measurement; thus Table I may be used as an indication of the resolution which could be achieved with a reactor having a flux of 5×10^{15} neutrons/cm^2/sec. Alternatively, if the higher flux were used to study the fission cross section of small quantities of rare fissionable materials, measurements with a resolution of about 0.1μsec/m, which now requires 20 mg to achieve refined results, could be done with $100 \mu g$ and measurements of lower resolution, which might cover the energy range below 10 ev, could be done with as little as 1μ mg of material. It is obvious that an ability to work with such a small quantity of material would in time increase the number of fissionable nuclides for which the resonance structure could be studied by a large factor.

Argonne Bent Crystal Spectrometer

The radiation following neutron capture plays a role in nuclei roughly analogous to the optical spectra of atoms. There is considerable hope that precise measurements of these spectra will be very useful in increasing our understanding of nuclear behavior. Thermal neutron capture in a reactor is the best source of these spectra.

The information obtained with the 7.7 meter bent-crystal spectrometer (Fig. 3) has shown that these capture gamma spectra are very complex. The high precision and high resolution of the instrument are such as to provide energy measurements with an average accuracy of 1 part in 5000. The mathematical problem of fitting the gamma rays into a level scheme is comparable in magnitude to that of fitting the lines of a complex atomic spectrum. An improvement of accuracy and resolution of a factor of 2 or 3 would simplify the problem to such an extent as to make the fitting of a level scheme an elementary problem.

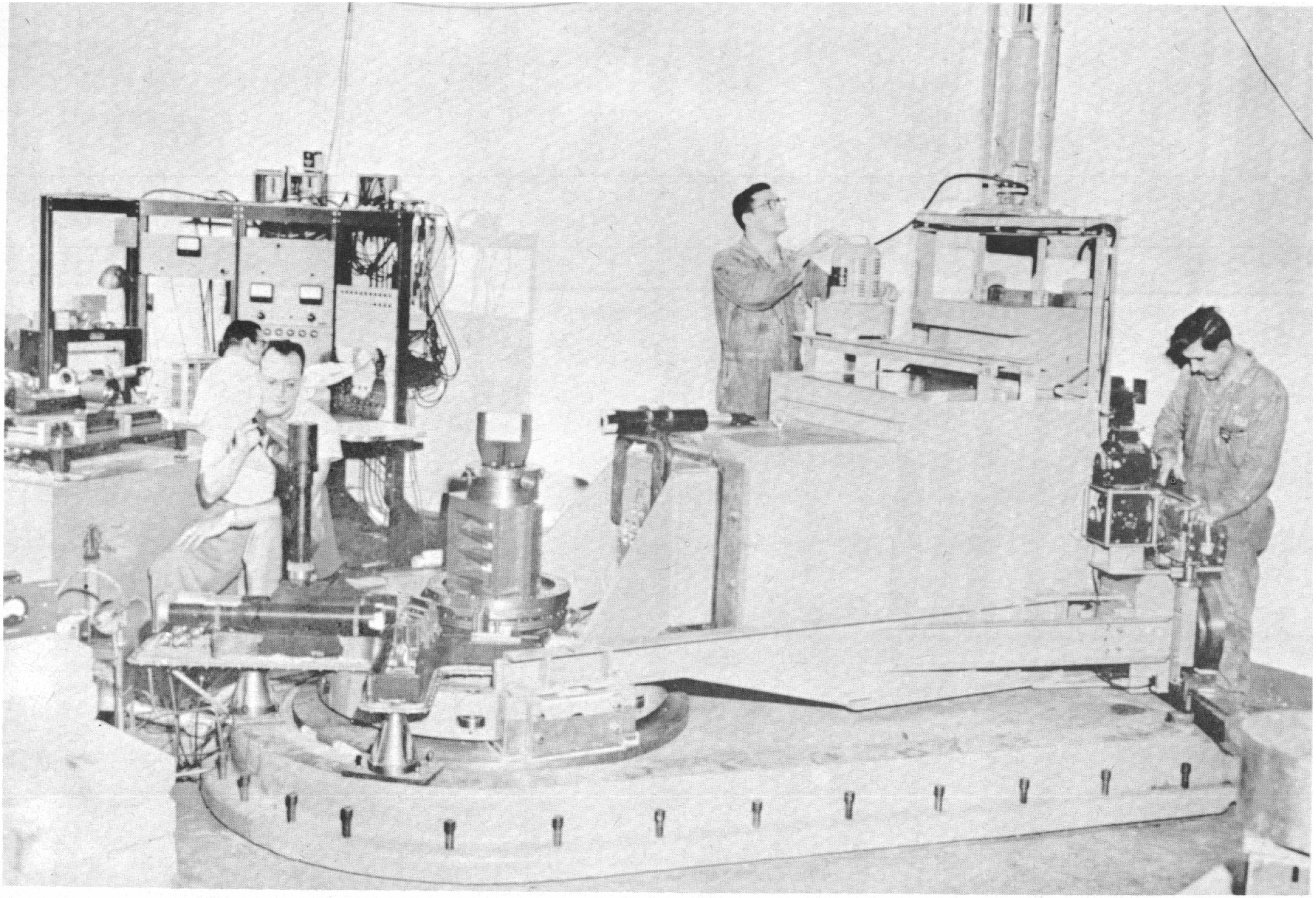

Fig. 3—ANL bent crystal spectrometer.

Such an improvement would be obtained very easily with a reactor having 50-100 times the flux of CP-5 (3×10^{13}). The 7.7 meter instrument has shown that the line widths are nearly proportional to the stress applied to the quartz crystal. If a 15 meter instrument using the same size of quartz were used then the stress would decrease by a factor of 2 and the line widths would approach those coming from unbent quartz plates. The loss of a factor of 4 in solid angle would be overcome by the 50-100 fold increase in flux. The greater distance of the detectors from the reactor would serve to decrease the general room background. As a fairly bold approach one may even think of a 30 meter instrument with the spectrometer located in a bubble outside the reactor room. A 4 mm thick quartz plate bent to a 30 meter radius would be virtually flat and should yield line widths close to 1-2 seconds of arc. It should be pointed out that the problem of bending the crystal in the above suggested instruments becomes very simple. There would be no danger of breaking the plates during bending.

Since the work with the bent-crystal spectrometer only yields data on the unconverted gamma radiation it is also necessary to study the conversion electrons. The best instrument for this purpose would be a magnetic Compton spectrometer. Such an instrument can also be used to study the higher energy portion of the unconverted gamma ray spectrum. The higher flux would make it possible to make the high energy measurements with a resolution well below the present limit of $\frac{1}{2}$ to 1%.

Much larger source strengths than are now available would make much more practical the rather primitive studies that have so far been conducted on the polarization of gamma produced by capture by polarized nuclei and on the anisotropy of beta emission following the same kind of capture.

Other Experiments: Internally Converted Capture Gamma Rays

As we have mentioned, an experiment complementary to those performed with the bent crystal spectrometer involves experiments on internal conversion. The study of gamma-ray spectra resulting from capture of thermal neutrons is by now a highly developed science. However, almost no work has been done in the closely related area of studying the beta-rays resulting from the internal conversion of these gamma rays. This neglect of an apparently fruitful field is a direct result of an inadequate source of neutrons. If a source strength of 5×10^{15} neutrons/cm^2/sec were made available, however, the study of internal conversion could become at least as important as the study of capture gamma rays because of the usefulness of internal conversion coefficients in assigning spins and parities to the states involved in transitions.

To be really powerful as a tool for studying the low energy state of a compound nucleus through observation of internally converted beta-rays, it is highly desirable to eliminate some of the complexity that is present when all of the beta-rays are detected. One technique (which has not yet been tried) of obtaining this greater simplicity is to detect only those beta-rays which are in coincidence with a high energy gamma-ray. In this way, only transitions between low-lying states are permitted to be observed. With the present reactor (CP-5) this technique would be limited to materials having a thermal capture cross section greater than about 200 barns, so that very few nuclides could be studied. With a neutron source strength of 5×10^{15} neutrons/cm^2/sec, however, a large fraction of the stable nuclides could be studied. Thus a new and powerful tool would become available for determining the energies, spins and parities of the low-lying states in most of the stable nuclides.

Possible New Experiments

Examples of new experiments which become feasible or which move from the status of barely observable to quantitative studies are those involving the elementary particles. A short summary of three of these is:

Neutron Decay

This is one of the most interesting kinds of beta decay because it is free of the usual nuclear structure complications. A high flux reactor, besides permitting much greater accuracy in such important measurements as lifetime, spectrum, correlation of electron momentum and neutron spin and others, would probably permit entirely new measurements to be made such as correlation of electron spin and neutron spin. In spite of the notable recent advances in our understanding of beta-decay, some of these quantities are poorly explained and none is completely accounted for.

Neutron Interactions

The neutron-proton and electron-neutron interactions are of considerable interest for their bearing on the structure of the particles involved and as a basis for a theory of nuclei. While low energy data alone on these may not lead to any major new developments, this data has a considerable value taken in conjunction with data from accelerators. There is no obvious limit to the precision that would be desirable in so fundamental a system. High intensity beams would help greatly in improved measurements.

Neutrino Interactions

In studies of neutrino interactions total reactor power divided by distance of experiment squared appears to be the most important quantity. In a high flux reactor there is at least a possibility of maximizing this quantity by special constructions in the shield. In addition, greater sophistication in detection schemes might well make flux the most important quantity in determining the number of measurable neutrinos. More measurements on this very elusive elementary particle are obviously badly needed.

While we have by no means made an exhaustive study of physics experiments suitable for a high flux reactor, we have given a sufficient number of examples to draw some general conclusions. After about 5 to 10 years work with fluxes in the 10^{15} to 10^{16} range, we would expect to have a reasonably complete picture of nuclear resonance levels up to a few hundred electron volts so that the accuracy of prediction of cross sections of most stable isotopes would be greatly increased. The average radiative capture cross section in the unresolved resonance region and the nuclear level schemes of most of the important heavy isotopes would be well known so that prediction of neutron spectra, breeding gains, and cross sections in general in fast reactors would become a matter of routine computation. Considerably more would be known about the neutron itself and the quantitative aspects of parity non-conservation would be on firm ground.

SOLID STATE WORK

Work on the solid state is now so highly specialized that it deserves treatment as a separate discipline. The principal beam experiment associated with this field is neutron diffraction.

Neutron Diffraction

This method of studying solids has a unique value for determination of hydrogen positions, magnetic structures and a few other problems. However, neutron diffraction patterns are much poorer in intensity and resolution than x-ray diffraction patterns and the work is badly handicapped as a result. The reason for this is that a flux of upwards of 10^{18} would be needed to give a monochromatic intensity comparable to that obtainable from an x-ray tube. While there are possibilities of using certain ingenious tricks to improve the situation relative to x-rays, it will clearly remain true that every increase in neutron flux will make it possible to analyze more complex crystals and materials available in smaller crystal sizes.

If finally neutron diffraction patterns become comparable in quality to x-ray patterns, it opens up the possibility that neutron diffraction may be a powerful aid to x-ray diffraction in the solution of the main structures of the complex organic materials which are of great importance in biochemistry. There are even techniques which make neutron diffraction, given sufficient flux, look more powerful than x-ray diffraction because it does not suffer from the smearing associated with the atomic scattering factor. Thus the Patterson projection technique[1] might be useful for more complex structures than it is with x-rays. Despite its advantage of not requiring the notoriously elusive phase information it is not very helpful with x-rays. This is because its complexity makes identification of structures quite difficult in all but fairly simple materials for the details are lost in the smearing arising from the atomic scattering factor. With neutrons of short wavelength (to minimize series-termination effects) the Patterson projections should have much greater clarity.

Another general technique available in neutron diffraction but not in x-rays is the isotopic replacement method which is the same as the isomorphous replacement method[1] with different isotopes playing the role of different elements. It is probably of wider applicability than isomorphous replacement because a change in isotopic composition is possible at least in almost all complex molecules. In addition, it is safer since isotopic changes generally have less effect on a structure.

The question has been raised of whether or not the real bottleneck to progress in diffraction studies is the number of qualified investigators available, so some remarks on this may be in order here. There are three comments that seem pertinent: (1) the high flux could profitably be used for better patterns rather than just more, i.e., for higher resolution, for work on smaller samples (separated isotopes, small single crystals, etc.) and for greater statistical accuracy; (2) given the neutron diffraction data there are probably many people in university and other research laboratories without high-flux reactors who would be interested in analyzing it, perhaps in conjunction with their own x-ray data; (3) there is a growing mechanization of the analysis of diffraction data by use of computers. This tendency will almost certainly go much further and will reduce the need for skilled manpower although in this as in all fields more geniuses would admittedly be useful.

Thermal Inelastic Scattering

This is a technique of measuring the vibration spectrum of solids and liquids which, were sufficient flux available, would be much the best method. It involves the measurement of spectra of monochromatic slow neutrons scattered in a particular direction by a crystal. It is competitive with x-ray methods at a flux of about 10^{14}. At a flux of 10^{15} it should be greatly superior in resolution and speed. The reason for the superiority is basically that neutrons are much more affected by the vibrations of a crystal than are x-rays because neutrons of the appropriate wave-length (ca. 1A) have energies near that of vibration phonons. The vibration spectra are the key to the understanding of the dynamic properties of solids - specific heat, compressibility, etc. This class of experiments includes the "Moderator Law Experiment" in progress at NRU. This tripartite experiment was planned and the equipment built at Harwell in the U.K. and is installed at NRU (Chalk River). The data will be processed at Argonne National Laboratory. The aim of this experiment is to find the scattering matrix for a variety of moderators. The main advantages of higher flux are higher speed (the experiment is involved and will take some time) and higher resolution.

ADDITIONAL BEAM EXPERIMENTS

There are a number of experiments which are much more convenient or only feasible in a region external to the reactor proper; these are perforce beam experiments. Some examples are:

Effects Following Neutron Capture

Studies of the effect of neutron capture (and the subsequent rapid emission of gamma rays) on the chemical or electronic state of molecules or atoms would be put on a new basis if high flux beams were available to permit measurements of the states of the ions left after capture in low pressure gases.

Radiation Damage in External Samples

A high flux reactor would permit studies of radiation damage at exposures of up to 10^{19} neuts cm^{-2} in samples external to the reactor. This has obvious advantages, as compared to radiation inside the reactor, in control of temperature and environment in general and in ease of measurement of changes. For example, x-ray diffraction studies of the sample are impossible inside the reactor and easy outside.

Separation of Effects of Different Kinds of Radiation

In a reactor it is not possible to separate the different kinds of radiation whose effects may be of interest as well as in a beam. For certain systems which are quite sensitive to radiation (animals for example) it should be possible to study radiation effects in beams of almost pure neutrons in controlled energy groups. Certainly this would be possible with thermal neutrons but it might also be done with neutrons in the resonance regions which are interesting because (a) radiation effects are difficult to estimate in this region on the basis of available data, (b) the resonances give a possibility of achieving chemical selectivity in capture which raises some interesting possibilities particularly in biological work.

Cold Neutrons and Cold Sample Experiments

Work on solid state effects often times involves maintaining the sample at temperatures of the order of liquid He (4.2°K) for prolonged periods. High source strength considerably simplifies cryostat design and exposure time while also lowering He consumption. There is a considerable amount of work being done with cold neutrons: the measurement of vibrational spectra has already been mentioned. Current sources of beams of such neutrons are usually beams of thermal neutrons which have been passed through cool Be filters. A new technique is under investigation at Harwell and Argonne. This technique consists of moderating thermal neutrons in a sample of H_2O or D_2O which is maintained at very low (4°K) temperatures. In this way a portion of the neutrons rejected by a Be filter are transferred to a lower energy and are passed through. These experiments are complicated by the difficulties of building a large cryostat and, at high power, by the necessity of removing the gamma heating. The preliminary results appear promising. Since the reaction rate in a $1/v$ sample is inversely proportional to the neutron temperature, a cold neutron facility of the cryostat type represents a possible way to increase further the reaction rate of small samples in high flux reactors.

Again some general conclusions can be drawn about the probable results of 5 to 10 years of experimental work at high fluxes. The increase in knowledge of vibrational spectra of solids, the successful prosecution in a precise fashion of the "Moderator Law Experiment" and greatly increased knowledge of the behavior of interstitial hydrogen in crystal lattices, coupled with only recently developed computational methods, will lead to good representations of the epithermal neutron spectrum so important for the proper evaluation of advanced power reactors. The mechanism of diffusion of vacancies and other lattice defects should be on a firm, quantitative basis. Marked advances should be made in identifying the structure of organic molecules and similar complex structures.

INFLUENCE ON REACTOR TECHNOLOGY

Turning now from the basic to the applied sciences, we can make some observations on the influence of high flux reactors on reactor technology and the reverse. Current high flux reactor designs call for heat releases of the order of 1 to 10 MW/ℓ . This is the region of heat release in fast reactors currently being designed or built, and it is reasonable to expect that thermal power reactor designs will start to approach these levels especially in designs calling for cool moderator. Certainly if planetary or inter-planetary propulsion with atomic energy is to be accomplished very high power densities must be attained. This means that facilities must be built which are capable of testing high power density fuel elements. Clearly, high flux test reactors are needed for this purpose. An important problem in advanced engineering test reactors is the problem of producing the proper neutron spectrum in the test sample. This can be accomplished in part by shielding with natural uranium, cadmium, iron, or a variety of other substances, depending on the facility and the spectrum desired. Unfortunately, these shields attenuate the neutron flux level so that a high source strength is a necessity.

The pressing problem of fuel behavior at high burn-up is with us and will continue to be a problem for some time. Current thermal power reactor designs often call for UO_2 fuel with burn-ups of 10,000 to 20,000 MWD/T. Performance of fuel at high burn-ups is essentially unknown so that the designs are often predicated on extrapolations which may be nothing more than wishful thinking.

We would be remiss not to mention the long history of the influence of test and research reactor design on power reactor design. The most prominent examples are MTR, CP-3, X-10 and HYPO. MTR was almost directly responsible for the BORAX experiments, which in turn led to the development of the boiling water reactor. Another project deriving great benefit from the MTR program was the submarine propulsion reactor and hence, the pressurized water reactors. CP-3 is the ancestor of CP-5, to be sure, but it is also the ancestor of a host of natural uranium heavy water power reactor designs. X-10, of course, is the arch-type of gas cooled reactor designs among which is KIWI-A, designed to lead to nuclear rocket propulsion. HYPO has given rise to a series of experiments on aqueous homogeneous reactors and designs which hold forth a great deal of promise, especially as thermal breeders.

Left unsaid so far is the fact that in a thermo-nuclear reactor the neutron flux is bound to be high by virtue of the high power of the machine. Conventional reactors operating at high flux levels will be required to establish material properties in the environment of a thermo-nuclear reactor.

CURRENT POSSIBILITIES FOR ULTRA-HIGH FLUX ($>10^{16}$) REACTORS

In conclusion we wish to discuss some of the current possibilities for ultra high flux reactors ($>10^{16}$). Experiments that might be performed at such levels include environmental tests for thermo-nuclear reactors, as we have just mentioned. Certainly the neutron-neutron interaction comes to mind as an object of study, and simulated low order nuclear explosions are another. Fundamental research in quantum electrodynamics will continue to demand ever larger sources of neutrons, gamma-rays, and neutrinos.

The problems of ultra-high flux reactors are: heat removal, fuel cycle (and operational cost), and control. Heat removal is probably the biggest stumbling block. Power generation at the rate of 10-100 MW/ℓ is necessary to produce ultra high fluxes. At present such power densities are encountered only in systems cooled largely by radiation or pulsed systems. One example is a 100 W tungsten filament lamp which has a power density of about 150 MW/ℓ. While radiation cooling is a promising method of heat removal, other methods that suggest themselves are: boiling metal or molten salts as coolants, or dust carrying gases.

The fuel cycle is unavoidably short and the cost high, but some methods can be applied to reduce the magnitude of the fuel handling problem. Rather than use a single core with a low maximum to average power generation to act as a source for a variety of experiments, it may be possible to use a loosely coupled system in which a large, low power density core is coupled with a small, high power density source for each facility. The total power may very well be as large or larger as for the compact system, but the fuel handling problem will be reduced in magnitude.

The control problem is vexing. At ultra-high fluxes conventional materials burn out too rapidly and create serious hot spot difficulties through rapid changes in blackness. The solution probably lies in the development of fast acting fluid systems.

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THE DESIGN AND USES OF HIGH FLUX RESEARCH AND TEST REACTORS

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ABSTRACT

The need for thermal and fast neutron fluxes in the range of 1-5 x 10^{15} neutrons/cm^2(sec) for the production of heavy elements such as Cf-252 and other special isotopes and for improving the quality of beam experiments, reactor materials testing work, and solid state research has led to plans for the construction of three ultra high flux research reactors by the U. S. Atomic Energy Commission. The designs of these and other high flux research reactors are based on the general technology of enriched tank-type reactors; however, they utilize the principle of separate fuel and moderator regions to achieve flux peaking in these regions. Thus, proposed designs take the form of an annular fuel region with internal and external moderating regions or an undermoderated core with an external moderator or reflector. In such arrangements, the thermal flux peaks in the moderating regions and the fast flux peaks in the fuel region, which results in maximum fast and thermal fluxes per unit of power. Since most of the moderation of fast neutrons takes place outside of the fuel region, the thermal flux peaking depends on the number of fast neutrons leaking from the reactor core, which in turn depends on the power density in the core. Thus, maximum fluxes of neutrons at all energy levels are proportional to the power density in the reactor core. Interest in high flux reactors, therefore, has led to a consideration of ways in which the power density in the reactor core, and thereby the flux, may be increased.

A review of the designs of various high flux research reactors and an advanced test reactor being considered for construction in the United States indicates a definite trend toward single purpose, rather than multipurpose, facilities. The Oak Ridge National Laboratory's high flux reactor (HFIR), for example, is designed specifically to provide facilities for the production of heavy elements and other isotopes, with lesser consideration given to other uses. The Brookhaven high flux beam reactor (HFBR) is also aimed at a specific objective; namely, neutron beam research. Finally, the Argonne high flux reactor (AHFR) is designed primarily for physical science research with chemistry experiments playing a secondary role. The combined usefulness of these three reactors is believed to be much greater in relation to their total cost than that of a large multipurpose reactor.

INTRODUCTION

The need for neutron fluxes above 10^{14} neutrons/cm^2(sec) for carrying out advanced basic research programs and for testing reactor fuels and other materials has led to the construction of some eleven high flux research reactors in the United States and eighteen in other countries. These reactors have yielded excellent results so far, and will continue to provide facilities which are adequate to meet most neutron flux needs for at least the next decade; however, for certain types of experiments and for other special uses, these facilities are not adequate. In particular, fluxes above 10^{15} neutrons/cm^2(sec) are required for the production of transplutonium elements and other special isotopes and for extending the range of accurate cross section measurements to the high epithermal regions. Because of these and other needs, plans have been initiated by the U. S. Atomic Energy Commission for the construction of three, and possibly four, ultra high flux reactors capable of providing fast and thermal neutron fluxes in the range of 2-5 x 10^{15} neutrons/cm^2(sec). Each of these reactors is designed to meet a different objective, thus the reactors complement, rather than supplement, one another. This approach, it is believed, will result in nuclear research and test facilities which in total will cost less, and be far more useful, than a single large multipurpose facility. The designs of these reactors, moreover, are representative of the best that can be done with present day technology, and provide the basis for extending this technology if desired.

The four reactors involved in this program are: (1) the Brookhaven High Flux Beam Reactor (HFBR) for neutron cross section measurements and other beam research, (2) the ORNL High Flux Isotope Reactor (HFIR) for the production of transplutonic elements and other isotopes, (3) the Argonne High Flux Reactor (AHFR) for physical science research, and (4) the Advanced Engineering Test Reactor (AETR) designed by the Internuclear Company for the Atomic Energy Commission for experiments with gas, liquid metal, and water cooled loops.

It is the purpose of this paper to review the uses, designs and characteristics of these reactors to show how the design differs if one wishes to build an isotope producer, a beam research reactor, a physical science reactor, or a loop testing facility.

USES OF NEUTRON FLUXES ABOVE 10^{15} NEUTRONS/CM^2(SEC)

1. Production of Transplutonium Elements

A most pressing need for high thermal neutron fluxes exists in connection with the production of isotopes of americium, curium, berkelium, and californium. These are formed in a reactor through successive neutron captures and beta decays, the primary reactions for which are as follows:

$$\underset{\sigma_a=52b}{\text{Pu-242}} + n \longrightarrow \text{Pu-243} \xrightarrow[5\ hr]{\beta} \underset{\sigma_a=138b}{\text{Am-243,}} + n \longrightarrow \text{Am-244} \xrightarrow[26\ m]{\beta}$$

$$\underset{\sigma_a=25b}{\text{Cm-244,}} + n \longrightarrow \underset{\substack{\sigma_a=200b \\ \sigma_f=1500b}}{\text{Cm-245,}} + n \longrightarrow \underset{\sigma_a=15b}{\text{Cm-246,}} + n \longrightarrow \underset{\substack{\sigma_a=200b \\ \sigma_f=200b}}{\text{Cm-247,}}$$

$$+ n \longrightarrow \underset{\sigma_a=2b}{\text{Cm-248,}} + n \longrightarrow \text{Cm-249} \xrightarrow[short]{\beta} \underset{\sigma_a=1100b}{\text{Bk-249,}} + n \longrightarrow$$

$$\text{Bk-250} \xrightarrow[3.1\ \text{hr}]{\beta} \underset{\sigma_a=1500\text{b}}{\text{Cf-250}}, + n \longrightarrow \underset{\substack{\sigma_a=3000\text{b}\\ \sigma_f=3000\text{b}}}{\text{Cf-251}}, + n \longrightarrow \underset{\sigma_a=30\text{b}}{\text{Cf-252}}$$

Thus it is seen that ten successive neutron captures are required to produce Cf-252 from Pu-242. Because of this, as well as the spontaneous fission of some of the heavy element isotopes, the initial production rates of Cf-252 and other heavy elements in a reactor are very flux dependent. This is shown in Figure 1, which gives the production of Cf-252 from 100 grams of Pu-242 at varying fluxes. The maximum values shown in this figure are based on the assumption of no losses due to side chain reactions, and the minimum values on the basis that the side chain reactions do not contribute at all to the production of Cf-252. Assuming that the cross sections used in the calculations are correct, actual production rates will lie within the ranges shown.

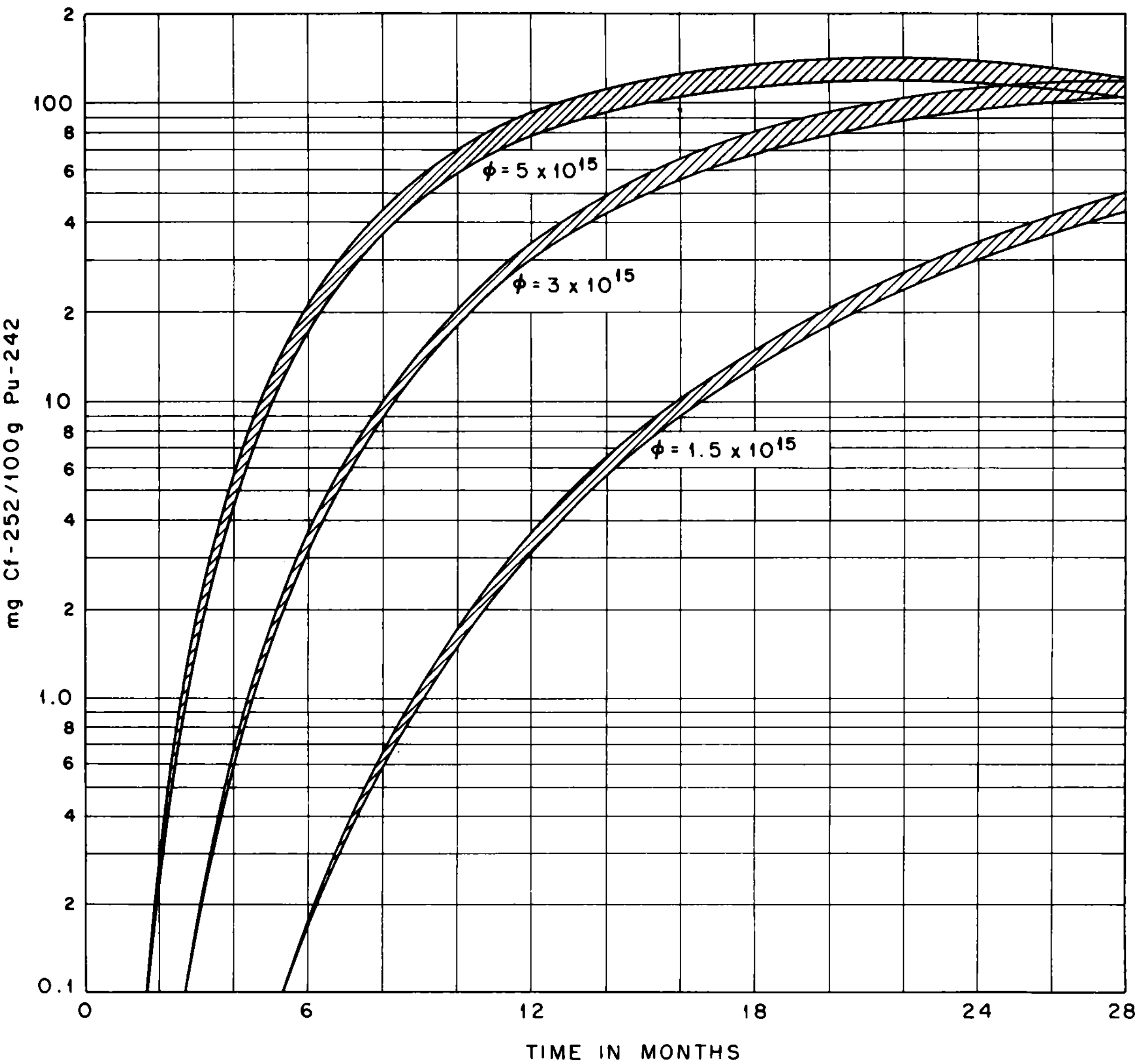

Fig. 1—Production of Cf^{252} at varying thermal neutron fluxes.

Although for short term irradiations the production of Cf-252 is proportional to about the third or fourth power of the flux, the time required to build up a given concentration of any particular heavy element in the target material varies inversely as the flux. At 5×10^{15} flux, for example, 12 months' exposure of 100 grams of Pu-242 will yield about 100 milligrams of Cf-252, whereas an exposure of 20 months is required at 3×10^{15} flux. The maximum number of milligrams of Cf-252 that can be produced annually from 100 grams of Pu-242, therefore, is proportional to the flux and is equal to $16.4\text{-}19.5 \times 10^{-15}\ \phi$, where ϕ is the effective thermal neutron flux in the target material.

These production rates are based on the use of thermal neutrons only; however, there may be some merit in exposing heavy element target material to neutrons at intermediate, rather than thermal, energies. Although quantitative data needed to verify this conclusion are not available, there are some theoretical reasons why intermediate fluxes may yield higher production rates than thermal fluxes(1). For example, one "bottle neck" in the production of Cf-252 appears to be Cm-248 which may have a thermal neutron absorption cross section of only 2 barns. Since Cf-252 has an absorption cross section of 30 barns, in the equilibrium between production and burn-out, only 2/30 of the Cf-252 formed is present. At intermediate fluxes, on the other hand, the statistical theory of the nucleus leads one to expect that the cross sections of even-even nuclides are about equal since the neutron spectra would extend over several resonances. Thus, the equilibrium between production and burn-out of Cf-252 would show a much greater fraction of the Cf-252 formed to be present. Also, the rate of approach to equilibrium would be increased.

A second reason why high intermediate fluxes may be desirable is that some of the transplutonium nuclides are fissionable and only the fraction, $\alpha/1 + \alpha$, is transformed into the next isotope, where α is the ratio of capture cross section to fission cross section. For known nuclides, α is greater at intermediate than at thermal energies, and one might guess that the same is true for the transplutonium elements. Thus intermediate energy neutrons would yield less fission and more higher isotopes than thermal ones. The design of a reactor for the production of transplutonium isotopes, therefore, must take into consideration the possibility that irradiating Pu-242 at intermediate, as well as thermal, energies may yield higher production rates of heavy elements.

The reasons for producing milligram quantities of heavy element isotopes are: (1) to study the long term reactivity of recycled fuels (α, ν, etc.), (2) to study the chemical, nuclear, and biological properties of the heavy elements, (3) to produce target material for charged particle bombardment, resulting in elements above atomic number 102, and (4) to serve as spontaneous fission neutron sources for neutron standards, measurement of reactor constants, study of short-lived fission products, biological studies of radiation damage, and portable high intensity neutron sources.

2. The Production of Special Isotopes(2)

For the production of isotopes of lighter elements, there are a number of reasons why thermal neutron fluxes in the range of 1×10^{15} neutrons/cm^2(sec) are desirable. These are: (1) the ability to obtain higher specific activities of short-lived isotopes, (2) the speeding-up in the production of long-lived materials, (3) the burning-out of undesired atoms in an isotopic mixture, (4) the conservation of high cost target materials, (5) the production of carrier-free isotopes, and (6) the reduction in cost of isotopes. Specific examples of the use of the HFIR for isotopic production are as follows:

1. The production of high-specific-activity short-lived isotopes,

such as 12.8 hr Cu-64, 14.2 hr Ga-72, 12.6 hr I-130, 12.5 hr K-42, and 15 hr Na-24, may be improved by up to a factor of 10 by irradiating the parent atoms at 10^{15} flux instead of 10^{14} flux.

2. The production of long-lived isotopes such as 7.2 yr Ba-133 and 5.3 yr Co-60 may be speeded up by irradiating at higher fluxes. The time to reach the maximum activity of Ba-133, for example, is decreased from 19.8 yrs at 10^{14} flux to 6.6 yrs at 10^{15} flux. Also, for Co-60 the time for maximum activity is decreased from 7.8 yrs to 1.6 yrs at these two flux levels.

3. The HFIR can also be used to burn out undesired atoms in an isotopic mixture. For example, at 10^{15} flux 99% of Co-59 could be burned out of Co-60 in about five years. Similarly, Eu-152 and Eu-154 could be removed from fission product Eu-155.

4. Many isotopes are prepared by irradiating isotopically-enriched target material, such as Cr-50 to make Cr-51. At 10^{15} fluxes, less target material would be needed; this would reduce the cost of the product.

5. The isotope irradiation facilities in the HFIR can also be used to advantage in producing carrier-free isotopes. For example, Ca-45, now made by an (n,p) reaction with Sc-45 followed by chemical separation, could be made from Ca-44 at 10^{15} flux at considerably lower cost.

3. Nuclear Chemistry Experiments

Thermal neutron fluxes above 10^{15} are of interest in the field of nuclear chemistry primarily as a means of improving the quality of the experiments that can be done. For example, fluxes at this level would permit neutron diffraction experiments to be made more rapidly, more easily, and with greater accuracy, due to increased resolution of the neutron crystal spectrometer. In general, for neutron diffraction work, however, a high ratio of thermal to fast neutrons and thermal neutrons to gamma rays is equally as important as higher fluxes per se. High thermal fluxes are also required by the chemists to measure fission cross sections of isotopes such as Np-237 which have low cross sections, to measure capture cross sections of isotopes such as C-14, deuterium and O-16, to investigate the problem of stored energy in graphite at high temperatures, and to improve the sensitivity of activation analyses.

4. Physics Research

In order to make neutron powder photographs of a quality and resolution comparable to those obtainable with X-rays, a beam of about 10^9 monoenergetic (0.06 ev) neutrons/cm^2(sec) is required. This corresponds to a total thermal neutron flux of about 3×10^{15}.

For cross section measurements using a velocity selector, fluxes above 10^{15} are necessary to extend the range of measurements to the high epithermal region; i.e., 1-10 kev. Above 10 kev, good measurements can be made using a Van de Graaff generator. Since the improvement in resolution of the neutron beam is proportional to about the cube root of the increase in flux, very high fluxes are needed to obtain even reasonably accurate data above 100 ev. It might be pointed out that a reactor designed for cross section work should provide beams of neutrons as nearly monoenergetic as possible. Consideration should also be given to methods of improving the performance of the detecting equipment.

In the field of solid state physics, the main interest is in fast, rather than thermal, fluxes. To determine the effects of radiation on the mechanical properties of reactor structural materials, integrated fast flux doses of about 5×10^{21} are required. At fluxes of 2×10^{15}, this point is reached after one month's exposure; consequently, there is little incentive to provide fluxes of greater magnitude. For most solid state research, small, capsule-type experiments are adequate; however, if fast flux beams of sufficient intensity are available, radiation damage studies might be done outside of the reactor, cutting down on gamma heating and permitting insertion and withdrawal of samples during operation of the reactor.

5. Loop Experiments(3)

The maximum unperturbed flux requirements for loop experiments to test the behavior of various fuel element-coolant systems seem to be in the range of $1\text{-}1.5 \times 10^{15}$ thermal neutrons/cm^2(sec). Adequate burn-ups and power densities can be achieved at this level without encountering the problem of excessive gamma heating. Relatively large diameter holes accessible to this flux must, however, be provided for loop work, and particular attention must be given to the design of the facilities for operating the loop and removing it at the end of the experiment. Each loop facility, moreover, should be designed so that the presence of the loop does not affect the performance of adjacent loops. As will be shown later, these objectives can be achieved by use of a multiple core reactor.

The maximum unperturbed neutron flux requirements for various research and test reactor uses are summarized in Table I.

TABLE I

MAXIMUM FLUX REQUIREMENTS FOR NUCLEAR RESEARCH

Type of Research	Required Unperturbed Flux n/cm^2(sec)	Desired Neutron Energies
Production of transplutonium isotopes	5×10^{15}	Thermal (possibly epithermal)
Production of special isotopes	1×10^{15}	Thermal
Neutron crystal spectrometer work	1×10^{15}	Thermal (monoenergetic)
Activation analyses	1×10^{15}	Thermal
Cross section measurements with velocity selector	$1\text{-}2 \times 10^{15}$	Thermal to 10 kev
Radiation damage in metals, etc.	2×10^{15}	Fast
Loop experiments with fuel-coolant systems	$1\text{-}2 \times 10^{15}$	Thermal and fast

COMPARISON OF REACTOR TYPES FOR ACHIEVING HIGH FLUXES

A consideration of possible reactor systems most suitable for obtaining high fluxes leads one invariably to an undermoderated, enriched core surrounded by a good reflector which has the advantage of thermal flux peaking in the moderating regions and fast flux peaking in the fuel region. Such an approach has been used for all of the ultra high flux research reactors being built. In the case of the HFIR, AETR, and AHFR, the reactor core consists of a light-water-cooled annular fuel region surrounding an internal H_2O moderating region, or "flux trap" as it is called. A beryllium or D_2O reflector around the core not only serves to keep the critical mass of the reactor reasonably low, but provides a larger volume for experimentation at fluxes only slightly lower than those in the flux trap. The Brookhaven HFBR, though based on the same principle, differs from these reactors in that the core consists of a closely packed lattice of plate-type fuel elements cooled, moderated, and reflected by D_2O. In this reactor the fast and epithermal flux peaks in the undermoderated core and the thermal flux in the reflector, providing an ideal facility for cross section measurements and other beam experiments which require varying neutron energies.

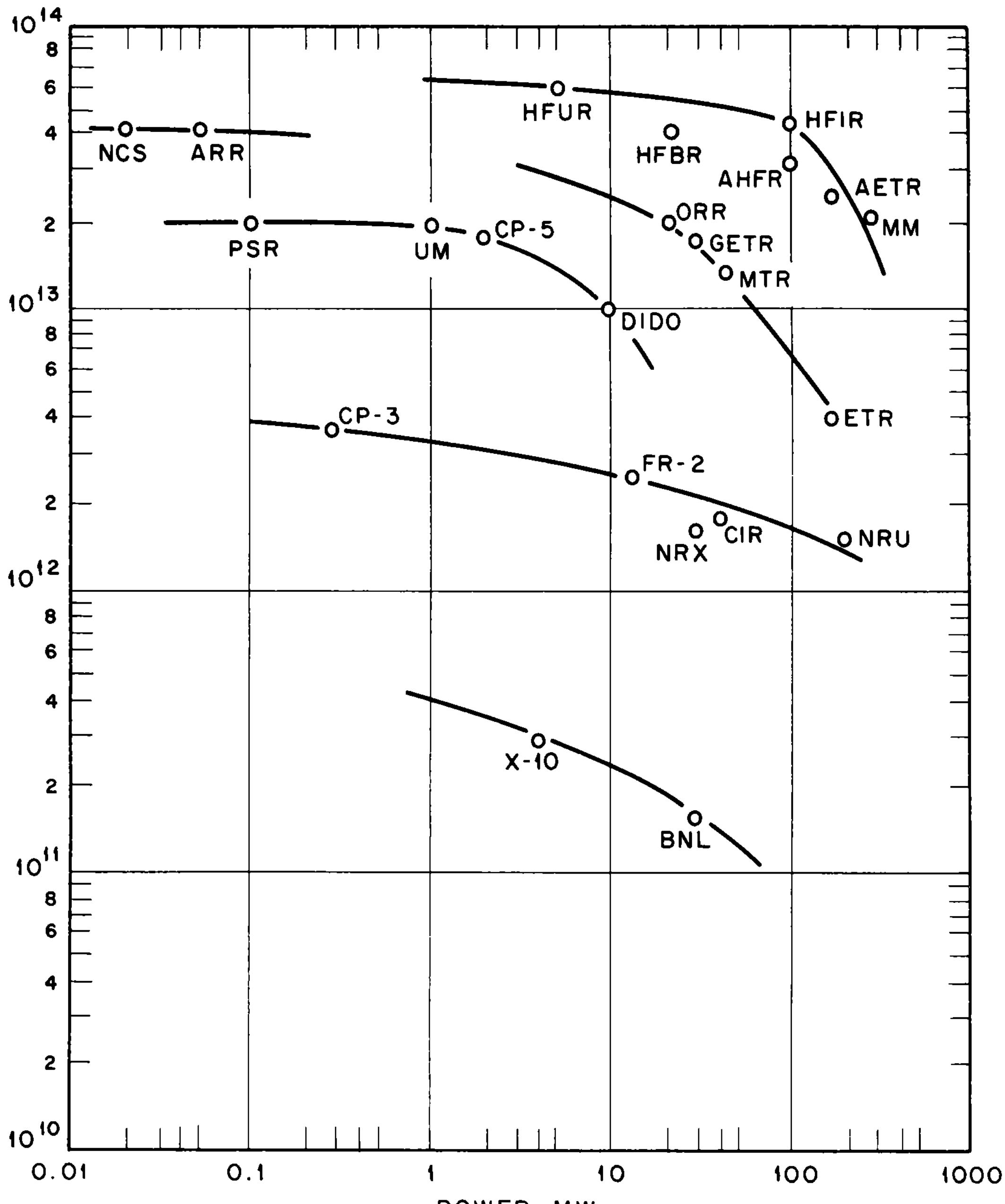

Fig. 2—Maximum thermal neutron flux per unit power for various reactor types.

The thermal flux characteristics and costs of these reactors are compared with those of existing and proposed research reactors in Figures 2 and 3. It is seen in these figures that the reactor concepts involving separate fuel and moderating regions have the highest thermal flux per unit of power and, consequently, the maximum thermal flux for a given cost. (Data for each of the reactors shown in Figures 2 and 3 are summarized in Table II.)

TABLE II

CHARACTERISTICS OF EXISTING AND PROPOSED RESEARCH REACTORS

Reactor	Designation	Power (Mw)	Cost (millions of $)	Maximum Unperturbed Thermal Flux ($n/cm^2 sec$)
ORNL - graphite	X-10	3.8	5.2	1.1×10^{12}
Brookhaven - graphite	BNL	28	19.7	4×10^{12}
Chicago Pile No. 3	CP-3	0.3	2	1×10^{12}
Karlsruhe Research Reactor	FR-2	12	5.2	3×10^{13}
Canadian - nat. U-D_2O	NRX	30	10	5×10^{13}
Canada-India - nat. U-D_2O	CIR	40	15	7×10^{13}
Canadian - nat. U-D_2O	NRU	200	57	3×10^{14}
North Carolina State	NCS	0.02	0.5	8×10^{11}
Armour - water boiler	ARR	0.05	0.7	2×10^{12}
Argonne - heavy water	CP-5	2	3.0	3.5×10^{13}
Harwell Dido	DIDO	10	5.7	1×10^{14}
Materials Testing Reactor	MTR	40	18	5×10^{14}
Engineering Test Reactor	ETR	175	17.2	7×10^{14}
Advanced Engineering Test Reactor*	AETR	980 (7 cores)	32	3.3×10^{15}
Mighty Mouse*	MM	250	60	5.5×10^{15}
Pennsylvania State Reactor	PSR	0.1	0.31	2×10^{12}
University of Michigan	UM	1	0.85	2×10^{13}
High Flux University Reactor*	HFUR	5	2.6	3×10^{14}
Oak Ridge Research Reactor	ORR	20	4.7	4×10^{14}
General Electric Test Reactor	GETR	30	4	5×10^{14}
High Flux Isotope Reactor*	HFIR	100	12	4.7×10^{15}
Argonne High Flux Reactor*	AHFR	100	13.7	$3.5\text{-}4 \times 10^{15}$
Brookhaven High Flux Beam Reactor*	HFBR	20-40	10	$0.8\text{-}1.6 \times 10^{15}$

*Design data only.

DEVELOPMENTS IN FUEL TECHNOLOGY

Interest in high flux reactors has led to a consideration of ways in which the power density in the reactor core, and thereby the flux, may

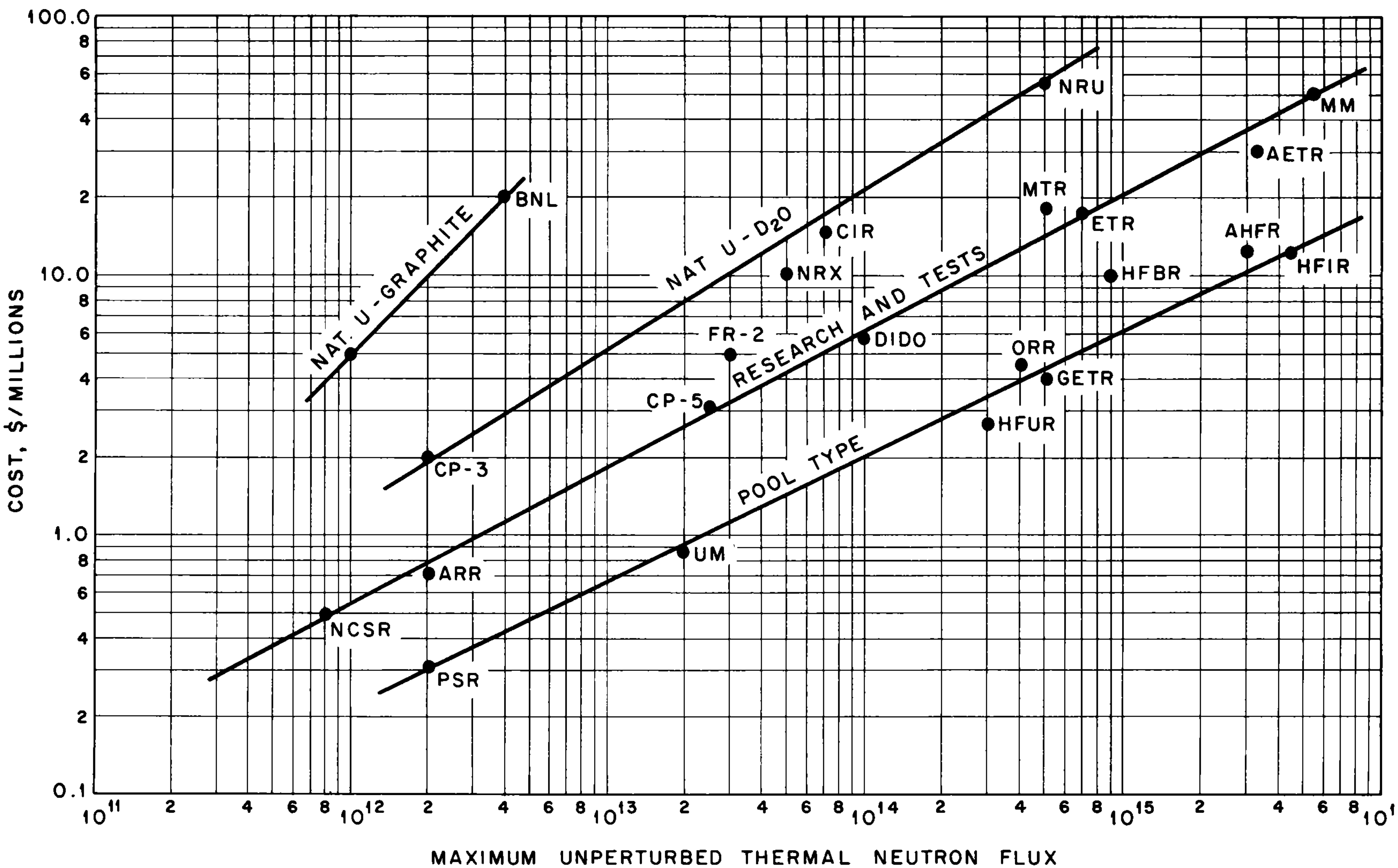

Fig. 3—Cost of various reactor types as a function of maximum thermal neutron flux.

be increased. A unique feature of the high flux reactor designs is that most of the moderation of fast neutrons takes place in the internal and external reflectors. The thermal flux peaking in these regions depends on the number of fast neutrons leaking from the reactor core, which in turn depends on the power density in the core. Thus, maximum neutron fluxes at all energy levels are proportional to power densities.

The approach to high power density fuel elements has taken two directions; first, toward an extension of the performance of MTR-ETR plate-type fuel elements by going to thinner, more closely spaced plates, and operating at higher water velocities and temperatures; and second, the development of more novel fuel elements utilizing woven screens or tubes or cylindrical channels containing twisted tapes.

In the case of the plate-type elements, the achievable average power density is about 2.5 megawatts per liter. To go above this point requires excessive coolant velocities or plates too thin for mechanical stability. It is believed that this point is reached at a plate thickness of about 50 mils. These same limitations apply to curved plate elements such as shown in Figure 4 though permissible thicknesses may be as low as 40 mils. The AETR fuel element plates shown in Figure 4 have an involute curvature to permit complete filling of the cylindrical annulus without changing the width of the coolant passages.

Fuel elements made up of cylindrical channels or tubes containing twisted tapes are of particular interest as they show promise of higher

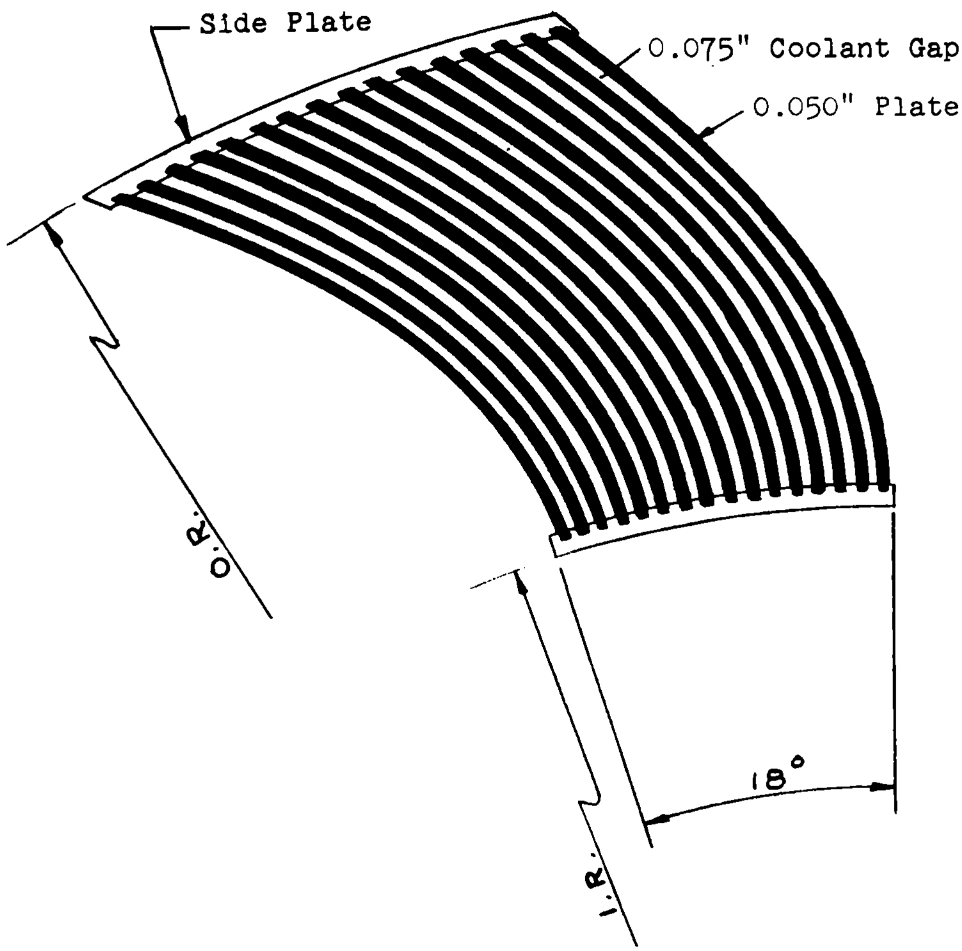

Fig. 4—Involute plate type fuel assembly (AETR design).

burn-out heat fluxes and greater mechanical stability at high flow velocities than plate-type elements. Such tubular fuel elements have the additional advantage that they can operate at surface temperatures above the boiling point without encountering unstable film boiling conditions. Thus, system discharge pressures can be held close to atmospheric pressure resulting in a lower cost external heat removal system. A typical tubular fuel element is shown in Figure 5.

One disadvantage of twisted tape fuel elements is that heat is removed from the inside of the tube only, resulting in a decrease in heat transfer area. Although this loss can more than be made up for by increasing the flow velocity and thereby the burn-out heat flux, the coolant pumping power requirements are also increased significantly. Thus, the relative merits of plates and tubes depend on the balance of lower construction costs on the one hand vs. higher operating costs on the other.

A considerable experimental program has been carried out at the Oak Ridge National Laboratory to measure the effect of variables such as velocity, pressure, temperature, tube diameter, tube length, and tape twist ratio on burn-out heat fluxes in tubes containing twisted tapes. The minimum burn-out results to date can be fairly well represented by the equation[4],

$$(q/A)_b = 4.03 \times 10^5\ V_{rw}/(L/D)^{1/3}$$

where, $(q/A)_b$ = burn-out heat flux, Btu/hr(sq ft)

V_{rw} = actual flow velocity at the tube wall, ft/sec

L/D = ratio of length of tube to diameter.

It is noted in this equation that pressure and temperature have no effect on burn-out heat fluxes which differs from the case of plate-type elements. The term V_{rw} is the velocity of the liquid making a spiral path along the tube. At a tape twist ratio of 2.5 diameters per twist, the value of $V_{rw} = 1.18\ V$, where V is the axial velocity, ft/sec. Measurements of burn-out heat fluxes in both tubes and plates are continuing at ORNL in order to provide a firmer basis for comparing the merits of these two approaches.

THE BROOKHAVEN HIGH FLUX BEAM REACTOR (HFBR)[5]

The first new high flux research reactor to be built in the United States is the HFBR which is scheduled for operation in late 1962. Because of the short time scale for construction, the reactor will use conventional ETR-type flat plate fuel elements (50 mils thick - 110 mil gap); however, they will be reduced in length from 36 in. to 17 in. The unique feature of the HFBR is that the 74 liter core, shown in Figure 6, will be a compact lattice of these fuel elements but cooled, moderated, and reflected with D_2O, rather than H_2O. This results in a very high leakage rate of fast and epithermal neutrons from the core and a factor of ten peaking of thermal neutrons in the reflector compared to the center of the core. This is shown in Figure 7, giving the radial neutron distribution in the HFBR. Horizontal thermal flux beam holes will terminate at the point of thermal flux peaking (about 15-20 cm from the face of the core) and epithermal flux beam holes at a point about 5 cm from the face. A vertical thimble will penetrate to the core center for fast flux irradiations. For each megawatt of power, these facilities deliver 3-4 x 10^{13} thermal neutrons/cm^2(sec) (low to 0.24 ev); 2.5-3.5 x 10^{13} epithermal neutrons/cm^2(sec) (0.24 ev up); and 2 x 10^{13} fast neutrons/cm^2(sec) (fission spectrum). By

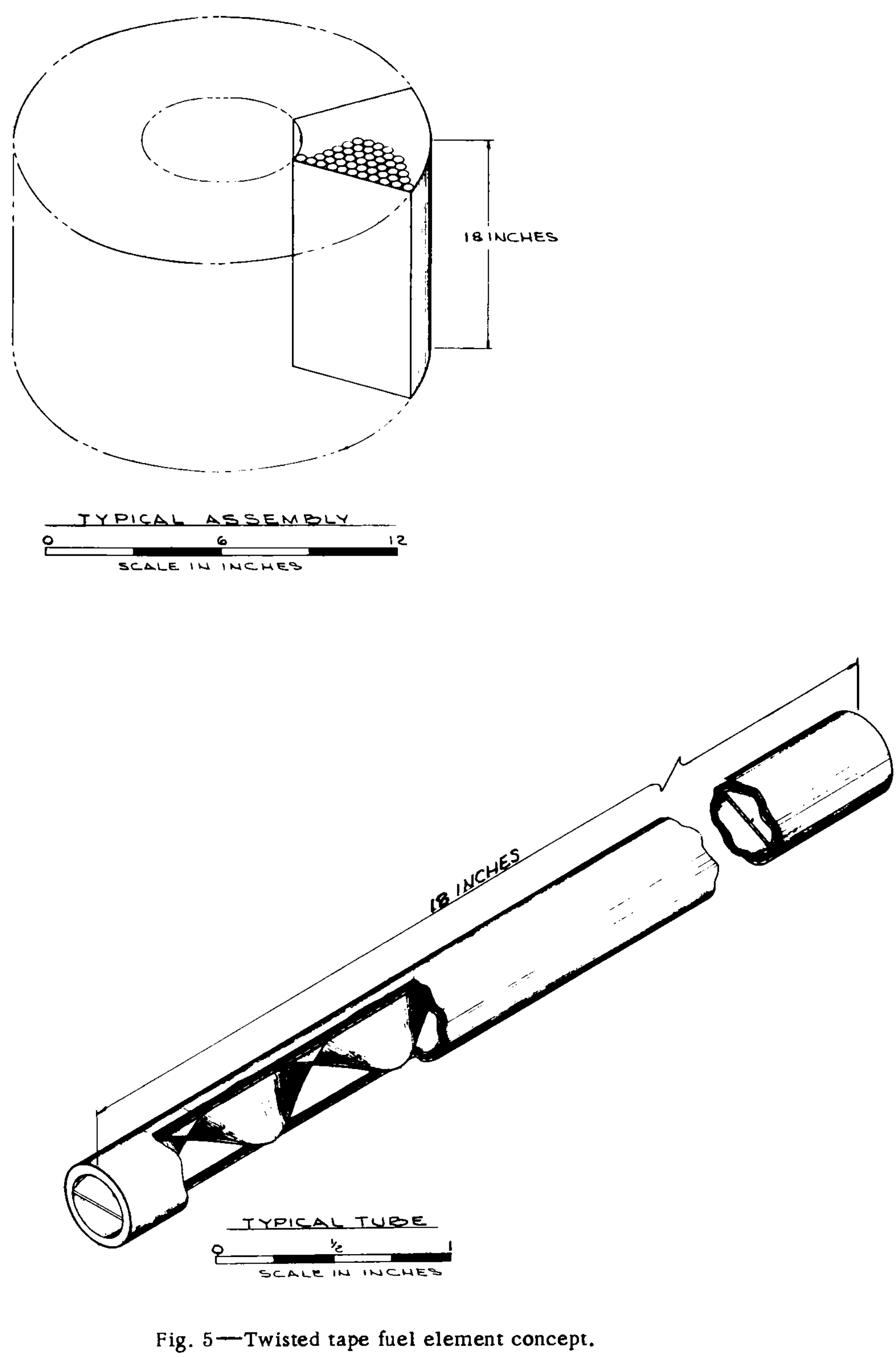

Fig. 5—Twisted tape fuel element concept.

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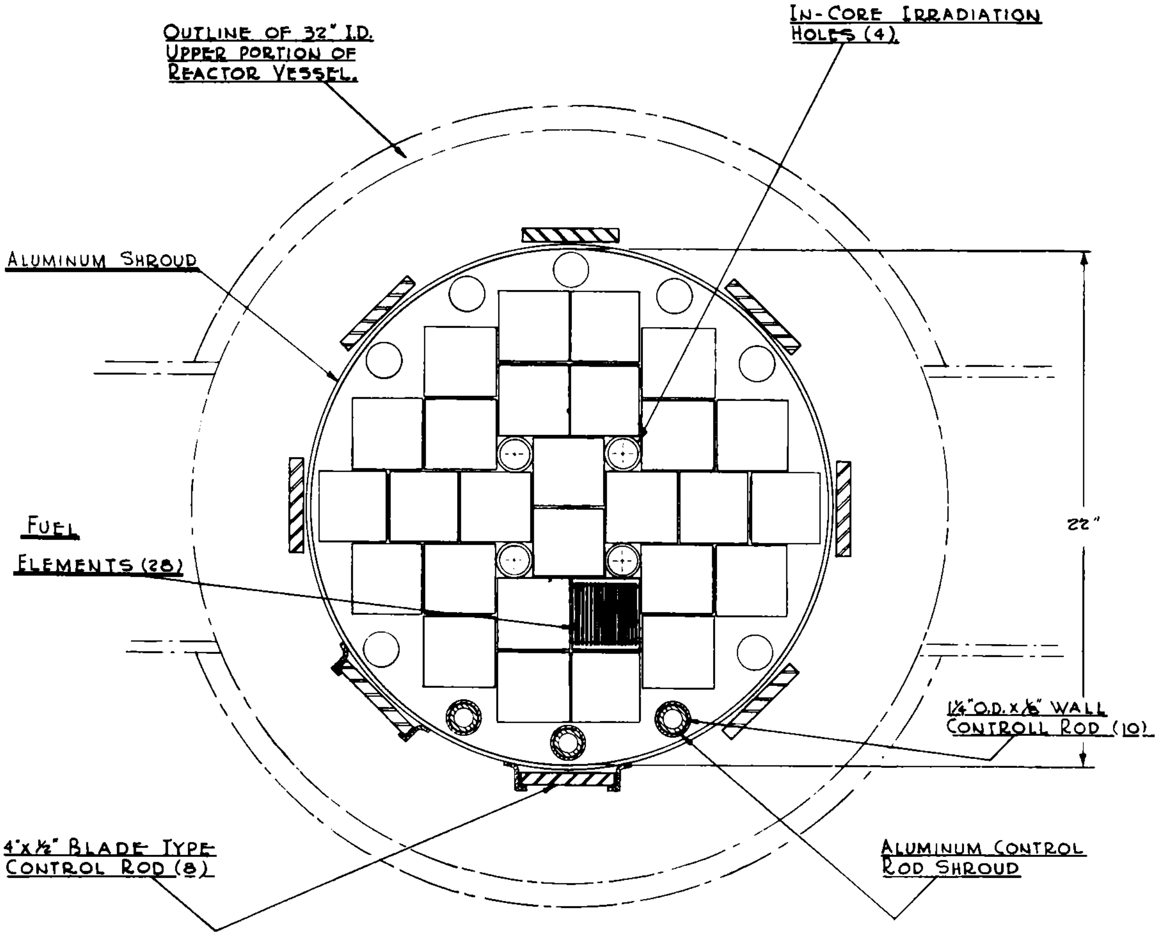

Fig. 6—High Flux Beam Reactor, core layout plan.

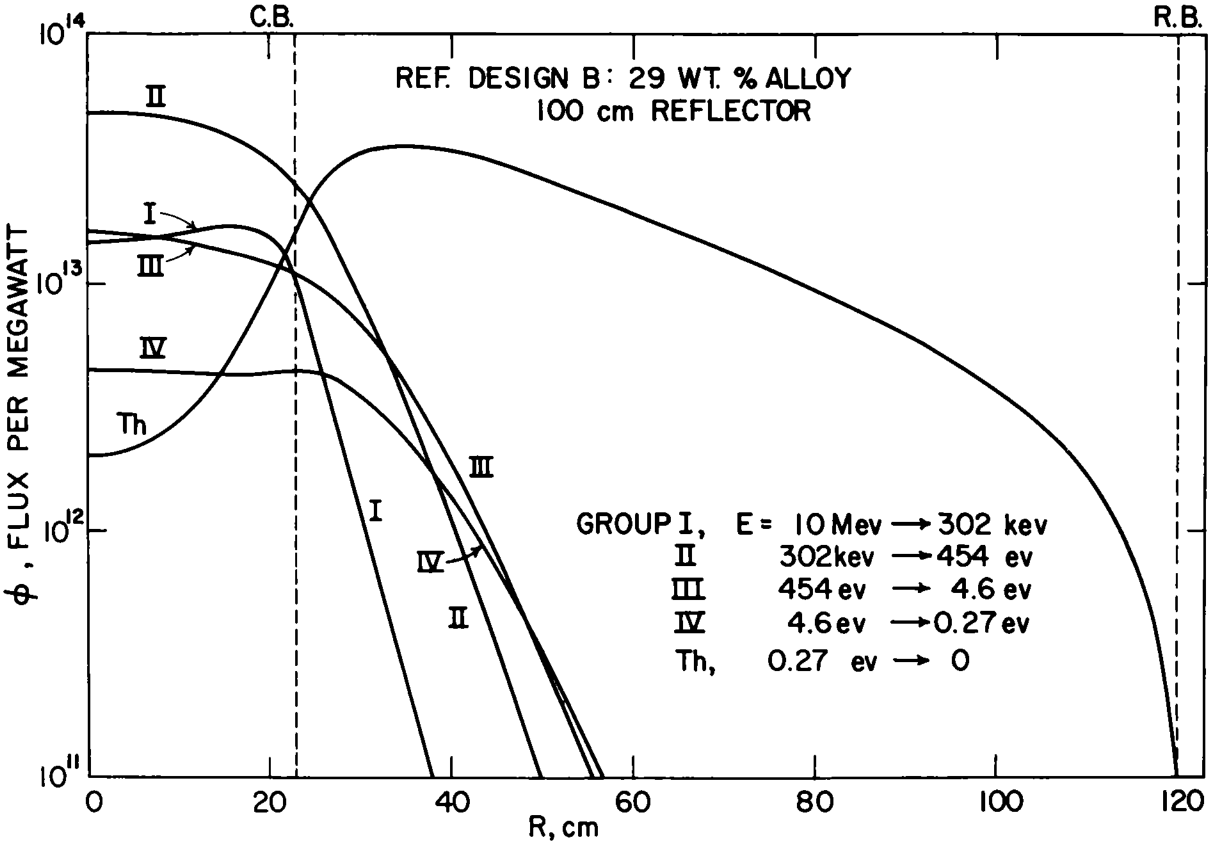

Fig. 7—Typical neutron flux distribution in the High Flux Beam Reactor (HFBR).

operating the reactor at a pressure of 200 psi inside the 8-ft-diameter spherical pressure vessel shown in Figure 8, maximum power densities up to 1.5 Mw/liter are possible, resulting in a total power level in the range of 20-40 Mw. The exact power level has not yet been established; however, it will be set as high as possible within the budgetary limitation of $10 million for the reactor and its associated facilities. Critical experiments now underway at Brookhaven have checked the estimated critical loading of 6 kg. The building arrangement for the HFBR is shown in Figure 9. Table III summarizes preliminary design data for the reactor.

THE ORNL HIGH FLUX ISOTOPE REACTOR

The design of a second high flux reactor for the production of heavy elements and other isotopes is presently underway at the Oak Ridge National Laboratory, with completion of construction scheduled for October 1963. This reactor will operate at power levels up to 100 Mw and produce thermal neutron fluxes up to 5×10^{15} neutrons/cm^2(sec). Details of the design of the reactor have not yet been firmly established; however, one conceptual design has been carried out based on curved plate fuel elements to establish preliminary cost information and performance criteria[6]. As mentioned previously, the design concept of the HFIR is based on the use of an internal flux trap to achieve maximum thermal fluxes. The proposed arrangement, however, also achieves very high fast fluxes in the core region as shown in Figure 10. The reactor core and pressure vessel as conceived in this design are shown in Figures 11 and 12. The over-all features of the reactor will be similar to those of the ORR based on the use of a pool and canal for fuel loading operations, handling of isotopes, and storage of radioactive materials. This is shown in Figures 13 and 14, illustrating the reactor and hot cell buildings. More complete containment than that shown in Figure 14 may be necessary to assure against

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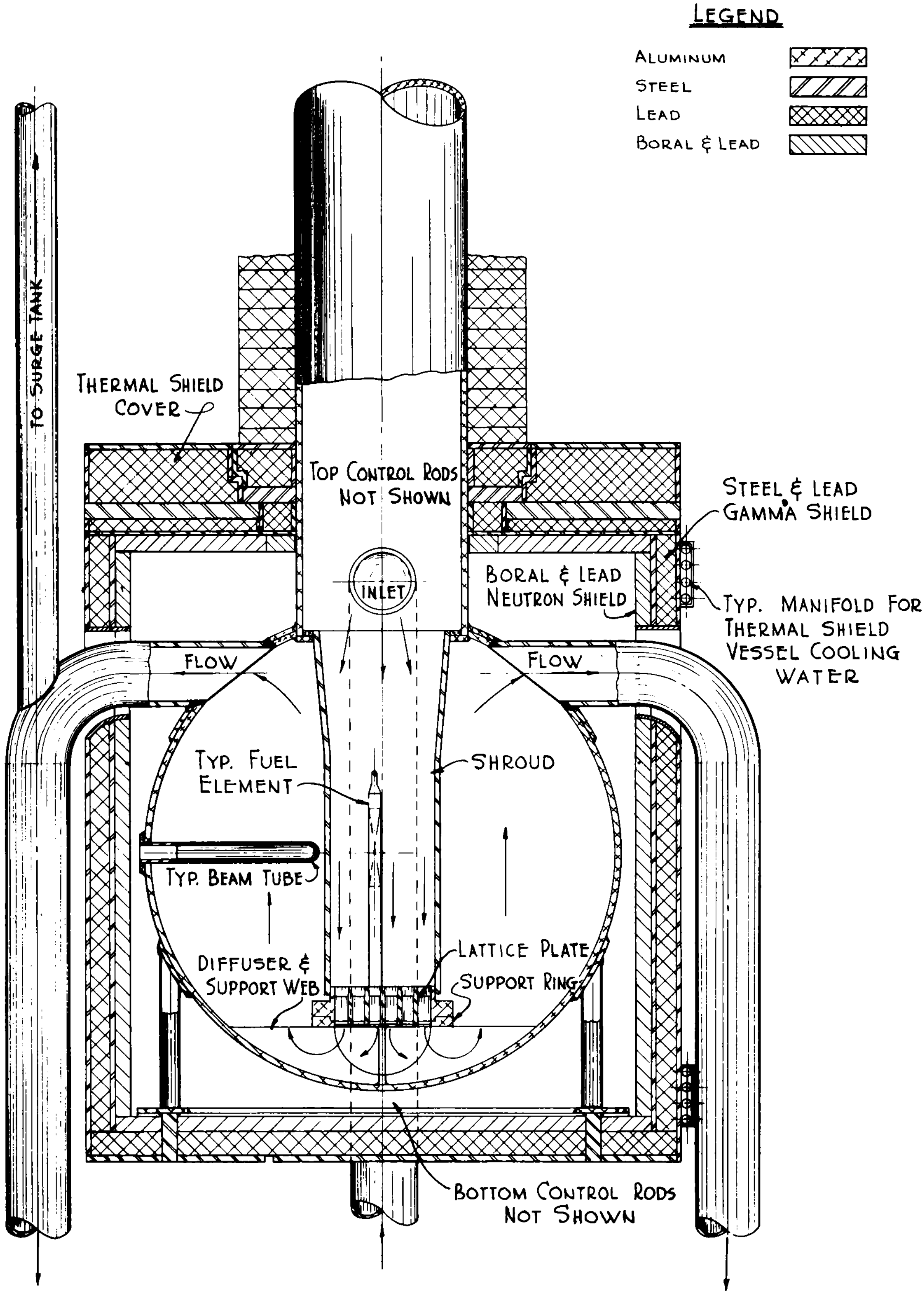

Fig. 8—High Flux Beam Reactor, core tank and thermal shield section.

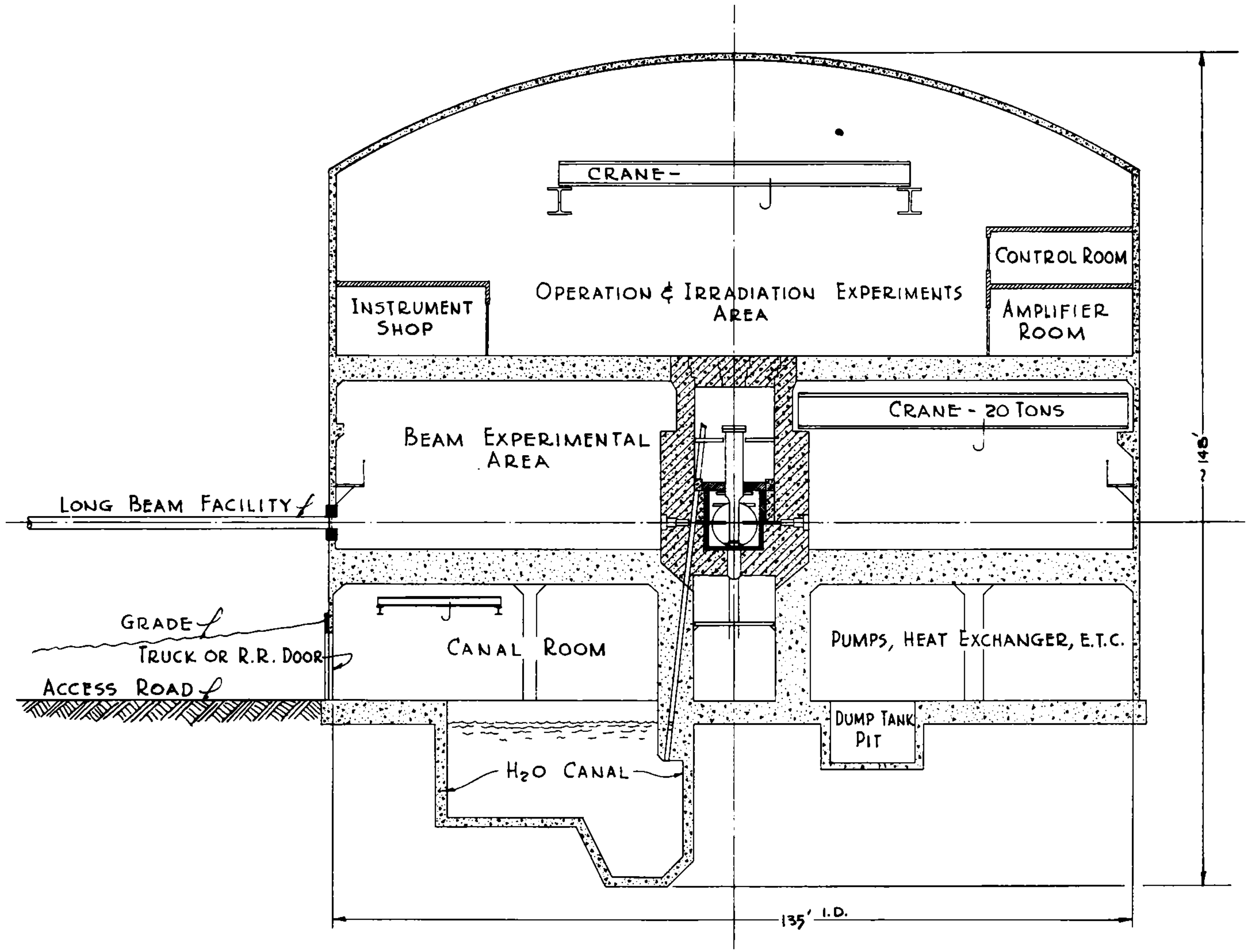

Fig. 9—High Flux Beam Reactor, reactor building elevation.

TABLE III

PRELIMINARY DESIGN DATA FOR THE BROOKHAVEN HIGH FLUX BEAM REACTOR

Characteristic	Design
Reactor power, Mw	20-40
Maximum fast flux in fuel region, neutrons/cm^2(sec)	$3.4\text{-}6.8 \times 10^{14}$
Maximum unperturbed thermal flux in reflector, neutrons/cm^2(sec)	$0.7\text{-}1.4 \times 10^{15}$
Average fuel cycle time, days	50-100
Specific power, Mw per kg of U-235	3.3-6.6
Maximum power density, Mw/liter	1.5
Average power density, Mw/liter	0.27-0.54
Maximum heat flux, Btu/hr(sq ft)	1.25×10^6
Maximum surface temperature, $^\circ$F	210-334
Coolant inlet temperature, $^\circ$F	110-120
Coolant outlet temperature, $^\circ$F	127-134
Coolant velocity, fps	35
Coolant circulation rate, gpm	17,500
Pressure drop across core, psi	40
Pressure at core exit, psi	50-150
Fuel plate thickness, in.	0.050
Coolant channel thickness, in.	0.105
Length of active core, in.	17
Diameter of fuel region, in.	18.4
Fuel loading, kg of U-235 (90% enriched)	6.03

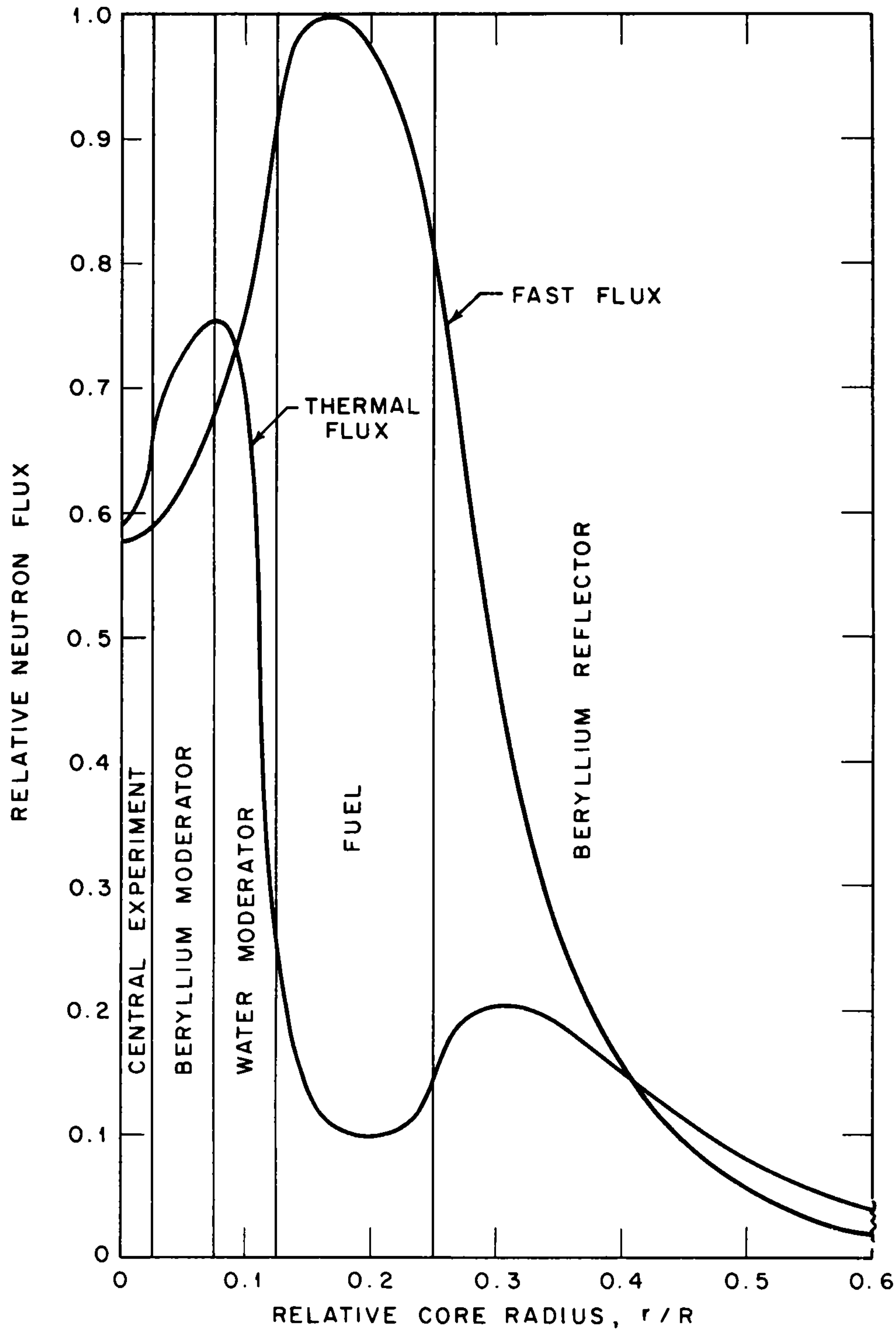

Fig. 10—Schematic representation of HFIR core and typical spatial neutron flux distribution.

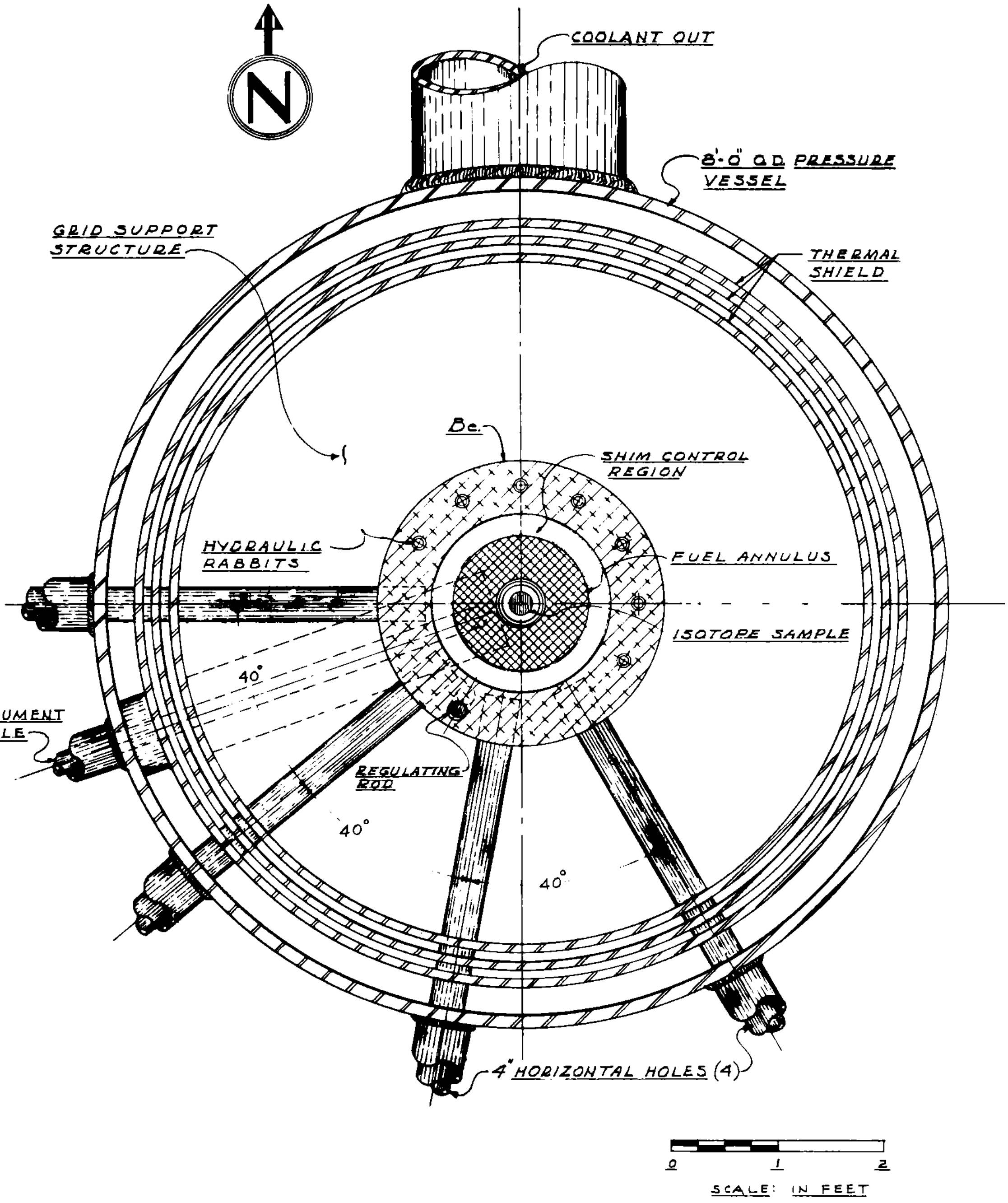

Fig. 11—HFIR vessel and core, horizontal section.

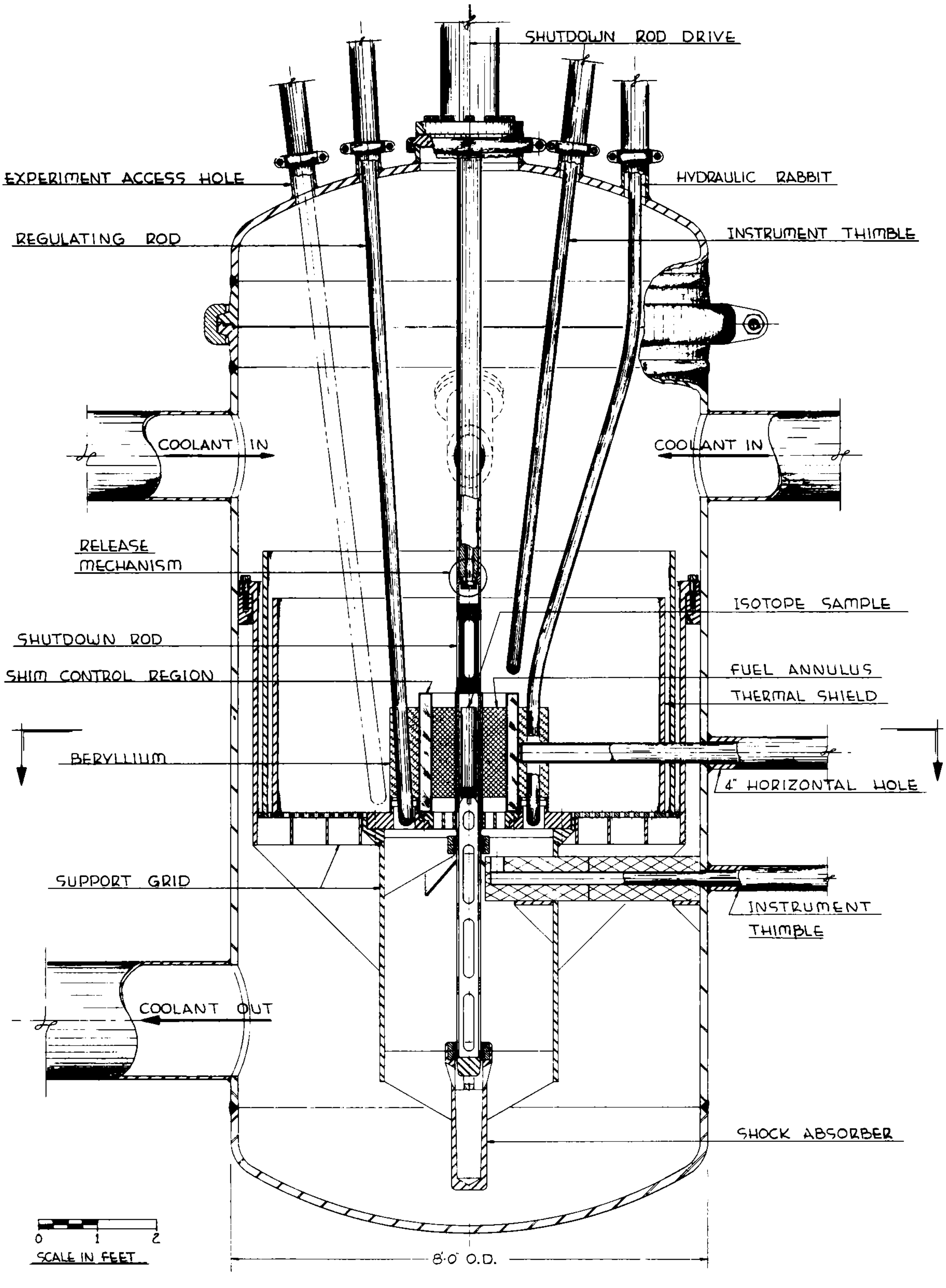

Fig. 12—HFIR vessel and core, vertical section.

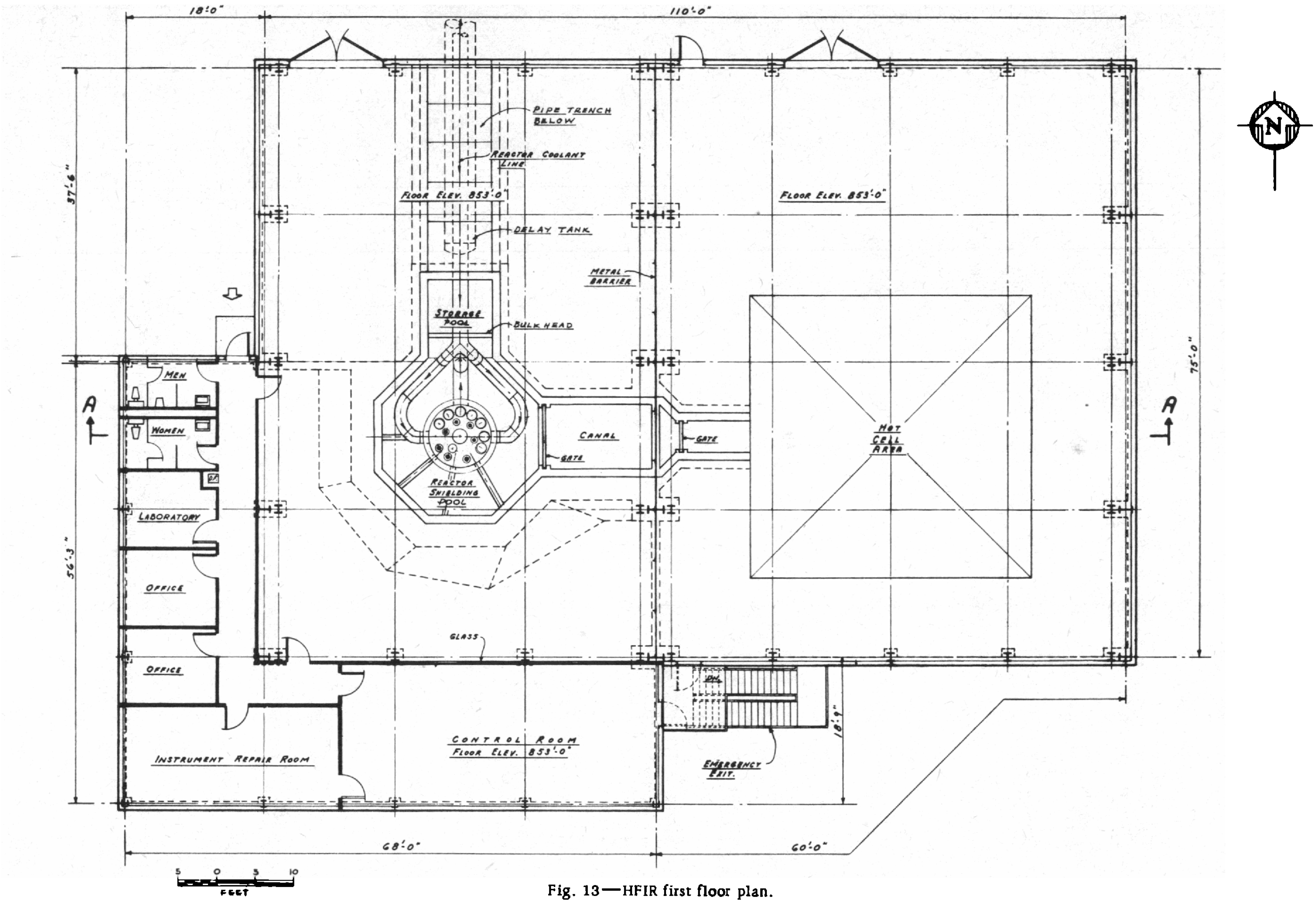

Fig. 13—HFIR first floor plan.

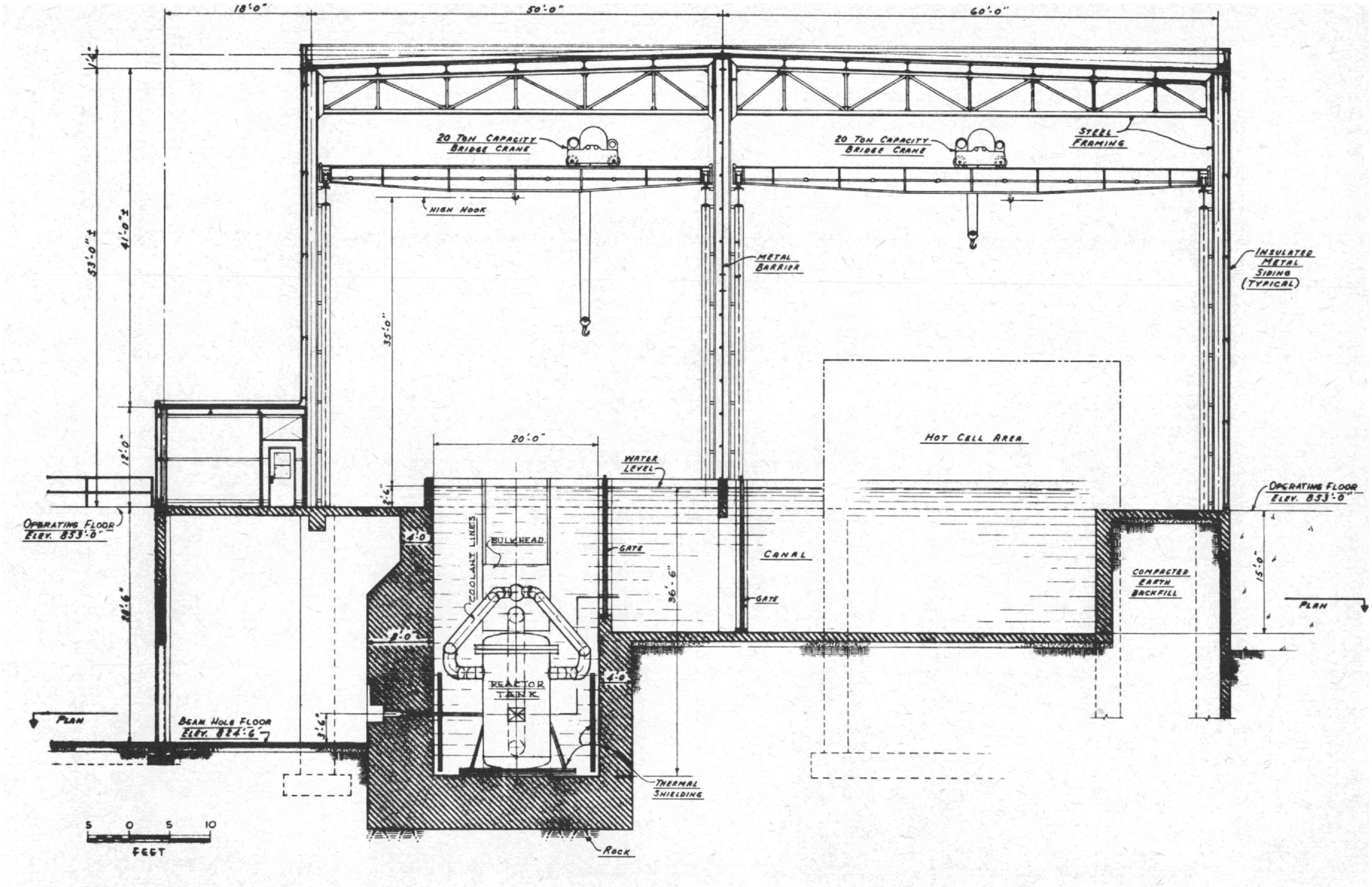

Fig. 14—HFIR elevation section.

possible accidents. This has not been evaluated at the present time; however, the proposed arrangement based on a controlled ventilation system such as that used for the ORR building can easily be modified to provide complete containment at a minimum added cost. No direct access is provided from the reactor room to the adjacent control room, office wing, experimental areas, or hot cell and isotope handling building. A water barrier is used in the canal entrance to the hot cell building to prevent the spread of contamination due to air circulation between the buildings. Pertinent characteristics of the HFIR are summarized in Table IV.

THE ARGONNE HIGH FLUX REACTOR (AHFR)

A high flux reactor for physical science research will be built at the Argonne National Laboratory. Present plans are to initiate construction in fiscal year 1962, with completion scheduled for 1964. The proposed arrangement of the reactor core is illustrated in Figures 15, 16, and 17 taken from Reference 7. As seen in these figures, the design employs a beryllium reflected annular fuel region surrounding a central moderating region as in the case of the HFIR. Since the primary objective of the AHFR is to provide the maximum number of beam holes and irradiation ports possible, rather than the maximum flux in one facility as in the HFIR, the core is larger (89 liters compared to 45 liters) and the flux in the central facility is somewhat less. The larger core does, however, permit the use of conventional ETR-type fuel elements operating at power densities only slightly higher than in existing reactors. The preliminary design data for the AHFR are summarized in Table V.

The major experimental facilities for the AHFR, as shown in Figure 15, consist of one central thimble 1.5 in. in diameter, twelve horizontal beam holes, including two 6-in.-diameter through holes, sixteen vertical irradiation holes or thimbles, and two hydraulic rabbits.

THE ADVANCED ENGINEERING TEST REACTOR

An advanced high flux test reactor to meet an AEC requirement for unperturbed thermal neutron fluxes of 1.5×10^{15} neutrons/cm^2(sec) and greater in water, gas, and liquid metal cooled loops has been under consideration since 1957. Optimization studies and a preliminary conceptual design have been carried out by the Internuclear Company[8], the results of which are summarized as follows:

The AETR specifications call for seven 30-in.-long test sections of 3 to 7 in. in diameter for loop irradiation experiments. A study of a single vs. a multiple core reactor led to the choice of seven separate reactor cores of the flux trap design located within a common shield, as shown in Figure 18. Four of these have 7-in.-diameter central vertical test holes (Type A core), and the other three have 4-in-diameter holes. Except for different radial dimensions, Type B core is the same as Type A core shown in Figure 19. As seen in this figure, each core has a light water annulus surrounding the central test hole which is inside an aluminum pressure tube. Six 3/4-in.-diameter capsule test tubes are located in this water annulus for small sample irradiations. The light-water-cooled annular fuel region is surrounded by a D_2O reflector, as shown in Figure 20. This figure also shows the over-all arrangement of reactor and test facility and location of cooling lines. As in the case of the HFIR, use is made of a pool and canal for handling and storage of radioactive materials.

The fuel elements proposed for the AETR consist of plates with an involute curvature to permit complete filling of the annular fuel region. In order to obtain a flat radial power distribution, each fuel element is loaded so that there is more fuel at the center of the plate than at the

(Text continues on p. 56.)

TABLE IV

PRELIMINARY DESIGN CHARACTERISTICS OF THE ORNL HIGH FLUX ISOTOPE REACTOR[a]

Characteristic	Design
Reactor power, Mw	100
Average thermal flux in 100 gram sample, neutrons/cm^2(sec)	3.0×10^{15}
Maximum unperturbed thermal flux in island	4.7×10^{15}
Maximum fast flux in fuel region	4.0×10^{15}
Maximum unperturbed thermal flux in outer reflector	1.2×10^{15}
Fuel cycle time, days	10
Specific power, Mw per kg of U-235	16.7
Maximum power density, Mw/liter	4.1
Average power density, Mw/liter	2.2
Maximum heat flux, Btu/hr(sq ft)	1.49×10^6
Burn-out heat flux, Btu/hr(sq ft)	4.0×10^6
Maximum surface temperature, °F	397
Coolant inlet temperature, °F	120
Coolant outlet temperature, °F	186
Coolant velocity, fps	40
Coolant circulation rate (fuel region), gpm	10,500
Pressure drop across core, psi	53
System pressure at pump discharge, psi	500
Length of active core, in.	18
Inside diameter of fuel annulus, in.	5.5
Outside diameter of fuel annulus, in.	15
Outside diameter of outer reflector, in.	41
Fuel loading (for aluminum fuel elements), kg of U-235 (as ~ 90% enriched)	~ 6

a. Based on plate type elements

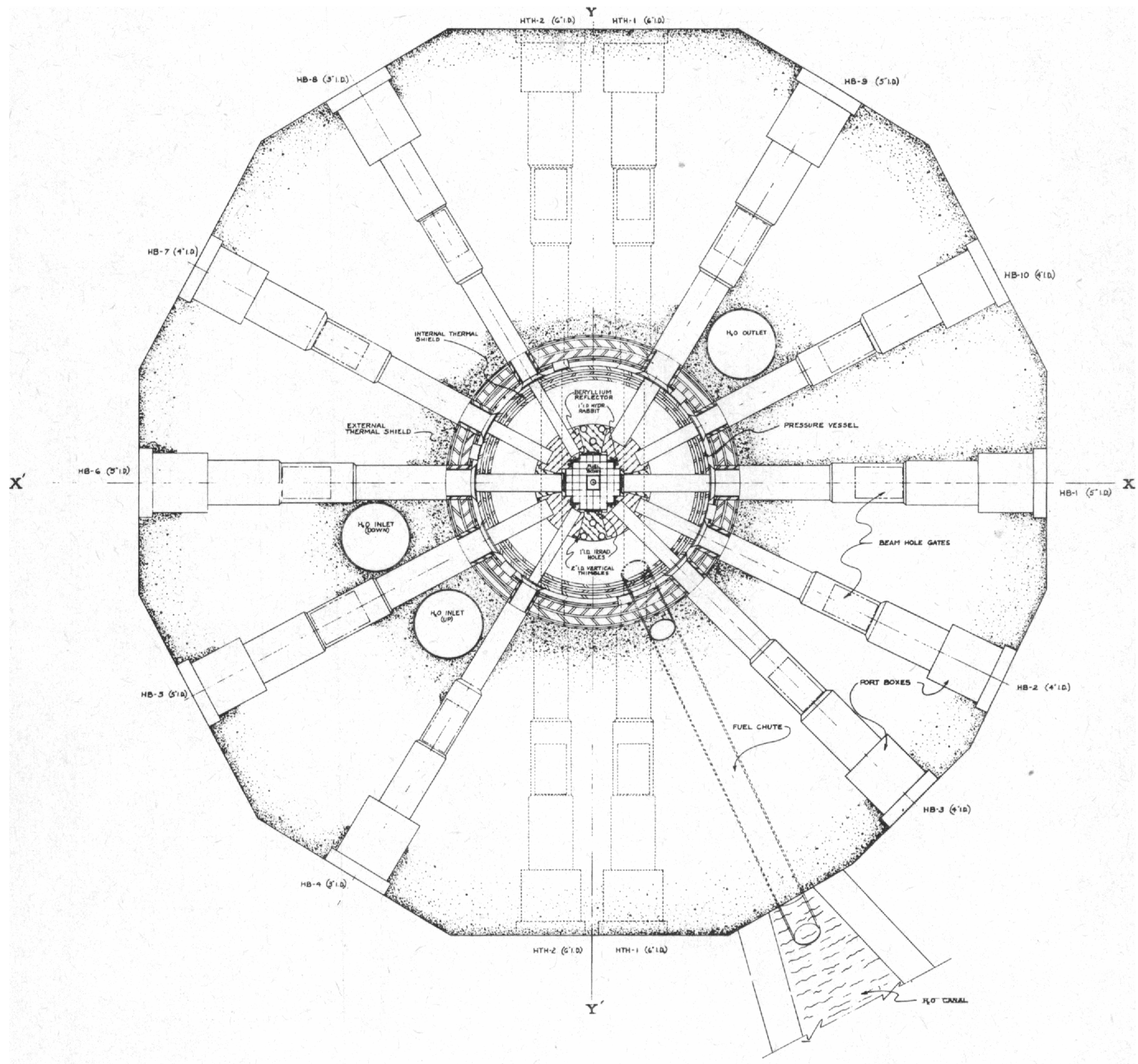

Fig. 15—AHFR section at horizontal centerline.

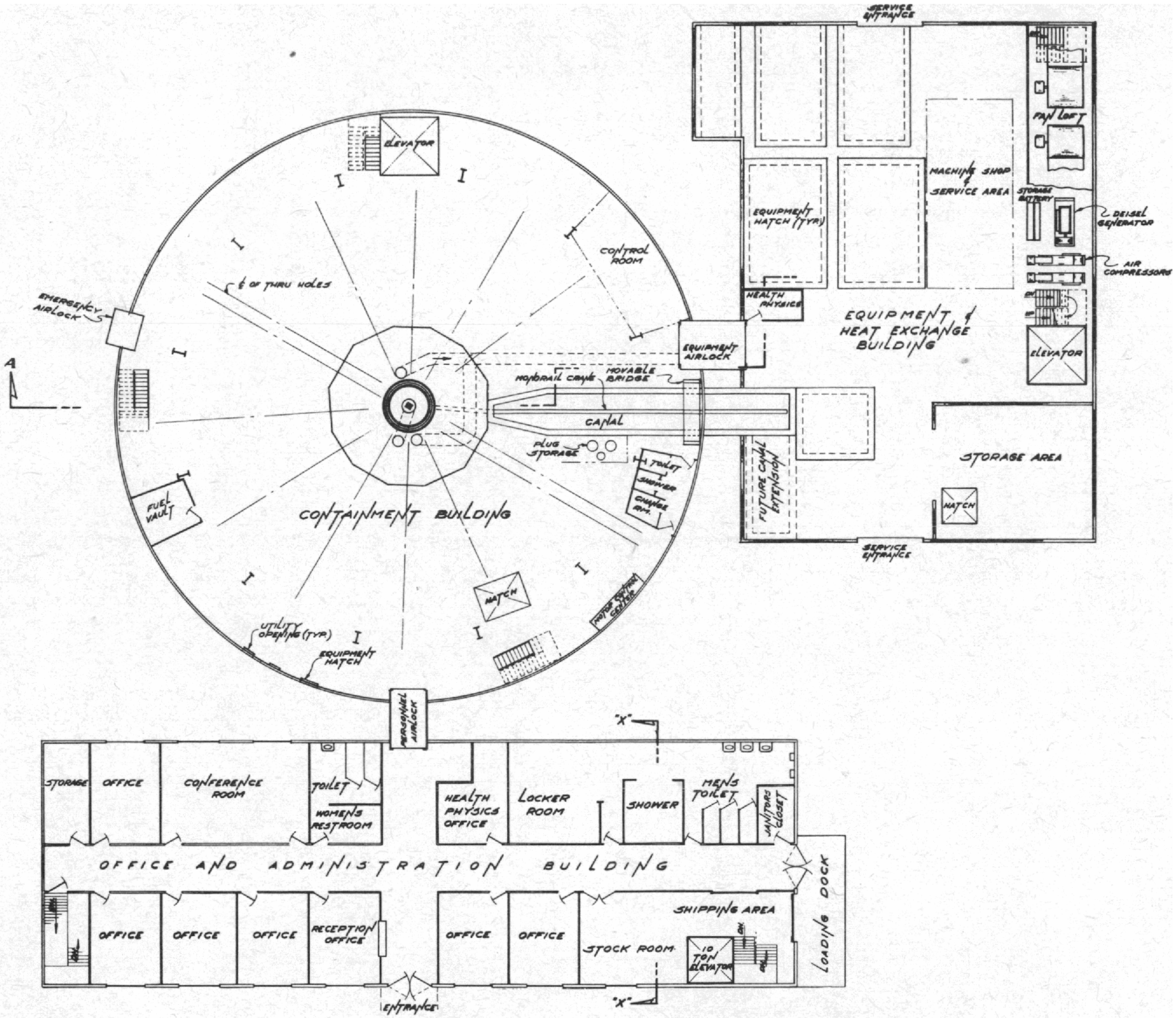

Fig. 16—AHFR main floor plan.

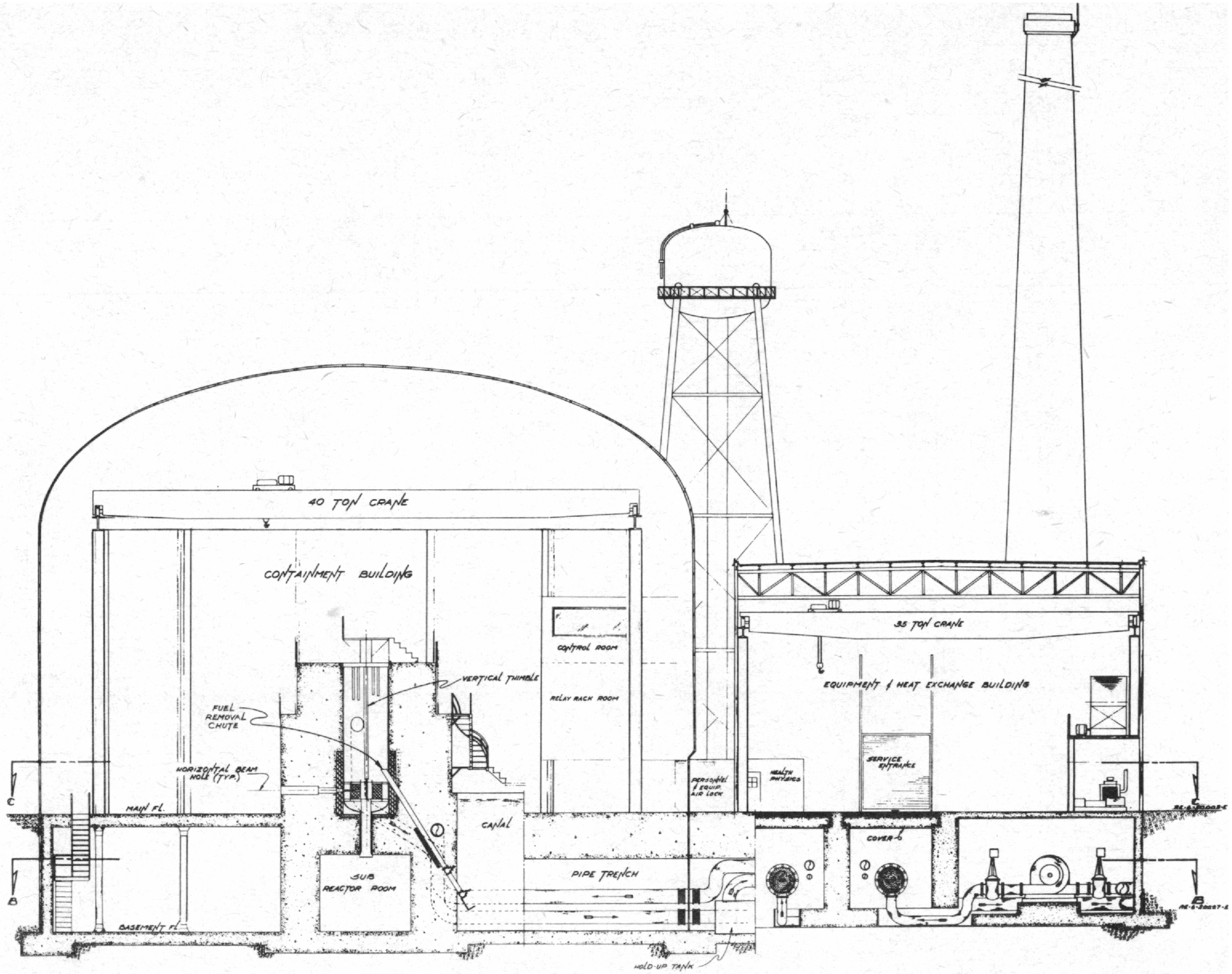

Fig. 17—AHFR building elevation.

TABLE V

PRELIMINARY DESIGN DATA FOR THE ARGONNE HIGH FLUX REACTOR

Characteristic	Design
Reactor power, Mw	100
Maximum unperturbed thermal flux in island, neutrons/cm^2(sec)	$3.5\text{-}4.5 \times 10^{15}$
Maximum fast flux in fuel region, neutrons/cm^2(sec)	3.0×10^{15}
Maximum unperturbed thermal flux in reflector, neutrons/cm^2(sec)	1.6×10^{15}
Average fuel cycle time, days	10
Specific power, Mw per kg of U-235	23
Maximum power density, Mw/liter	1.9
Average power density, Mw/liter	1.13
Maximum heat flux, Btu/hr(sq ft)	0.98×10^6
Maximum surface temperature, °F	315
Coolant inlet temperature, °F	135
Coolant outlet temperature, °F	188
Coolant velocity, fps	35
Coolant circulation rate, gpm	15,000
Pressure drop across core, psi	37
Pressure at core exit, psi	200
Fuel plate thickness, in.	0.050
Coolant channel thickness, in.	0.057
Length of active core, in.	20
Inside dimensions of fuel annulus, in.	4.75 x 4.75
Outside diameter of fuel annulus, in.	19.4
Fuel loading, kg of U-235 (90% enriched)	4.3

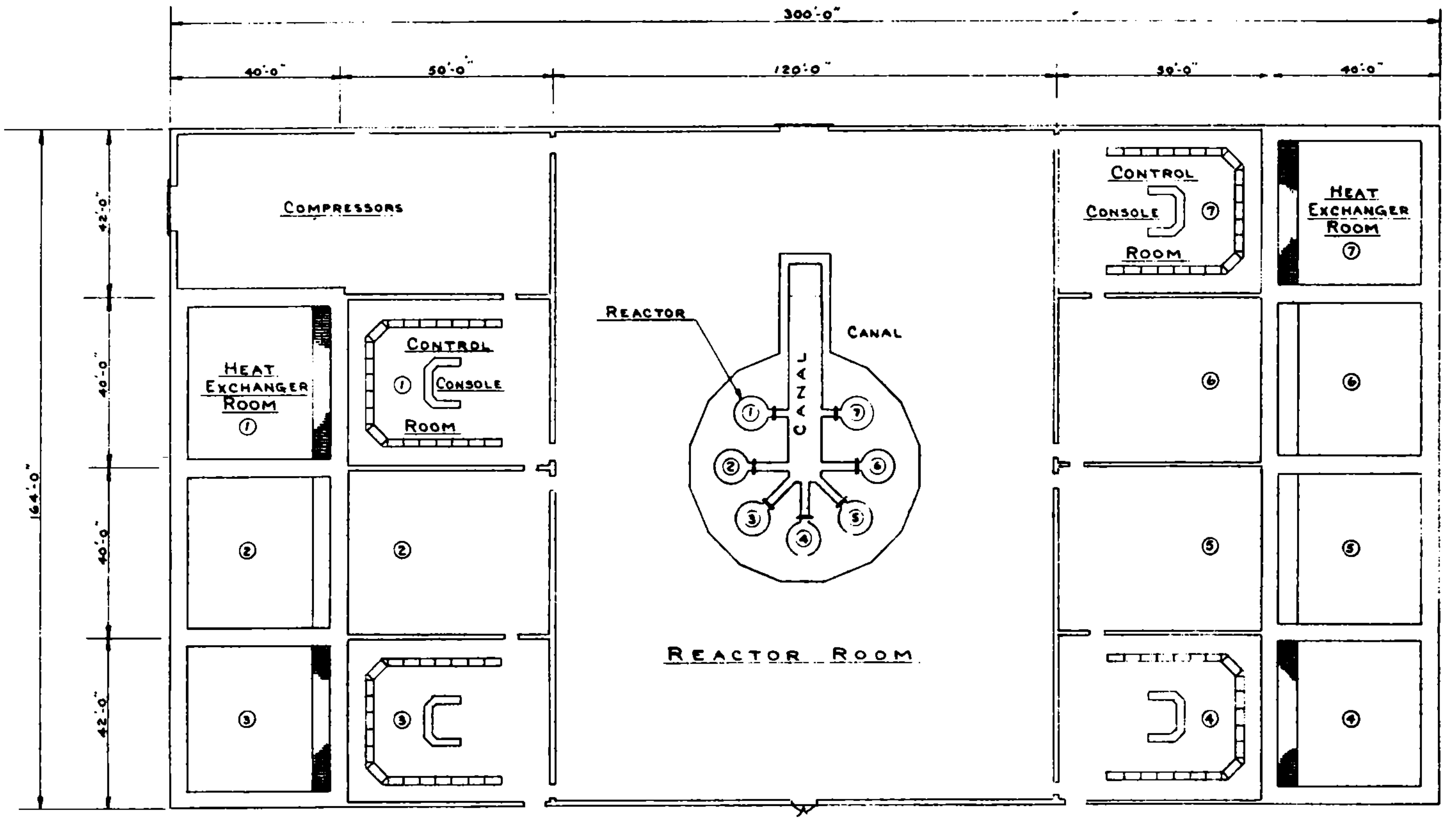

Fig. 18—Advanced Engineering Test Reactor building plan.

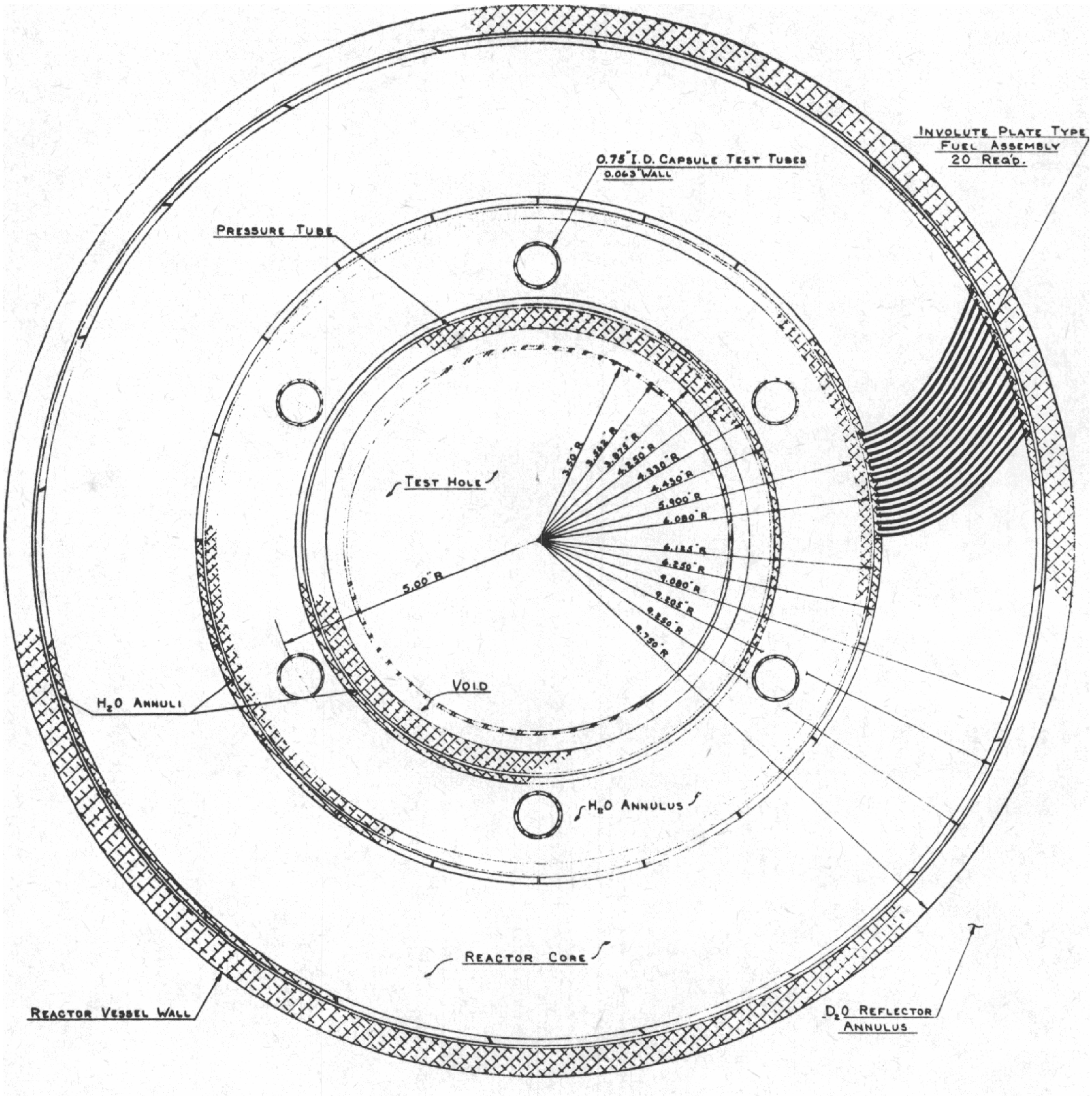

Fig. 19—Advanced Engineering Test Reactor, type A core cross section.

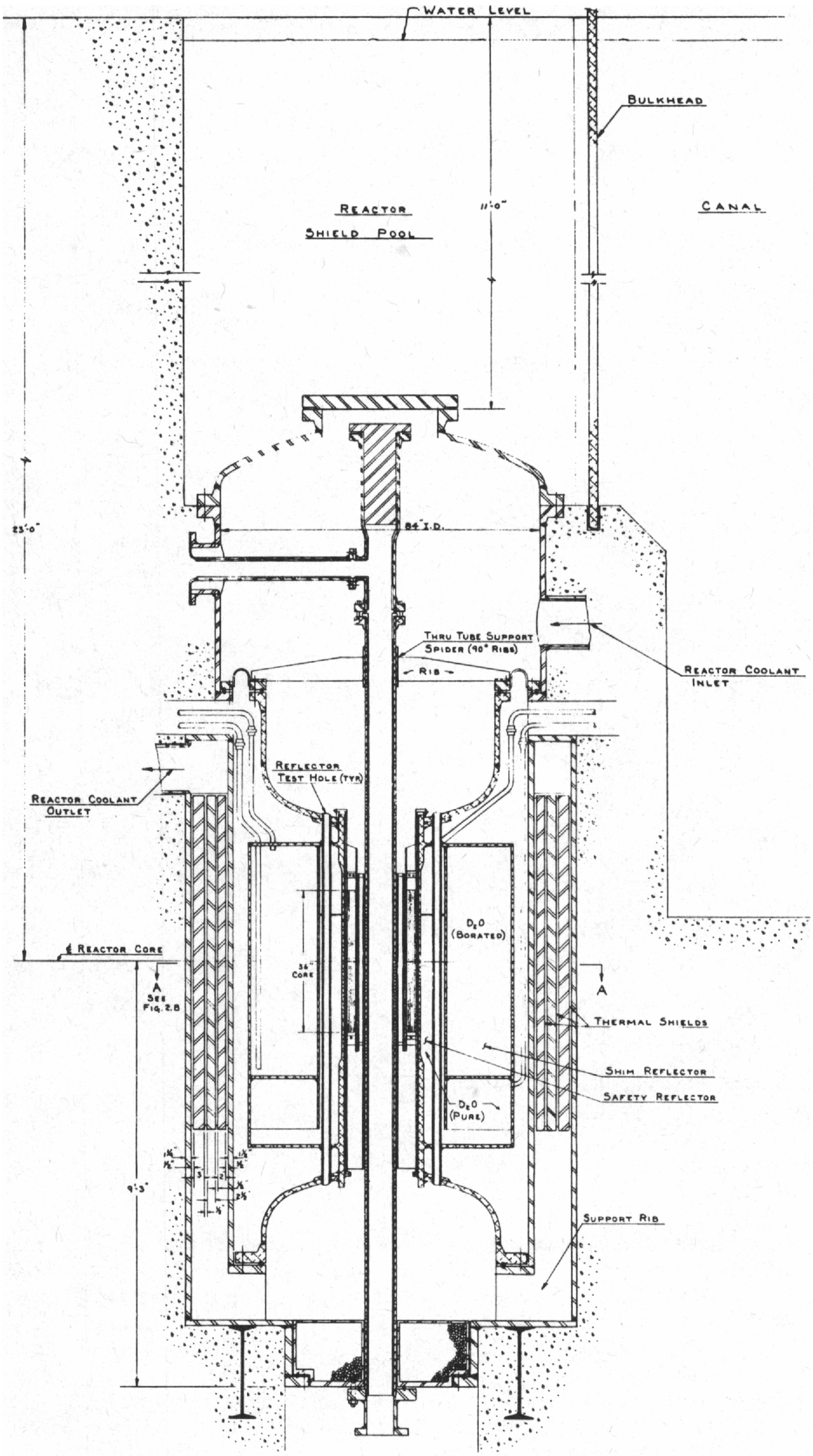

Fig. 20—Advanced Engineering Test Reactor elevation.

edges. As a result, an average power density of 2100 kw/liter is achieved with a 40 ft/sec water velocity, as shown in Table VI summarizing the AETR design characteristics.

TABLE VI

DESIGN DATA FOR THE ADVANCED ENGINEERING TEST REACTOR

Characteristic	Type A Core	Type B Core
Reactor power, Mw	170	100
Test hole coolant	Sodium	H_2O
Average thermal flux in test hole	1.5×10^{15}	2.1×10^{15}
Average intermediate flux in test hole	0.75×10^{15}	0.5×10^{15}
Average fast flux in test hole	1.6×10^{15}	1.0×10^{15}
Average fuel cycle time, days	19	30
Specific power, Mw/kg of U-235	15	11
Maximum power density, Mw/liter	2.9	2.9
Average power density, Mw/liter	2.1	2.1
Maximum heat flux, Btu/hr(sq ft)	1.44×10^6	1.44×10^6
Maximum surface temperature, ^oF	380	380
Coolant inlet temperature, ^oF	130	130
Coolant exit temperature, ^oF	252	252
Coolant velocity, fps	40	40
Coolant circulation rate, gpm	10,700	6300
Pressure drop across core, psi	79	79
Pressure at core exit, psi	380	380
Fuel plate thickness, in.	0.073	0.073
Coolant channel thickness, in.	0.050	0.050
Length of active core, in.	36	36
Inside diameter of fuel annulus, in.	12.5	9.5
Outside diameter of fuel annulus, in.	18.2	14.2
Fuel loading, kg of U-235 (90% enriched)	11.3	9.0

One particularly novel feature of the proposed design is the method of control. As in the case of all annular core reactors which have to operate at high power densities, the use of solid neutron absorbing control rods close to the core is unsatisfactory since these cause local perturbations in the neutron flux and create hot spots elsewhere in the core. Since this lowers the permissible power density in the core, other methods of control are necessary, such as the use of burnable poisons in the fuel or through the use of soluble poisons in the coolant or reflector. In the system proposed for the AETR, shim control is achieved by varying the concentration of soluble poisons in the D_2O reflector. Rapid shutdown of the reactor is achieved by sudden release of gas pressure which dumps a portion of the D_2O reflector adjacent to the core. To insure smooth operation of the reactor at steady state, however, conventional regulating rods located in two of eight capsule tubes in the D_2O reflector are included in the design.

CONCLUSION

Thus, it is seen that the need for neutron fluxes above 10^{15} neutrons/ cm^2(sec) is great enough to warrant the construction of not one but several high flux research reactors in the U. S. AEC laboratories. The development of fuel elements for these reactors, moreover, may lead to further advances in technology, permitting the construction of research reactors in the 10^{16} flux range. Experience at 10^{15} fluxes, however, will indicate whether the construction of the more expensive, higher flux facilities is justified.

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THE HIGH FLUX MATERIALS TESTING AND RESEARCH REACTOR OF THE REACTOR CENTRUM NEDERLAND

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ABSTRACT

A 20 MW, enriched uranium fuelled, light water moderated and cooled, research reactor, patterned after the Oak Ridge Research reactor, is under construction at the R.C.N. research centre. This centre is situated near the village of Petten, about 60 km NW of Amsterdan on the coast of the North sea.

The reactor is housed in a gastight building capable to withstand an inside overpressure of 0,5 atm and an underpressure of 0.02 atm. Access to the building is provided for by means of two personnel airlocks and one vehicle airlock.

The reactor consists of a concrete pool which can be divided into three parts by means of aluminum doors. The aluminum reactor vessel is placed in one of these pools. The two other pools are meant for storage of used fuel elements and activated materials. On the inside the pools are aluminum clad.

The reactor core is located on the axis of the reactor vessel, and is composed of 35 fuel elements, 36 beryllium reflector elements and 10 aluminum filler pieces. The core elements are arranged in a 9 x 9 grid. The fuel elements are modified MTR elements.

Six of the fuel elements are control rods which contain a cadmium poison section on top of a fuel section. These rods can be moved up and down by drives that are mounted in a sub-pile room in the basement. A ball lock mechanism releases the rods in case of a scram.

A maximum unperturbed fast flux of $7.8 \cdot 10^{14}$ n/cm^2.sec and a maximum unperturbed thermal flux of $2.6 \cdot 10^{14}$ n/cm^2.sec is expected in the centre of the core.

The reactor is shielded by the water in the pools and by a magnetite filled concrete shield.

The cooling of the reactor by demineralized light water has a total flow of 12,000 gpm.

The primary water circulates through the heat exchangers in a closed circuit and is in its turn cooled by secondary water taken from the North Holland Canal and disposed of in the North Sea.

A separate cooling system with a pump and a heat exchanger is provided for cooling of the pool water.

Two demineralizer units are used to purify the reactor water and one unit to purify the pool water.

GENERAL ELECTRIC TEST REACTOR

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As a representative of the General Electric Company's Atomic Power Equipment Department, I am honored to have the opportunity to participate in your symposium on high flux test reactors. It is always gratifying to exchange thoughts and ideas concerning our mutual problems in research and development associated with the nuclear power industry. I am confident that you will agree that our technical problems are numerous; however, the solution to our problems of today will permit us to continue at an ever increasing pace towards plentiful and economical electrical energy which is not dependent on our planet's decreasing supplies of suitable fossil fuels. Throughout the world we find ourselves in an era of great technological advancement, and we may reasonably conclude that today's efforts toward nuclear power will provide technological advancements having the greatest potential benefit to mankind in his search for a higher standard of living. Historically, man has sought sources of energy to relieve his burdens and provide leisure. In the world today, a rather reliable measure of the standard of living is the availability and cost of electrical energy. Perhaps 90% of the available energy on this earth which is suitable for use in power generation within present technology is concentrated in fissionable materials. We, therefore, have a great incentive in our very young nuclear industry and I am confident that we now stand on the threshold

of a period in the nuclear industry during which we will witness outstanding technical advancement, tremendous industry growth and unparalleled benefits to mankind.

One of the potentially most fruitful areas for significant strides towards economical nuclear power is the area of fuel and material development. The test reactor is certainly a primary tool in our fuel and material development efforts; and great emphasis is being placed on providing suitable test reactors and supporting facilities. In the United States, we have observed with keen interest as your new test reactor has progressed through design and construction since in many ways, our efforts in designing and constructing the General Electric Test Reactor have paralleled your efforts. We wish you success as you embark on programs utilizing your reactor.

When the General Electric Test Reactor (GETR) attained criticality on December 26, 1958, and full power February 24, 1959, it represented a significant advance in test reactor technology. The solutions to the challenges involved in designing and constructing this new concept in test reactor facilities spanned several years of study and work.

The GETR grew out of the need for additional test reactor facilities in the United States. We realized that an accelerated program for economic nuclear power would not be possible without increased test facilities for use in the development of reactor materials and long-exposure fuel.

Plans were laid toconstruct a test reactor at the Vallecitos Atomic Laboratory as an addition to its multi-million dollar nuclear research facility. The estimated cost of this test reactor was $ 4 million, exclusive of land, but including site development, parking facilities, mock-up shop, office building, and main reactor structure and facilities. It was felt that not only could an adequate facility be constructed at this cost, but with ingenuity and experienced engineering, a reactor could be designed which was much more flexible than existing designs.

A small design team of engeneers and physicists at the Atomic Power Equipment Department was selected to study the requirements, to recommend facility capabilities, to study various possible designs, to estimate time and cost requirements, and to recommend the optimum conceptual design.

In the test and research reactor field, General Electric had completed, at the start of design team study, the nuclear design of the AEC's Engineering Test Reactor (ETR), and was working on the Radiation Effects Reactor (RER) and the Shielding Development Reactor (SDR) for the USAF-Lockheed Aircraft Corporation facility in Georgia. In addition, work was proceeding on various open pool research reactors. With this experience, the following concepts were initially investigated :

a. Light water vs. heavy water (CP-5 type)

b. Extremely high power open pool reactor

c. ETR-type design

d. RER-type design

Various aspects of the above concepts ware investigated, with the following conclusions :

a. Light water was selected over heavy water for physics and engineering reasons form a test reactor standpoint. Some of the many engineering reasons for selection of light water include :

1. Higher fast flux per unit thermal flux,
2. Total neutron spectrum more closely approximating that obtained in light water moderated power reactors (of primary interest to the General Electric Company),
3. Higher thermal flux at experimental locations (result of higher fast flux per unit thermal flux; i.e., a better thermal flux stiffness),
4. More ease in handling experiments as compared to heavy water resulting from high cost of heavy water and necessity of keeping heavy water purity high,
5. Ability to use light water canal and move experimental facilities from canal to reactor without danger of contaminating heavy water,
6. No possibility of tritium buildup in light water,
7. Less significance of minor leakage around pumps and valves,
8. Less significance of isotopic contamination of water,
9. Less concern with contaminating experiments

b. High power pools were investigated at power levels up to 15 mv. It was decided that maximum power density limitations ruled out the high perturbed fluxes desired. (High fluxes obtained in pool reactors by "flux trap" principles are difficult to hold up in the perturbed state.) The tremendous flexibility of the Open Pool Reactor, though, was noted to be of real value for experimental use. The high leakage characteristics of the G.E. Open Pool Reactor design also were considered desirable.

c. The ETR-type design was investigated for powers up to 75 mv. Basically, the designs considered were very similar, except for total core size, to the ETR.

Although the general active core configuration was considered to be very good, the limitations on the use of the reflector region (requiring vessel penetrations for access) and the high cost of the ETR type of vessel, were considered as not meeting the specifications for the simpler facility desired. This design also lacked the capability to change reactor core and vessel after use without major disruption of the facility. The high power density of the core and the bottom drive concept were considered as highly desirable. It was recognized that the ETR was designed with different objectives, exhibited by its high power output (175 mv) and its numerous and large facilities (i.e.; up to 9 inches). While we felt the ETR was a significant influence in test reactor design, we also felt a continuing design advancement was necessary to make test reactors applicable for commercial use.

d. The RER was studied, and it was decided that ist design concept was not applicable to the type facility desired.

The study resulted in the selection of general concepts for a reactor type. Its specific requirements were as follows :

Required fluxes and power density determined the power level. Initial studies showed that 15 mv would probably produce adequate fluxes. A cost comparison, though, indicated that the power could be doubled to 30 mv without significant cost increase or additional system complexity. This resulted from a minimum core size required to provide in-core loops completely surrounded by fuel. A 30 mv level was selected as the desired power.

Experimental facilities presented a problem. Although a survey of our Company's long range requirements was made, it was recognized that we could not adequately estimate all our future needs in this rapidly changing business, nor economically construct all the facilities indicated by the maximum numbers of our estimates. We decided to have a minimum number of high flux in-core facilities, and a large number of flexible reflector facilities. This resulted in three in-core facilities with an expandable reflector region. The in-core facilities were sized at 3" OD.

To promulgate the above, we initially prepared two conceptual designs, and carried them sufficiently along to enable a good comparison of cost and performance to be made.

a. <u>GETR Mod. I (ETR-Type)</u>

This design was carried along as a reference point for performance, and to provide a basis for cost comparison. In our opinion, the ETR core conditions represented the maximum attainable under existing technology, and should not be exceeded. It was decided that any deviation from the ETR design concept would have to be justified on the basis of cost, design simplicity, and performance.

b. GETR Mod. II (Tank-Type - Pool Reactor)

A new concept was conceived, combining the high performance active core of the ETR with the high leakage and pool flexibility of the open pool reactor. This design showed many advantages over the Mod. I design.
Some of the advantages are as follows :

1. Substantially lower cost;
2. Easier handling of facilities due to smaller vessel and vessel head;
3. Readily accessible pool area for reflector experiments;
4. Wide use of aluminum (including vessel), thus reducing reactor component activation and cost;
5. Greater ease of refueling (removing of fuel over vessel flange).

This design was particularly effective in that it provided a fixed in-core set of facilities, and also provided a high degree of flexibility in the reflector region, permitting complete change of out-of-core requirements with minimum difficulty. It also permitted change of vessel and core after prolonged use without difficulty. The design retained the ETR bottom drive concept to reduce obstructions to the experimental facilities.

After comparisons were carried out, it was decided that a Mod. II type reactor could be built. It provided most of the desirable features for a test reactor, had comparatively high fluxes, was exceptionally safe, and relatively low in capital and operating cost. Mod. II was selected for the GETR design basis, and detailed plant design begun. To assist in the building construction, the Ralph M. Parsons Company was selected as Architect-Engineer-Constructor.

The reactor design consists of a high leakage core, tightly enclosed in an aluminum pressure vessel. The core is water-reflected, with certain beryllium and aluminum filler pieces utilized between the core and the vessel as flow restricters. This design permits the use of vessel penetrating high flux in-core through loops, together with a completely flexible pool or reflector experimental space which does not require vessel penetration.

The fuel is ETR flat-plate type, with 3-foot active sections. The 3-foot active length was determined to be the optimum active length.

Control rods are approximately 2-$\frac{1}{2}$" by 2$\frac{1}{2}$" by 72" in length, and utilize boron stainless steel poison material with fuel element followers. Work has been done on the possibility of replacing fuel followers with beryllium followers for later loads. No final decisions have been made on this point. It should be noted that the control elements are separated from all loops by at least one fuel element, thus reducing rod perturbation of loop fluxes.

Internal core beryllium filler pieces provide flexibility in core loading spaces (allowing for variations in loading without requiring numerous special loaded fuel elements), to provide space for in-core capsule experiments and to reduce the total number of fuel elements in the core matrix.

The drives are bottom-mounted nut and lead screw type, utilizing magnet release gravity actuated scram. There are a total of six shim drives, with one of the shim drives being equipped with a differential over-riding servo control regulating drive. This concept eliminated the requirement for a separate core space for the regulating rod. Total servo over-ride travel is limited to 4 inches.

The final GETR design resulted in a reactor with design power levels available up to 100 mv. The reactor is available in different module designs, and it is necessary to select the maximum desired power level during design. The GETR is designed for a nominal rating at present of 30 mv, as previously noted. Actual maximum power level will be determined after operational experience has indicated the actual degree of design conservatism.

The reactor vessel is in a pool of water. Sufficient water is maintained over the vessel head to permit manual refueling by removing the elements over the top of the vessel and into the canal.

The experimental facilities consist of three 3" through loops and 16 internal capsule positions. Externally, there is complete flexibility. The pool wall penetrations are designed to accommodate pool hairpin loops up to 6 inches in diameters. The internal loop sizes were determined based upon the requirements for testing, plus the economics of loop construction. A hydraulic shuttle tube and an 8" beam port are also provided. Internally, the GETR is all aluminum, with the exception of the stainless steel grid plate and core support structure. The vessel is aluminum, with stainless steel heads and spool piece. The primary coolant loop is all aluminum except for trim, etc., in the pumps, and stainless steel in certain valves. Secondary systems and clean-up loops are conventional. The wide use of aluminum resulted in low cost, minimum corrosion, low materials activation, and a system in which the coolant chemistry can be optimized for one material.

The spool piece on the vessel and the extensions are provided to enable experimental leads to be brought out above the pool surface.

The entire high power pool area is closed by a sliding upper shield when operating at power. The opening to the canal is closed by a gate during power operation. The high power pool is lined with aluminum, while the storage canal is coated with a phenolic resin.

The entire reactor facility including the biological shielding and a four story reinforced concrete structure is enclosed in a complete containment building, representing on of the cost-saving advancements for this reactor. Because of the design safety of the reactor, plus the presence of the pool of water surrounding the vessel, containment building design specification amounted to less than 5 psi building overpressure.

The containment vessel is approximately 66 feet in diameter by 105 feet high. Skin thickness generally varied from 0.3125 inches at the bottom courses, to 0.250 inches at the upper. The building is cylindrical, with a hemispherical head and flat reinforced concrete base. All space on all floors other than that occupied by the reactor cooling equipment and the pool and biological shield is used for experimental equipment associated with the use of the in-pile and reflector loops. The basement, first, and second floors each provide two experimental areas of a approximately 800 square feet each and in addition on the second floor is provided space to permit use of the beam port facility. The third floor of the containment building provides one experimental space in addition to the space taken up by the canal, the pool and its upper shield, and building service equipment. The third floor is served by polar crane of 15 ton capacity. This crane provides service for all other floors through the 8' x 10' equipment hatch which is expandable to 8' x 14' by removal of the demountable stairs. It should be noted that on each floor manholes are located to permit extension of loop piping from any of the experimental facilities to any one of the seven experimental areas throughout the containment building. This is accomplished by the simple expedient of removing plugs that are normally in the manholes, extending loop piping and providing the necessary shielding. Entrance to the containment building is by means of airlocks. Two airlocks are provided, one for personnel entrance and one for equipment entrance. Plant services are provided at "Service centers" which are accessible from each experimental area. These service centers provide secondary water supply and return, demineralized water supply, compressed air supply, a clean drain, a hot drain, a raw water supply, 440 volt normal power for equipment, 440 volt emergency power, normal power for instrumentation and emergency power for instrumentation. Emergency power is provided by a 150 kw diesel generator set located in the auxiliary equipment building.

Instrumentation is conventional for a reactor of this type. Both nuclear and process instrumentation consoles are located in the control room, which is in the administration building.

Emergency cooling for the reactor during complete service power failure is provided by coolant flow reversal in the core and the forming of a convection cooling loop through the pool and core. Sufficient heat sink is available here to adequately handle all emergency shutdowns. If there is a loss of pool water, the 500,000 gallon fire tank on the site can be diverted into the pool by gravity feed.

Complete flexibility for future plant modifications has been provided. It is possible to remove the reactor completely and install a new vessel and core into the plant at any time. The vessel, which represents the largest piece, can be taken out and handled in the canal, for disposal through a port installed in the containment roof. The remaining components can be handled by conventional methods. This is particularly noteworthy when it is realized that the entire core structural members and the vessel represent and investment of less than $100,000, because of the unusual design. This is another advantage of the pool-tank-type design.

All experimental facilities are handled from the top of the reactor. It is only necessary to remove the top head for access to the core internal facilities. Since the drives are bottom-mounted, the facilities are readily accessible, with the head removed, as are the fuel elements for refueling. Water shielding is sufficient to permit equipment to be moved back and forth between reactor and storage pool without danger of personnel over-exposure.

Initially, the GETR has gone into operation with several loops and numerous capsule experiments. One internal 3-inch annular-type pressurized water loop has been installed in the core, and a 3-inch boiling water loop has been placed in the pool region. In addition, and Oak Ridge National Laboratory 5-loop gas-cooled facility has been installed in the pool region.

In any discussion with the available fluxes in the test reactor, it is always desirable to define the basis on which the fluxes are presented however, to preclude the necessity of detailed definitions, I would like to discuss the flux data as approximate perturbed fluxes available in the various irradiation positions with a reasonable reactor loading. The GETR core contains provision for one nominal 3" OD through-loop in the center of the core. In this position the peak thermal flux is approximately 2×10^{14} neutrons per square centimeter per second and the peak above thermal flux is approximately 8.5×10^{14} neutrons per square centimeter per second. In all reference to above thermal fluxes, these are considered as being defined as flux having energy greater than 0.17 EV. The two side core through-loop positions are also suitable for nominal 3" OD loops and in these positions, the peak thermal flux is approximately 1.7×10^{14} and the peak above thermal flux is approximately 5.8×10^{14}. In addition within the reactor pressure vessel 16 - $1\frac{1}{2}$" diameter by 36" long capsule positions provide peak thermal flux of approximately 1.7×10^{14} and peak above thermal fluxes of approximately 4×10^{14}. In the reflector region of the reactor which is, of course, outside the pressure vessel and within the reactor pool spaces are available; 4 hairpin loop positions and 31 - $1\frac{1}{2}$" diameter by 36" long capsule positions. The peak thermal flux in this region varies from approximately 6×10^{13} to 1×10^{14} and the peak above thermal flux available is approximately 5×10^{13}. The calculated values for the critical physics parameters were determined by measurements to be accurate. In the detailed measurements made, we ascertain that the flux varies around the reactor in the reflector region such that there are four areas of relatively high flux and four areas of relatively low flux. We have deemed this flux distribution as a "clover-leafing" effect for lack of a better definition.

This flux distribution represented somewhat of a surprise to us since our physics calculations were concentrated primarily on the portion of the reactor within the pressure vessel and somewhat less attention was paid to the calculation of the flux distribution in the reflector. We have found, however, that this "clover-leafing" effect provides us with great flexibility in that we can select an appropriate flux for a given experiment to be conducted in the reflector region. By measurement, we have determined that the peak to average flux ratio is approximately 1.8 at the beginning and approximately 1.4 at the end of the cycle. During the cycle, we experience a thermal flux increase of approximately 10%. In our opinion one of the great problems of utilizing a test reactor is maintaining knowledge of the flux levels and flux distribution in the reactor as reactor loadings bearing. We have already made, and continue to make, numerous measurements as the occasion demands and our experience to date indicates that we have been

quite successful in predicting the conditions of irradiation from the standpoint of flux delivered to the samples.

In our use of the reactor to date, we have found that the flexibility inherent in the design of the reactor has been of paramount importance to us. The design of capsule experiments is becoming more and more complex and the requirements of the experiments are becoming more demanding due first, I believe, to our need for accurate data and also due in large part to the very high cost of loop facilities. We have worked with capsules utilizing gas blankets for temperature control, positioning devices to maintain an approximate constant thermal flux during the reactor cycle, capsules involving gas flow over the test pieces for cooling and to permit determination of fission product release and capsules providing a means of simulating the difference of linear expansion between the fuel meat and the fuel cladding. Of course, we have also been faced with accommodating numbers of relatively simple capsules; however, for the more exotic designs which require tubing and devices within the reactor, we have found that we can accomodate these successfully in the pool areas, as accommodating this equipment within a reactor pressure vessel would be very difficult. We have also used the reflector region of the reactor for the installation of large tubes into the flux region which will permit the installation and withdrawal of large devices into the flux region during the reactor cycle and further permit the cycling of the experiment as required. We have also found that the reflector region permits the installation of loop facilities which would not be possible within the pressure vessel of most reactors. As an example, we have a gas loop installed in the reflector region of the GETR which provides five independently controlled facility pipes served by one group of supporting equipment. During the design, we felt that it was almost impossible to predict the future use of test reactor facilities and our experience to date has certainly confirmed this viewpoint in our opinion. New requirements for unusual irradiations seem to arise almost daily and we have found that the flexibility of our reactor permits the accommodation of unusual irradiation requirements with relative ease.

Although a research and development laboratory in a young and growing industry is never complete, the operation of the GETR has provided the General Electric Company with a balanced facility capable of successfully accommodating all aspects of an irradiation program. As you know, vast and expensive facilities are required to support a test reactor and in very brief review our laboratory facilities include a Radioactive Materials Laboratory having four large high level test cells, a metallographic cell, hot and cold chemistry laboratories and Metallogra-

phy and Ceramics Laboratories. In Addition, we are very fortunate in having an experimental physics laboratory which includes a 30 KW nuclear test reactor, critical assembly and a computer facility. The Vallecitos Atomic Laboratory also is the location of the Vallecitos Boiling Water Reactor which I am sure you are familiar with. All of these facilities complement and support each other in the work which we have undertaken.

In taking an objective look at the experience to date with our test reactor, I believe that it is significant to point our that we found our reactor to be relatively simple to design and build and we can conclude that our test reactor technology is sufficiently advanced for the fluxes which we presently require. We have found, however, that we do have problems in the area of design of experiments and in the evaluation of data obtained from the experiments. The demands of experimental programs are becoming more exacting as we seek to satisfy our urgent need for accurate data.
We believe that much work remains to be accomplished in the areas of experimental design and data evaluation and I submit these problem areas to you for your consideration as you proceed with your research and development work.

It has been a pleasure to have the opportunity to briefly describe the GETR and present some of the concerns that we have after several months of operation. I am confident that you will meet with success in your research efforts and I sincerely hope that the results of all the investigations on an international scale in the nuclear energy area can be applied successfully in the very near future to the provision of economic nuclear power throughout the world.

SPECIAL DESIGN CONSIDERATIONS FOR HIGH FLEXIBILITY IN BR-2

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FOREWORD

The Reactor BR-2 was designed as a cooperative effort between the Centre d'Etudes de l'Energie Nucléaire (CEN) and the Nuclear Development Corporation of America (NDA). The reactor will serve as a basic research tool for the growing Belgian nuclear industry and will be one of the world's major research reactor installations.

1. INTRODUCTION

The reactor BR-2 is a water-cooled and moderated, beryllium-reflected high flux research reactor. When operated at a power of 50 MW, it will have a maximum thermal neutron flux of about 6×10^{14} n/cm^2-sec. It has been designed to accommodate a wide variety of experimental facilities including radial and tangential beam holes, through-loops, irradiation capsules, and rabbits. The reactor will be used in both an extensive materials and engineering test program and a physical research program.

2. DESIGN CONSIDERATIONS

The most important feature of the design of BR-2 is the inclusion of a large number of different types of experimental facilities arranged so that the operation of one facility will have a minimum effect on the neighboring facilities.

The reactor contains 64 holes 3.3 in. in diameter which are usable for fuel, control rods, or experiments; 5 holes 8 in. in diameter for experiments; 10 holes 2 in. in diameter for rabbit experiments; 5 radial beam tubes and 4 tangential beam tubes with ports at each end, each 12 in. in diameter. In addition, experiments can be placed in the large pool of water surrounding the reactor.

The neutron flux to be expected in each type of facility is shown in Table 1.

The reactor proper contains 79 lattice positions, and it became apparent early in the design program that each position should have an independent access through the reactor vessel; since operating convenience requires that experiments be straight rather than curved or offset, the accesses should be through the vessel head. The provision of 79 openings in a vessel head each with a sealing device required the use of a large diameter head. However, nuclear requirements of high flux, approximately 2×10^{15} n/cm^2-sec at 50 MW, made it necessary to keep the core volume small. These conflicting requirements were met by adopting the present skewed or hyperboloid geometry which maintains close spacing at the center of the core yet provides wide spacing of the lattice position ends to accommodate the individual seals. The vessel was made in an hourglass shape to enclose the reactor but still permit the beam tubes to approach the core without the necessity for complicated vessel openings. This design offered particular advantages in the placement of the tangential beam tubes. The hourglass-shaped vessel was used to good advantage by placing seals in the bottom head so that the 5 large 8 in. diameter holes and 13 of the 3.3 in. diameter holes can be used for through-loops.

Table 1 — BR-2 Performance Characteristics

Power, MW	25	50
Maximum Thermal Flux, n/cm^2-sec		
Core	3.1×10^{14}	6.2×10^{14}
Reflector	3.1×10^{14}	6.2×10^{14}
Test holes	1.4×10^{14}	2.8×10^{14}
Beam holes	0.3×10^{14}	0.6×10^{14}
Maximum Epithermal Flux, n/cm^2-sec		
Core	1.2×10^{15}	2.4×10^{15}
Reflector	8.6×10^{14}	1.7×10^{15}
Test holes	4.7×10^{14}	9.4×10^{14}
Beam holes	0.3×10^{14}	0.6×10^{14}
Center Beryllium Island (without neutron absorbing experiments)	36 MW	
Maximum thermal flux, n/cm^2-sec	8.6×10^{14}	
Maximum epithermal flux, n/cm^2-sec	1.4×10^{15}	

The arrangement of the reactor is illustrated in Fig. 1, which shows the relationship between the cover seals, reactor core, and beam tubes. The guide tubes extending between the top cover and the core are also shown; they serve to guide the experimental equipment during installation and to protect it from damage during operation or during the installation of adjacent equipment. The experiments can be installed or removed without removing the vessel head or disturbing the connections to the already installed experiments or the control rods.

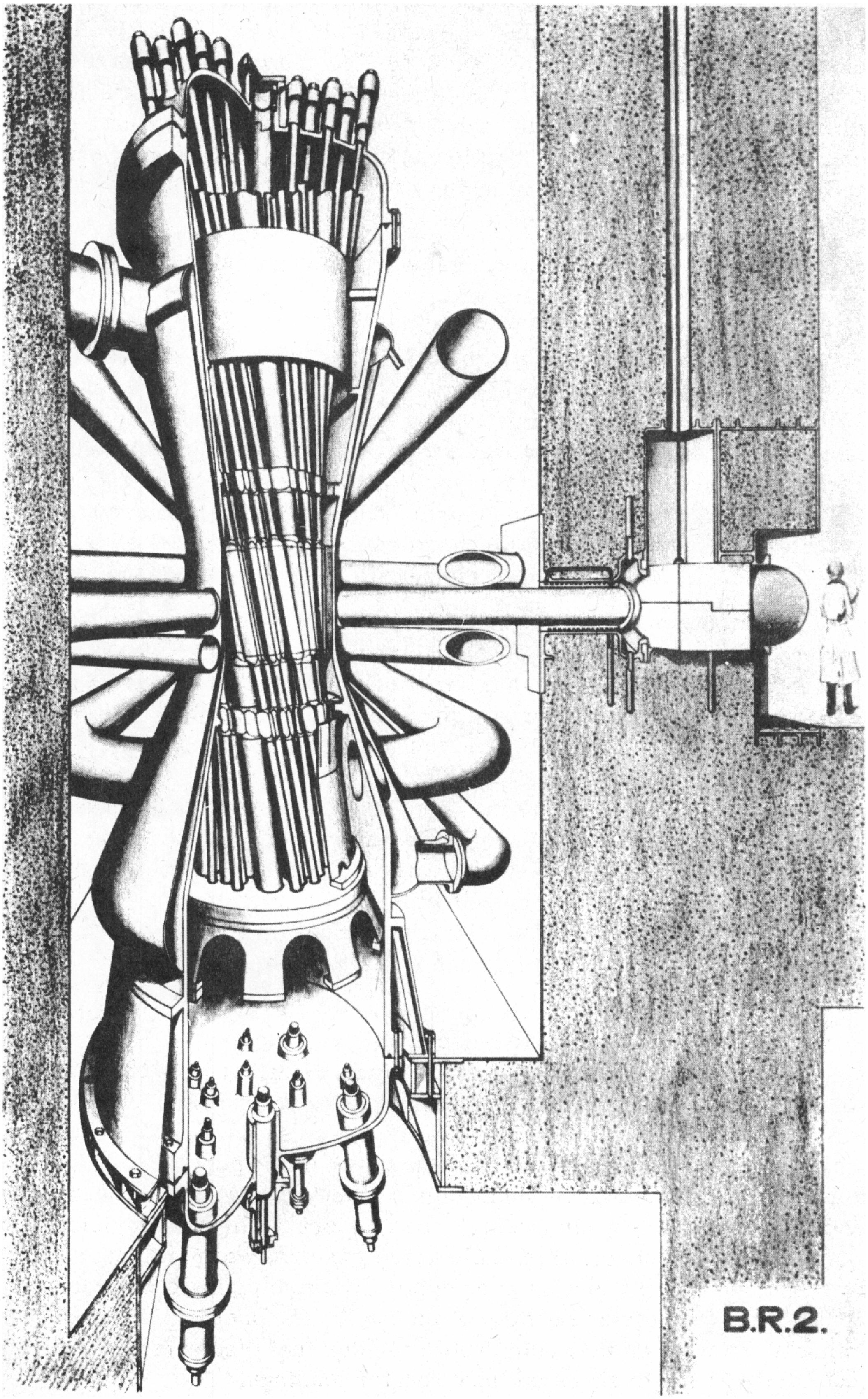


Fig. 1—BR-2 reactor arrangement.

Fig. 2 is a cross section of the reactor core at the midplane. It shows a typical loading of experiments, fuel elements, and control rods. It illustrates the close packing achieved by the hyperboloid geometry and shows unused holes filled by beryllium plugs. These plugs are constructed of a hollow beryllium cylinder 3.3 in. in diameter surrounding a solid beryllium plug 1.35 in. in diameter. This construction permits the installation of small experiments without introducing the nuclear and hydraulic problems resulting from large water-filled passages in the core.

In the core, all 64 of the 3.3 in. holes are identical, making it possible to arrange the distribution of fuel, control rods, and experiments so as to control the neutron flux available in the experiments or the beam tubes. This is a degree of flexibility never provided before in any high flux test reactor. Previous designs permit only relatively minor changes in core arrangement. In order to take full advantage of this type of core arrangement, a new type of concentric cylinder fuel element was developed. This type of element provides a central hole which can be used for small experiments requiring a high fast flux. When required, some of the central cylinders can be removed to accommodate larger experiments, the remaining cylinders acting as converters to increase the fast flux. This fuel element is shown in Fig. 3.

The five radial and four tangential beam tubes are arranged so that they may be used for either physics or engineering experiments. The 12-in. diameter of the tubes is larger than that installed in any other reactor of this type. The size of the tubes is large enough to permit installing much of the experimental equipment inside the tube, thus reducing the congestion of external equipment often installed around existing beam facilities. The large size, coupled with the expected 10^{13} n/cm^2-sec flux in the tubes, makes them useful for irradiation experiments. If special tubes are required for experiments, the existing tubes can be changed without draining the shielding water from the pool. Each beam port is equipped with internal utility and instrument connections so that the problem of shielding numerous piping and instrument connections at the outside of the shield can be simplified.

The shielding pool is utilized to increase the flexibility of the reactor in many ways. Its prime use is as transparent shielding for the reactor. By operating through the water the reactor may be loaded and the experiments installed or removed without the use of heavy shielding casks. Long-handled tools are provided for operating the reactor cover seals and for the manipulation of the various pieces of equipment installed in the pool. Provision has been made to bring instrument connections from experiments through the pool water to control panels in the reactor containment building. Provision has also been made to connect large engineering experiments requiring separate cooling systems to their auxiliary equipment through piping connections installed in the side walls and bottom of the pool. The pool is also used to shield a gamma irradiation facility using spent reactor fuel elements and a core mockup facility used to check out new reactor loadings.

The relationship between the reactor and the pool is shown in Fig. 4. Here are illustrated the installation of loop experiments connected through the pool

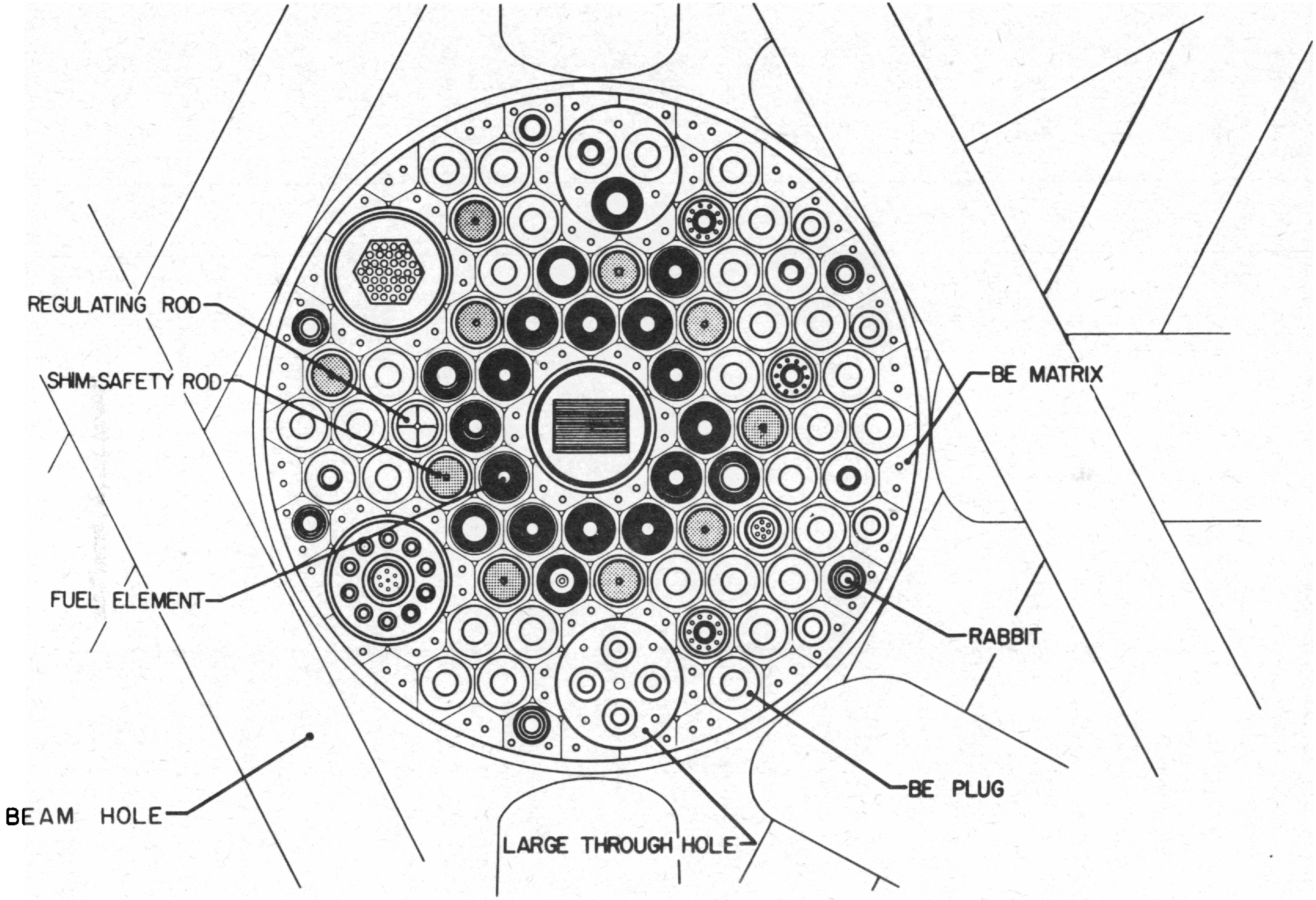


Fig. 2—BR-2 core cross section.

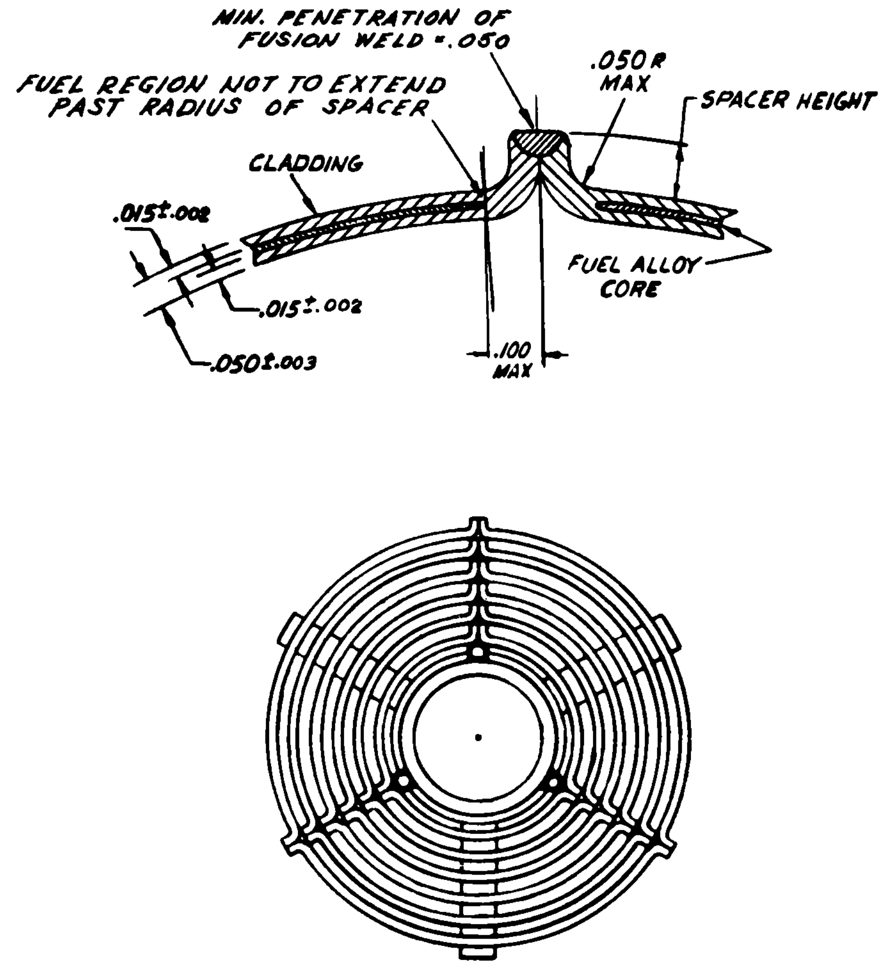


Fig. 3—BR-2 fuel element.

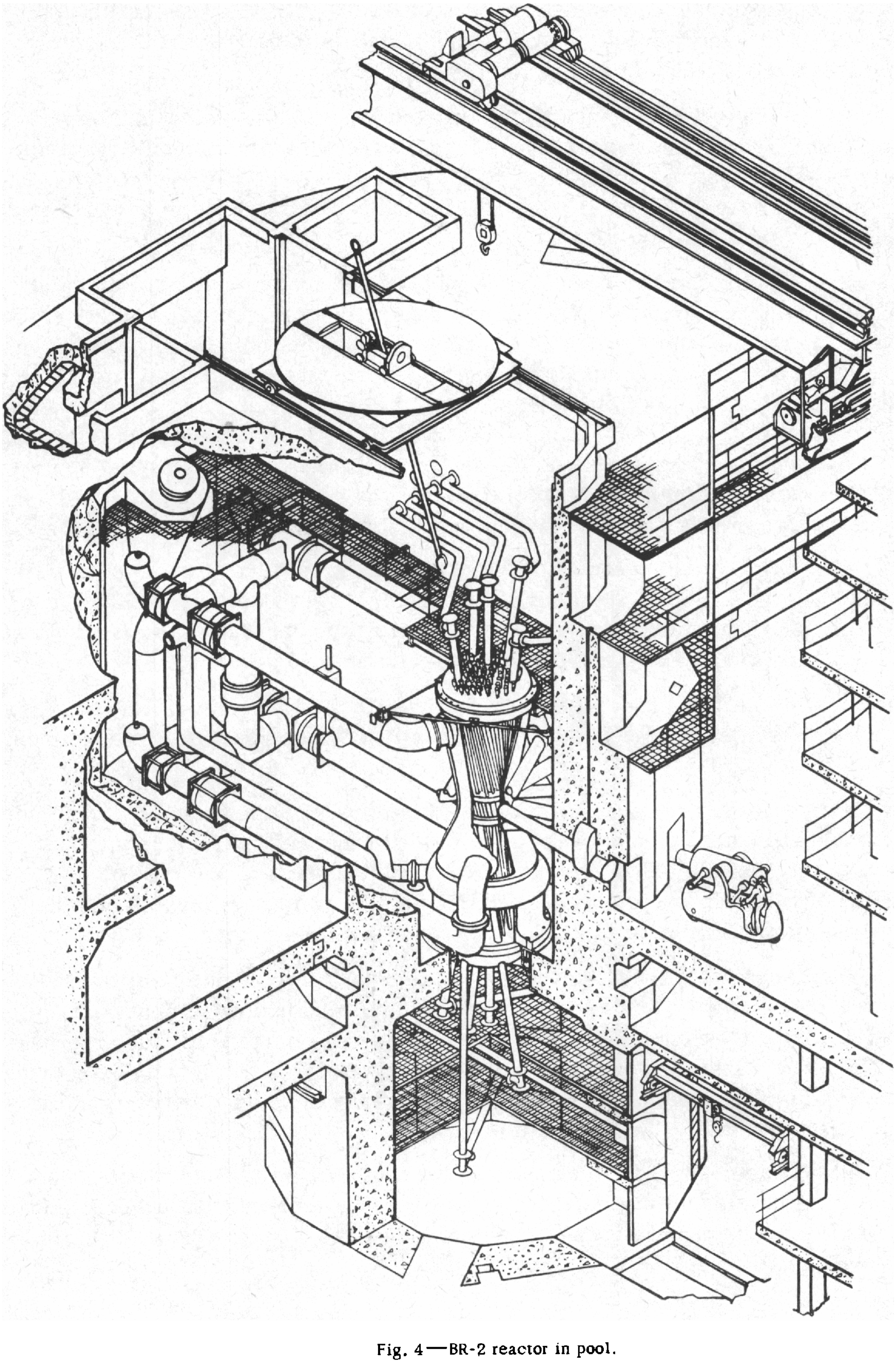


Fig. 4—BR-2 reactor in pool.

wall and through the sub-pile room, the beam ports, and the large cooling water lines which are shielded by the pool water. The gamma irradiation facility and the core mockup facility are installed in the right and left branch pools shown in the upper left of the figure.

Large experiments (such as the irradiation of full-sized equipment), certain types of biological experiments, and engineering experiments requiring only relatively low neutron fluxes can be installed in the pool adjacent to the reactor. If needed, some of the beam tubes may be removed to permit a closer approach to the reactor vessel.

In a reactor facility where large numbers of instrumented experiments and engineering experiments requiring separate cooling systems are installed, space for the auxiliary equipment is often inadequate. In BR-2, an attempt was made to provide the maximum amount of space for such equipment in the reactor containment building, consistent with considerations of cost and construction problems. For example, all of the floors were designed to permit the construction of shielded enclosures for equipment and to provide space for the instrument panels. Two of the floors are so constructed that only shielding side walls need be erected around the equipment.

Simplified cross sections of the containment building are shown in Figs. 5 and 6. Fig. 5 indicates the large number of levels available for the installation of experimental equipment and shows the pipe connections in the floor of the pool for routing loop experiments to the sub-pile area.

Fig. 6 is a floor plan at the level of the first instrument galley and shows the large area of floor space available for installing experimental equipment and instrument panels.

It is quite likely that experiments which are larger and more complicated than any attempted to date will be built as the nuclear industry grows and advances technically. To accommodate such future experiments in BR-2, provision has been made to connect experiments to equipment located outside the containment building through pipe trenches in the floor of the sub-pile area.

In the design of BR-2, certain facilities were intentionally omitted. No provision has been made for a thermal column, medical treatment port, shielding facility, or exponential assemblies. They were omitted because their operation might interfere with the primary purpose of operating an engineering test reactor; experience in the U.S. has shown that other types of reactors are better suited for these facilities.

BR-2 was designed to incorporate the best features of the present generation of materials and engineering test reactors and to anticipate, where possible, the future needs of the European nuclear industry.

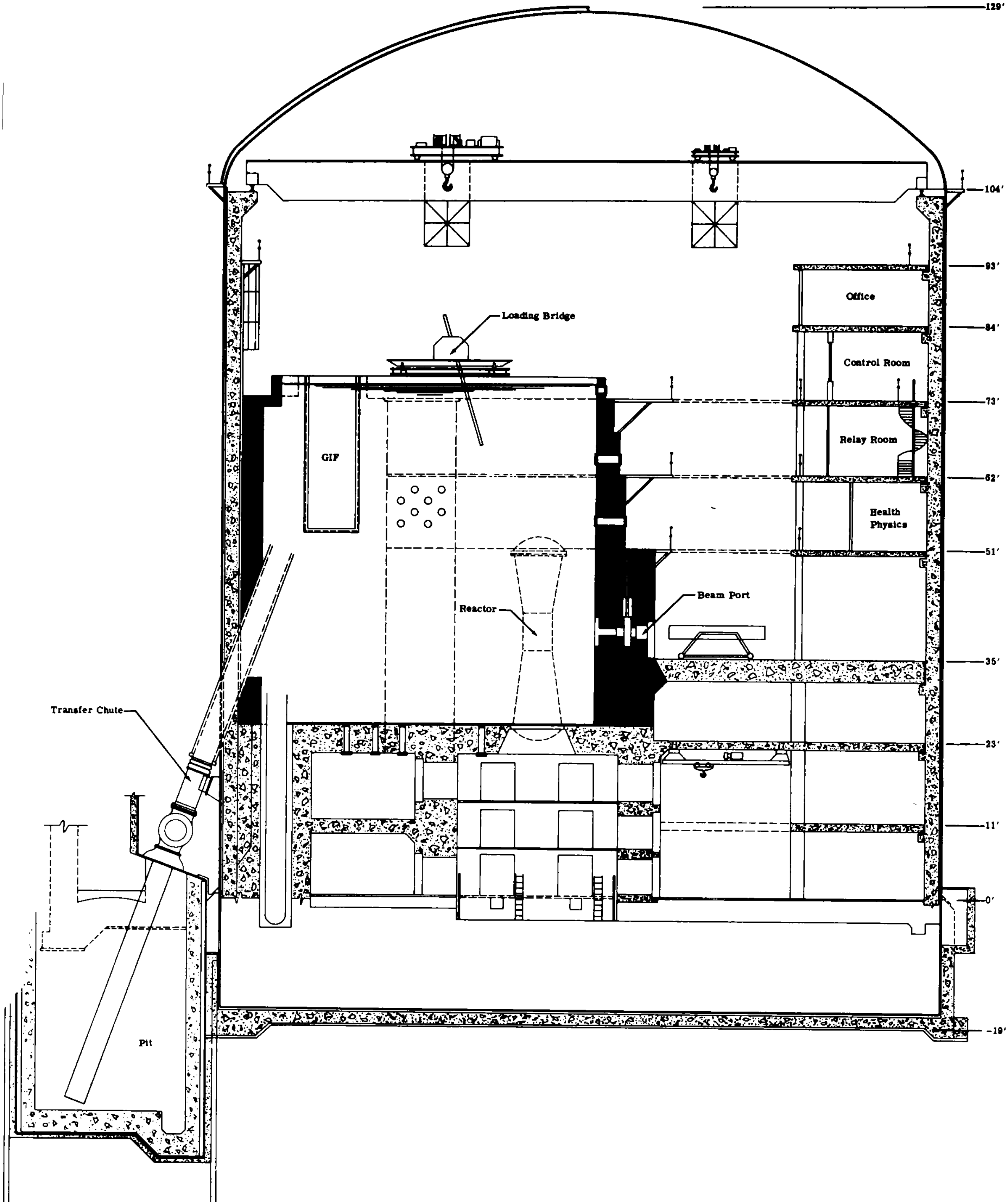


Fig. 5—BR-2 containment building, elevation.

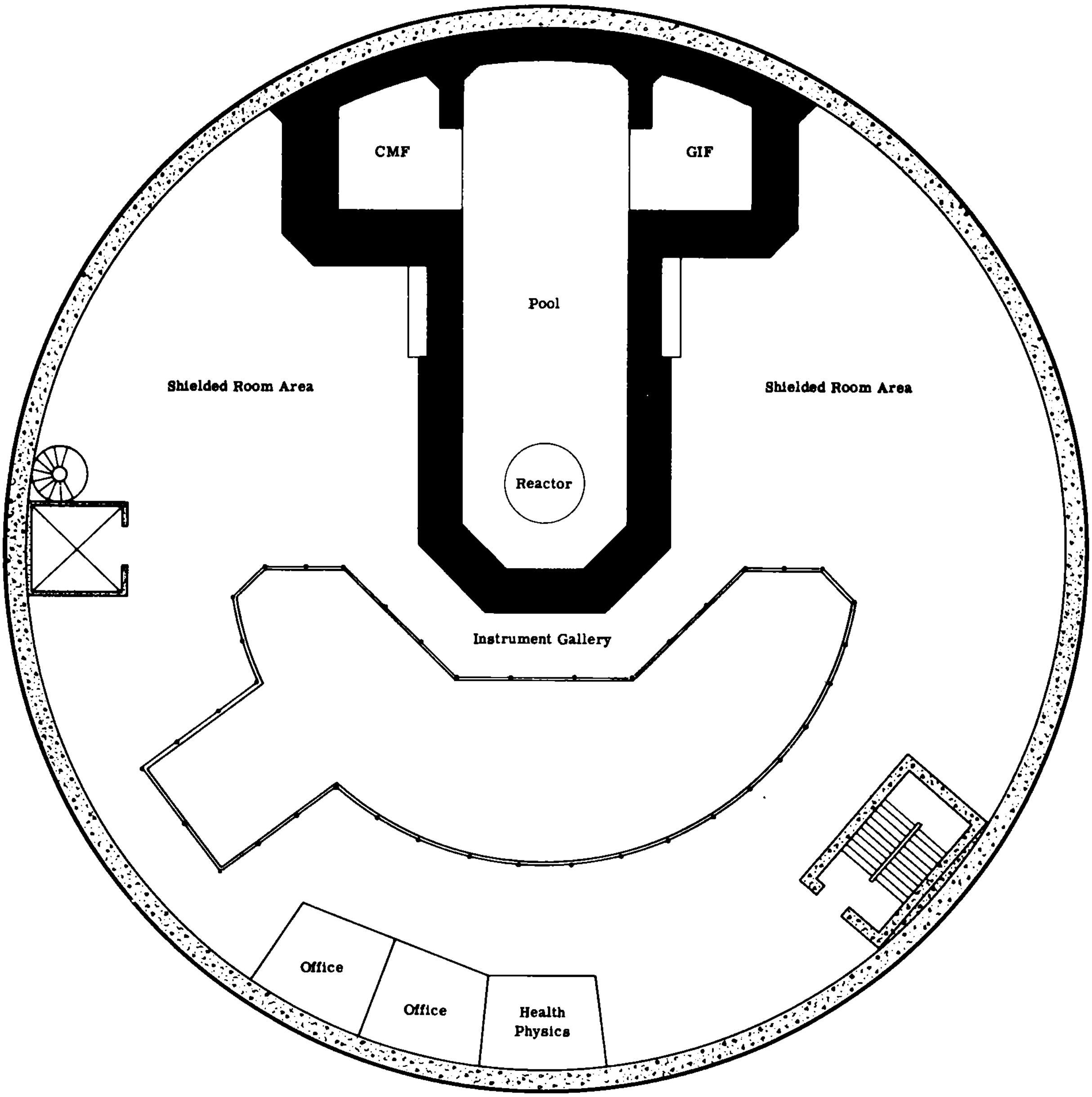
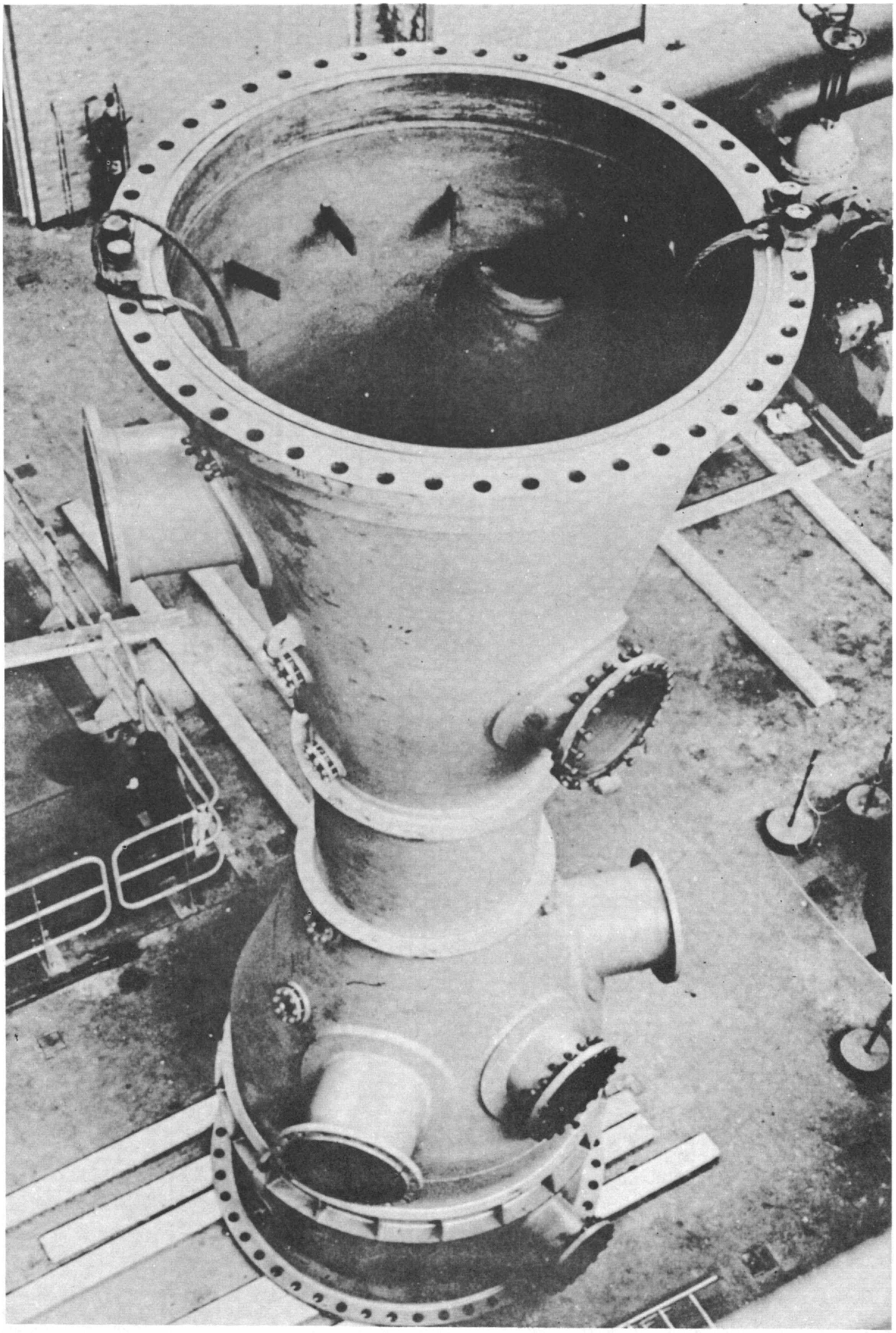


Fig. 6—BR-2 containment building, plan at 51 ft.

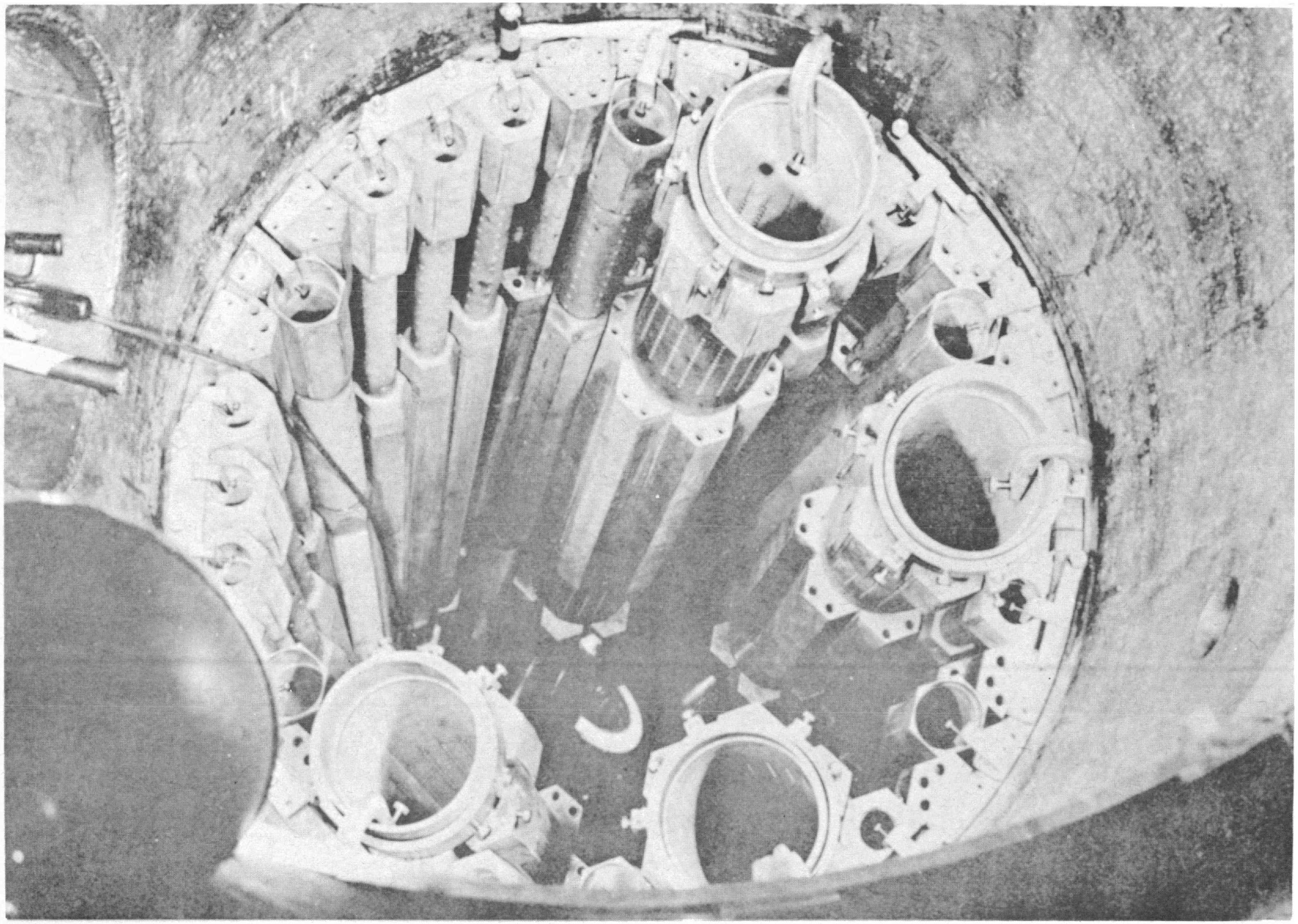
3. SUMMARY

The design of BR-2 was completed with the active participation and co-operation of CEN. It incorporates design features which provide maximum flexibility and utility in meeting the special requirements of CEN, at the lowest practical capital and operating costs. It was possible to manufacture all of the reactor facility equipment in Belgium except for the machined beryllium parts and the fuel elements.

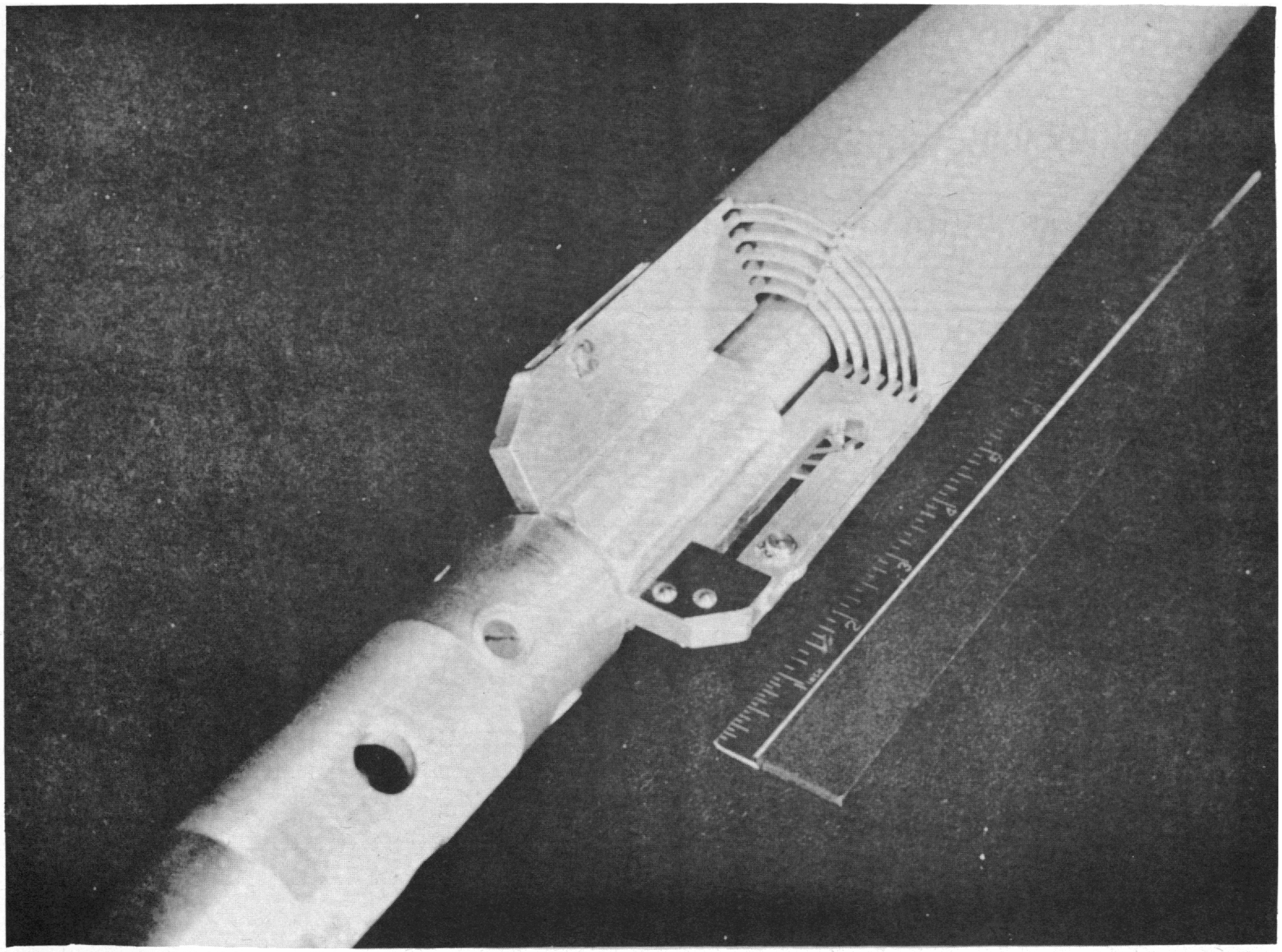


NEG No 2089

BR-2 test vessel during installation.



BR-2 test matrix during installation.



NEG No 1637

BR-2 fuel element.

DIFFICULTIES MET IN THE CONSTRUCTION OF THE BR-2 AND SOLUTIONS ADOPTED

ARMAND WALRAVENS
Bureau d'Études Nucléaires, Brussels, Belgium

The construction of the BR-2 reactor has posed and is still posing problems in the most varied fields. The solutions adopted, frequently original, have required either certain developments of a theoretical or technological nature, or have resulted in special designs or preliminary tests tending to show the practicability of the former.

In the course of this report we shall examine several of these problems, choosing them, on one hand, on the basis of technical developments and, on the other hand, on the basis of special ideas and, perhaps, specific to the BR-2.

A first problem, the testing of whose solution will perhaps contribute to the knowledge of the treatment of materials designed to be used as or in the core of a reactor and to remain geometrically stable, was raised by the development of the inside core equipment of the BR-2 reactor.

Such equipment consists of an assembly of channels whose axes, with the exception of the central axis which is vertical, are orientated in accordance with the generatrices of several concentric hyperboloids.

The channels are made up of five sections, respectively from top to bottom, and, in the case of channels of the nominal size of 3 1/2":

- a guide tube, pierced laterally by holes destined to admit most of the flow of the water in the channel;
- an upper extension, in the form of a tube pierced by lateral holes and provided with two appendices of hexagonal cross-section.
- a central part, of beryllium, of hexagonal cross-section;
- a lower extention, similar to the upper extension;
- a prop tube.

These five sections, once assembled, form a rigid unit.

The upper and lower extensions are of stainless steel.

As the AISI 304 type steel was chosen as the basic material, two possibilities were offered for its fabrication: either cast or forged steel.

The reasons for prefering forged stainless steel to cast steel are in particular the following:

1. The Radiographic Standards for Steel Castings, ASTM E 71, in the form of plates, and for the various categories of defects which may be found in cast steels, renders a classification according to the importance of such defects.

An examination of these defects shows that the zones of discontinuity of the material may have fairly large dimensions and be in relatively concentrated form.

It is an established fact, moreover, that all discontinuity in the material brings about hot points with the release of heat in the material. In the present case, such heat release has been estimated to be 10 W/cm^3.

The magnitude of such discontinuities being evidently less in the case of forged steel, the adoption of this solution therefore gave the advantage of reducing the amplitude of temperature gradients.

2. Repairing cast austenitic steels by welding must be considered as impracticable without very special precautions. In effect, the cooling of a weld brings about microscopic fissures in the underlying metal.

By way of information, let us mention here that certain foreign firms have perfected special methods making such welding repairs possible, for example, by judicious hammering designed to break the austenite grain. These methods are protected by special patents.

To conclude on the matter of these two points, the standards which would have to be met in the case of cast steel would have been very strict ones; the difficulty of adhering to them would have led to its inevitable rejection.

The next question was that of the choice of C contents in the 304 steel.

According to certain specialists, the high tempering practiced on stainless steels causes in them residual stresses of about 15 to 20 kg/mm^2.

Such strong stresses appeared unacceptable to us. In fact the heating undergone and the temperatures attained, although moderate, in the core of very thick sections would, in the course of the life of the reactor, have produced a relaxation of such stresses and thus brought about deformations in the extensions which, although slight, would have resulted in a jamming of the matrix.

As a relaxation treatment, following high tempering and performed at 900°C, is, moreover, without any danger to the metallurgical properties of an 18/8 low carbon stainless steel or one stabilized and reducing by 75 percent, according to the same specialists, the amount of the residual stresses, resulted in the choice of a low carbon steel.

It was necessary, finally, to indicate precisely the practical methods which would lead to the determination of the magnitude of the residual stresses actually found in the parts and to discover the influence of the relaxation treatment recommended and the actual machining of the forged rough shapes on the stress relief.

Spectroradiographs of austenitic steels could indicate only local variations in stress and only qualitatively. Quantitatively, no indications could be given regarding the size of the residual stresses below about 10 kg/mm^2.

Destructive methods, particularly those of Mathar and Mesnager-Sachs, are capable of giving a precision of the order of a few kg/mm^2. Mathar's method, by the way, was developed by Professor Soete of the University of Ghent, and may be applied at the present time and in certain cases in a non-destructive form. This Mathar-Soete method, proceeding by the partial relief of stress, gives a precision of 2...3 kg/mm^2.

Finally, it is also possible, by working with a control sample, to relieve entirely the constraint gauges used in the preceding method and thus push the precision of the measurement to $\pm$ 1 kg/mm^2.

In considering the state of the problem and the information available on the importance of residual stresses, their relief and the method of their measurement, we were led to conclude that if it should be found by experience that relaxation annealing greatly reduced the magnitude of the residual stresses, no quantitative result giving the precise values which they might attain after different heat treatments, was available.

Consequently and in order to clarify this problem of residual stresses in austenitic steels, the Centre d'Etude de l'Energie Nucléaire undertook to collect data on residual stresses, after various stages of fabrication, on a sample similar to the one from which we made the 3 1/2" extensions destined for the nuclear model of the reactor core.

We recommended that the control part be divided into four sections to be examined in the following states:

1st piece: after high tempering
2nd piece: after high tempering and machining
3rd piece: after high tempering and annealing
4th piece: after high tempering, annealing and machining.

(Fig. 1 shows a sample after machining).

A comparison of the measurements made on the four sections should permit one to show:

a) The magnitude of the residual stresses after high tempering;

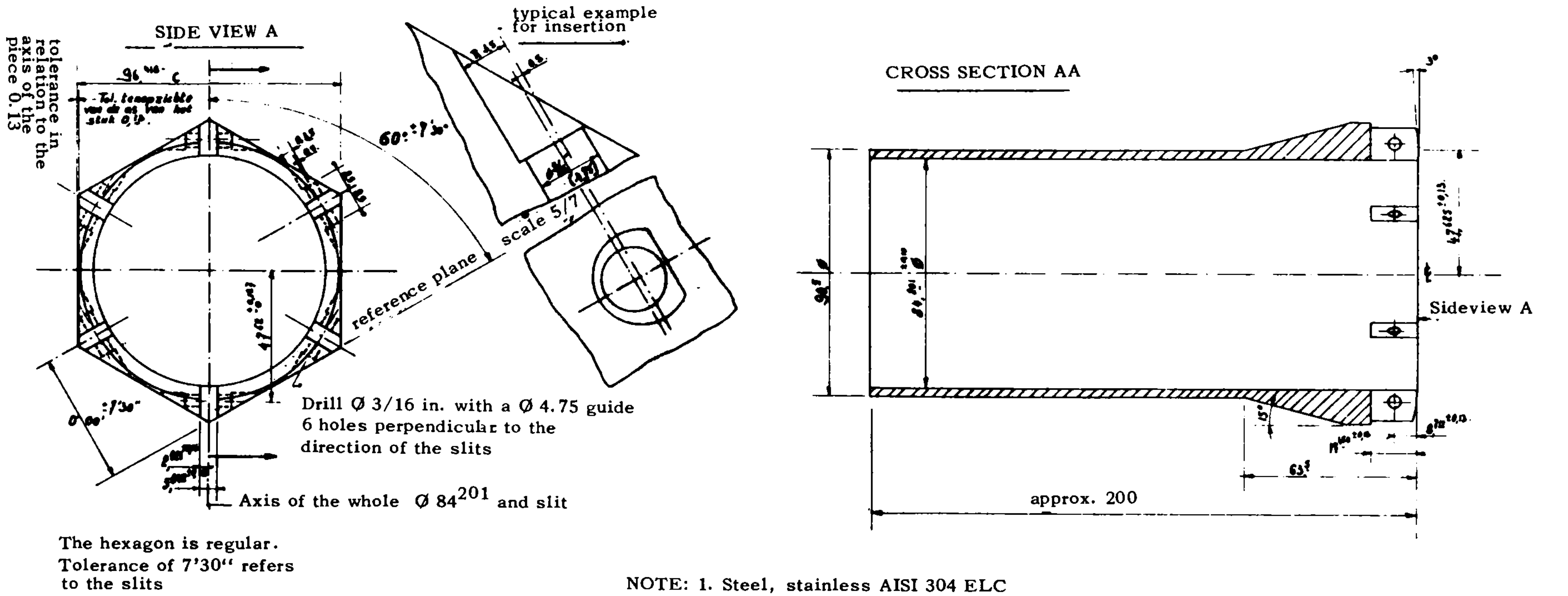
b) The rate of relaxation brought about by the annealing;

c) The influence of machining on the relief of stresses after high tempering and after annealing, respectively.

As this part is sacrified, it is permitted to bring about the total relief of stresses and then compute the results of these measurement with a precision of $\pm$ 1 kgs/mm^2.

The results of these tests which are proceeding now, are not as yet known to us, so that it is not as yet possible for us to give you any ample information on this subject.

Another example regarding which we have encountered a technical problem still being pursued, is that of the execution of the central collar of the reactor tank.



NOTE:
1. Steel, stainless AISI 304 ELC
2. Finish ∇63
3. External corners cut, R=0.25
4. Internal corners cut, R=0.8
5. Plane A perpendicular to the axis of the hexagon: tolerance 0.013
6. 0.25 maximum slant of the hole Ø 84.201
7. Tolerance ± 0.4 for lengths } unless otherwise specified
8. Tolerance ±30' for angles } unless otherwise specified

According to Plan No 731 of Brugeoise and Nivelles

BR2	CMF Extension. Test Sample for the determination of residual tensions
Echelle: 1/1	Office of Nuclear Studies Brussels, Belgium
	2/2077

Figure 1

To give an idea, the approximate dimensions of these collars are 1525 mm in height and 1100 mm internal diameter. Its thickness, after shaping, welding, and assembly to the other sections of the body should be brought to 21.4 mm. Moreover, strict tolerances were imposed on the builder, the Association Momentanée ACEC-La Meuse, both as to the thickness as well as the interior diameter of the collar.

One of the difficulties of this problem lies in the fact that the finishing of this collar could take place only after the assembly of the various sections that make up the body of the tank, which should then have a length of about 7.50 m.

Moreover, the boring and lathe operations of this central collar would have relieved part of the stresses which would have formed during its shaping, and the deformations which would have followed would have been difficult to control. The problem thus arose of annealing this collar before its assembly with the other parts.

The material constituting the body of the tank is an aluminum alloy of the type AA 5052.

Now, the Boiler Code mentions that relief annealing is not necessary or desirable in the case of non-ferrous materials.

Moreover, in the course of the "Symposium on Aluminium Pressure Vessels" which took place in London on 28 October 1958, the same point of view was adopted although no explanation was given for it.

However, such an explanation appears to us plausible and we are giving it here in brief:

The shaping of a collar, such as the central cylindrical collar of the reactor tank for the BR-2, gives the latter a rate of cold working equal to several percent.

Now it happens that light alloys with Mg, Mg-Si,... are, in the cold-working state (1...5%), subject to a sudden grain enlargement if relief annealing is not carried out with the greatest of precautions. This phenomenon has been described by saying that alloys are sensitive to grain enlargement in critical cold working. The critical value of the deformation, moreover, is smaller, the higher the temperature of the final annealing and the longer its duration.

The recrystallization temperature of the Al-Mg family is about 330°C. One should therefore avoid approaching this temperature, even if only locally, and one should keep at 20 - 50 C below that figure. Such restoration treatment would cause the stresses to disappear without resulting in a recrystallized state.

This problem is still a difficult one at the present time.

Indeed, the builder proceeded to effect annealing at 390°C with the help of blow torches; the temperature was maintained at that value for a half hour. As for the value of the maximum local cold working, it is difficult to affirm that it stayed everywhere below 5 percent.

Samples were taken from the edge of the collar and submitted by the research laboratory ISSOIRE of the CEGEDUR to micrographic tests.

A magnification of x 400 of a longitudinal section in one of the samples, after electrolytical polishing, yielded the micrograph No 1 (Fig. 2). Let us note that the intermetallic constituents are distributed not only in a very fine fashion but even very regularly for a thick sheet.

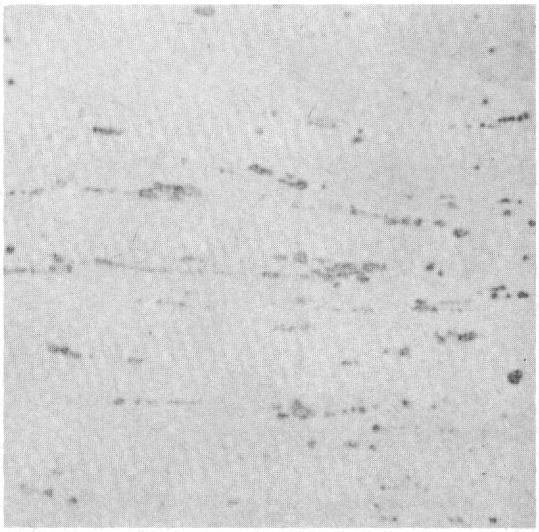


Figure 2

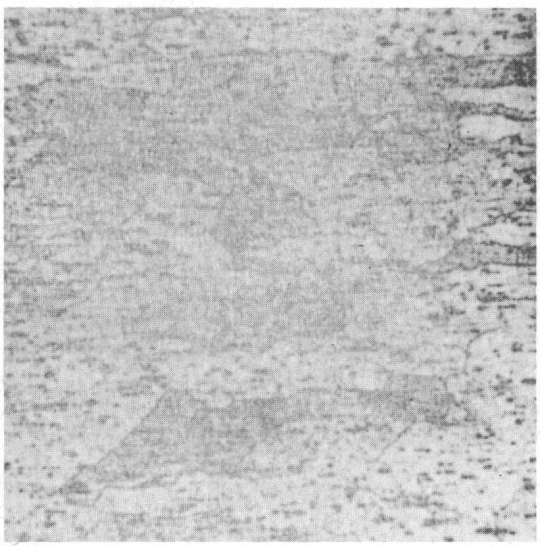


Figure 3

The micrograph No 2 (Fig. 3), after being treated by an HCl-HF reagent and a 100 x magnification to show a sufficient number of grains, indicates that treatment at 390°C did not result in any enlargements of the grains and that their size is normal and regular.

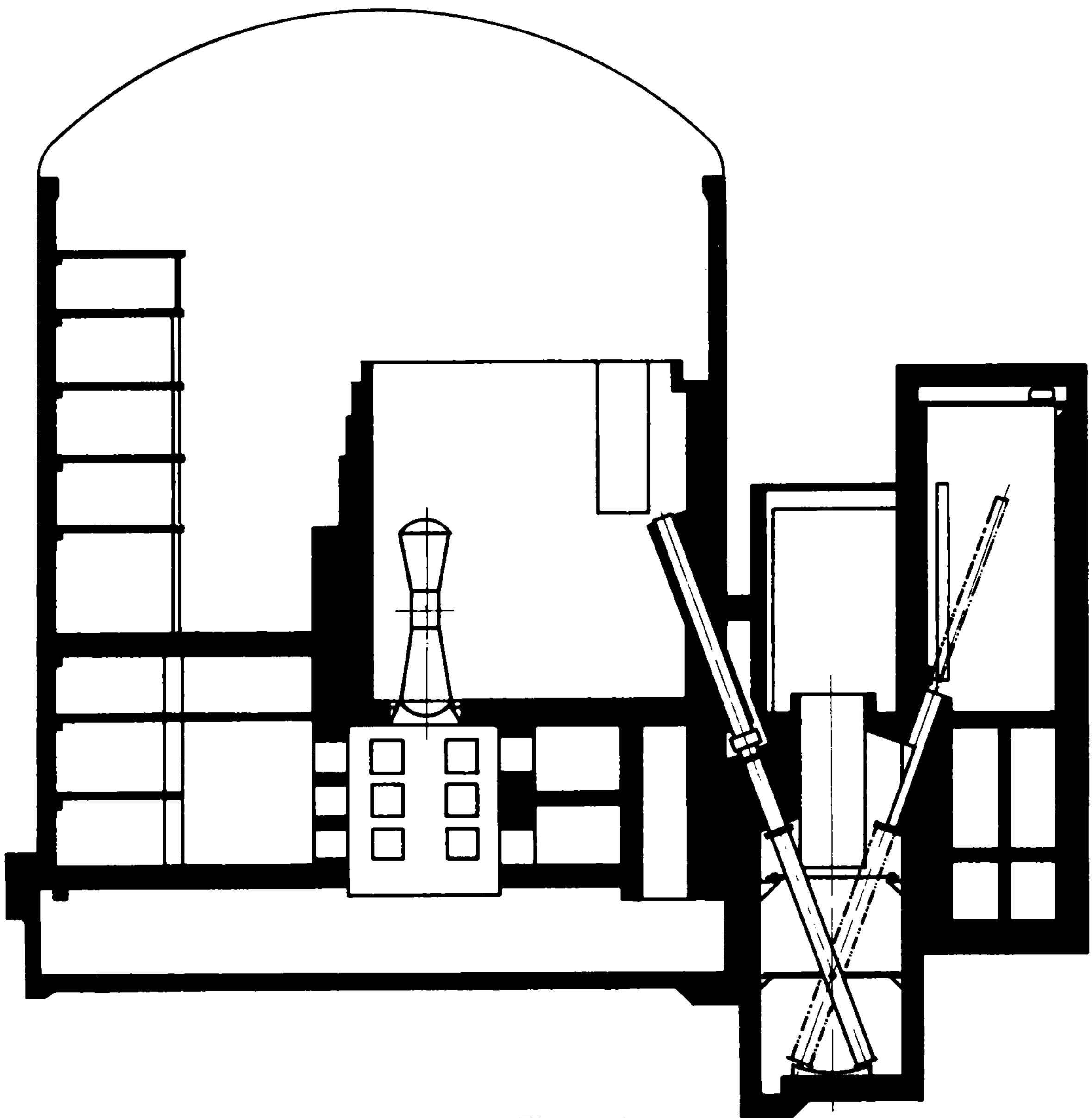


Figure 4

In conclusion, annealing at 390°C has given satisfactory results from the metallurgical standpoint. Moreover, the stresses having been relaxed, the finishing of this central collar could be effected with a better control of deformations.

Let us now study the special designs adopted for the execution of certain equipment of the Br-2. Here we shall chose two examples in the waste disposal channel for the waste disposal of radioactive elements via water.

A first sketch (Fig. 4) gives the layout of the assembly and makes it possible to locate:

- the transfer tube connecting the pool of the reactor structure with the well provided in the auxilliary structure;
- the transfer machine which is to be immersed in the well;
- the transfer tube connecting the well with the hot cell;
- the guidance casing and the rolling bridge provided in the hot cell and serving to introduce elements in the latter. This rolling bridge moreover, serves the entire dismantling cell.

The service conditions of this waste disposal channel are mainly:

- as for the medium: demineralized, non-degassed water whose temperature is between 10°C and 40°C;
- as for requirements: transfer equipment whose maximum weight will be 15 tons, distributed equally over a length of 4 m minimum. Other, much lighter equipment, could have a length of 11.3 m, maximum.

Let us begin by giving some details of the transfer tube connecting the pool of the reactor structure with the well of the auxilliary structure (Fig. 5): its design had to take into account the following points:

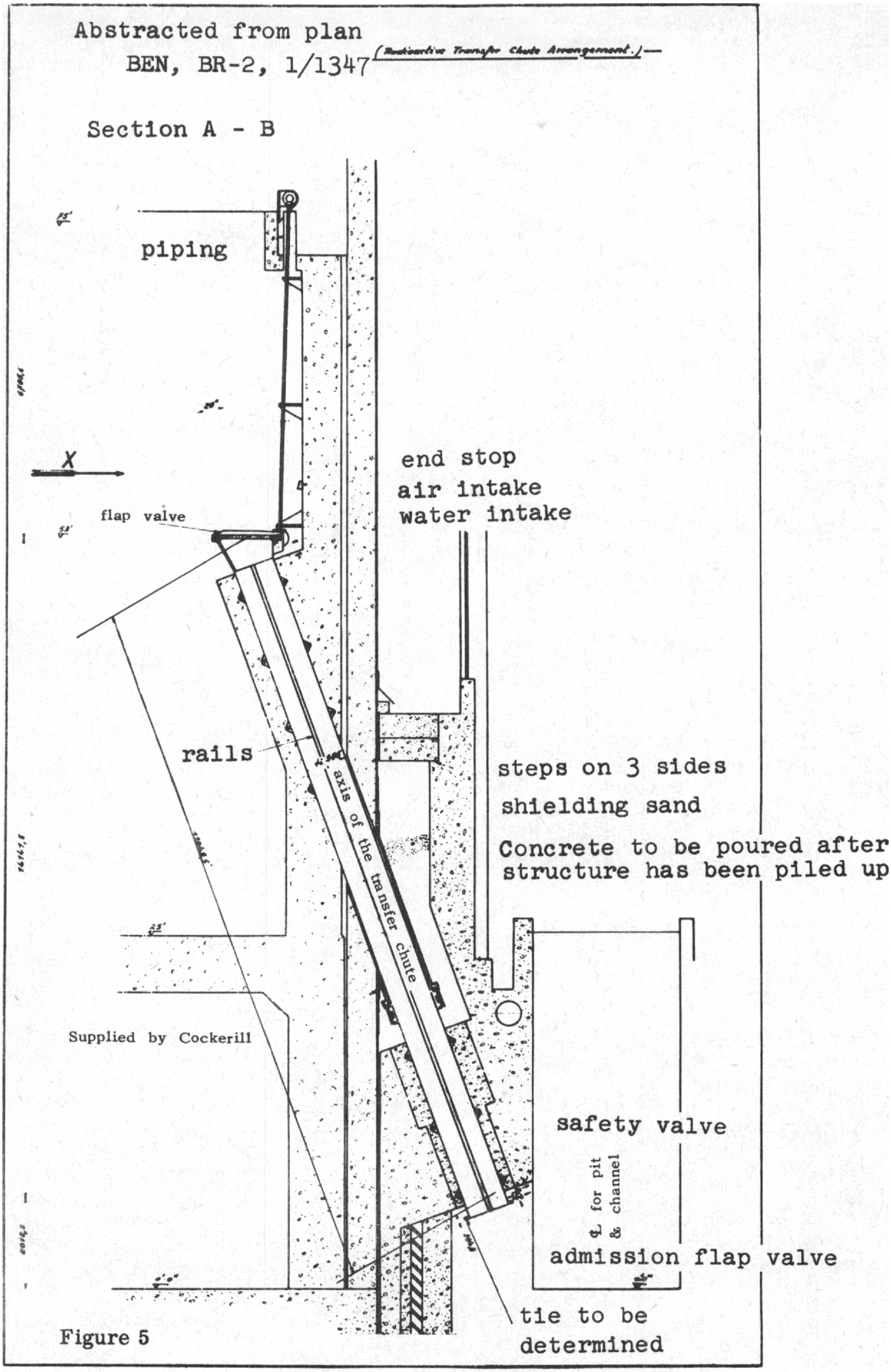
1. The foundation section of the reactor structure situated under the transfer tube will, in the course of construction and after finishing, undergo a differential loading in relation to the adjoining foundations of the auxillary structure. These foundations are independent. Now, the upper end of the tube is to be connected in a radiation proof manner to the facing of the pool of the reactor without any intermediate weak, mechanical pieces. A simple way of carrying out this condition is to anchor the tube near the facing of the pool and to fasten the latter by welding.

The lower end of the tube, on the other hand, should likewise be fastened in a radiation-proof manner to the facing of the well.

It therefore follows that the length of the tube should be variable and the difference in levels between the two ends will be subject to fluctuations.

2. As the extreme temperatures of the tube in the course of the operation deviate by a certain value from its temperature at the time of anchoring, it should be possible for it to expand freely, particularly in its longitudinal axis.

3. As the radiation-proofing of the shells of the reactor structure should be secured at the opening for the passage of the tube, these two elements should be connected in a suitable manner. Now, the variations in the atmospheric conditions will bring about a radial and vertical displacement of the solid penetration of the shell and surrounding the tube about its central point. Moreover, no appreciable force can be applied to the shell locally. Its unification with the tube should be elastic.



Abstracted from plan
BEN, BR-2, 1/1347
Section A - B
piping
X
end stop
air intake
water intake
flap valve
rails
axis of the transfer chute
steps on 3 sides
shielding sand
Concrete to be poured after structure has been piled up
Supplied by Cockerill
safety valve
℄ for pit & channel
admission flap valve
tie to be determined

Figure 5

The solution found consists in a tube made up of two sections: the upper section is anchored in the wall of the reactor pool as we have just stated while the lower section is anchored at the bottom in the wall of the well. For reasons which we cannot go into here, each of the sections comprises an overhang length.

The ends of the two pieces made facing each other, end up in a tie on which joints of natural rubber are fastened; they are fixed to piping connected solid, in the penetration.

The two sections of the transfer tube are thus connected in a radiation-proof manner and the radiation-proofing of the reactor structure is at the same time assumed with respect to the outside of the tube.

As far as the inside of the tube is concerned, this radiation-proofing is maintained during the transfer of elements by means of two gaskets placed at the ends of the tubes and forming an air lock.

Given the conditions of the medium and the temperature, such natural rubber gaskets may last for many years. Their replacement, however, should easy: by emptying the tube, they can be replaced from the inside, the tube diameter of 905 mm allowing easy access.

As for the two sections proper (Fig. 6), they make up two longitudinal plates, machined in U-shape and destined to receive the roller holders of the charges to be transferred. A length of detachable rail assures to the joints continuity of guidance between the two sections, at the same time giving them freedom of expansion as well as relative displacement resulting from the differential loading of the foundations.

In conclusion, we shall examine the case of the rolling bridge of the hot cell located at the other end of the waste disposal channel.

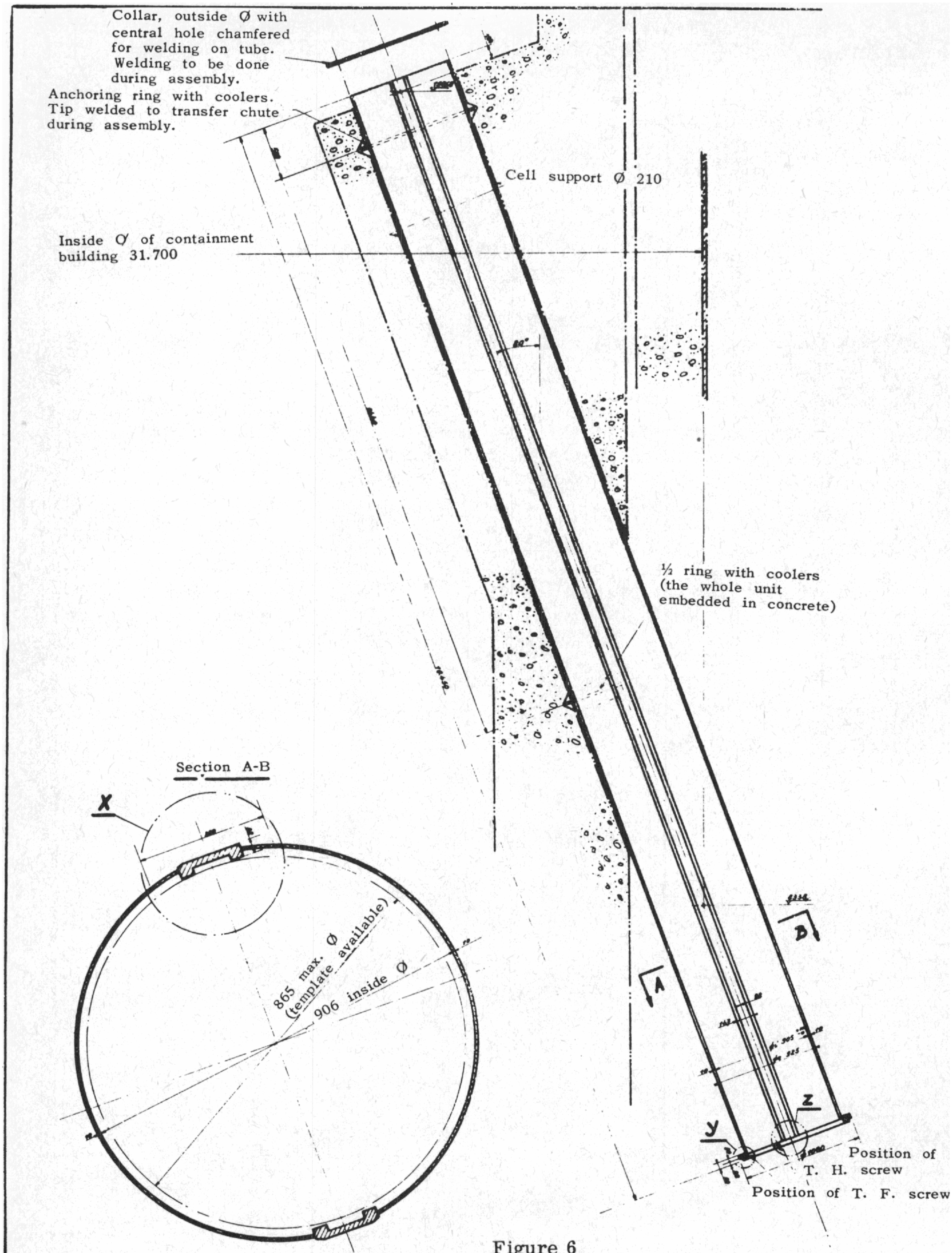
The characteristics of that cell, particularly as regards the rolling bridge, are the following:

1. Calculation of the cell for an activity of 60,000 C Na at 2.75 Mev; this poses serious problems for the insulation of the electrical leads which must be enclosed there.

2. Transfer tube opening in the cell at an angle of 20^{o} with respect to the vertical; this involves oblique tractions from the bridge.

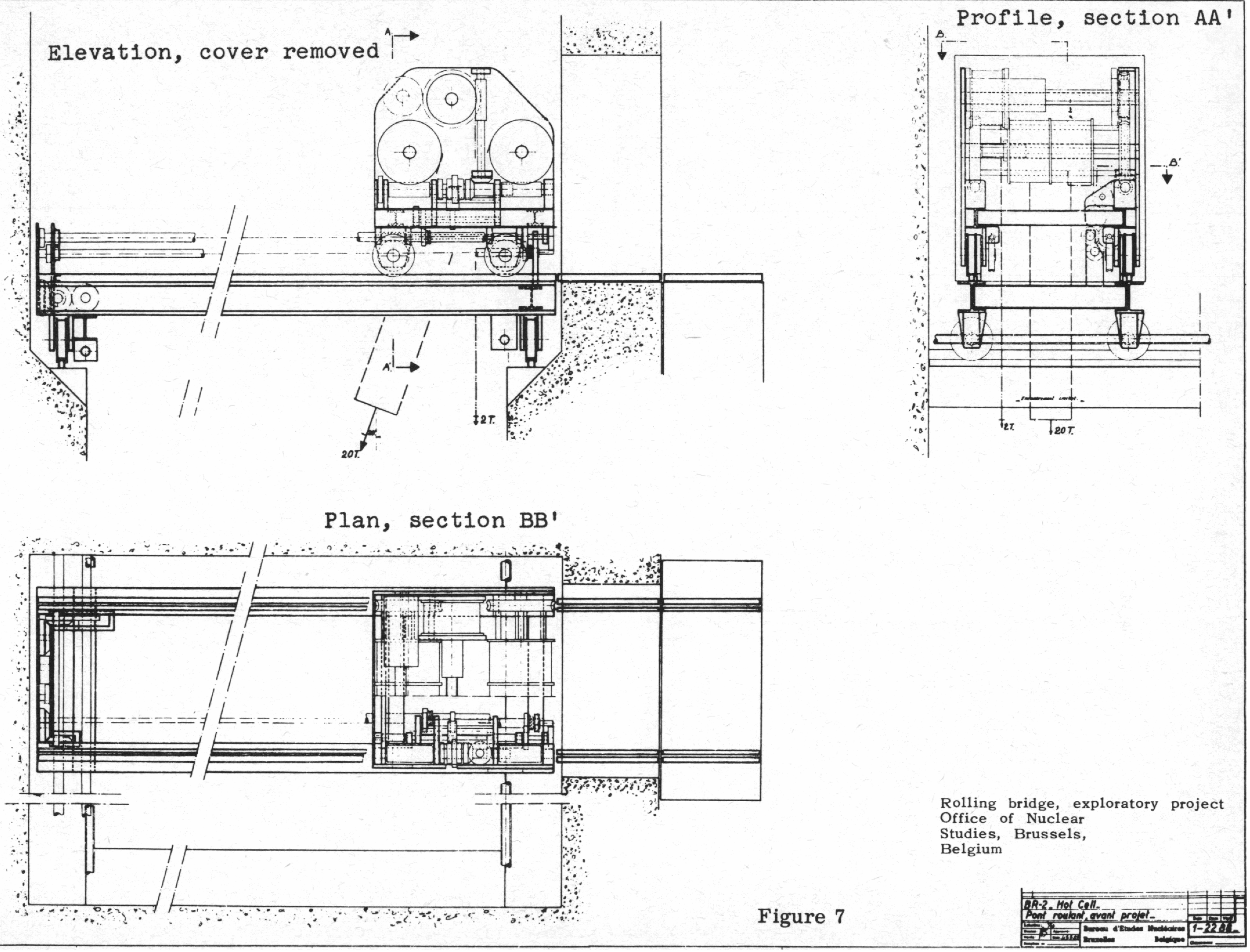
3. Need of the bridge to serve an area as large as possible; this involves reduced approaches.

The first of these characteristics gave the preference to a rolling bridge of the standard type whose drive motors are outside of the cell (Fig. 7). There are only purely mechanical transmissions inside the cell. Transmission is effected by means of bevel pinions and either worm screw and nut or rack bars or by means of splined shaft and idle pinions. These mechanisms, moreover, present advantages which are directly workable. Indeed, they permit the carriage to be buttressed on its rack-bars and thus effecting oblique tractions without any difficulty, thus satisfying the second characteristic which we mentioned. Moreover, the winches of the carriage activating the two hooks -- respectively 20 T for the main hook and 2 T for the auxilliary hook -- which the bridge is provided with, are affected, in the last



Collar, outside Ø with central hole chamfered for welding on tube. Welding to be done during assembly.
Anchoring ring with coolers. Tip welded to transfer chute during assembly.
Cell support Ø 210
Inside Ø of containment building 31.700
½ ring with coolers (the whole unit embedded in concrete)
Section A-B
X
865 max. Ø (template available)
906 inside Ø
A
B
Z
Y
Position of T. H. screw
Position of T. F. screw

Figure 6



Rolling bridge, exploratory project Office of Nuclear Studies, Brussels, Belgium

Figure 7

analysis, by a worm screw system. The irreversibility of such system makes it possible to uncouple one or the other of the hooks when they are on. This also makes it possible to provide only one shaft for the drive of the hoisting movement, the passage from one hook to the other being by means of coupling.

Another noteworthy particularity is the following: in order to permit an almost vertical hoisting of the charges and at the same time safeguard the desired approaches, a special device of winches was devised in which the drums are displaced laterally while they rotate so as to leave motionless the axis of each of the hoisting ends.

Attention is also drawn to the facility of maintaining the bridge. The frame, holding the drive shafts for the travel of the carriage and the hoisting of the hooks -- whose maintenance is extremely limited -- requires special measures before it is accessible. On the other hand, the carriage, holding the winches and the hoisting hooks, may be extracted without any difficulty. For this purpose, it suffices to bring the bridge opposite the door on top of the cell and extending outwardly the rolling track and the splined shaft activating the carriage displacement.

We hope that these few examples may have served to illustrate to you the nature of some problems met in the construction of the BR-2 and the solutions that have been applied.

ANALYSIS OF THE METHODS OF MEASURING THE SHIELDING EFFECT OF THE RADIATIONPROOF SHELL OF A BR-3 NUCLEAR REACTOR

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1. INTRODUCTION

1.1. Purpose of Shielding Tests

The radiationproof shell that houses the reactor is intended to hold back the radioactive products that could otherwise be spread into the atmosphere in case of a serious accident of the reactor.

In such an accident a certain pressure develops in the shell; this is called the accident pressure. For the BR-2 this pressure is estimated at $1 kg/cm^2$, effective pressure.

Due to this pressure a certain amount of radioactive products may escape through the leaks of the shell and can create, near the reactor, a dose of radiation whose maximum permissible value limits the maximum permissible value for the leak rate of the shell.

The specified value for this maximum leak rate corresponds to a relative decrease of the amount of active substances of 21×10^{-4} in 24 hours.

If it is assumed that the active substances are uniformly mixed with the gases present in the shell, then this leak rate will also represent the relative drop in absolute pressure, that can be tolerated in 24 hours.

1.2. Difficulties of Shielding Tests

Only to mention that a decrease of $0.5^{\circ}C$ in the mean temperature will cause a pressure drop identical to that resulting from the maximum tolerable leakage in 24 hours, would show clearly the difficulties of shielding tests.

2. METHODS OF MEASUREMENT

2.1. Fundamental Relation

In order to disclose the principles that underly the methods of measuring the effectiveness of shielding, we will write the relation between the leak rate and the variations of other quantities that control the pressure of the system. This relation, derived in appendix 1 of this report is of the following form:

$$- \frac{dP}{P} = - \frac{dp}{p} + \frac{dp_{vs}(\theta_{\sim})}{d\theta_{\sim}} \frac{d\theta_{\sim}}{p} - \frac{dV}{V} + \frac{dT}{T} \qquad (1)$$

which shows how the relative leakage can be calculated from the measurement of pressure variations, the mean air temperature and the dew-point of the shell volume.

The analysis of the errors involved in these measurements will enable us to evaluate the possibility of obtaining sufficient accuracy for measuring a leak $- dP/P$ of 21×10^{-4} per 24 hours at the end of one day of measurements.

This analysis will also indicate to us the most suitable method of measurement.

2.2 Analysis of the Errors of Measurement

Let us assume that we investigate the feasibility of measuring a leakage of 21×10^{-4} per 24 hours with an observation of one day. For such a measurement to have significance, the various errors in the evaluation of the terms of equation (1) should be small compared to 21×10^{-4} viz. less than 2×10^{-4} each.

a) Measurement of the Pressure

In the course of the test we brought the pressure up to 0.6 kg/cm^2, effective, or $p = 1.6$ kg/cm^2

The condition: $\varepsilon_p = \frac{\Delta p}{p} < 2 \times 10^{-4}$

Where Δp is the permissible error for the pressure measurement gives:

$$\Delta p < 1.6 \times 760 \times 2 \times 10^{-4} = 0.24 \text{ mm Hg},$$

This is possible with a mercury barometer with vernier.

b) Volume Variations of the Shell

The expansion of the shell may be either thermal or mechanical.

For the thermal expansion we have:

$$\left(\frac{dV}{V}\right)_\theta = 3\alpha d\theta_{cl}$$

where α is the linear thermal expansion coefficient of the metal and is on the order of $1.2 \times 10^{-5}/^{\circ}C$,
$d\theta_{cl}$ the heating of the shell metal.

The condition: $\mathcal{E}_v = 3\alpha\Delta\theta_{cl} < 2 \times 10^{-4}$ gives

$$\Delta\theta_{cl} < \frac{2 \times 10^{-4}}{3 \times 1.2 \times 10^{-5}} = 5.5^{\circ}C.$$

or, in other words, a variation in shell temperature of $5^{\circ}C$ corresponds to an error of 2×10^{-4} and correction is not necessary unless the temperature variation markedly exceeds $5^{\circ}C$. This correction does not give any difficulty.

The temperature of the shell metal was measured with thermistors, devices that are extremely sensitive and reliable, provided they have been previously aged and calibrated.

For the mechanical expansion we may write similarly:

$$\frac{dV}{V} = 3\,\frac{\sigma}{E},$$

where now E represents the modulus of elasticity and σ the mean mechanical stress.

For $E = 2 \times 10^6$ kg/mm^2 and $\sigma = 20$ kg/mm^2 we have:

$$\frac{\Delta V}{V} = 3 \times \frac{20}{2 \times 10^6} = 3 \times 10^{-5}.$$

The effect of the mechanical expansion is therefore negligible.

c) Variations of the Humidity

Let us now examine the effect of variations of the humidity represented by the term $\frac{dp_{vs}(\theta_r)}{d\theta_r}\,\frac{d\theta_r}{p}$ of equation (1).

Let us assume that the dew-point is $10^{\circ}C$, as was actually the case during the measurements.

$\frac{dp_{vs}(\theta_r)}{d\theta_r}$ is in effect the angular coefficient of the vapor curve giving $\frac{dp_{vs}(\theta_r)}{d\theta_r} = 0.635$ mm Hg/$^{\circ}$C

The condition $\mathcal{E}_h = \frac{dp_{vs}(\theta_r)}{d\theta_r}\,\frac{\Delta\theta_r}{p} < 2 \times 10^{-4}$,
where $\Delta\theta_r$ represents the error in dew-point measurement gives

$$\Delta\theta_r < \frac{2 \times 10^{-4} \times 2 \times 760 \text{ mm Hg.}}{0.635 \text{ mm Hg}/^{\circ}C} = 0.48^{\circ}C$$

Hygrometric tables for the case of room temperature $15^{\circ}C$ show that such a variation in dew point corresponds to a humidity variation of three percent only!

This means that in case the air humidity varies during measurements, either because of condensation due to a temperature drop, or because of water evaporation from the equipment in the shell, it is important to measure hunidity in order to apply the required correction.

It should be noted that, contrary to the mean temperature measurements, dew-point measurements may be made in only one spot because the vapor pressure corresponding to the dew point is assumed to be uniform throughout the shell. The measurement of this dew-point within 0.5°C may be carried out by means of ventilated psychrometer.

d) Variation of the Mean Temperature

Finally, let us examine the effect of the mean temperature, represented by the last term $\frac{dT}{T}$ in equation (1).

The condition $\mathcal{E}_T = \frac{\Delta T}{T} < 2 \times 10^{-4}$ gives, for instance, for $T = 300^{\circ}K$

$$\Delta T < 2 \times 10^{-4} \times 300 = 0.06^{\circ}C!$$

for $\mathcal{E}_T = 4 \times 10^{-4}$, representing 20 percent of the leak rate per 24 hours we would obtain $\Delta T \cong 0.1^{\circ}C$.

Here we face the most difficult part of the measurement, that is to measure a variation of the mean temperature with an error of the order of one tenth of one degree centigrade when temperatures may vary as much as six degrees between different points of the shell!

All the proposed methods of measurement concentrate on this problem.

2.3 Compensating Method, Called Reservoirs Method

A solution for this problem was found by the compensating method using reference reservoirs, a method that was tried in the U.S. during the tests of the Pressurized Water Reactor Shell of Shippingport and the Boiling Water Reactor of Dresden and that was also used by us for the BR-2 shell.

As illustrated in Figure 1 it is based on the comparison of the pressure inside the metal shell with the pressure in interconnected reference reservoirs dispersed throughout the shell.

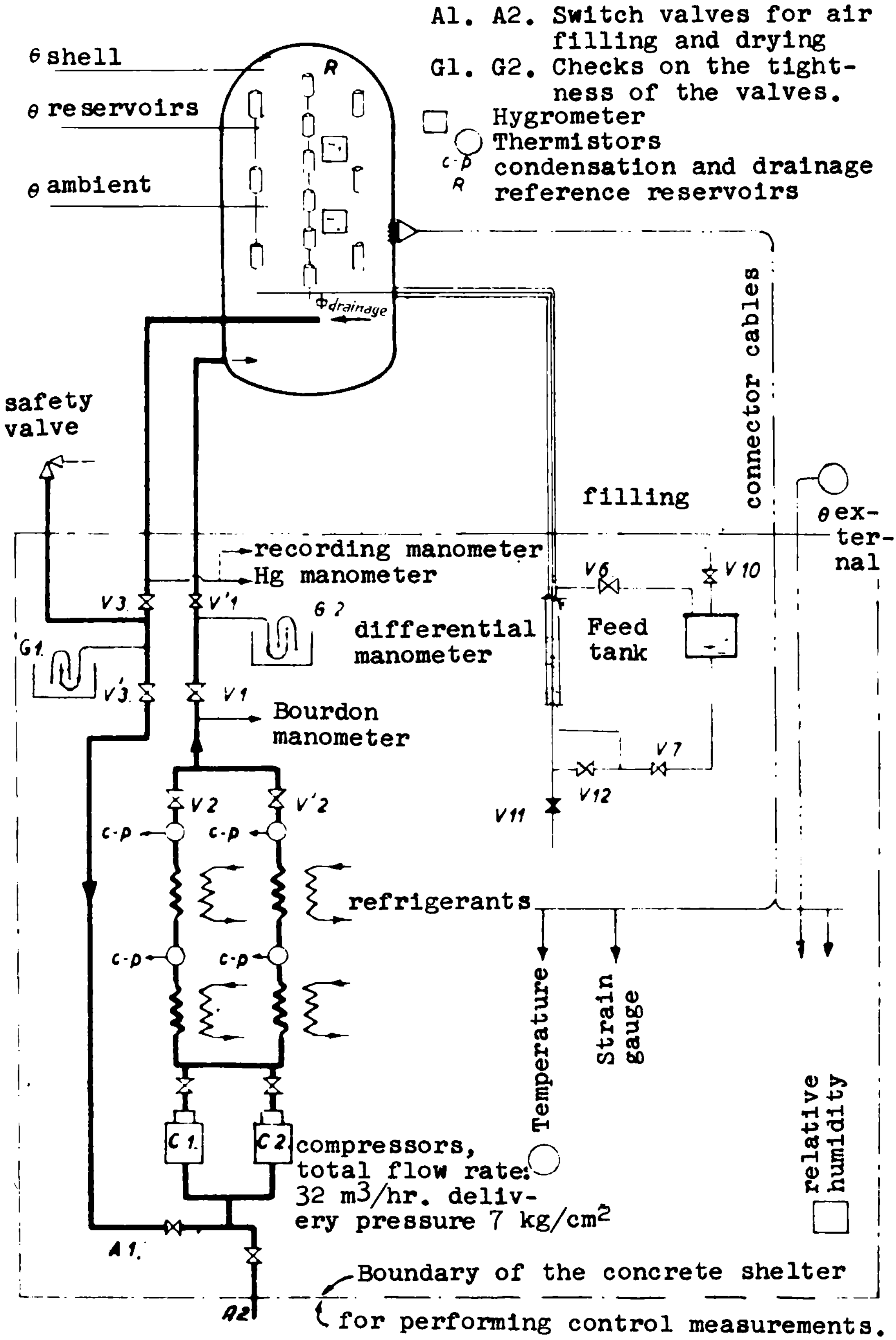
The pressures are compared by means of a differential manometer, filled with water.

The shell and the reservoirs are made to communicate before the measurements in order to balance their pressures. This takes place through the differential manometer that has been drained of its liquid.

If the temperature inside the reference vessels would at any given instant equal the temperature in the shell, pressure fluctuations due to temperature variations would be the same in the two systems and would compensate each other exactly, and thus the differential pressure measured at the manometer would correspond exclusively to the pressure drop attributed to leakage, to the thermal expansion, and to the humidity, in case the latter two are not the same in the two systems.

The great advantages of this method is that instrumentation is considerably simplified. The reading to within 0.15 mm

Figure 1: Schematic Representation of the Control Systems Used For Testing the Metal Shells of the Two Reactors BR-2 and BR-3, at Present Under Construction at Mol, Belgium



of a precision barometer is replaced by a reading of a differential manometer that may be made to within 0.15 x 13 = 2 mm because water is used.

And in addition the delicate temperature measurements needed for the calculation of the correction $\frac{dT}{T}$ are eliminated.

The practical drawback is the laborious installation of the reservoirs and their connections, as they have to be absolutely tight.

The practical value of this method becomes apparent when we examine to what extent it enables us to eliminate the corrections and what are, on the other hand, the sources of errors associated with it and that we had to face in the course of the measurements.

The relative excess pressure read on the differential manometer may in principle be put in the following form, derived from equation (1):

$$\frac{\delta p}{p} = -\left(\frac{dP}{P}\right)_{\ell} + \left(\frac{dP}{P}\right)_{r} + \frac{\delta T}{T} - \frac{\delta V}{V} + \frac{dp_{vs}(\theta_{\nu})}{d\,\theta_{\nu}}\;\frac{\delta\theta_{\nu}}{p} \qquad (2)$$

where δx represents this time in a general way the excess of the mean value of X in the reference system in relation to its value in the shell.

Thus: $\frac{\delta p}{p}$ is the relative excess pressure of the reference system,

$\frac{\delta T}{T}$ the relative "thermal" excess pressure,

$\frac{\delta V}{V}$ the relative thermal expansion,

$\frac{dp_{vs}(\theta_{\nu})}{d\,\theta_{\nu}}\;\frac{\delta\theta_{\nu}}{p}$ the excess pressure due to a difference $\delta\theta_{\nu}$ between the dew-points of the two systems.

$-\left(\frac{dP}{P}\right)_{\ell}$ is the total leakage in the shell, taken from the start of the measurements, when all δ are assumed to be zero.

$-\left(\frac{dP}{P}\right)_{r}$ is the leakage of the reference system itself, an additional source of error introduced by the method.

The symbols δ denote thus in general the various shortcomings in compensation of the temperature, humidity or expansion effects of the systems.

Let us analyze these different errors briefly, using the results obtained in the analysis of errors made previously.

a. Thermal Compensation

We have seen that in order to obtain a meaningful thermal correction for a 24 hours test we require a measurement of the mean temperature within 0.1°C.

For the case of the compensation method this condition becomes that the mean temperature of the reservoirs should not differ more than 0.1°C from the mean shell temperature.

There are two sources of error δT.

a.1 The first cause is the non-uniform temperature distribution. In this respect the reference system acts as an enormous gas thermometer giving automatically the mean temperature of the various reservoirs, an average that is balanced with respect to the volumes of these reservoirs.

This brings up the following problem: how to select the number, size, and distribution of the different reference reservoirs in the shell in such a way that δT will not exceed $0.1^{\circ}C$ for any assumed temperature distribution.

We have found (reference 1) a limiting solution for this problem by investigating the unfavorable case where each reservoir behaves as a source of heat; this would happen in the case of a sudden rise in air temperature.

A simplified solution of this problem shows that if the distribution of the reservoirs is such that the temperature difference between two adjacent reservoirs be DT, then the errors δT is smaller than 3/2 DT.

There is actually little chance that there should be a temperature inversion in the vertical direction because of convection currents. 3/2 DT is thus reduced to 2/2 DT.

It is therefore desirable that the temperature difference between any two adjacent reservoirs be not higher than $0.1^{\circ}C$.

The distribution selected is shown in Figure 2.

Measurements have shown that this distribution does not satisfy the condition mentioned above, even during periods of exceptional thermal stability, such as are found at night with an overcast sky, and it may rather be expected that this difference is of the order of $0.3^{\circ}C$. This indicates that a measuring time of three times 24 hours is needed in order to obtain significant results.

a.2 The second cause of the error δT is due to thermal time lags connected with the temperature exchanges between the air and the reservoirs that occur during temperature changes.

It is possible to write: $\delta T = \tau \frac{dTr}{dt}$

giving the relation between the difference δT in the mean temperatures of a reservoir and the surrounding air, τ, times constant connected with the thermal exchange and $\frac{dTr}{dt}$ the derivative of the reservoir temperature with respect to time. The temperature variations observed during the day were of the order of $1^{\circ}C$/hour. To have $\delta T = 0.1^{\circ}C$ τ should be equal to six minutes.

In view of the relatively large distance between two reservoirs (approximately six meters) and the immobility of the air outside and inside the reservoirs we can see that such a small time constant was not reached and that it was necessary to wait until night for making significant measurements.

It is actually this high value of the time constant that limits the reservoirs method.

Forced convection with ventilators, as used in the U.S.A. would have undoubtedly accelerated thermal exchange and made the temperature more uniform.

We did not make use of ventilators because we wanted to

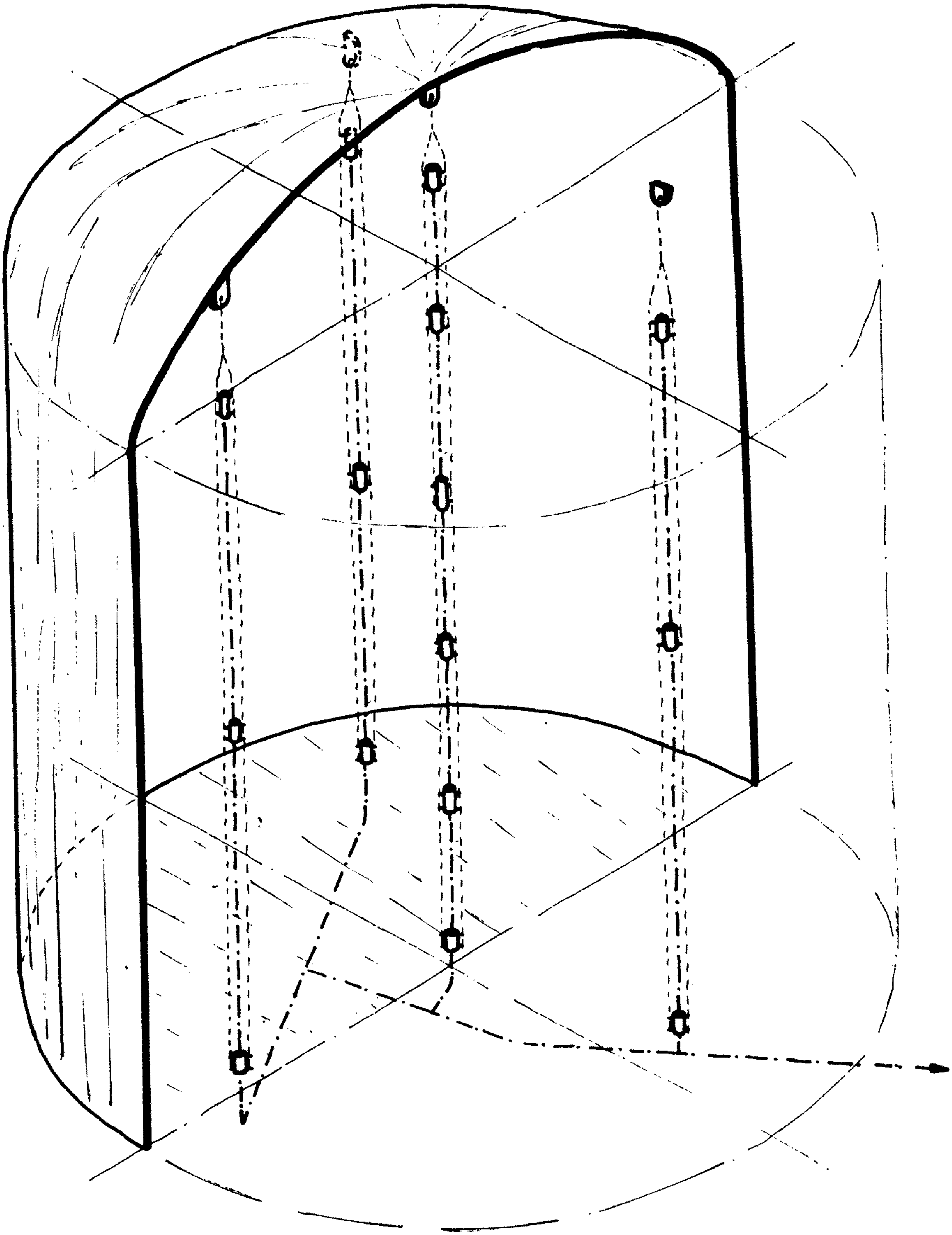


Fig. 2

Scale 1/250 along the axes

try a method that would be applicable to a container divided in cells where forced convection would be difficult.

b. Error $\delta V/V$

As for the error due to the different expansions of the two systems, it was found that, during the useful period of measurements, temperature differences between the two systems remained below $5^{\circ}C$; corrections were therefore not needed.

c. Humidity Correction

Humidity variations require, on the other hand, particular attention because of their important effects.

Because the inside of the reservoirs is not accessible for measuring the degree of humidity, it is essential that the reference air should be sufficiently dried and that we can be sure that no condensation whatsoever will occur during measurements.

We have calculated in this connection that (reference 1) the evaporation of manometer water is liable to change humidity beyond permissible limits. To avoid evaporation we have introduced in the manometer tubes the oil before the water so that it would be covered with an oil layer that would stop evaporation.

For humidity measurements we have used a Hartmann and Brauwn psychrometer that should in principle be sensitive within the required tolerances but that gave wrong readings, apparently because of too little ventilation.

This instrument was supplemented by a vapor pressure hygrometer TESA that transmitted readings by closed circuit television. The latter broke down during the measurements on BR-2 after behaving well during the tests on BR-3.

We had as a last recourse two hair hygrometers whose readings were not in agreement.

Because of the uncertainty of humidity measurements and in order to avoid making corrections, it was decided to dry the air in the shell and in the reservoir, thus avoiding all condensation during the measurements.

Drying was carried out as follows:

The compressors used compressed the air to 7 kg/cm^2. The air was then cooled in water heat exchangers and it was freed of condensate before expansion. This would eliminate 6/7 of the humidity present in principle.

Preliminary drying was carried out by circulation of the air in a closed circuit.

Contrary to expectations, it was impossible to reduce the dew-point as much as was desired and thus the test pressure was forcibly limited to 0.6 kg/cm^2, effective, in order to avoid reaching the dew-point. In appendix 2 it is explained that the rate of leakage corresponding to this pressure is $21 \times 10^{-4} \times 0.6$ or 12.5×10^{-4} per 24 hours.

d. Leaks

A new source of error introduced by the reservoirs method is the effect of a leak in the reference system.

All possible precautions were taken to reduce these leaks to a minimum:

- All reservoirs and their connections with the differential manometer were tested with soapy water.
- All connections were soldered.
- The differential manometer was arranged (see Figure 3) so that no part whatsoever of the system was in contact with the atmosphere.

The total volume of the reference reservoirs was made relatively large, of the order of 1 m^3, to reduce the effect of a leak.

It should be noted, however, that a rather large volume is unfavorable for the time constant τ that limits the method, as we have seen above.

One can note in Figure 3 the action of valves allowing to reach the two systems separately and to fill or drain the manometer at will without establishing any contact between the reference system and the atmosphere.

Finally, the whole reference system was tested for tightness by bringing the pressure in the system up to 1 kg/cm^2 and observing the change in this pressure.

It can be shown (see appendix 3) that if it is required that the error $(\frac{dp}{p})_r$ due to the leakage of the reference system be less than 2×10^{-4} per 24 hours, the criterion of tightness becomes the following formula:

$$(\frac{dp}{p})_r /24h < 4 \times 10^{-3}/24\ h$$

where $(\frac{dp}{p})_r$ is the maximum relative drop per 24 hours of the system of reference during its test.

A variation in temperature of 1°C, on the other hand, gives:

$$\frac{dT}{T} = \frac{1}{300} \cong 3.3 \times 10^{-3}$$

It is thus seen that it is possible to test the reference system for effectiveness of shielding during 48 hours provided it is possible to measure temperature variations of the system within 1°C, which is perfectly feasible.

The reference system used for BR-3 has been also used for Br-2. The operations of transporting the system from one container to the other has undoubtedly been harmful to the soldered joints; when the system was retested for tightness leaks were discovered in such numbers that the use of the system had to be confined to the central column only.

Under these conditions compensation was much poorer than for BR-3.

2.4. Thermometric Wire Method

The considerations of the preceding sections and the insight gained from the previously made tests on the shielding effect on the Br-3 shells have shown us where lie the weaknesses

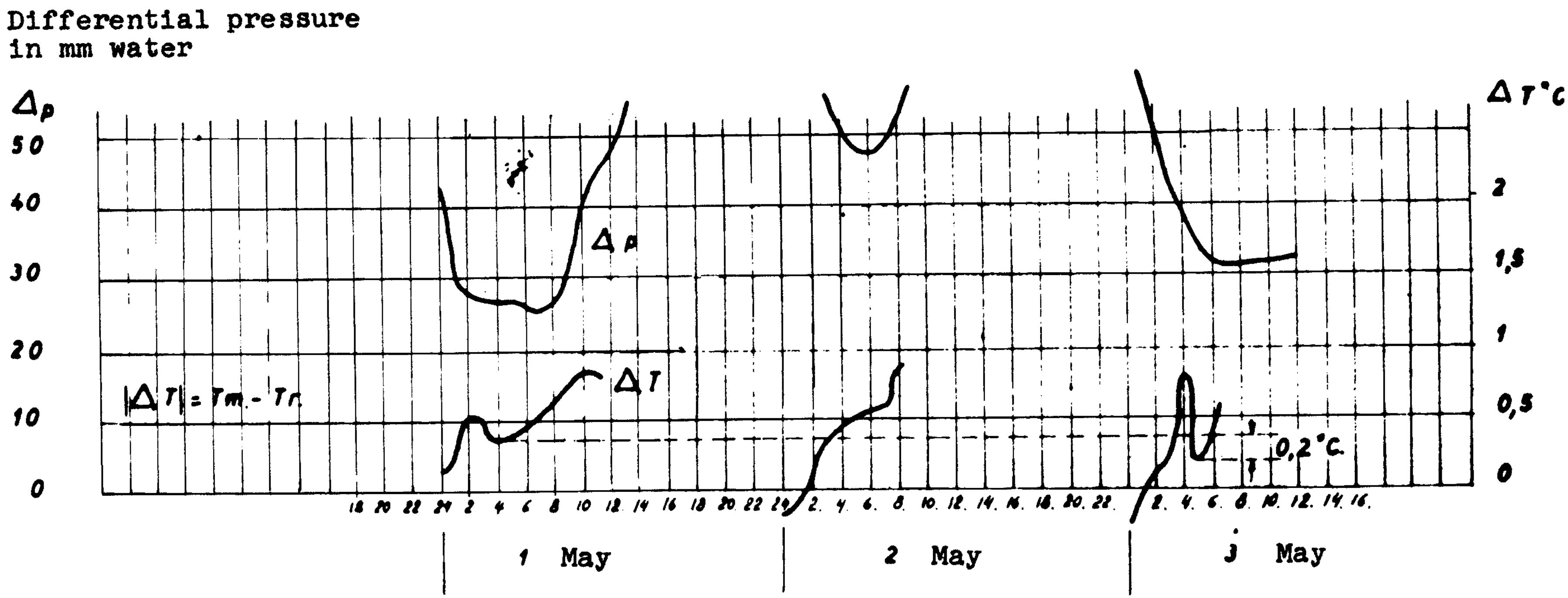


Figure 3. Test of the Quantitative Method of Measurement of Leaks in the Shell of Reactor BR-2

Δp = pressure variation of the shell relative to the pressure of the reference system.
$|\Delta T|$ = temperature variation of the shell relative to the temperature of the reference system.

of the reservoirs compensation method and how it is necessary to reduce as much as possible the thermal time constant of the system, intended for measuring the mean air temperature in the shell.

This consideration caused us to use for this measurement a long copper wire running through the entire shell. Its resistance, measured without difficulty with a precision bridge indicates the mean temperature along the wire.

This method was utilized with BR-2 concurrently with the reservoirs method, so that the two methods could be compared, and could supplement each other.

3. RESULTS OF THE MEASUREMENTS

3.1. Compensation Method of the Reservoirs

The diagram in Figure 4 illustrates the measurements executed during three consecutive nights. In it we find: the differential pressure δp and the difference δT.

It is found that the reference system does not give the required compensation even approximately, especially outside the periods of stability at night.

During the BR-3 tests a complete reference system was available but only a few accurate temperature measurements in the shell were available.

The opposite was true for BR-2 where the reference system was reduced to only the central column, but the mean air temperature was measured with improved accuracy by the thermometric resistance. The curves show that the correlation δp-δT is less good than for BR-3. The explanation of this apparent discrepancy lies in the fact that the regions of measurement of the two temperature-sensitive systems are not the same.

Taking the results of the measurements made the first night at 4 o'clock and the third night at 5 o'clock we find a variation of $\delta p = 8$mm water.

The temperature correction would induce us to increase this value by $\frac{0.2^oC}{300} \times 16 \times 10^3$ mm water or 10.5 mm water and we would obtain a total of 8 + 10.5 = 18.5 mm water. The maximum permissible leakage would on the other hand result in a change of

$$21 \times 10^{-4} \times 1.6 \times 10^4 \text{ mm water} = 20 \text{ mm/24 hour},$$

or 40 mm in 48 hours.

We thus conclude that the measured leakage equals zero, with an accuracy of 50 percent in 48 hours.

3.2. Method of Thermometric Resistance

In the first and the third nights the following values were tabulated:

	1st. night (4 a.m.)	3rd. night (5 a.m.)	Difference
p (mm Hg)	1174.3	1160.2	14.1
resistance (Ω)	38.75	38.23	0.52

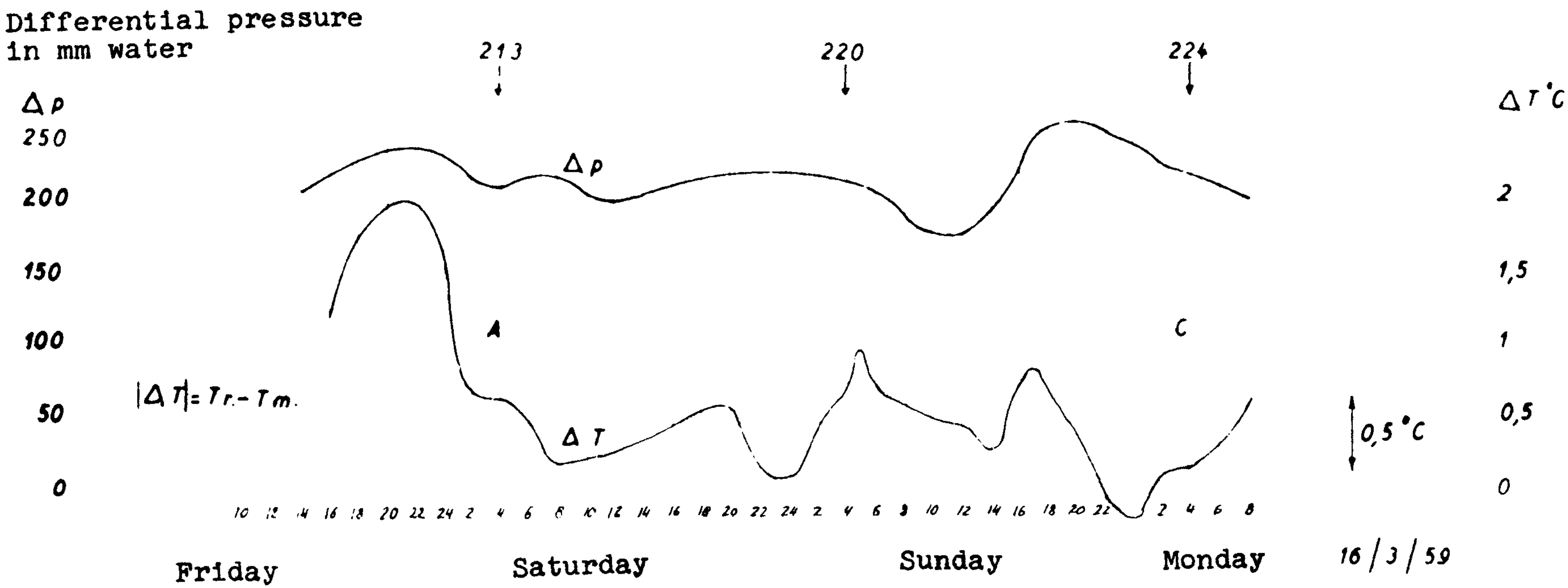


Figure 4. Test of the Quantitative Method of Measurement of Leaks in the Shell of Reactor BR-3.

Δp = pressure variation of the shell relative to the pressure of the reference system.

$|\Delta T|$ = temperature variation of the shell relative to the temperature of the reference system.

Taking for the value of the temperature coefficient 4.25×10^{-3} per ^{o}C we have:

$$\Delta T_m = 0.52 \times \frac{1}{37.2 \times 4.25 \times 10^{-3}} = 3.29^{o}C$$

This difference in temperature accounts for a pressure difference of $\frac{3.29}{300}$ or 12.1×10^{-2},

whereas $\frac{\Delta p}{p} = \frac{14.1}{760 + (760 \times 0.6)} = 11.6 \times 10^{-3}$

The difference between these two values, i.e., 5×10^{-4} representing the measured leak compares favorably with the following values of the maximum leak after 24 hours:

$$21 \times 10^{-4} \times 2 \times 0.6 = 25 \times 10^{-4}$$

The wire method brings us to the same conclusion that the leakage is zero but this time with an accuracy of $\frac{5}{25} = 20$ percent.

CONCLUSIONS AND RECOMMENDATIONS

From the tests conducted on BR-2 and BR-3 we may derive the following conclusions and deductions:

1^{o} The compensation method is delicate and time-consuming and it does not enable us to obtain significant results in less than two or three days of observation.

It would be of interest to replace the reservoirs by tubes of 3 to 4 cm diameter, to fill the reference system with a gas that is dry and a good conductor of heat and to increase the convection of the air by a few ventilators.

If these improvements were to be carried out we think that the method could give results in 24 hours.

2^{o} It is preferable and much more economical to perfect a suitable method for measuring the humidity, rather than trying to dry the air.

3^{o} We think that the method of the thermometric resistance of which we have not yet used all the possibilities is capable of replacing the reservoirs method advantageously, if it is improved by the use of forced convection. It is more simple, more rapid, and more economical.

We are convinced that a judicious distribution of the wires and forced convection would make a measurement of the rate of leakage feasible within less than 24 hours and at a reduced pressure.

We would recommend that it be demonstrated, before starting any further experiments, that it is possible to reduce the time constant associated with the heat exchange of a copper wire centered in an air column of 6 meter diameter with forced convection to approximately one minute. This can be done by simple experiments.

Forced convection could be provided by the ventilation of the container.

APPENDIX 1

Derivation of the Fundamental Relation

The ideal gas law, applied to the air in the shell is:

$$p_a V = P r_a T \tag{1}$$

where: p_a is the absolute partial pressure of the air,
V the volume occupied by the air in the shell,
P the weight of this air,
T the mean absolute temperature of this air,
r_a the ideal gas constant for air.

For the absolute partial pressure p_v of the vapor it is possible to write:

$$p_v = p_{vs}(\theta_r) + \delta(\theta_r)\, r_v\, (T-T_r) \tag{2}$$

where: θ_r is the dew-point of the vapor, in ^oC,
T_r the same, but in ^oK,
$P_{vs}(\theta_r)$ the saturation vapor pressure at θ_r,
$\delta(\theta_r)$ the specific gravity of the vapor at θ_r.

This relation states that we obtain the pressure $p_v(T)$ by heating the weight of air $\delta(\theta_r)$ that exerts a pressure $P_{vs}(\theta_r)$ at θ_r, by an amount of $(T-T_r)^oC$ at a constant volume; the coefficient $(\partial p_v / \partial T)_{V=constant}$ may be expressed by $\delta(\theta_r).r_v$ where r_v is approximately equal to the ideal gas constant for the vapor.

Finally we have:

$$p = p_a + p_v \tag{3}$$

Differentiation of the relations (1), (2) and (3) gives respectively:

$$\frac{dP}{P} = \frac{dp_a}{p_a} + \frac{dV}{V} - \frac{dT}{T} \tag{4}$$

$$dp_v = \frac{dp_{vs}(\theta_r)}{d\theta_r}\, d\theta_r + r_v[(T-T_r)d\delta(\theta_r) + \delta(\theta_r)d(T-T_r)] \tag{5}$$

$$dp = dp_a = dp_v \tag{6}$$

By means of (6) we substitute in (4) for dp_a/p_a:

$$\frac{dp_a}{p_a} = \frac{dp-dp_v}{p_a} \cong \frac{dp-dp_v}{p}$$

(5) on the other hand gives:

$$\frac{dp_v}{p} = \frac{dp_{vs}(\theta_r)}{d\theta_r}\frac{d\theta_r}{p} + \frac{r_v(T-T_r)}{p}\, d\delta(\theta_r) + \frac{r_v\,\delta(\theta_r)}{p}\, d(T-T_r) \tag{7}$$

and if in the denominators of the last two terms of this equation we substitute $\delta_a r_a T$ for $p \cong p_a$ we obtain:

$$\frac{dp_v}{p} = \frac{dp_{vs}(\theta_\sim)}{d\theta_\sim}\frac{d\theta_\sim}{p} + \frac{r_v(T-T_r)}{\delta_a r_a t}\, d\delta(\theta_\sim) + \frac{r_v\,\delta(\theta_\sim)}{\delta_a r_a t}\, d\,(T-T_r) \quad (8)$$

It is readily seen that the two last terms are infinitesimals of a higher order than the first term because of the smallness of $\frac{T-T_r}{T}$ and of $\frac{\delta(\theta_\sim)}{\delta a}$;

By neglecting the two last terms and carrying out the two substitutions we obtain the fundamental relation:

$$\frac{dP}{P} = -\frac{dp}{p} + \frac{dp_{vs}(\theta_\sim)}{d\theta_\sim}\frac{d\theta_\sim}{p} - \frac{dV}{V} + \frac{dT}{T} \quad (9)$$

APPENDIX 2

Calculation of the Correction in the Leak Rate, Required When the Test is Performed at a Pressure Different from the Pressure for Which the Leak Rate was Specified

Symbols Used:

p : absolute pressure
p_e: gauge pressure
p_b: barometric pressure
δ : specific gravity of air
μ : kinematic viscosity
v : speed of air in the crack
L : length of the crack
D : equivalent width of the crack
G : rate of flow by weight
Q : rate of flow by volume
V : volume of the shell
P : weight of the air in the shell
T : absolute temperature of the air.

Two extreme cases may be hypothetically considered in relation to the way of air loss through cracks:

a - Laminar Flow

would undoubtedly occur in the case of porous spots or cracks in the metal.

Laminar flow is governed by the equation

$$\frac{p_e}{\delta} = \frac{32\mu v L}{\delta D^2} \quad (1)$$

that is: $p_e \sim Q = \frac{G}{\delta}$ (2)

b - Turbulent Flow

This would occur for instance in case of a leak in one of the connections. Here we have:

$$\frac{p_e}{\delta} = \xi \frac{v^2}{2g} = \xi \frac{Q^2}{2gS^2} \tag{3}$$

or

$$p_e \sim \frac{G^2}{\delta} \tag{4}$$

Let us now express the rate of flow G in terms of the leak rate

$$\frac{1}{p}\frac{dp}{dt}.$$

The ideal gas law pV = PrT gives us by differentiation:

$$G = \frac{dP}{dt} \sim \frac{dp}{dt} \quad \text{for constant V and T} \tag{5}$$

We also have $p = \delta rT$, or $p \sim \delta$ for constant T (6)
Substituting the values of G and δ obtained from (5) and (6) in (2) and (4) we obtain:

$$p_e \sim \frac{1}{p}\frac{dp}{dt} \quad \text{for laminar flow} \tag{7}$$

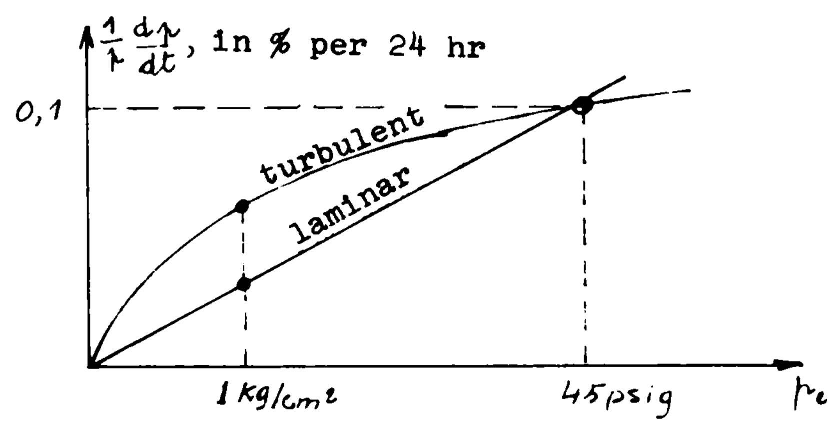
and

$$p_e \sim \frac{1}{p}\left(\frac{dp}{dt}\right)^2 = \left(\frac{1}{p}\frac{dp}{dt}\right)^2 (p_e + p_b) \quad \text{for turbulent flow,} \tag{8}$$

and these expressions give the relation between test pressure and rate of leakage under the two hypotheses made.

The leak rate specified for the BR-3 at 45 psig is 0.1 percent per 24 hours and the test pressure was 1 kg/cm²

The following diagram shows the change of the leak rate as a function of the pressure p_e for the two hypotheses about using the rate of leakage of 45 psig as the common starting point.



It is seen that the assumption of laminar flow is the more conservative one.

The leak rate at 1 kg/cm² will equal under these conditions:

$$1\% \text{ per 24 hr.} \times \frac{14.223}{45} = 0.0316\% \text{ per 24 hr.}$$

The test for BR-2 was performed at 0.6 kg/cm^2 whereas the specification of 21 x 10^{-4} per 24 hour was given for 1 kg/cm^2.

For 0.6 kg/cm^2 we should take $21 \times 10^{-4} \times \frac{0.6}{1} = 12.5 \times$ $\times 10^{-4}$ per 24 hour.

APPENDIX 3

Setting up a Tightness Criterion for the Reference System

During the test an excess pressure of δp is applied to the reference system. The leak of this system itself after a certain time is given by $-(\frac{dP}{P})_r$ and it should be small compared to the maximum permissible leak from the shell at the end of the same period.

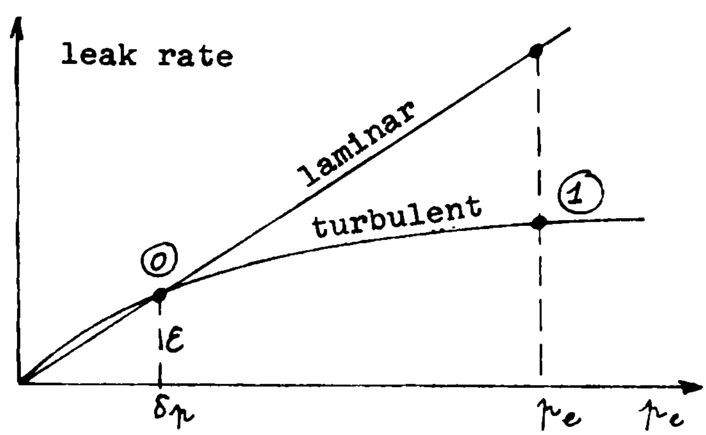
In the case of BR-2, for instance, the reservoirs shall be considered tight if

$$-(\frac{dP}{P})_r < 2 \times 10^{-4}/24 \text{ hr.}$$

(the maximum permissible leak of the shell at the pressure 0.6 kg/cm^2 is 12.5 x 10^{-4} per 24 hours)

To make the measurement of the leakage of the reference system easier its pressure is brought to $p_e \gg \delta p$.

What is the maximum leak rate permissible at the pressure p_e so that at the excess pressure δp the leak rate be less than $\mathcal{E}$?



Let us plot the curves of the leak rates as a function of the excess pressure p_e for the two hypotheses of flow and let us assume that the leak rate is the same for both cases at δp and that it is there equal to $\mathcal{E}$; (see appendix 2).

Let the numbers 1 and 2 indicate the conditions at δp and p_e in that order.

It is seen that here it is the turbulent flow that is the more conservative choice, contrary to what happens in the problem discussed in appendix 2; the assumption of turbulent flow is more conservative since it requires a leak rate $(\frac{1}{P} \frac{dP}{d\ell})_{r,1}$ smaller than the one that would follow from the other hypothesis.

Relation (8) of appendix 2 gives us:

$$\frac{\left(\frac{1}{P}\frac{dP}{dt}\right)_{r,1}}{\left(\frac{1}{P}\frac{dP}{dt}\right)_{r,o}} = \sqrt{\frac{p_e}{\delta p} \cdot \frac{p_o}{p_e+p_b}} \qquad (1)$$

and the criterion of tightness is expressed by:

$$\left(\frac{1}{P}\frac{dP}{dt}\right)_{r,1} < \mathcal{E} \sqrt{\frac{p_o}{\delta p} \cdot \frac{p_e}{p_e+p_b}} \qquad (2)$$

For BR-2 we will have $\mathcal{E} = 2 \times 10^{-4}$ per 24 hour, for instance, $\frac{\delta p}{p_o} = 12.5 \times 10^{-4}$, $p_e = 1$ kg/cm^2 $p_e + p_b = 2$ kg/cm^2

with these values:

$$\left(\frac{1}{P}\frac{dP}{dt}\right)_{r,1} < 4 \times 10^{-3}/24 \text{ hr}$$

A temperature change in time of 1°C gives on the other hand:

$$\frac{dT}{T} = \frac{1}{300} = 3 \times 10^{-3}$$

It is thus shown that the temperature correction which must be made to apply the tightness criterion of the reference system, is much easier than that to measure the tightness of the shell.

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TESTS FOR THE RADIATIONPROOF METALLIC SHELL OF THE BR-2 REACTOR

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Part One

TESTS FOR RESISTANCE

INTRODUCTION

Before approaching the problem of the tests themselves, we should recall the general characteristics of the radiationproof shell of the BR-2 reactor.

The shell is in the form of a cylindrical body with a flat bottom and torispherical dome. The diameter of the cylindrical portion is 31.700 meters, the total height about 44 meters. Soudotenax 41 steel is used.

Thicknesses used are:

8 mm for the bottom
18 and 22 mm for the cylindrical portion
26 mm for the dome

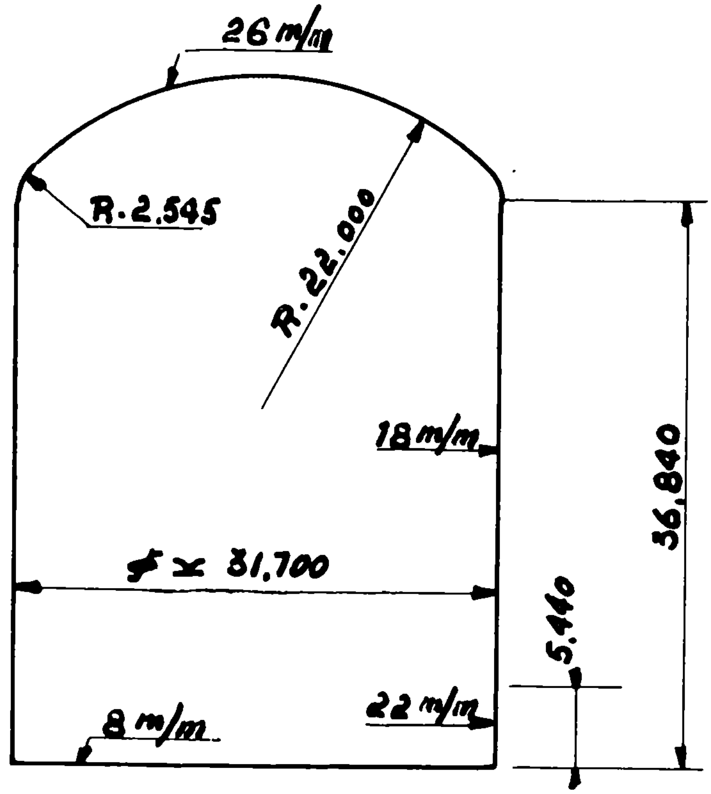
Execution Tolerances

Si D = theoretical diameter

$\phi = D \pm 0.1$ percent D.

Several openings -- carriage lock chambers, staff lock chambers, entry chute, ventilation, pipe passage-ways, etc... -- are allowed, at different levels, in the cylindrical body. These shells, designed to make the reactor almost completely radiationproof with respect to the atmosphere in view of preventing any dispersion of radioactive products that might be released in case of accident, require great care and control in their construction.

This is one of the reasons why all of the welding was checked 100 percent by radio and by ultra-sonics.



II. OBJECTS OF THE TESTS

After long discussion and with the unanimous agreement of the Chief of Operations, the CEN, its advisors, the BEN, Sofina and the Vincotte Association, and of the builder, the Cockerill-Ougrée Company, it was decided to proceed to a compressed air test as soon as the major work was completed and before the internal assembly.

The following objectives were pursued:

1. - to be completely sure that the shell would resist a pressure 20 percent greater than that developed in case of accident, i.e., 1 kg/cm^2.

2. - Check whether the welds are perfectly radiationproof and allow any defects discovered at the time of examination to be corrected before work on the inside and outside of the shell interferes with proper accessibility.

3. - Make tensometric measurements to establish the absence of dangerous stresses and a reasonable agreement between the results as previously calculated and obtained on a reduced model with experimental reality.

4. - Allow the perfecting of the method for measuring the leakage rate, which method will have real meaning only at the time of the second test when the shell will be completely finished with the various lock chambers and openings outfitted with doors and permanent covers.

The summary of this fourth point will be taken up in the second part of this report.

III. RESISTANCE TEST

The air test was adopted because of the practical impossibility of performing a hydraulic test which would have imposed demands on the metallic shells too different from those for which the shells had been calculated.

It is appropriate to emphasize that such air testing was not without some danger, and, for ordinary reservoirs, this method is not permitted, and water testing is imperative.

In fact, in the case of hydraulic testing any rupturing resulting from escaping water causes a rapid drop in pressure, which is not the case for air, which is much more compressible. Consequently, there is a danger that the rupturing will be propagated in an explosive manner, and there is a real danger of accident. However, a detailed study showed that this danger, whose probability is infinitesimal, would be limited in its effects to a small area of the order of a few tens of meters around the container. Safety considerations, then, do not justify abandoning these highly interesting tests which alone could dispel any apprehensions regarding the behavior of these essential structures in the event of nuclear accident. Obviously, measures had to be taken to eliminate any risk of personal accidents. These measures were taken by responsible departments of the C.E.N. Close approach to the containers during periods of rising pressure was strictly limited to a small number of technicians in charge of supervision and measurements for whom concrete shelters, suitably protected, had been prepared in the structures under construction near the shells.

IV. PERFORMING THE TESTS

All existing openings were closed by a series of temporary covers. Most of the latter were welded to assure their being leakproof. Openings with drilled flanges were closed by means of bolted lids made radiationproof by a plastic joint tightened between the flange and the lid.

The inlet and outlet of compressed air was accomplished by a special lid provided with two tubes welded directly to the compressed air circuit.

The passage of electric cables (cables for strain-gages, thermistors and other measuring devices) through the shell wall was assured by a terminal plate especially designed for this purpose.

The shell was put under pressure by means of two compressors in parallel with a total output of approximately 32 $N/m^3/min$.

The filling and drain pipe was provided with a series of stop valves used when during measurements at constant pressure.

An adjustable trip valve directly connected to the radiationproof enclosure was used for safety.

Among the different types of control to be carried out on the shell during testing under pressure, were, principally, acoustic and soapy water controls.

Acoustic controls proved ineffective.

Soap control, on the other hand, gave complete satisfaction because the sensitivity of this method is very great, and it can be carried out relatively easily.

Two controls were decided on at approximately 1/2 the test pressure: the first when the pressure rose and the second when it fell.

In order to insure the safety of the technicians assigned to this task, it was agreed that all soap testing would be performed after a halt of approximately two hours at constant pressure.

In addition, at the time of increasing pressure, the pressure should be raised to 25 percent above that which is desired for the soapy water tests and maintained at this level for one hour before being reduced to the amount desired where a new halt of two hours, as we have just said, preceded the test itself.

These measures were carried out scrupulously; only the halt times were reduced on certain occasions.

The theoretical and practical test diagrams bring out clearly, however, these various precautions.

At the time of "soaping" properly speaking, a number of teams of three men, a workman and two controllers were spread out on the scaffolding erected for this purpose.

After an energetic brushing with a metallic brush, the welds were carefully washed with soapy water. Immediately thereafter, the welds were examined by the two, above mentioned controllers.

All the welds including the temporary ones on the radiation-proof covers were controlled.

For example, the time necessary to control all the welding on the shell of the BR-2 runs approximately six hours with ten teams of three men each.

It is interesting to note that in addition to the two controls of the shell welding with soapy water, several intermediate controls were performed locally on the spots which required it the most: welding of angles, welding of face plates, all the reinforcement rings and openings.

V. TENSOMETRIC MEASUREMENTS

We shall elaborate on the tensometric measurements made by the departments of Cockerill-Ougrée in collaboration with those of Professor Soete of the University of Ghent.

We have already emphasized that Cockerill-Ougrée has great experience in the technique of tensometry with strain-gages with very diverse applications: diesel motors, rolling stock, cranes, buttresses, frameworks, etc...

However, in the present case, the problem presented special aspects and individual difficulties which required detailed studies and minute precautions in order to obtain coherent results: difficulties of access and, therefore, the unfavorable position of the technicians who install the gages, great length of the connections, great length of testing time, exposure of gages to weather extremes -- wind and rain, etc...

The whole method of measuring by ohmic extensometers had to be analyzed point by point in the light of these particular conditions as well as the manner of requirements and distribution of stresses that could be predicted. Tests bearing on well determined points were carried out first in the shop: effect of the length of connection cables, effect of humidity, effect of contact resistances at the switch, radiationproof state of the terminal plate allowing cables to pass through to the gages inside the tank, etc... Let us state right now that all this preparatory work was fruitful.

Perfectly coherent results were obtained for the BR-2 with an insignificant amount of variation.

In fact, 97 percent of the gages responded coherently, and the precision of reading obtained was of $\pm$ 50 microdistortions, which corresponds to $\pm$ 1 kg/mm^2.

Let us recall briefly the principle of ohmic extensometers. A fine metal wire wound several times is stuck so that it adheres perfectly to the wall whose state of stress we wish to know. During the test, it follows that strain of the wall faithfully, expanding and contracting with it; a proportional variation of the ohmic resistance results. The latter is measured by the method of the Wheatstone bridge by comparing with another gage as comparable as possible located nearby at a spot which is not being tested. The reading on the bridge is directly proportional to the strain of the wall according to the direction of the gage. Equations derived from the theory of elasticity lead from the deformations to the stresses.

We should emphasize two points. First, the above method only brings out the resistance variations of the gage from the time of installation to that of measurement and, consequently, the variations in the state of stress of the wall in this interval. Preexisting stresses, for example, lamination stresses or residual stresses from welding, are not registered at all. Secondly, the relations connecting stresses with strains are only valid in the elastic region. When the elastic limit is reached in measurement, the gage continues to faithfully measure local strains, but the value of the stresses can no longer be directly deduced.

This elementary review will permit us to state more definitely some of the precautions which had to be taken to obtain really valid measurement results.

First, the gages must be stuck on with the greatest care. The wall must be first cleaned mechanically and any grease removed. The sticking must be done sheltered from rain which forced us, for the outside gages, to provide protective mantels of light sheet-metal to the wall above the different regions of measurement. These mantels were maintained during the drying of the gages which was accelerated and reduced to about 24 hours by using infra-red lamps; very thorough drying is essential to the value of the measurements.

On the other hand, the gages once dried must be protected from atmospheric humidity which is susceptible of being absorbed by the cellulose glue which swells as a result. It is equally important to reduce the insulation between the windings of the resistance wire; the insulation deprives the measurements of all accuracy. This protection was achieved by means of Philips 9240 rubber caps especially designed for this purpose which stick to the wall covering the gage and contain a little bit of silica gel designed to absorb residual humidity. As for the connections of the measuring cables, they were protected by means of insulating gauze and melted wax.

The lengths of the connection cables from the gages to the measuring devices were, in the case studied, exceptionally important because of the large size of the tank and the necessity of keeping the devices in the shelter provided for the technicians in charge of measurements. Certain connections reached 80 meters and many kilometers of wire were used for all the measurements. The following precautions were taken to avoid affecting their accuracy. We adopted gages of the proper high resistance, in this case, the Philips 600 ohm gage. We used two-strand faradized wire to remove from these connections the inductive and capacitive effects to which their great length rendered them particularly susceptible (let us point out that the measuring bridge is fed by alternating current). We adopted cables of the same length following the same path for the measuring and compensation gages in the same area. Finally, at least in the BR-2, we practically eliminated the possible effect of a difference in temperature or any other condition from one cable to another by adopting for each measuring gage and corresponding compensation gage the mounting shown schematically in Figure 1. One of the terminals of each of the two gages was connected to one of the strands of the same faradized wire, while the two other terminals were connected to the joined strands of a second wire used as a return cable.

Other precautions had to be taken with respect to the compensation gages. These should not be stressed but have conditions of temperature as near as possible to those of the measuring gage to which they correspond. In the case of a similar test, it was judged acceptable to use one compensation gage for a group of active gages placed near each other in the same region of measurement.

Experience showed, however, that noticeable differences in temperature could result, especially during the day or during the period of pressure drop, between this compensation gage and the measuring gage. A difference of $1^{o}C$ corresponds appreciably to an error in stress of 0.3 kg/mm^2. That is why in the BR-2 we adopted the principle of one compensation gage for every measuring gage; precautions were taken to place the two gages as near together as possible in order to improve the thermal contact between the wall of the container and the plate on which the compensation gage is set. This allowed us, in addition, to use the mounting of Figure 1, described above, with all the advantages it implies.

In the BR-2 we used 72 measuring gages and as many compensation gages. A third of the measuring points were located near the opening of the carriage lock chamber and were entrusted to the care of the department under Professor Soete; two-thirds were located in the dome which was under the supervision of the departments of Cockerill-Ougrée.

The principle used for choosing locations was the following: to retain only a small number of points which could be considered critical beforehand, either on the basis of theoretic considerations or in the light of tests on the reduced model. At these points, to place the gages in the two principal directions following the parallels and meridians, and to provide these gages at the same time on the inside and outside surfaces in order to dissociate by taking the half-sum and half-difference of the experimental results, the membrane stresses and the bending stresses. Finally, to place such groups of gages, as much as possible,

Proposed Mounting

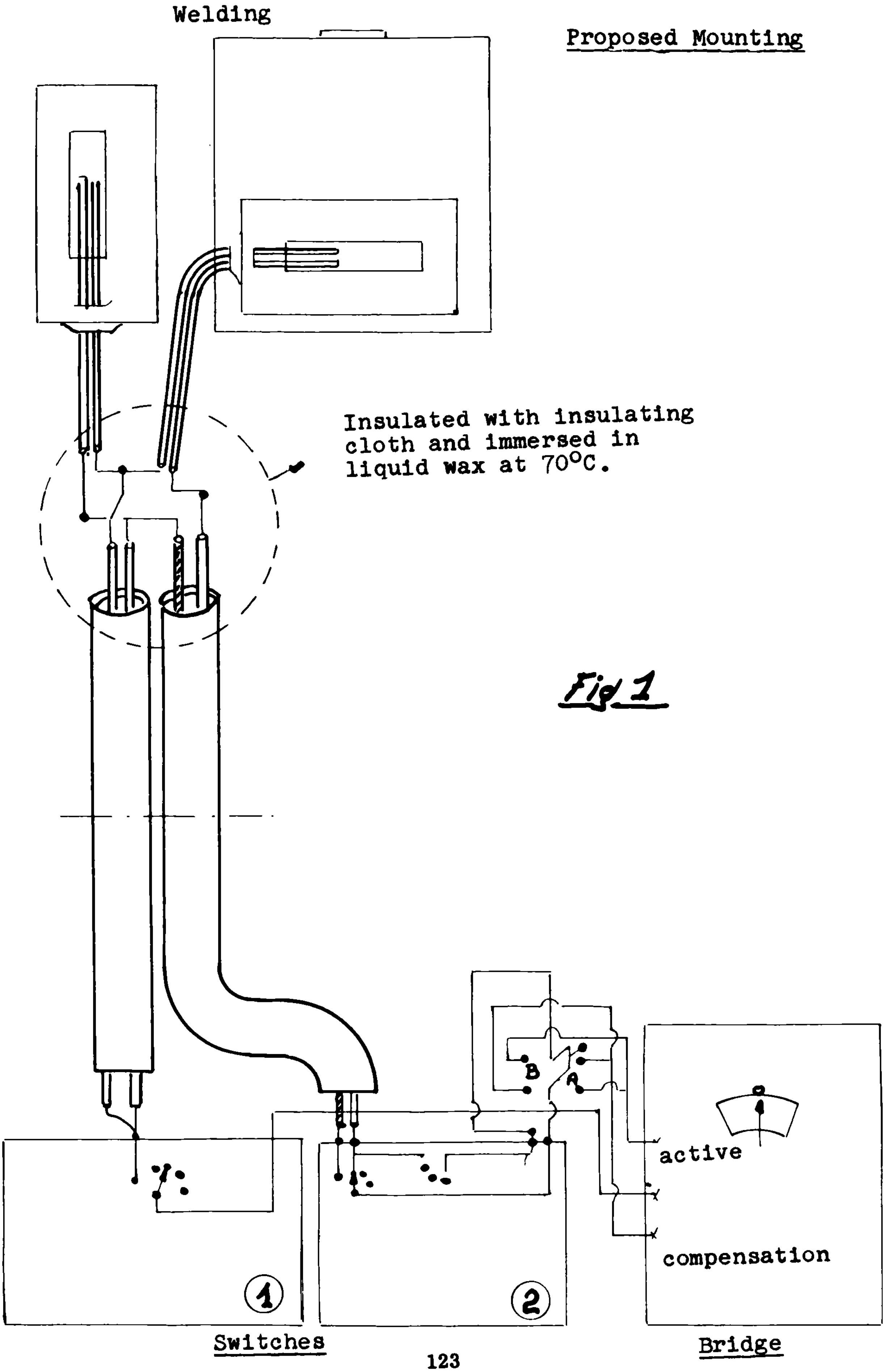


Fig 1

at two or three similar points under theoretically similar test conditions in order to be able to proceed with controls and cross-checking.

From a practical point of view, a series of recorders were placed in the cellar serving as a shelter for the technicians in charge of conducting the test.

At each instant the stresses indicated by the different strain-gages could be read off. A periodic control allowed us to follow their evolution which would have rendered possible a halt or change in the tests if alarming values were obtained.

It remains to say a word about the quantitative value and the interpretation of the measurement results.

Without being able to go into detail, we point out that the measurements taken near the carriage lock chamber yielded maxima stresses satisfactorily in harmony with those deduced from the tests on the reduced model of which we spoke above.

With regard to the stresses in the dome, they are generally very moderate; they are less than 10 kgs/mm^2 under 1.2 kg/cm^2, except where the toroidal portion joins the spherical portion where they reach higher values, moderately above those calculated, but remain within the requirement limits and are still largely acceptable, in view of their local character: under 1.2 kg/cm^2, in the direction of the circumference, the membrane stress is 13.1 kg/mm^2 in compression, and the bending stress 4.2 kg/mm^2; in the direction of the meridian, the diaphragm stress is 3.75 kg/mm^2 under traction and the bending stress 9.8 kg/mm^2. The above values are the averages of the results of three homologous points. The formulas for calculating the surfaces of revolution with thin walls and double curvature:

$$\sigma_1 = \frac{p\ r_2}{2\ e} \text{ and } \sigma_2 = \frac{p}{e}\ (r_2 - \frac{r_2^{\ 2}}{2\ r_1}),$$

where σ_1 and σ_2 are the membrane stresses of the meridian and circumference respectively,

r_1 and r_2, the corresponding radii of curvature,

p, the internal pressure and

e, the thickness of the wall

allow us to calculate from the results of tensometric measurements, the real, local radii of curvature of the dome under stress. The radii of curvature thus calculated differ from the values given in the plane and differ also from one point to another homologous one for two reasons: distortion due to pressure, primary imperfections of form occurring in stamping, assembly or welding. It can be verified, however, that the discrepancies with respect to the theoretical values are of reasonable size and in a logical direction: we were able to convince ourselves of this in the case of the BR-2.

RESULTS OF THE TESTS

These tests took place from 25 to 30 April 1959.

The different objectives enumerated above were attained satisfactorily.

At no time did the resistance of the metallic shell give any cause for worry.

The examination of welds with soapy water yielded excellent results which established without question the remarkable quality of the work performed.

The only leak registered, due to porosity, was very small. It was located in a K weld where radiographic and ultrasonic controls are impossible.

Results of tensometric measurements must be held to be extremely satisfactory, very reassuring, coherent and logical. Their agreement with the values calculated or measured on the reduction model was excellent.

They thus provide a sure basis for new developments.

OPERATIONAL SAFETY CONSIDERATIONS FOR HIGH FLUX TESTING REACTORS

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ABSTRACT

The operation of a large test reactor presents peculiar problems not encountered in the operation of power or production reactors. The MTR and ETR were designed and built for the purpose of carrying out experimental programs, many of which approach or exceed the reactor itself in engineering complexity. It is entirely possible that failures of experimental components could result in hazards to the reactor and operating personnel which would exceed those resulting from the failure of reactor components. To safeguard against the possibility of a serious incident, an operating philosophy has been developed, and an operating organization has been established to carry out this philosophy. The important features of this operating philosophy are outlined in the following paragraphs.

Management has clearly defined the lines of supervisory responsibility not only with regard to engineering and operating problems, but also with regard to investigations and reporting of incidents resulting from malfunction of the reactor or experimental equipment.

The organization is supplemented by special committees which insure that feasibility and safety have been adequately studied prior to insertion of material into the reactors. The organization is so constituted that it can maintain constant surveillance over the reactor and experiments and be capable of recognizing and dealing with long-term changes which could lead to hazardous situations.

Management has clearly defined the areas of responsibility of customer personnel, operations staff, and the special committees in order to eliminate possible gaps which could lead to serious incidents.

The safe operation of high flux test reactors is based not only on good engineering design of the reactor and its associated experiments, but also on an operating organization with sound safety practices which provide for constant surveillance of the reactor and its associated experiments.

COMPLEXITY OF THE EXPERIMENTAL PROGRAM

The operation of a high flux test reactor presents problems not associated with the operation of power and production reactors. The MTR and ETR were designed and built for the purpose of carrying out experimental programs, many of which approach or exceed the reactor itself in engineering complexity. In order to more fully understand and appreciate this complexity, one must consider what is involved in such an "operating plant". The Engineering Test Reactor, which was built at a cost of approximately $15,000,000, can serve to illustrate this point. This reactor is capable of handling seventeen major engineered experiments and hundreds of minor irradiations of varying degrees of engineering complexity. A large loop experiment will cost in the neighborhood of $1,000,000, require some 30 lineal feet of control panel space, approximately 800 square feet of floor space for out-of-pile equipment, and may handle heat loads of several million watts. When one considers a fully loaded ETR, the number of drawings necessary to describe the experiments will greatly exceed the number required to describe the reactor, and the operation of the experiments will be a much more formidable task than the operation of the reactor itself. The test reactor, its supporting facilities, and its experimental equipment form a single operating plant and must be operated by a single operating organization for efficiency and safety. The problem is even more complicated by the fact that the experimental program is continuously variable because of changes in experimental operating conditions, field changes in equipment, addition of new experiments, and removal of those experiments which have been completed.

Fully loaded, the ETR will contain six high-pressure water loops, three single-pass air loops, one liquid metal loop, one liquid metal fuel loop, and one recirculating air loop.

The situation in the MTR is equally complex, since it is currently operating with seven high-pressure water loops, two intermediate-pressure water loops, one fused salt loop, two single-pass air loops, a crystal spectrometer, a time-of-flight spectrometer; and a liquid metal fuel loop is planned in the near future.

OPERATING SAFETY

This paper concerns itself mainly with the safe operation of the test reactor and its associated experimental programs. It is entirely possible that failure of experimental components could result in hazards to the reactor and operating personnel which would exceed those resulting from failure of reactor components. To safeguard against the possibility of a serious incident, an operating philosophy has been developed and an operating organization has been established to carry out this philosophy. This philosophy and the organization which resulted from it has been discussed by Mr. J. P. Lyon in his paper, "Management Aspects of MTR/ETR Operations", and we will here consider how the organization handles the irradiation program.

Phillips established a group of project engineers to evaluate the engineering of experiments and the hazards associated with them early in the history of the MTR. This same organization has been expanded to handle the experimental programs in the ETR. Since experimental programs are constantly pushing into unknown or little known areas, careful evaluation of hazards is an absolute necessity. Attention is therefore given to all possible causes of experiment malfunction which could lead to incidents involving personnel, the experiments, or the reactor.

To best serve this purpose, an engineer (or engineers) is assigned to each irradiation program to work with customer research personnel and design engineers. This engineer will follow the experiment from its earliest stages of development to its completion.

It is during the early phases of the development work that feasibility is established, and the Phillips engineer works with customer personnel checking design, construction, methods of operation, insertion and removal, safety aspects and controls to be maintained; and a complete hazards study is prepared for presentation to the MTR/ETR Safeguard Committee. In many respects this hazard survey is similar to a full-scale reactor hazard study and would contain information concerning all or part of the following items depending on the nature of the experiment.

Heat Transfer

The mechanism of heat transfer represents one of the most likely areas of experiment malfunction. In order to evaluate these hazards, it is necessary to establish the validity of the mathematical model used to describe the physical setup. It must be ascertained that the various correlation equations used are applicable to the experimental system, and the uncertainties associated with various parameters such as flow temperature, etc., must be established. Changes in experimental operating conditions caused by changes in the system occurring progressively or suddenly must be analyzed.

The principal hazards associated with heat transfer are the destruction of containment due to high temperature and the consequent release of foreign material to the reactor, subjecting experimental equipment to thermal shock and stress which could result in component failure, metal water reactions whose explosive violence could damage the experiments and the reactor, and indirect effects on the reactor and experiment such as the effect of steam voids in an experiment on the reactor or the effect of boiling in an experiment on the system hydraulics.

Chemical Reactions

Chemical effects which could result in the release of large amounts of energy or could result in deleterious effects on the reactor represent another area of concern to safe operation. These considerations are important in those experiments using liquid metal coolants or which might release corrosive materials to the reactor or result in corrosion in experimental equipment.

Nuclear Coupling

Nuclear coupling becomes an important phase of safety considerations since test samples can form an important part of the reactor core. The gross effects of placing an experiment in the reactor core must be determined. These effects can manifest themselves in changes in reactor temperature coefficient, void coefficient, rod worth, etc. These effects must be evaluated for each individual experiment for each reactor loading. In addition to the gross effects on the reactor itself, the effect of the experiment as a control element must also be determined. Changes in experiment temperature, voids in the experimental system, moderator or fuel material in the experiment, all couple directly into the effective k of the test reactor. The effect of any possible displacement of the experimental sample must also be considered.

A reactivity measuring facility and an ETR critical facility are operated in support of the MTR and ETR irradiation programs. The RMF can be used for measuring reactivity effects and perturbations of flux produced by insertions of experimental irradiations. In experiments where heat generation is a problem, these flux measurements are very important.

The ETR critical facility is a duplication of the ETR core and beryllium reflector. Design data necessary for ETR experimental irradiations are developed in this facility. In addition, the facility is used for evaluating hazards associated with experiments, their coupling with the reactor and with each other. Some of the parameters which are investigated and which are significant with regard to operation are excess reactivity, flux distribution, and flux perturbation. Complete flux plots are developed in the ETRC for each ETR loading. The effect on reactivity of displacements of samples or filling voids in the facility with water, etc., are also determined in the ETRC.

Mechanical Failures

Mechanical failures of both in-pile equipment and out-of-pile equipment may result in damage to equipment or injury to personnel. Displacement of reactor structure could result in jamming of control rods, rearranging of fuel with consequent change of critical mass, blocking of fuel coolant channels, etc. This displacement could result from mechanical instabilities, thermal expansion, explosion or breakage due to pressure or thermal stresses or both. All these effects are carefully studied for inclusion in the hazard survey. Other mechanical failures might seriously affect the experiment itself. Failure of experimental equipment could compromise the heat transfer characteristics of a given experiment resulting in release of fission products to the experiment coolant stream and loss of the experiment.

Out-of-pile equipment failures can result not only in the normal industrial hazards to personnel and property such as fire, explosion, toxic vapors and gases, but also in the probable release of radioactive material outside the reactor. The highest safety standards with regard to non-nuclear hazards must be exercised since the probability of a non-nuclear incident becoming a serious nuclear incident is quite large.

Handling

All handling procedures which are to be used in the course of the experiment are carefully reviewed. Procedures for handling radioactive components must be worked out and handling problems which could result in hazards to personnel or equipment must be considered. Many experimental irradiation programs are concerned with the behavior of defected fuel elements, and special attention must be given to the handling of these as well as to those that may fail on test.

Radiation Hazards

Incidents resulting from any of the situations we have discussed can in the final analysis lead to radiation hazards to personnel. These hazards can be either direct radiation from contained materials or radioactive materials released to the atmosphere or a combination of both. Analyses of these hazards are made for both normal and abnormal situations in order to determine shielding effectiveness. In places when personnel are required to carry out operating functions, total integrated doses must be computed if the dose rate is above 1 mrem/hr or where levels might be expected to change with time.

Normal radiation levels are held below those required for biological reasons in order that personnel monitoring instruments may maintain low threshold levels. Any radiation level above this instrument tolerance is considered hazardous since the increased level may obscure the precursors of a serious nuclear incident.

Studies of the possibility of the release of material to the reactor building or the atmosphere include both accidental and scheduled releases. Such things as atmospheric dilution factors, quantitative and qualitative characteristics of the contaminants, and weather conditions must be covered. Here again, it is necessary to maintain "instrument tolerance" because the wide range of biological tolerance of air borne material could result in saturating instruments, which are set to respond to low levels of hazardous materials, with relatively innocuous material.

Uncertainties in Calculations

In all cases calculations are made using the most probable center values of the experimental parameters. Calculations may be conservative, but this does not justify relaxation of operating conditions without quantitative evaluation. Uncertainty factors which take into account the worst possible conditions are used to determine the relationship between these conditions and the base calculations. Experiments which have small uncertainties are allowed greater operating lattitude than those whose calculations involve large uncertainty factors.

Time Sequence Studies

Each hazard analysis contains a time sequence study. This study outlines the sequence of events which occur when various parts of experimental equipment fail or function improperly. A good example of such a study would be the loss of experiment coolant flow in a loop. In this case, consideration of coolant temperature versus time, experiment metal temperature versus time, and coolant flow verus time must be carefully studied and correlated with respect to reactor power versus time in order to set the reactor scram set point to prevent any temperatures exceeding the saturation temperature of the coolant at any time during the reactor shutdown. An analysis is required for each power reduction set point and must indicate that unsafe conditions do not develop during the transient phase of operation following any malfunction of equipment. These studies are also used to establish the rate at which reactor power levels are reduced since reactor operating schedules would be seriously compromised if more drastic action were taken than required to maintain safe conditions.

Specific Troubles

The hazard survey submitted to the MTR/ETR Safeguard Committee contains, in addition to the considerations given above, an analysis of specific incidents. Two types of analyses are presented. The first is a study of the maximum credible accident, which postulates the worst accident which could conceivably occur regardless of cause. One then studies the radiation levels, the release of radioactive material to the building and/or the stack and the consequences of such an accident in terms of contamination, exposure to personnel, etc. The second type of analysis would include specific accidents such as loss of system pressure, loss of system flow, instrument failure, power failure, rupture of equipment, or a variation of any other parameter considered important enough to initiate reactor power reductions. This analysis includes the sequence of events from the time of the malfunction until the situation is again under control. The cause, action of experiment sensing equipment, power reductions and the condition of the experiment during and following the incident, the effect on reactor or personnel, and necessary work required to restore normal operation or disposal of the equipment are presented.

Once the hazard analysis is completed, the final insertion approval is based on the probability that the various accidents associated with the equipment will occur and the consequences of such an accident should it occur. It is not always possible to describe in absolute terms in all cases which experimental situations are considered acceptable and which are not. Wherever possible, absolute standards have been developed and distributed to customers in the form of Standard Practices Guides. Those situations which are not covered in such guides are resolved by close liaison between MTR/ETR project engineers and sponsor personnel.

HANDLING OF INCIDENTS

In spite of the best engineering effort, incidents involving experimental equipment will occur; and in order to profit by these incidents management has clearly defined the responsibilities for reporting incidents.

The Project Engineering Section which is responsible for the proper design and inspection of all experimental equipment prior to insertion in the reactor, the issuing of all instructions necessary to properly handle the experimental equipment, and the maintenance of experimental equipment while it is in the reactor, is also responsible for the proper functioning of the equipment while it is in the reactor. The Chief Project Engineer is therefore charged with the prompt investigation and reporting of all incidents resulting from failure or malfunctioning of experimental equipment with two exceptions. Should the incident be the result of an operating error or some failure on the part of Operations personnel to carry out normal operating functions or special instructions, the Operations Branch is responsible for the investigation and reporting of the incident. Since the final decision on controls to be tied to the reactor control circuit resides with the Safeguard Committee, this Committee must investigate and report any incidents associated with power reduction devices including any unusual downtime resulting from false scrams.

When an accident occurs, it is the responsibility of the Operations Shift Supervisor and personnel to take immediate and direct action in accordance with their best judgment as to what should be done and to notify as soon as practically possible responsible line supervision that an incident has occurred.

The method of reporting an incident is as follows: within two to three days following an incident a report is prepared for the Manager of the Atomic Energy Division, Phillips Petroleum Company. This report is written by the responsible individual: Chief Project Engineer, Operations Supervisor, or Chairman of the MTR/ETR Safeguard Committee, and will include a description of the nature of the experiment; its desired operating conditions; a description of the incident and its effect on personnel, reactor, and other experiments; actual operating conditions prior to the incident; and as near as possible the cause of the incident. At this time causes are generally speculative, and definite information is not available until more detailed examination of the experiment is made in the MTR-ETR laboratory or in the sponsor's laboratory. Copies of the report are sent to the Safeguard Committee and the Accident Review Committee. Within a week after the incident, representatives of Project Engineering and Operations Branch meet with the Safeguard Committee for further review of the incident. Further investigation may be conducted by the Accident Review Committee which concerns itself primarily with non-nuclear accidents. The presentation to the various committees is not for disciplinary action, which is left for line supervision when such action is necessary, but simply to attempt to arrive at the causes for incidents and to insure that knowledge gained from such incidents is added to the store of experience being developed and made available to personnel responsible for safe operation of the reactor and its experiments.

PROCEDURES

In order to illustrate the relationship between the various MTR/ETR groups and the Safeguard Committee, a description of the handling of a typical engineered experiment from its conception to its completion is now presented. Handling of experiments which are inserted in the MTR/ETR reflector and which share cooling water with the reactor will also be summarized.

Engineered Experiments

Many of the experimental programs at the MTR/ETR are for the purpose of developing fuel elements of advanced design. Most of these require that the test samples be irradiated in environments not found in either of the test reactors. An example of such a study would be the development of fuel elements for pressurized water power reactors for which it would be necessary to evaluate performance of fuel at high pressure and high temperature. Pressures up to 2500 pounds per square inch and temperatures up to 750°F might be desirable for such a test. These conditions are not available in the MTR/ETR, and it is necessary to insert a pressure vessel into the reactor to achieve these objectives. Problems such as space limitations, pumping requirements, pressure vessel design complicated with thermal stresses resulting from high gamma heating, and the possibility of steam explosions are immediately apparent.

How would such an experiment be handled by the various MTR/ETR groups? The initial proposal usually is presented in a meeting between Project Engineering personnel and the sponsor's representative, where the objectives of the experiment and the general plans for accomplishing these objectives are discussed. The MTR/ETR Safeguard Committee Chairman regularly attends these meetings and advises the group of any special requirements which might be indicated from the nature of the test. No formal Safeguard action is taken at this time.

The Safeguard Committee reviews the experiment again when the preliminary engineering has been accomplished and the design has become more or less firm. This meeting is held on the request of the MTR/ETR project engineer who has been assigned to work with sponsor's research and engineering personnel developing the experiment. At this point, the Safeguard Committee, project engineers, operations personnel, and sponsor personnel discuss in detail the various design problems such as pressure vessel design, methods of calculating total stresses, corrosion problems, pumping and pressurizing, water treatment, instrumentation, cooling requirements, and safe operating conditions. The project engineer then works closely with sponsor personnel in the execution of the design incorporating any recommendations which developed during the Safeguard review. When the design is completed, the experiment is rechecked before construction is started. The Safeguard Committee is by now fairly familiar with the details of the in-pile loop and its associated facilities, including the instrumentation.

The experimental facility is then fabricated and made ready to insert in the reactor. At this point, the project engineer makes a formal request for Safeguard approval of the experiment and transmits to the Safeguard Chairman the hazards survey.

The Safeguard Committee, together with any necessary technical consultants, project engineers, operations personnel, and sponsor representatives, will review the experimental facility and the first series of tests to be conducted. If all the preliminary phases of experimental approval have been complied with and there has been no major change in scope of the program, the final approval of the experiment is straight-forward. Setpoints and operating conditions are also approved for the first test at this time.

During the life of the facility, each new experimental sample is separately approved by the Safeguard Committee as are the appropriate setpoints for alarm and reactor power reduction for the new test.

Tank Experiments

The primary considerations given experiments which are inserted in facilities existing in the reactor tank are reactivity effects, heat generation, explosion hazards, and potential changes which might arise from exposure to conditions existing in the reactor. Some of these experiments are equipped with lead tubes, and operating conditions with respect to temperature and pressure can be monitored; many are done in sealed capsules, however, and here a great deal of emphasis is placed on the calculation of conditions. In each case, the experiment is reviewed by the Safeguard Committee prior to insertion.

On completion of the irradiation, samples are returned to the sponsor by the responsible project engineer along with all pertinent details concerning the irradiation.

CONCLUSION

The safety of a test reactor plant operation is not a simple matter of good engineering design, but is rather a combination of good engineering, sound safety practices in operation of the reactor complex, and constant surveillance of that complex.

SAFETY OF THE BR-2, THE BELGIUM MATERIALS TESTING REACTOR

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GENERAL CONSIDERATIONS.

The safety of the BR-2 is of prime importance, because the reactor is located in a populated area.

The population inside a circle of 15 km radius around the site is of 112.000. Inside a radius of 30 km the population is of 463.000 people of which 76.000 live in Holland.

The closest boundary of the C.E.N. site is located at 350 m from the BR-2.

The residential area of the laboratory extends from 500 to 1.000 m from the reactor.

The probability of any serious accident must be kept to an absolute minimum.

The general approach taken toward BR-2 safety centers upon two basic principles :

1. Prevention of an accident.
2. Containment of fission products in the event of an accident.

The first principle is implemented by :

- insuring that the reactor is inherently safe,
- making a close check of each loading, by the use of the core mock-up facility.
- keeping close administrative control to insure safe operation.

The second principle is satisfied by placing the reactor within a containment building designed to prevent the escape of fission products, and to attenuate the radiations, in the event of any conceivable accident.

INHERENT SAFETY OF THE REACTOR.

There are three reactor parameters which are very important in reactor safety considerations.

- void coefficient : this coefficient varies from −0,04 to −0,20 $\frac{\Delta k}{k}$ per fractional change in water volume, depending on the loading of the reactor. For MTR the quoted number is −0,24.
- temperature coefficient : it varies with the core loading. An average value is $- 10^{-4}\ \frac{\delta k}{k} /\ ^{\circ}C$.
- neutron life time : depending on the loading, this number can vary from $5\ 10^{-5}$ sec to $2\ 10^{-4}$ sec.

POSSIBLE ACCIDENTS.

Many possible accidents have been investigated.

These accidents can be caused by :

1. Failure of the reactor plant machinery, such as the cooling system piping and pumps, or the control system.
2. Failure of experiments, which can result in explosions or changes in reactivity.
3. Improper start up procedures, which can result in large power excursions.
4. External damage from fire, explosion or the forces of nature.

We will not review all of the accidents which have been investigated.

All of them have been rendered as improbable as possible by the use of fail-safe systems, or by backing up the sensing elements by an other element of the same nature, or by the use of more than one safety channel (for instance, the use of a flowmeter and of a differential pressure meter to sense the flow in the primary circuit).

Some of the most dangerous accidents are discussed here under.

Primary pump failure.

If the primary pumps would fail, the reactor scrams and an auxiliary pump is started.

If this auxiliary pump would fail to start, a valve opens which connects the pressure vessel with a heat-exchanger immersed in the pool. The fuel elements would reach 350°F. The pool water would reach 120°F after 30 hours.

The Borax-type accident.

The Borax-type accident is initiated by the sudden insertion in the reactor of a large amount of excess reactivity of the order of 1 % or more. This accident could conceivably be produced by the ejection of a control rod out of the pressure vessel.

In this case, the regulation of power is accomplished by boiling of the water during the power excursion, which results in the ejection of some of the moderator from the core. This, in turn, results in a decrease in reactivity, provided that the reactor has a negative void coefficient.

Calculations have shown that, depending on the value of the void coefficient, the reactor is capable of sustaining the instantaneous insertion of 1,3 to 2 % excess reactivity without melting of the fuel elements.

The control rod worth may be as high as 4 %, but its withdrawal could not be instantaneous, due to its mass and the mass of water it would have to displace.

Hence, it is believed that the reactor could sustain the ejection of a control rod.

Malfunction of experiments.

When an experiment fails in the reactor, the following can result :

1.- Contamination of the primary water.
2.- Contamination of the experimental loop outside the tank.
3.- Change in reactivity.

The contamination is detected by the activity of the ion-exchanger fission product detector, or of the gases out of the degasser.

The reactor scrams, and the activity must be reduced to 10^{-2} c/m^2 before the accident is removed.

The water treatment flow is increased from 120 g.p.m. to 400 g.p.m. If the accident releases 10^4 curies (100 m^3 lead to 10^2 curies/m^3), it would take 10 hr before the primary circuit can be cleaned.

The change in reactivity can be handled as for the Borax accident. For instance, the flooding of several beam holes could introduce as much as 1 to 2 % reactivity, but not instantaneously.

Start-up accident.

A serious accident could occur, due to error (human, electrical or mechanical) during the start-up of the poison free reactor, if the removal of control poison were not stopped when either the limiting power or period was reached.

The reactor period would then become so short that dangerously high power levels would be attained in brief time intervals.

Start-up of the reactor is accomplished by withdrawing the control rods on an intermittent basis of 0,5 sec of travel for every 2,5 sec. This intermittent withdrawal is to be an integral part of the control mechanism, and not subject to the operator's discretion. During the 0,5 sec out of every 2,5 sec, k is increased at a rate of 0,074 %/sec.

If it is assumed that the intermittent operation of the control rods fails, that the reactor period scram trip fails but, that the power level trip operates at 50 MW, it is given 0,050 sec for this last trip to operate, and still prevent the melting of the fuel elements.

THE MAXIMUM CREDIBLE ACCIDENT.

1. Description of the accident.

The accident under study is initiated by an uncontrolled nuclear excursion, which melts the fuel elements of the core.

This we call the maximum credible accident, although it is somewhat difficult to visualise how the necessary circumstances could be met. A step change reactivity insertion of 1,3 % to 1,9 % or greater is needed ; these excess reactivity are increased if the instantaneous insertion is not postulated. The introduction of excess reactivity in such amounts would in fact demand some ingenuity, and its fortuitous happening seems improbable.

After the melting of the fuel elements, a metal-water chemical reaction is postulated. This, also, is very unlikely as experiments indicate that a violent metal-water reaction can occur only under very special conditions.

It is then assumed that the hydrogen released in the Al-water reaction reacts with the oxygen of the air.

Finally, we have postulated the presence in the building of 670 lb (about 100 gal) of sodium, which burns in the air.

The containment building will withstand the last pressures. The concrete reactor and pool structure will be damaged but will not be completely disintegrated.

As a consequence of the accident, a maximum pressure of 0,19 kg/cm^2 or 2,8 p.s.i. results from the heating of the atmosphere in the building.

II. Automatic procedure following an accident.

When the contamination in the building attains a preset value, an alarm will sound which orders the evacuation of the building.

Simultaneously the normal ventilation system of the building is closed in less than one second by a preliminary valve and then in about three seconds by a leaktight valve. The waste pipings are closed in less than one second.

At the same time, scrubbers are connected into the ventilating systems of the air-locks. These scrubbers are designed for an air flow of 5.500 cfm.

After a quarter of an hour, the scrubbers are connected automatically in closed circuit on the containment building.

If the γ radiations in the building exceed 10^3 roentgens per hour, a shower is automatically put in use, spraying water from the top of the containment building. The flow of water is 30 liters per second, the diameter of the droplets is specified between 0,5 and 1 mm. The water is pumped from an underground tank of 120 m^3.
The level of more than 10^3 roentgens per hour is read by three out of four counters located in the building.

Before one hour of operation, the pressure in the building will be reduced below atmospheric.

The scrubbers are then maintained in operation during 5 to 6 days, after which all the building air can be evacuated through the scrubbers and chimney.

Some considerations have been given to the possibility of actuating the water spraying by manual operation from a point located outside the containment building.

This solution would have the advantage of minimizing the probability of an accidental actuation of the spraying system. The choice between the two solutions has not been made as yet.

The consequences of a delay of t minutes in the actuation of the sprayers are to multiply the iodine inhalation and deposition by the following factors f. :

t min	f
5	1,3
10	1,6
15	1,9

As will be seen later, the other doses due to external radiations, strontium and caesium, are small and less affected by the delay.

III. Assumptions used in establishing the maximum pressure.

1) Sodium reaction.

If the reaction happens in the reactor or in the pool, only part of the heat will be transmitted to the air. Similarly, all the hydrogen liberated in the reaction will not react with the oxygen of the air. Hence it is assumed that the sodium burns in the air and is subsequently hydrated by the shower or vapor in the atmosphere.

$$2\ Na + 1/2\ O_2 \rightarrow Na_2O$$

$$Na_2O + H_2O \rightarrow 2NaOH.$$

This is the only way by which all the energy could be transmitted to the air.

The heat releases is then 900 Mcal. This heat is so large compared to the heat released by other sources, that any assumptions made on these sources can be quite arbitrary.

2) Aluminium-water reaction.

$$2\ Al + 3\ H_2O \rightarrow Al_2O_3 + 3\ H_2$$

We assume that half of the aluminium in direct contact with uranium (meat + cladding) reacts with water.

We have assumed further that only half of the heat is transmitted to the air of the building.

This metal-water reaction releases then 39 Mcal to the building air.

3) Hydrogen-air reaction.

$$2\ H_2 + O_2 \rightarrow 2\ H_2O$$

The hydrogen is released by the reaction of half of the aluminium with water.

It is assumed that 1/3 only of the hydrogen can burn before its concentration be reduced under the combustion limit.

The hydrogen-air reaction releases then 31 Mcal to the building atmosphere.

4) Nuclear reaction.

This reaction releases about 100 MW.sec, whatever the excess reactivity. One half of this comparatively small quantity, or 12 Mcal is assumed to be transmitted to the air.

5) Total.

These quantities add up to 982 Mcal.

The combustion of sodium cannot be instantaneous, and will be pursued while the spraying is going on. The spraying would saturate the air extremely fast (in a period of the order of two minutes) and certainly a lot faster than the sodium can possibly burn.

Hence, we have made the very conservative hypothesis that the sodium burns instantaneously, but that the energy released is used partially to saturate the air in humidity.

IV. Fission products dispersed in the building.

When the maximum accident happens, the reactor is loaded with 4 kg of U-235, and has been operating during 30 days at an average flux of $2{,}7\ 10^{14}$ n/cm^2 sec.

As a consequence of the accident, it is assumed that all the volatile fission products are dispersed in the building atmosphere. No allowance has been taken for the solid products contained in the core, for the three following reasons :

- most of them would be retained by the fuel elements or by the pool water, and sufficiently shielded.
- the rest of them would be settled down by the water spraying, by the scrubbers, or by gravity.
- it is impossible that all the volatile products get into the atmosphere, because some would be retained in the undestroyed fuel elements and some iodine would be dissolved in the pool water.

However the isotopes Sr-89, Sr-90, Cs-137 and Cs-138, born from the gases Kr-89, Kr-90, Xe-137 and Xe-138, after the accident are considered as gaseous.

Also, because of their potential danger, 0,1 % of the solid products Sr-89, Sr-90 and Cs-137 contained in the core are assumed to be dispersed in the atmosphere. This factor of 10^{-3} is not critical as the consecutive doses are far below tolerance.

V. Leakage out of the building.

A leakage rate of 0.1 % per p.s.i. overpressure and per day, or 1,47 % per atmosphere and per day is assumed.

The total volume of the building is 24.000 m^3. Only 0,77 m^3 leak out of the pressure-tight shell.

To maintain this high leaktightness, a periodic check and maintenance procedure will have to be set-up.

It is suggested actually that the following periodicity of test be decided : once every 3 months for the first year of operation and every 6 months thereafter.

VI. Weather conditions.

1) Wind velocity.

The smaller the wind velocities the greater the doses.

The frequencies of small wind velocities in 1957 were :

from 0 to 0,5 m/sec : 0,43 %

from 0,6 to 1 m/sec : 1,6 %.

We have assumed a wind velocity of 1 m/sec. Hence, a larger velocity would be present 98 % of the time. The probability of a smaller velocity, happening during more than one hour, should be considered as negligeable.

2) Sutton parameters.

a) dry weather.

The following values were used :

$n = 0,5$

$c_y = c_z = 0,06\ m^{n/2}$ at ground level (for leakage calculations)

$c_y = c_z = 0,032\ m^{n/2}$ at a heigh of 100 m (for evacuation of contaminated air from the building).

These values correspond to a strong inversion.

b) rain.

Rain is practically impossible during an inversion. The values taken are at the limit between inversion and unstable conditions.

$n = 0,25$

$c_y = c_z = 0,07\ m^{n/2}$

VII. Consecutive irradiation doses outside the building.

1) External radiation.

Four sources of direct radiations have been considered :

a) Direct radiation from the building.

b) Immersion dose in the leakage stream.

c) Direct radiation from iodine deposition on the ground.

d) Direct radiation from the fission products evacuated through the chimney, 5,4 days after the accident.

In establishing the total doses, the beta doses have been divided by 5.

distance (meters) roentgens	100	350	500	1.000	5.000
Gamma					
building	160	6,0	1,6	0,050	-
leakage	0,10	0,08	0,07	0,049	0,01
deposition	0,61	0,42	0,33	0,20	0,035
evacuation	0,40	0,30	0,27	0,27	0,27
Beta					
leakage	1,62	0,80	0,56	0,25	0,018
Total	160	6,9	2,4	0,62	0,32

2) Inhalation doses.

The doses accumulated by inhalation are consequences of the leakage of fission products from the building.

Three elements have been considered : iodine, strontium 89 and 90 and caesium 137.

distance (meters)	100	350	500	1.000	5.000
I (μ c)	220	115	82	40	4,6
Sr-89 (m μ c)	60	32	23	11	0,14
Sr-90 (m μ c)	0,42	0,23	0,16	0,079	0,009
Cs-137 (m μ c)	0,48	0,25	0,18	0,092	0,011

3) Deposition.

The deposition on the ground has been studied for the following isotopes : I-131, Sr-89, Sr-90 and Cs-137.

distance (meters)	100	350	500	1.000	5.000	10.000
I-131 (μ c/m^2)	260	180	140	87	12	3,2
Sr-89 (m μ c/m^2)	800	350	240	100	9	2,3
Sr-90 (m μ c/m^2)	5,6	2,4	1,7	0,70	0,063	0,016
Cs-137 (m μ c/m^2)	4,7	2,1	1,4	0,59	0,053	0,014

All the numbers cited above must be considered as maxima.

1) The doses are computed for extreme weather conditions.

2) All the doses, except from direct radiations from the building, are localised in the axis of the plume. As soon as one moves out of this plume, these doses are greatly decreased.

3) It should be recognized that inhalation and direct radiations from leakage on one side, deposition on the other, can not be achieved together as presented above. Indeed for deposition rain has been assumed and rain is incompatible with the type of inversion assumed for the other leakage effects.

4) Finally, the deposition, as computed, corresponds to the maximum possible deposition at each individual point, based on a certain deposition rate which is different for each distance. Hence the results are incompatible in the sense that these depositions, integrated over the earth surface, would give a higher total activity that actually leaked out of the building.

VIII. Allowable doses.

1) Outside radiations.

The "once in a lifetime" permissible dose usually admitted is 25 r.

2) Iodine inhalation.

The continuous presence of 0,6 μc of I-131 produces a dose of 0,3 r per week in the thyroïd gland. From this number it can be deduced that 1 μ c of I-131, introduced in the gland at a single time, produces 0,778 rep/ μ c.

However, for a child, the number above has to be multiplied by 4, for the thyroïd gland of a child is assumed to weight 5 g (20 g for an adult). There is an other factor of 3 due to the non-uniformity of distribution of iodine in the thyrdïd. The number above is then 9,3 rep/ μ c.

From the American reports, a "once in a lifetime" dose of 2.000 rep is admissible. Although these reports recognize that a dose of about 200 rep could conceivably produce cancer, this probability is extremely low under 2.000 rep.

For a child, the cancer threshold is at 250 rep and this dose should not be overpassed.

a) the limit admitted for adults is :

$$\frac{2000}{0,778} = 2\ 570\ \mu\ c.$$

b) the limit, as computed for children, is :

$$\frac{250}{9.3} = 27\ \mu\ c.$$

It should be noted about this last limit, that it does not account for a difference of breathing rate between children and adults.

3) Strontium and caesium inhalation.

The activity which produces 0,3 rep per week in the body is 2 μ c of Sr-89, 1 μ c of Sr-90 and 98 μ c of Cs-137.

As the doses due to the maximum credible accident are far below these limits, no further calculations are needed.

4) Iodine deposition.

The computation of the allowable deposition of iodine can be based on the following data :

for rich pastures, 1 μ c/m^2 will lead to 0,1 μ c per liter of milk.

- the half life of I-131 on pastures is 4 days.
- a child drinks 1 liter of milk per day (fresh milk is assumed so that the 4 days half life is used)

Then the criteria described for iodine inhalation give the following results :

a) limit for adults : 4 470 μ c/m^2

b) limit for children : 47 μ c/m^2.

Now, the deposition doses cannot be considered as accident doses, as it is possible to monitor the milk and forbid the consumption when the activity is too high.

c) we then divide the admissible activity b) by a factor of 10, leading to 4,7 μ c/m^2.

d) after the Windscale accident, some rapid evaluations lead to the decision of forbidding the use of milk when its iodine activity exceeds 0,1 μ c/liter. This number corresponds to 1 μ c/m^2.

5) Strontium deposition.

Concentrations of 3,6 μ c/m^2 for Sr-89 and of 0,2 μ c/m^2 for Sr-90 are permissible without any precaution. (AERE HP/M 10).

6) Caesium deposition.

To evaluate the admissible deposition of caesium, Cs-137 is compared to Sr-90 on the following grounds :

- their half lives are about the same.
- the maximum admissible body burden is 98 times greater for Cs-137 as for Sr-90.
- a cow, eating contaminated grass, will concentrate in the milk 8,5 more Cs-137 than Sr-90 (J. Nucl. Eng. 5, n°3/4, 1957).
- the tolerable deposition for Sr-90 is 0,2 μ c/m^2.

The permissible deposition is then : $\frac{0{,}2 \times 98}{8{,}5} = 2{,}35\ \mu\ c/m^2$.

IX. Conclusion.

The calculations have been made, based on extremely conservative assumptions, the principal of which are :

- the dispersion of all volatile fission products in the building,
- the complete burning of 670 lb of sodium,
- the extreme weather conditions,
- the possibility that somebody may remain for an hour in the axis of the very narrow plume, leaking out of the building.

Despite these pessimistic characteristics, none of the doses could lead to an accident outside the C.E.N. site.

1) Direct radiations.

The effect of direct radiations is quite admissible, except on some locations inside the C.E.N. site, close to the BR-2 building.

All regions inside a radius of 150 m should be evacuated as soon as possible. No rapid evacuation is necessary at a distance greater than 200 m. No evacuation is necessary outside the C.E.N. site.

2) Inhalation.

Strontium and caesium do not lead to any problem.

Although the more stringent British norms may not be fulfilled within a radius of 1,5 km, all other criteria indicate that the doses are quite acceptable everywhere.

Considering the fact that the plume is quite narrow, and the doses are given in its axis, it is believed that, even based on the British criteria, the maximum permissible doses would not be overpassed.

3) Deposition.

Some steps for the limitations on milk consumption would have to be taken if the maximum credible accident would occur. However, it should be kept in mind that the calculations have been performed under the assumption of rain. On other circumstances, the deposition would be far less. Again, the numbers correspond to the concentration in the axis of a narrow plume.

Strontium and caesium concentrations are quite acceptable.

Iodine concentration could not lead to any serious accident, even if no survey on milk concentrations is made.

If the limit of 4,7 μ c/m^2 (leading to 0,47 μ c/liter of milk) is adopted, the consumption of milk may have to be prohibited up to a distance of 8 km.

The limit of 1 μ c/m^2 (0,1 μ c/liter of milk) adopted after the Windscale incident could lead to limitations on milk consumption up to a maximum distance of 20 km.

If BR-2 is operated by qualified personnel, and supervised with competence and strictness we believe that the maximum credible accident has a probability so low that it can be quoted as impossible.

Whatever these probabilities are, no damage could affect people located outside the C.E.N. site No evacuation is necessary and only some restrictions on milk consumption may have to be provoqued. As the peak concentrations of iodine in milk happen about 16 hours after ingestion by the cow some time is provided for the necessary steps to be taken.

MANAGEMENT ASPECTS OF MTR-ETR OPERATIONS

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ABSTRACT

The realization of maximum utility from high-flux test reactors requires an operating philosophy which is somewhat different from that applicable to reactors where no associated complex experimental program exists. The management policies under which the MTR and ETR are operated for the United States Atomic Energy Commission are designed to provide maximum utilization of the reactors for test purposes and at the same time insure adequate safety in reactor and experiment operation.

A combination of proven operating policies and procedures, well trained operating and technical surveillance personnel and thorough engineering liaison on experiments, has yielded seven years of MTR operation and two years of ETR operation without major incidents involving the release of radioactive materials, although several minor incidents have occurred.

The operating staffs of both reactors are supervised by technically trained and experienced personnel. Critical functions with respect to reactor and experiment operation are performed by technically trained or specially selected employees. Organization of the operating staff is presented.

Since irradiation space in the reactors is limited, each experiment is reviewed and approved by the Atomic Energy Commission for conformance with overall program objectives and is reviewed by Phillips for feasibility and compatibility with the reactors and existing experimental programs.

The engineering design and detail of each experiment is worked out with the experiment sponsor and is finally subjected to rigorous review for conformance with approved design and compliance with MTR-ETR safety and engineering standards.

Operating costs for MTR and ETR are presented, together with the economic basis for arriving at a full-cost recovery charge per irradiation unit of 10^{20} nvt per cubic inch of irradiation space that is in effect for commercial irradiations.

BASIC OPERATING PHILOSOPHY

The Materials Testing Reactor (MTR) and Engineering Test Reactor (ETR) are water-moderated, beryllium-reflected, enriched-uranium reactors employing plate type fuel elements of uranium-aluminum alloy clad with aluminum. These reactors provide a wide variety of experimental space with thermal neutron fluxes varying from 10^{12} nv to 5×10^{14} nv.

The prime objective in constructing the MTR and ETR was to provide tools for the development of knowledge on radiation damage through the conduct of experiments relevant to the AEC-sponsored reactor development program. A secondary, although highly important, objective was the use of the neutrons produced for solid state studies and the production of radioisotopes. The policies and procedures developed for handling MTR-ETR irradiations are designed to accomplish these objectives by assuring that available reactor space is being used to the maximum benefit of approved programs.

The basic philosophy governing the operation of the MTR and ETR is to conduct operations in as safe and expeditious a manner as possible consistent with the over-all objectives for which the reactors were constructed. This philosophy requires the acceptance of some calculated risks but the probabilities of major incidents are kept below what could be considered the taking of chances with respect to the reactors or their associated experiments. This method of operation is made mandatory by an ever increasing tendency to work at higher and higher temperatures and pressures and to subject materials to stresses under conditions not hitherto explored.

An experiment is accepted for irradiation only after it has been determined to the best of our knowledge that the results of the worst incident which could conceivably happen involving the experiment can be handled without significant hazard to personnel or to the reactors. A more restrictive policy would certainly mean that many experiments would fail to yield useful data.

Coupled with the safety considerations in operation is a programmatic requirement to supply the maximum number of neutrons for the largest number of experiments which can be accommodated usefully in the reactors.

The MTR and ETR are operated and maintained for the Atomic Energy Commission by the Atomic Energy Division of Phillips Petroleum Company. The Division is headed by a Project Manager who has over-all responsibility for all Division functions and three Assistant Managers who are respectively responsible for technical, administrative and operations aspects of the work. An abbreviated organization chart covering these segments of the Division organization applicable to MTR and ETR operation is shown in Figure 1.

OPERATIONS BRANCHES

The MTR Operations Branch has exclusive responsibility for the safe and efficient operation of the reactor, its supporting facilities, and associated experiments. Personnel in the Branch also maintain reactor instrumentation, refuel the reactor and insert and remove all experiments which can be handled by hand.

The function of the ETR Operations Branch is equivalent to that for the MTR except for the operation by the MTR Operations Branch of utility systems (steam, raw and demineralized water, compressed air, etc.) which are common to both plants.

Each Branch is managed by a Superintendent of Operations, Assistant Superintendent, Shift Supervisors and Shift Foremen with supporting personnel composed of technically trained and job trained men.

Branch supervision down through Shift Supervisors all hold technical degress and have from five to twenty years experience in research, engineering and operations. Shift Foremen and Technician's may or may not hold degrees in

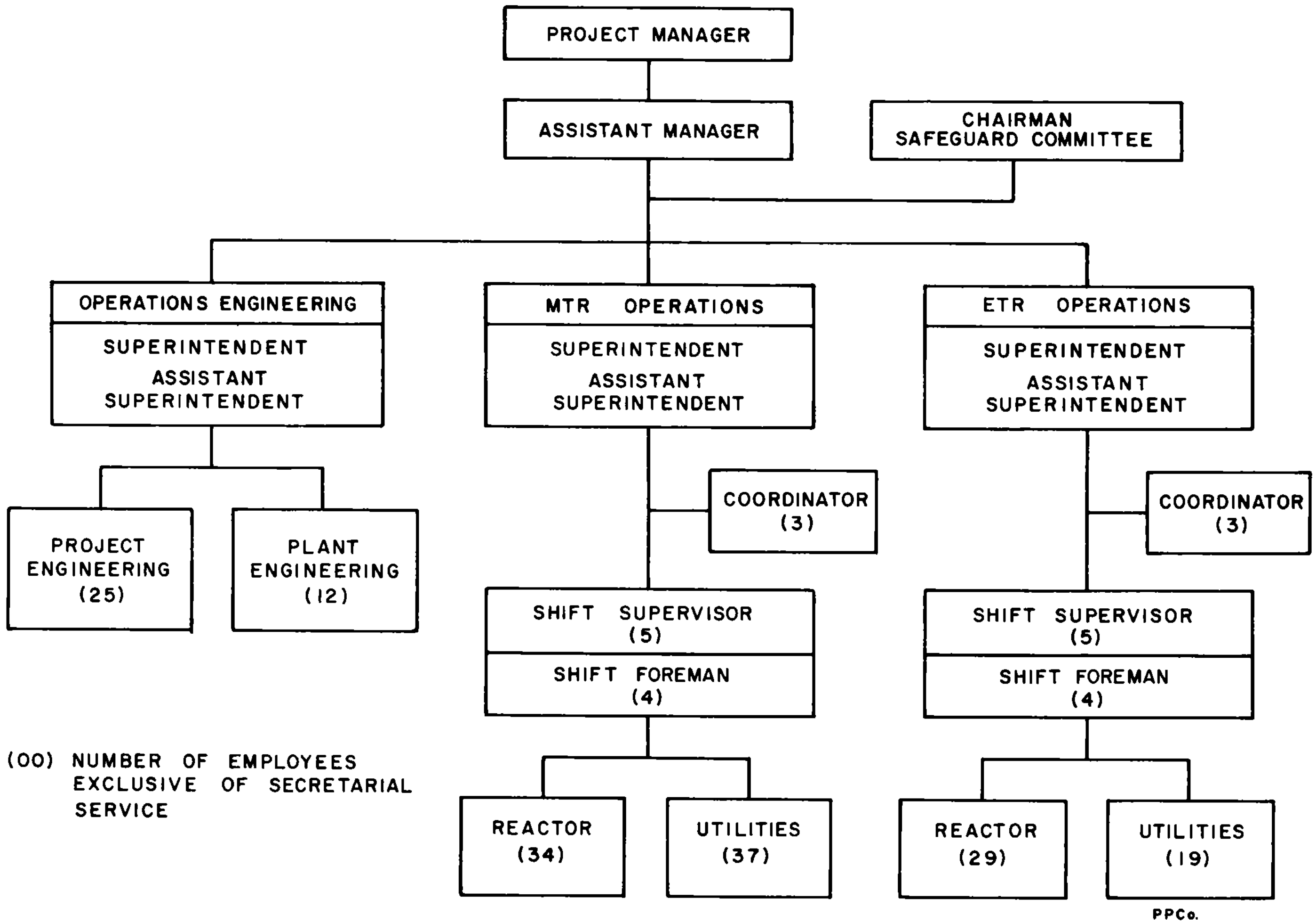

Fig. 1—MTR-ETR operations organization.

one of the physical sciences but in the absence of a full degree the men usually have two or more years of college training and are hand picked on the basis of general intelligence, aptitude, stability and interest in the types of work to be performed. Personnel who operate the utilities systems are screened for general intelligence and mechanical aptitude and are trained on the job through a series of progression steps.

OPERATIONS ENGINEERING BRANCH

The Operations Engineering Branch is responsible for all engineering aspects of the experimental programs conducted in the reactors and for engineering assistance to the operations branches on all non-nuclear problems associated with operation of the reactors, supporting facilities and experiments. The Branch is headed by a Superintendent who is an engineer with broad training and many years experience in metallurgy, strength of materials and corrosion. In order to effect a separation of the two principal functions of the Branch a subdivision into a Project Engineering Section having responsibility for all aspects of the experimental programs and a Plant Engineering Section having responsibility for plant engineering has been made.

(a) Project Engineering Section. The Project Engineering Section is responsible for all Phillips' efforts applicable to the irradiation programs carried out in the MTR and ETR. This responsibility includes program coordination, scheduling, engineering liaison on design, construction and inspection, supervision of installation and the issuing of special operating and handling instructions. In the event of significant malfunction of an experiment during irradiation, the Section conducts an investigation and prepares a written report covering the reasons for malfunction and recommendations for remedial action.

The Assistant Branch Manager is primarily responsible for Project Engineering; under his supervision a Chief Project Engineer has administrative and technical supervision over several groups of engineers who handle all phases of liaison on the experimental programs.

The men working in Project Engineering are required to have broad engineering and specialist knowledge. Included are men with formal training in physics, chemical engineering, mechanical engineering, engineering physics, metallurgical engineering, nuclear engineering, and specialized knowledge in fields of nuclear reactions, shielding, fluid mechanics, heat transfer, electrical circuits, instrumentation and others.

(b) Plant Engineering Section. This Section is responsible for providing the MTR and ETR Operating branches with engineering services required for safe operation of the reactors and supporting facilities. The Section operates test facilities for determining the hydraulic characteristics of reactor fuel assemblies, control rods and reflector components and as an aid in developing improved units.

MTR-ETR Reactor Safeguard Committee

Early in the operation of the MTR it became apparent that the experimental programs to be conducted in the reactor were of sufficient complexity and potential hazard to warrant a procedure by which each experiment would be subjected to hazard evaluation independent of that given by project engineers. A system of double checking was effected by establishing a Reactor Safeguard Committee comprising senior physicists and engineers having no direct responsibility for the experimental programs. The Committee functions under a full time chairman and is responsible for review and approval of all experiments which could by malfunction create a significant hazard to personnel or to the reactors. An additional responsibility of the Committee is the approval of changes in reactor operating procedures prior to effecting the changes.

When a technical problem arises with an experiment it is usually possible for the Committee to find a solution, since it is the purpose of the Committee to find safe ways to conduct experiments rather than merely reject them if they appear hazardous. In practice only a few experiments have been finally rejected but many experiments are modified extensively between the time they are first proposed and actual insertion in a reactor.

Supporting Groups

Maintenance of facilities and installation and removal of experiments requires the services of approximately 200 supervisors, engineers and craftsmen. Of these, about two-thirds are directly concerned with the experimental programs.

In addition to the services provided by the previously mentioned groups, the operations are supported by several groups within the Atomic Energy Division technical, safety and radiation protection organizations.

PROCEDURES AND APPROVALS REQUIRED FOR CONDUCTING EXPERIMENTS

It is apparent that in order to avoid confusion, well defined and carefully followed procedures are required to effectively carry out the numerous experimental programs being conducted in the MTR and ETR.

Procedures are in effect covering the approvals required to process an experiment from the time it is first proposed by a sponsor to completion by irradiation in one of the reactors. The procedures are divided into two parts, the first of which is primarily administrative and is designed to assure the AEC that the reactors are being used in conformance with program objectives. The second part insures that the scope and equipment of a specific experiment meet MTR-ETR safety and engineering standards.

All requests for irradiation programs are submitted by the sponsoring organization to the MTR-ETR Policy Board which functions under the Atomic Energy Commission's Division of Reactor Development. Initial approval and a priority assignment are made by the Policy Board and the irradiation request is sent to the Idaho Operations of the AEC for approval by Phillips as to feasibility before final approval by the AEC. Figure 2 shows the approvals required for an experimental program.

Many simple types of irradiations such as isotope production and irradiation of structural materials which involve no problems of temperature or health hazard are reviewed by a project engineer assigned to assist with the program and returned to the Phillips' Project Manager with a recommendation for action.

All engineered experiments and other irradiations involving problems in heat transfer, fluid flow, strength of materials, etc., are subjected to detailed study to arrive at a basic design which is compatible with available reactor space and which conforms to MTR-ETR safety and engineering standards.

When a design satisfactory to the sponsor and our project engineers has been agreed upon, the design and proposed operating conditions are reviewed by the MTR-ETR Safeguard Committee for conformance with safety standards and the program is then forwarded to our Project Manager for transmittal with Phillips recommendations and approval to the Idaho Operations Office for final AEC approval of the program.

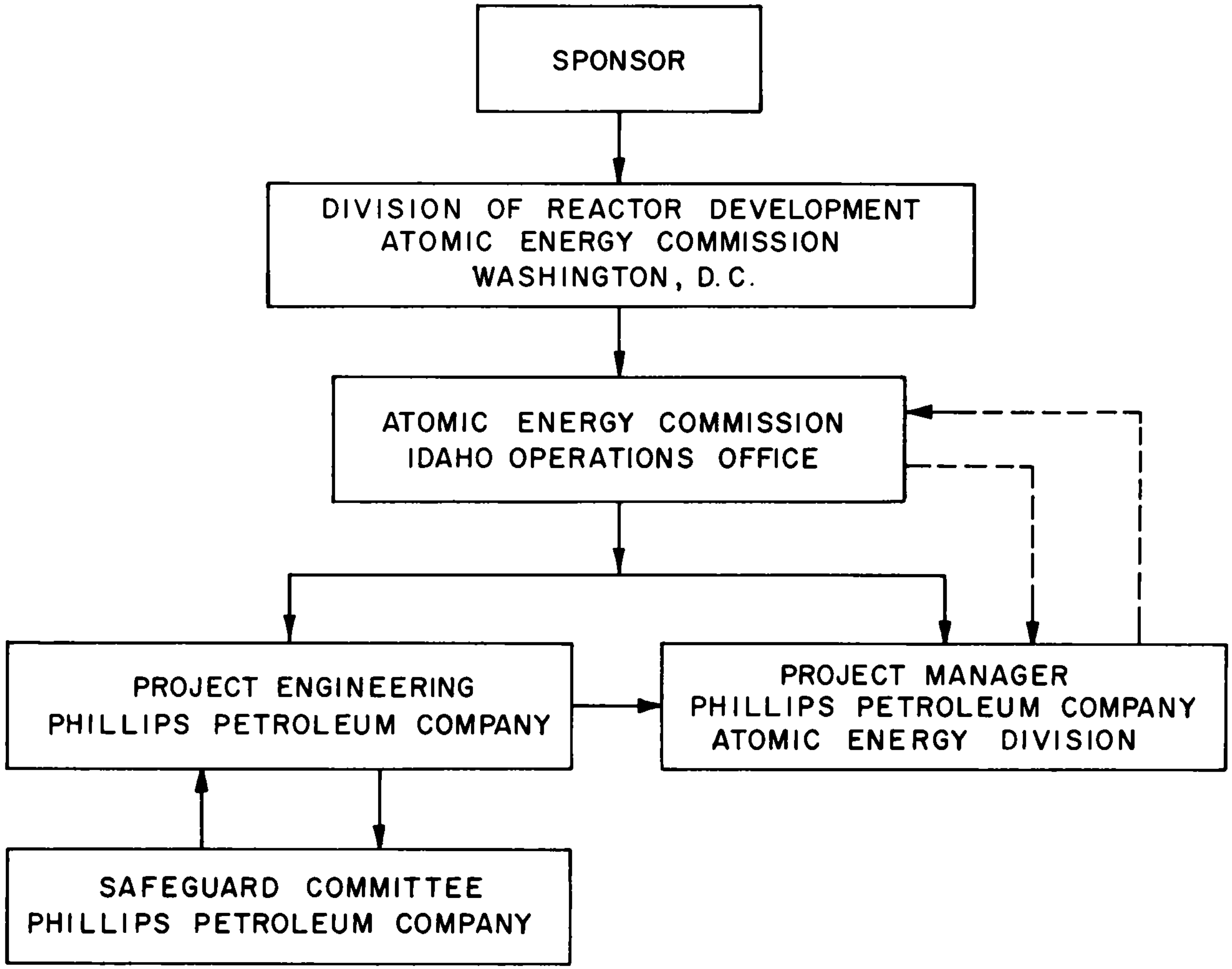

Fig. 2—Control of AEC sponsored irradiation programs.

The second procedural phase concerns the actual conduct of the experiment and involves inspection and approval of the equipment and test specimens furnished by the sponsor. When in-pile and out-of-pile equipment is received it is inspected and dimensionally checked to insure compliance with design and conformance with program approvals and safety and engineering standards. Upon final inspection and pressure testing the results are transmitted to the Safeguard Committee for approval to operate the experiment under specified conditions. If conditions are to be changed or the test specimen is to be replaced, Safeguard Committee approval of each change is required. In this manner each major experiment is under almost constant surveillance of the project engineer assigned to follow the program and by members of the Safeguard Committee. These procedures, while somewhat lengthy, have resulted in the absence of significant incidents during the conduct of hundreds of experiments in the MTR and ETR.

IMPORTANT CONSIDERATIONS IN DESIGN, APPROVAL, INSTALLATION AND OPERATION OF EXPERIMENTS

The successful conduct of any experiment in either reactor obviously requires that each be compatible with the reactor from the standpoint of safety, available space and operational feasibility. Simple experiments such as encapsulated target materials are fairly well standardized and generate few troubles but large engineered experiments such as high temperature water or liquid metal cooled loops sometimes provide formidable problems.

Since many of the engineered experiments are large, complicated and expensive it is necessary to insure to the greatest practical extent that each experiment will yield the data desired by the sponsor and at the same time be capable of safe and essentially continuous operation. These conditions and objectives require engineering liaison between Phillips and the sponsoring organization from the earliest stages of design through fabrication, installation, testing, operation and removal of the experiment. This liaison assures us and the sponsor that the equipment as designed and constructed will meet MTR-ETR engineering standards, is capable of installation in the in-pile and out-of-pile space provided and will operate with sufficient reliability not to interfere unduly with other experiments or normal reactor operation. Reasonable operational reliability in each experiment is imperative since failure of one experiment usually results in loss of neutrons by all programs while the malfunctioning experiment is repaired, removed or isolated.

When agreement has been reached between the sponsor and Phillips on the design aspects of the equipment, the loop components are assembled by the sponsor or a fabrication contractor into units which can be handled with reasonable facility in the available space around the reactors. Where practicable, the loops are completely assembled and tested in the sponsor's or vendor's shops prior to partial disassembly for shipment to the MTR or ETR. The loop equipment is inspected, reassembled, shielded and retested at the reactor prior to insertion of the in-pile component. All experiments are installed and removed by Phillips personnel.

As mentioned previously, conduct of many of the more complicated experiments requires the acceptance of some calculated risk that the experiments will not operate for the scheduled periods without some type of failure. The requirement for risk acceptance is brought about by such factors as incomplete data on the effects of radiation damage and inability to calculate within known limits the stresses which may be set up due to uneven gamma and fission heating and thermal cycling. Further, if the behavior of some experiments could be accurately predicted there would be little or no incentive to conduct the experiments.

Evaluation of the risk involved in conducting a particular experiment makes use of the best known and accepted methods of performing engineering and nuclear calculations on heat transfer, heat generation, fluid flow and stress, and is based on the knowledge that, in the light of the best available data, the materials used and fabrication methods employed are adequate for the conditions under which the experiment will be conducted.

As might be expected, the careful review which we apply to each design and the inspection to which all equipment is subjected does not completely prevent the failure of some experiments. As a minimum, a failure usually means that results from a portion of the program are lost and that other experiments are adversely affected while the failure is corrected or isolated. To date no serious failures have occurred from the standpoint of personnel exposure or equipment damage although on several occasions it has been necessary to evacuate the reactor building of most personnel for periods up to several hours while experiment removal and general cleanup were effected.

PROCEDURES FOR EXPERIMENTERS

Sponsors of the major irradiation programs usually desire to have permanent or part time representatives at the MTR and ETR. This arrangement not only assists the sponsor in maintaining close contact with his program but assists Phillips in obtaining quick decisions and action from the sponsor on program and equipment changes. Prior to the assignment of a sponsor representative to the MTR-ETR, Phillips and the sponsoring organization enter into an AEC approved agreement covering such matters as liability, workmen's compensation, supervision and conduct. This basic agreement permits observation but not participation in the sponsor's work. The representatives are under Phillips administrative supervision while at the reactor site and are expected to adhere to general rules for conduct and safety applicable to all Phillips employees. In the event a sponsor desires his representative to participate actively in the conduct of experimental work, a supplemental agreement is entered into between the parties which specifies further conditions under which the representative may perform work in connection with his own program. Active participation in experimental work by sponsor's representatives has the desired effect of reducing Phillips' requirements for experienced technical personnel.

OPERATING COSTS

The net operating cost for the reactors is approximately $675 per hour for the ETR and $600 per hour for the MTR including all direct and indirect costs but excluding the cost of experimental equipment and labor furnished by sponsors and the cost of installing and operating major experiments.

Figure 3 shows a breakdown of costs into major direct and indirect categories. Miscellaneous Direct Costs include travel and moving expenses, employee benefits, payroll taxes, telephone and telegraph costs and similar items. Administrative Services include administrative overhead, accounting service, library, purchasing and warehousing, personnel service such as recruiting and education, while National Reactor Testing Station (NRTS) services include bus transportation, cafeteria service, maintenance of roads and power distribution systems, radio networks and fire alarm systems. The Idaho Operations Office (IDO) Services include Site security, fire protection, medical services, personnel radiation metering service and NRTS radiation monitoring, all of which are provided by the AEC for all operating contractors at the Testing Station.

Indirect costs are apportioned on the basis of direct costs exclusive of depreciation and uranium cost. The AEC added factor covers Idaho Operations Office and AEC general overhead and is allocated as a percentage of all other operating costs.

	MTR	ETR
LABOR	$1,290	$1,151
MATERIALS AND SUPPLIES	281	477
PLANT UTILITIES	427	399
MISCELLANEOUS DIRECT	201	207
FUEL BURNUP	321	601
FUEL REPROCESSING	112	337
DEPRECIATION	731	590
ADMINISTRATIVE SERVICES	567	601
NRTS SERVICES	193	199
IDO SERVICES	258	274
NRTS DEPRECIATION	196	213
AEC ADDED FACTOR	687	757
	$5,264	$5,803

Fig. 3—Estimated MTR and ETR annual operating cost (thousands of dollars).

It is readily apparent that reactor operation is expensive and since most of the costs are incurred whether or not the reactors are fully occupied and in operation, the efficiency with which the experimental programs are scheduled and installed to insure maximum on-stream time assumes a position of major monetary importance. On-stream time for the MTR was 69.4 per cent in 1957 and 73.8 per cent in 1958. The operating time for the ETR will be somewhat less than that for the MTR because of the greater size and complexity of many ETR experiments and the attendant increase in down time required for installation of in-pile components.

IRRADIATION PRICING POLICY

Both the MTR and ETR are used for public service irradiations. In order to perform these irradiations it was necessary to establish a pricing basis which would fulfill the full cost-recovery requirements of the Atomic Energy Commission and provide an equitable price for the various experimental facilities. Fundamentally, the problem was one of distributing the total cost of operating the reactors over the available experimental space in such a manner as to assure full recovery of costs from the anticipated use of the irradiation space by public service and AEC-sponsored experiments. A second consideration was that the method of pricing should not contribute unduly to the cost of operating the reactors nor result in increased reactor down time. Of course, the price established for various irradiation spaces in the reactors must bear a direct relation to the neutron flux in each location since neutrons represent the product we are selling.

A common price has been established for irradiations in the MTR and ETR to permit flexibility in conducting experiments in either reactor as dictated by available space and the flux and space requirements of the experiment. This means that, even though separate cost accounting is maintained for each reactor, the "full cost-recovery" principle is applied to the complex and the experimenter is charged the same for equivalent irradiation in either reactor.

In order to establish a reasonable pricing policy it was necessary to define costs, define a standard production unit and spread the costs over the production units in such a manner that the estimated sales of production units yield the total cost of operating the facilities during a fiscal year. The costs involved are the operating costs discussed previously.

In developing a standard production unit we first selected a "normal" experimental piece in each MTR facility. This normal experimental piece is a 2-hole piece in the 3" x 3" reflector positions and a 1-hole piece in the core positions. The vertical length of the piece times the number of holes in the assumed standard is the "normal" experimental area. Although the diameters of the holes in the two "normal" pieces are slightly different, we have eliminated the diameter from price consideration and the price is the same for a vertical inch of sample at a given flux in either of the two types of facilities.

In accordance with this "normalizing" philosophy we have established irradiation volume for the larger facilities in the ETR by dividing them into 3" x 3" "normal" experimental pieces. Thus, disregarding flux, the charge for a 6" x 6" facility is four times the charge for a 3" x 3" piece and the charge for a 9" x 9" facility is nine times that for a 3" x 3" piece.

A determination of experimental volume in the MTR horizontal facilities is somewhat different inasmuch as they do not accommodate a standard 3" x 3" piece. For pricing purposes their volume is based on an assumed cylinder having a diameter of the facility and a height starting from the point nearest the core and proceeding from the core to that point at which the flux is one-half the maximum.

With the experimental volume thus determined, we next evaluated the thermal flux in the various facilities. These values were initially established by irradiating cobalt wires in key facilities throughout the reactors. We attempt to keep our flux data current through a continuing program of gathering data required for experiment loading purposes as well as for making adjustments in the pricing schedule.

Current flux data for the ETR are such that it is not apparent whether or not we can continue to "normalize" the thermal flux values with the same degree of accuracy that we have at the MTR. Because of the larger core and the experimental facilities penetrating the core, the flux pattern varies somewhat as the experimental load changes. If this variation is significant, it will introduce another variable into our pricing system.

Utilizing the standard experimental volume and the normalized average thermal flux, the number of irradiation units are computed for each experimental facility in terms of reactor megawatt-day production. An irradiation unit is defined as one vertical inch of experimental space irradiated to 10^{20} nvt.

Prior to the beginning of each fiscal year we estimate the number of megawatt days that will be produced during that year (on-stream time) and the percentage of the normal flux volume that will be occupied (occupancy factor). This yields the estimated number of irradiation units which will be produced and sold during the year. We divide this number into the estimated costs for the MTR-ETR complex to arrive at a unit cost which is used for the fiscal year.

The problem of pricing test reactor space has received attention over the past two years from people who either operate or contemplate the operation of test reactors. Most of the proposed pricing schemes accept the space-flux concept but propose more elaborate methods for measuring flux values. The final objective of a space-flux pricing method is, of course, a technique for measuring accurately and economically actual values for a given irradiation. We know of no method by which this can be accomplished at the present time. To be sure, we could use large amounts of reactor time making repeated determinations of flux values at the beginning, middle and end of fuel cycles and thus improve our values for normalized average flux but the benefits to be gained do not appear to justify the resultant loss in reactor full-power operating time.

It may appear that our procedures and policies for reactor operation and approval of experimental programs are somewhat lengthy and detailed but we are convinced from experience during seven years of operation that a high accident rate coupled with possible personnel injury and property damage might well result from the absence of these or similar policies.

ORGANIZATION OF THE BR-2 SECTION

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The BR-2 section is the unit in the laboratory of the Center for Nuclear Energy Studies at Mol, responsible for using the BR-2 installations according to the schedule of experimental work set by the management of the laboratory and following the safety rules approved by the management.

The BR-2 section consists of the following 6 groups:

Use,
Maintenance,
Analysis,
Health Control,
Experiments,
Administration.

The first three groups have to keep the BR-2 installations in operation. The fourth sees to it that no hazards are incurred by the personnel during operation, whereas the Experiments Group maintains liaison between the section and the experimenters. The Administration Group provides administrative support for the section.

The functions of each of the three first groups are defined as follows:

Use:
- this group must start the reactor, keep it at the operating power level and shut it down in accordance with the schedule set by the section chief.
- it must modify the charge of the reactor after all operations equivalent to the withdrawal of a fuel element.
- in general it must perform in the installations all the operations that take place more frequently than once a day.

Maintenance:
- it must perform in the installations all the operations that enable the Use Group to carry out its tasks.

Analysis:
- it must use the nuclear model in order to obtain for each reactor charge the values of the nuclear

parameters needed by the leaders of the Use and Maintenance groups to work out their schedule of operations.

- it must analyze information provided by the other groups in order to improve the operation of the BR-2 installations.

Each group is headed by a group leader, a civil engineer. The groups are divided in teams headed by a team foreman (a technician).

The Use Group consists of 5 teams, each having six men:

1 team foreman,

1 assistant foreman, in charge of the machines control room,

1 electromechanical technician,

1 attendant in charge of the reactor control room,

1 instruments technician,

1 radiation control technician.

The five teams are necessary for a 45 hour work week, but a number of hours are available to be used in the form of reinforcements for the operating day shift.

The team make up should be such as to make it easy to go the rounds through the installations, and above all to form an emergency subteam of two to four men for investigation, while keeping at least one man in each control room.

The Maintenance Group consists of three maintenance teams and a design office.

The three teams are: Instrument Maintenance,
Electromechanical Maintenance,
General Maintenance.

These teams have only very few technicians (2 to 5) because the team foremen use for their work the personnel of the maintenance units of the Center for Nuclear Energy Studies, either by directly taking charge of this personnel or by giving technical advice to existing organizational groups of this personnel.

The Analysis Group consists of two teams: one measurement team and one computation team, each team consisting of one foreman and three technicians.

The Radiation Control Group has only one team. Each technician is detailed to an operating team. The team foreman receives all the information obtained by the technicians and with the group leader he serves as additional support for the control of the operations accomplished during the day.

The Experiments Group also has only one team, its size being proportional to the schedule of experimental work of the reactor. Each of its members serves as a liaison worker between the BR-2 section and the experimenter. While the experiment is being set up it is the function of the group leader to prepare all the requirements for information for the other groups of the section, and after the experiment, he will transmit all the information obtained to the experimenter.

In the BR-2 installations every operation is carried out by the orders of a team foreman in accordance with a procedure or written work rules approved by the section chief.

Observations made by section personnel are recorded on the "Remarks" sheets of the operation procedures or in the diaries kept by each team.

The observations gathered during one day are summarized by the night team foreman in the daily report presented at the morning meeting.

The object of the morning meeting in which all the organizational units of the section participate is:

- checking the daily report,
- checking report on previous failures,
- giving out instructions for writing up the work orders,
- discussion and approval of the work orders.

Failure reports are written by the appropriate group leaders or by the Analysis Group, depending on the importance of the incident. They consist of three parts: a detailed and objective description of the failure, an analysis of its causes, and suggestions for their elimination. Normally a failure report brings about the working out of a new procedure, the modification of an existing procedure, or the writing of a work order.

As the observations accumulate they are subjected to analysis and recombined, to become the material for more general reports on different subjects that are of interest for high flux research reactors. This, obviously, does not refer to observations made directly as the result of an experiment. These observations are given to the experimenter, in the form indicated by him.

The BR-2 section was set up at approximately the same time as the construction of the installations was started. During the whole period of construction and part of the preliminary run the section has a reduced make-up.

The function of the reduced section is defined as follows:

- scanning the study and construction documents produced by the study groups and the builders working on the BR-2 project,
- supervision of the building site of BR-2,
- preliminary run of the BR-2 installations.

This function was defined as described above in order to furnish future users as complete a knowledge as possible of the installations and at the same time to help the Construction Department of the Center for Nuclear Energy Studies in carrying out its task.

In view of its function the reduced section consists of all its organizational units: group leaders and team foremen of the operational section and somewhat less than half the technicians and employees of the operational section.

Recruiting of personnel started in the supervisory ranks and proceeded towards the hiring of technicians gradually as each became familiar with the installation, and the site required more receiving and supervising personnel. In this way each member of the section was sure to acquire a detailed knowledge of the complete installation.

At present the section is recruiting the last technicians that are to form the reduced section. Three months before the preliminary run ends the rest of the section members that are going to be part of the operational section will be hired. They will be trained during these three months in the different functions they will have to fulfill, by the members of the reduced section.

It is expected that by proceding in this fashion personnel of the BR-2 section will acquire the best possible knowledge of the installations and will be able to perfect for the section a method of operating that best suits its duties. When the BR-2 reactor will become active it will thus be ready for use with a maximum of safety and efficiency.

Finally we want to say a few words on how an experiment can be performed in BR-2. Technical data available so far will be given to you by J. Planquart in his paper next Friday "Design and testing of experiments". Here we will only outline the administrative aspects of this question.

In an experiment we will distinguish the four following phases:

Establishing contact,
Setting up the experiment,
Irradiation,
Examination after irradiation.

In the first phase the experimenter contacts the laboratory management and submits his experiment project. The BR-2 section examins this request from the point of view of reactor work schedule, safety of use and feasibility of the experiment. The "Preparation of Experiments" group of the Technology Section makes a detailed investigation into the technical requirements of the experiment. These different examinations end in a report that decides on the feasibility of the experiment. If the conclusion is positive this phase ends by a decision to carry out the experiment, and in the BR-2 section by appointing the liaison worker.

In the second phase the experimenter can choose three different ways for setting up the experiment:

1. set up the experiment on his own,
2. have the experiment set up by another group,
3. have the experiment set up by the "Preparation of Experiments" group of the Technology Section of the Center for Nuclear Energy Studies.

In case methods one or two are chosen, the "Preparation of Experiments" group will act as technical adviser and as receiving agent for the materials used.

While the experiment is being set up, when the experimental design is completed, it must be submitted for investigation to the Nuclear Safety Council, and the responsibility for submitting it rests with the BR-2 section.

This phase of setting up the experiment ends with reception of the experiment in the hall for the assembly of experiments and its transmittal to the BR-2 section.

During irradiation the responsibility for the material belongs to the BR-2 section and the experimenter is part of the BR-2 section as technical adviser.

The experiment is removed from the reactor and dismantled at the end of the irradiation in accordance with instructions of the experimenter. Whatever the experimenter does not wish to take back is removed by the BR-2 section.

SOME SPECIAL PROBLEMS POSED BY THE OPERATION OF LABORATORY PILES

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GENERAL OBSERVATIONS

The problems posed by the operation of research piles or, more properly speaking, of "laboratory piles" are different from those posed by other piles and even more so in the classical devices (for example, coal electrical powerhouses) having a similar functional character.

This type of pile, generally with high or even very high neutron flux, constitutes essentially a "factory" of "high quality neutrons" which must be considered by us as a direct goal and not a means.

These devices have capacities of irradiation as numerous and varied as possible, which are used simultaneously for several researches or production (isotopes). They constitute a collective instrument some of whose experimental devices, moreover, when set up, must be considered from the standpoint of safety as an embryo pile in the womb of a feeding pile. The conditions resulting therefrom have the following essential difficulties:

- operation of the pile becomes more complex and more difficult;
- necessity of providing considerable and suitable handling apparatus;
- search for a daily program of operation giving satisfaction to all the users. This imposes a common discipline.
- etc...

Moreover, the characteristics of the irradiation zones (dimensions and number) were determined at the time of the study of the pile as a function, on one hand, of the general program of research of the Atomic Energy Commission at the corresponding time and, on the other hand, the existing technological possibilities.

This program evolving in the course of time by reason of the results obtained either by our own research teams on our other piles, or by foreign research workers whose works were published, the experimental possibilities offered no longer correspond at all times to the new needs resulting from such evolution. It is therefore indispensible to adapt, within the possible limits, the present apparatus to new uses provided for or anticipated; this involves numerous operations some of which may be very important; they must be performed in such a way that the pile be available as much time as possible. It is not exaggerating to state that a laboratory pile, before being put into service no longer corresponds to the formulated need, and even that the absence of changes might be interpreted as a sign of a halt in research.

To sum up, our laboratory piles may be differentiated essentially by:

- the high flux;
- the collective nature of utilization;
- the rapid growth of their uses.

The organization of the use of EL-2 and EL-3 piles takes into account these peculiarities and a six-year experience of continuous service.

The EL-2 and EL-3 piles have been designed to permit different irradiations of slow, thermal, rapid neutrons and of gamma (γ) rays.

The distribution of the experimental devices may be conceived as follows:

- fundamental (basic) research: 20 percent
- technological research : 60 percent
- manufacture of radio-isotopes: 20 percent

These figures make it possible to obtain a very general idea of the equipment necessary to permit the use of the channels (equipment of the pile halls). Moreover, the pile itself is a laboratory whose different circuits make up an experimental base on a true scale.

In addition, these piles also serve for the training of the personnel called upon to direct other reactors and, in a general way, to work near these apparatuses.

The use of the piles should take into consideration the difficult conditions which are directly involved in this type of apparatus. The most important of these conditions are the following:

a) high flux pile;
b) collective laboratory;
c) continuous changes and additions.

The high-flux type of pile, as far as we are concerned, has the drawback, on one hand, of requiring the utilization of the materials almost at their critical limits and, on the other hand, the recharging of the pile at a relatively high rate. It is therefore necessary to study the cells in very great detail and to improve them as much as possible. Moreover, the means of storing the rods in the course of cooling must be great (capacity of two to three sets of rods). In a like manner, the apparatus for observing the cells and for the separation of the uranium elements must be designed and used at such a rate that it is not likely to cause a failure in the reactor (broken rods) or the blocking of the fuel utilization cycle.

But the most important hardships basically result from the fact that these piles are utilized equally for testing the new types of elements intended for performing technological research and for the need to be able to win, under the most unfavorable conditions, the energy arising from fission or radiation, whether the pile is in operation or at rest. On the one hand, this requires a very special safety design (number, quality, reliability, etc.) and, on the other hand, the carrying out of strict orders on the part of the personnel, in order to avoid accidents.

The piles contain a large number of circuits or loops whose smooth operation is indispensable as this, as has been observed, affects the operation of the pile. Every auxilliary circuit should be treated like the main cooling circuit, especially the essential parts should be provided in a sufficient number (e.g., two in service and one at rest), their electrical supply coming from the priority circuits being always under voltage. Moreover, the basic safety rules (flow, temperature, pressure, etc.) are inserted in the channel controlling the drop of the pile safety rods.

The control and the supervision of these circuits are difficult to centralize because this would lead to an equipment which is too complex and therefore difficult to use. It is also to be noted that, as not all of the experimental loops are of a permanent nature, the control panels would always be in a state of change; this would have the risk of making them unavailable and would, as a result, limit the annual duration of use of the reactor.

In order to mitigate these difficulties, one is led to have multiple panels distributed in a certain number of locations; only the safety devices are centralized.

The personnel must know this equipment in detail and make periodic checks to make sure they are functioning properly. The personnel and the material should therefore grow at the same time so that the personnel would be able to interpret, to a certain extent, the abnormal conditions which may possibly be brought about.

The handling contrivances of the irradiated devices are likewise relatively complex and difficult to handle because a minor failure frequently takes on considerable proportions.

The experience acquired on the pile and the setting up of new experiments involve continuous work for adapting the installations. It is necessary to carry them out without interfering with the operation or the experimenters. Thus this work has to be carried out, at times, under conditions which are not rational from the industrial standpoint. The personnel should bear in mind this special "atmosphere" in their daily work which, at times, is complicated even more by delays in certain deliveries or experimental equipment.

The main technical difficulties to be solved are the following:

- handling of the active uranium and irradiated devices;
- handling of the continuous cooling of the pile and of certain loops;
- work on the contaminated materials.

The whole of the problems arising shows that the task assigned to the personnel in charge of the operation is not a simple one. The control and supervision of a pile are not limited to merely the nuclear part which generally does not involve any special or troublesome conditions.

The numerous disciplines involved compel us to look for a relatively numerous personnel with diversified training. As a rule they cannot be immediately employed and must undergo an adaptation period whose duration varies with their basic training and the roles with which they are to be entrusted. Of course, it would be desirable to organize systematic courses for the whole force in order to have a personnel of a homogeneous nature. But as far as we are concerned, this has not been possible because we are confronted with personnel recruiting difficulties and at the same time we have to carry out urgent tasks in short periods of time. We believe, nevertheless, that provided that one makes a good choice of applicants, an adaptation based on courses of instruction is not always necessary. However, lectures should be given in order to define radioactivity, state its dangers and prescribe the rules to be followed.

The personnel we need may be divided into three main categories:

- electronic technicians;
- electricians;
- mechanics;

those in the first category being technical personnel and those of the other two categories being technical personnel and workers. They can be entrusted either with the operation or the maintenance (up to heavy labor).

In our piles we have sought to give a responsibility to each individual. The materiel has in effect been designed and distributed in such a way that within the framework of definite rules, every individual must necessarily know the apparatus of which he is in charge so that he can act upon it in case of failure. Instead of a rigorous automatism we have chosen a less absolute and more human solution but, on the other hand, it would expose us to the hazards of human failure. Nevertheless, we feel that, bearing in mind the prototype nature of all our installations which always require continuous adjustments, the risks entailed are smaller, because we maintain intact the intelligence and the reflexes of the individuals who are in a position to react according to the anomaly which arises. This argument is valid except that the safety devices which would require an exceedingly rapid human reaction should be automatic (alarm channel). Among the latter one should note those tied directly to the reactor (power jump, reactivity, cell temperature, minimum flow in of coolant circuit, basic parameters of loops, etc.).

This method also permits a better growth in the quality of the personnel, a more exact estimate of their value and thus a professional development more in keeping with their aptitudes; this may to a large extent compensate for the lack of a systematic, theoretical training.

The technical problems to be solved and the conditions imposed by the experimentation cause one to provide for a relatively large personnel in the engineer, technician, and worker categories. We have been led to provide for the following distribution:

- a total personnel for EL-2

- a total personnel for EL-3
- a total personnel for the group of installations.

The total force attached directly to the piles has been calculated in such a way that their total utilization can be assured for the entire year. The base is made up of operating teams six in number.

The personnel common to the group of installations is destined to make the studies, the developments, and the repairs which are beyond the usual damages. They constitute a group of technicians very versant with the construction, design, and installation of piles. Generally speaking, these technicians have been selected from the pile operating force and they have a profound knowledge of the apparatus.

The uninformed visitor to our laboratory piles gets an impression of apathy and disorder.

Behind this screen, however, is hidden the constant work of adjustment in the service of all, requiring from those responsible a technical competence and a constant devotion.

In the following pages we have tried to indicate a certain number of special points illustrated by several examples showing how the organization is set up and how it functions.

CHAPTER A

PRESENT USE

I - GENERAL REMARKS

The Saclay EL-2 and EL-3 piles whose use is assured by the technical section of the S.G.P.S., have been operating, respectively, since 1952 and 1957.

The EL-2 pile is a natural uranium pile, with heavy water moderator cooled by CO_2, a present power of 2 MW, for a flux of 8×10^{12} n/cm^2/sec.

The EL-3 pile is a slightly enriched uranium pile, heavy water moderator, cooled by heavy water, with a rated power of 15 MW, for a flux of 10^{14} n/cm^2/sec.

The table below gives the principal characteristics of these two piles as of 1 September 1959.

	Total energy released in KWH	MWJ/T on the most charged cell	Possibilities of manufacture of artificial radio-elements	Number of experiments on the pile on 1 Sep 59
EL-2	40,612.000	600 (nat. Ur.) 1200 (nat. Ur.)	250	13
EL-3	43,750,000	4800	360	10

The experimental cells of different characteristics are not included in this table (25 cells in EL-3 - 5 cells in EL-2).

The few minor incidents which took place in connection with the use of these piles did not involve any accidents to an individual nor any irradiation of the personnel higher than the acceptable doses.

In the case of EL-2, these incidents were generally due to breaks in the jackets and in the case of EL-3, to contaminations of circuits or sites due to failures.

The operation of these piles have therefore permitted us to draw the following conclusions regarding:

- the operation proper;
- the routine maintenance;
- the changes brought about or planned;
- additions made or to be made.

II - USE

The use of a research pile should ensure:

- the continuity of its running at a level of predetermined power for the benefit of the experimenters and under the best safety conditions.

II-1. Continuity is assured by:

- the apparatus,
- the personnel,
- the work methods.

II-1.1. The Apparatus

A. The design of the apparatus, in number as well as in character, is such as to have a total safety of operation, that is why, generally speaking, two apparatuses are working and one is in reserve; and of the two running, a single one may be used.

Example: D_2O circuit: three pumps of which two are normally in use.

CO_2 circuit: three blowers of which two are normally in use.

Even if the power to be drawn may be supplied by a single machine.

B. Because it is indispensable to ensure the continuity of the operation, the time of setting the apparatus in a working condition is to be cut to a minimum, therefore:

- All parts of a circuit should be easily accessible, removable, be protected from the other active circuits and allow for isolation from the other circuits.

The control of the apparatus in active circuit should be by remote control (electrical controls or Teleflex type).

- The sites where the active circuits pass should allow for easy decontamination (removable paint, ground tightness).
- The apparatus designed for handling the pile at work must assure the necessary protection of the personnel.
- The continuity of electrical supply should be assured in case of external power failure for the principal apparatuses.

II-1.2. The Personnel

The pile operating personnel is grouped in shifts to ensure a continuous service. Six teams are provided for. The two supplementary teams take care of the maintenance and the possible complement of the operating teams.

This operating personnel ensures the operation of the pile. This gives the operation the advantage of having a force versant with the significance of an order and may mitigate against possible deficiencies in the apparatus when the cause is known and it has probably made it possible to avoid numerous failures or accidents.

II-1.3. Methods of Work

These work methods which will be treated further down in connection with maintenance, may at times appear disturbing and from the industrial point of view they may not always appear rational, but it must not be forgotten that the pile is an instrument of work that must be run no matter what the cost and that one cannot always have access to this apparatus during operation.

II-2. The maintenance of the power at the desired level is assured by the operation of the pile properly speaking and the partial recharging with fuel at the desired time.

II-2.1. The operation of the pile is assured by mechanisms of control and supervision.

- The control mechanisms (especially the regulating rods) are in duplicate but the fine control is effected only by a single rod so as to reserve the possibility of passing over the second one in case of failure in the first one. This is the principle of continuity of service.

- The crude control is done systematically in order to have a uniformly variable flux and to easily evaluate the rate of irradiation.

- The controls are at least double: four power channels of which one is a logarithmic channel that will soon be duplicated.

Knowing the temperatures of each cell is very important; for EL-3, this is effected by two methods: slow survey in a few minutes and rapid survey in a few seconds.

Such knowledge is more important, in our opinion, than the knowledge of the value of the flows which are likewise controlled.

A separate check to detect ruptures in the jackets is effected but there is an overall checking of the helium circuit in the EL-3 and of the CO_2 coolant circuit in the EL-2. These control rooms are in duplicate.

- The files of the experiments in progress are constantly at the disposal of the heads of the watches.

- It seems advantageous to have 2 control rooms, a main room grouping the main supervision and control elements of the pile properly speaking and an auxilliary one grouping all of the "supporting" control and checking devices.

The first room is above all reserved for the operators of the pile and includes a reduced number of records and a minimum personnel in spite of its dimensions which must remain large.

The second room is reserved for the electricians and mechanics.

The starting of the machines (action on push-turn buttons) is effected by the electrician on request of the mechanic who has set up the circuits, as the electrician is more likely to know whether the supply of the machine is ensured under the required conditions.

This solution, adopted for EL-3, results from difficulties found with EL-2 because of the complete separation of these teams.

- The separation of the control and the checking systems on two panels makes it possible

to separate the functions,

to relieve the principal control panel,

to reduce the risks of panic,

to permit the personnel of the machine room to act in a "non-blind" fashion,

to control, in case of possible accident, all the apparatus from a board located outside of the tight zone. The entire personnel is under the orders of a chief of the shift, who is responsible for the operation of the pile.

As the electronic section is the most complex, it is normal procedure to entrust this function to one who has a background in electronics.

A certain number of documents make it possible to ensure continuity of service among the various shifts, particularly the transmission of orders, which remains the most delicate point of all, bearing in mind the complexity of the pile, the frequent changes, and the operation which does not permit keeping the personnel for a sufficient length of time.

II-2.2.

The recharging of the pile is not assured in a simple manner. This is due to the fact that many fuels are tested on the pile.

Basically, this pile is recharged with a sufficient number of cells to ensure continuous running at the desired power for three or four weeks.

It follows from this that the operation is ensured up to the extreme limit of reactivity. Work has been done with a reserve of 100 pcm whereas the available reactivity for a virgin and cold pile is about 15,000 pcm; consequently, there is the drawback of immediate re-starting in case of the drops of rods (Xenon poisoning).

The curve shown herewith demonstrates that the reserve of reactivity does not vary in a linear fashion. It would be of interest to be able to push this experiment still further, up to the complete exhaustion of reactivity at zero power.

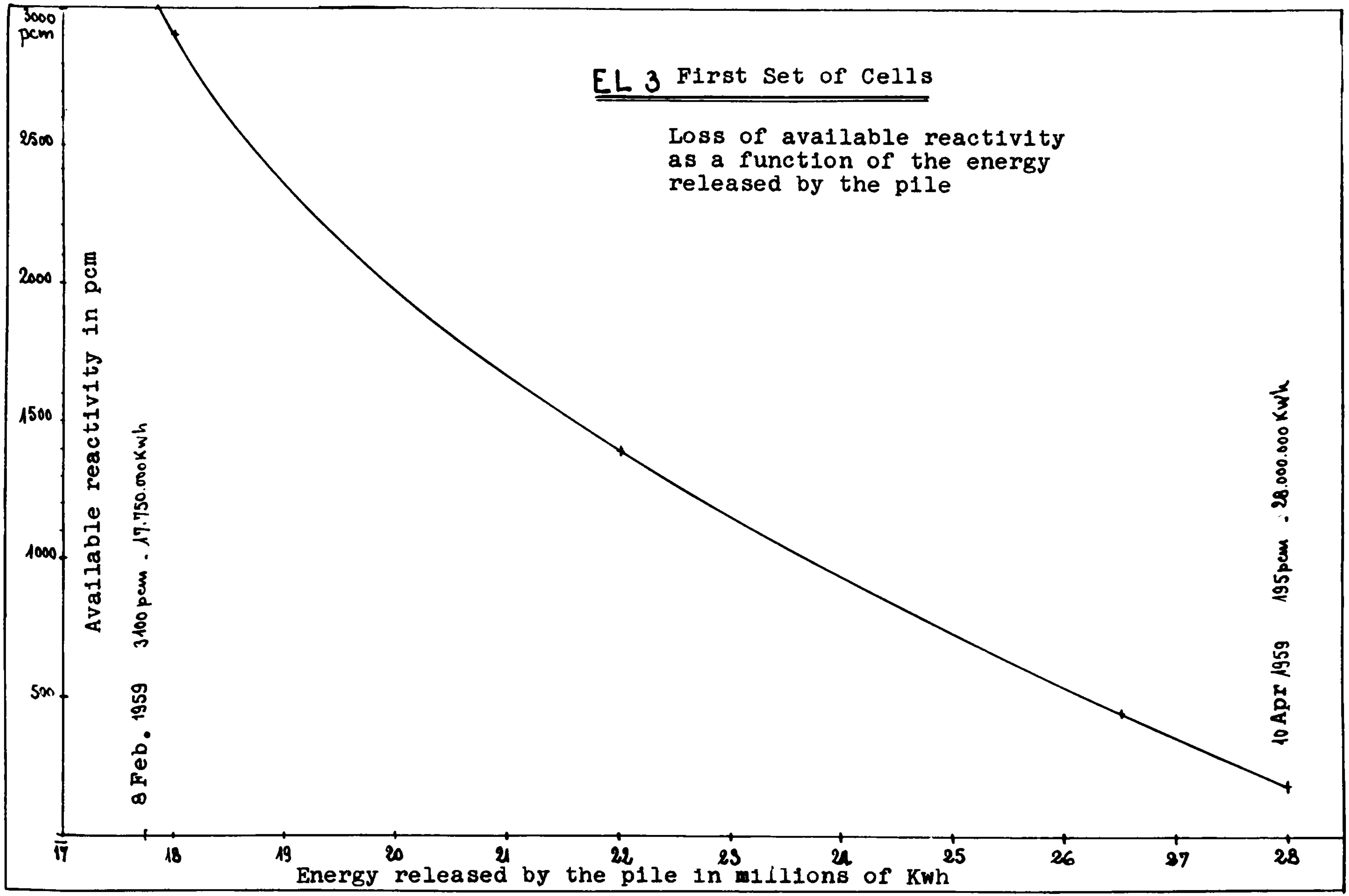
In the EL-2-pile the sets of rods are periodically changed except for the special experimental cells equipped for the metallurgy department.

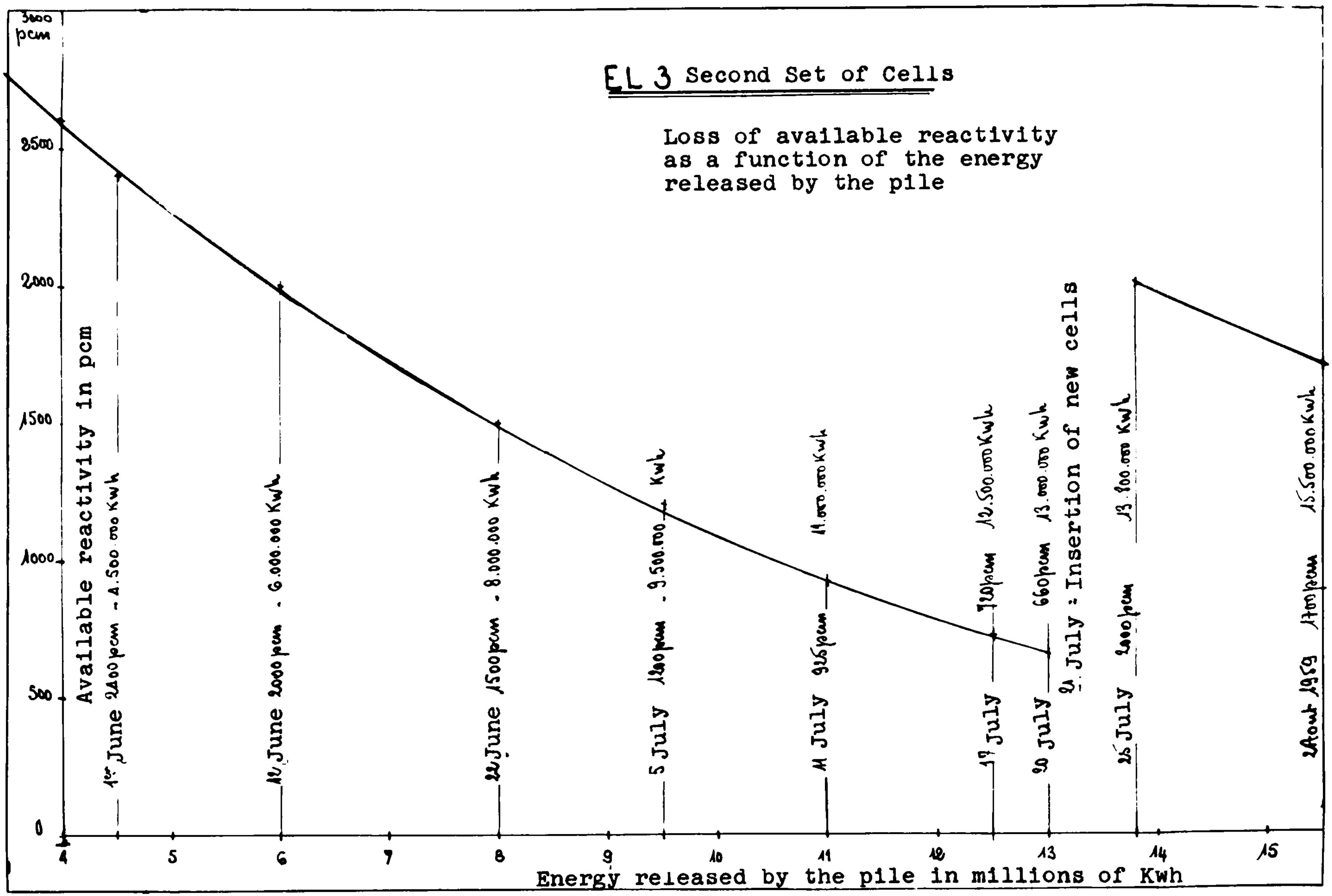
II-2.3.

This problem of operating the pile is a classical one with the condition that simple nuclear checking devices and perfect operation be provided. But minor classical failure become major failures because of the presence of fission products or activities.

II-3.

The pile is operated for the benefit of the experimenters. As far as our problems are concerned, their experiments may be classified into two categories: those which have an interaction with the piles and those which do not.


EL 3 First Set of Cells
Loss of available reactivity as a function of the energy released by the pile
Available reactivity in pcm
3000 pcm
2500
2000
1500
1000
500
8 Feb. 1959 3400 pcm . 17.750.000 Kwh
10 Apr 1959 195 pcm . 28.000.000 Kwh
17
18
19
20
21
22
23
24
25
26
27
28
Energy released by the pile in millions of Kwh


EL 3 Second Set of Cells
Loss of available reactivity as a function of the energy released by the pile
Available reactivity in pcm
Energy released by the pile in millions of Kwh
3000 pcm
2500
2000
1500
1000
500
0
4
5
6
7
8
9
10
11
12
13
14
15
1er June 2400 pcm - 4.500.000 Kwh
12 June 2000 pcm - 6.000.000 Kwh
22 June 1500 pcm - 8.000.000 Kwh
5 July 1180 pcm - 9.500.000 Kwh
11 July 925 pcm 11.000.000 Kwh
17 July 720 pcm 12.500.000 Kwh
20 July 660 pcm 13.000.000 Kwh
21 July : Insertion of new cells
25 July 2000 pcm 13.800.000 Kwh
2 Aout 1959 1700 pcm 15.500.000 Kwh

II-3.1.

The experiments which have no interaction with the pile are those which generally use a beam coming out of the pile. The only important problem for the operator is that of protection, frequently minimized at the outset.

II-3.2

The experiments which have an interaction with the pile are the most numerous. One may distinguish between:

- fuel tests;
- simple radiations;
- artificial radioelements which may be loaded and unloaded while the pile is running; and
- the loops.

The fuel tests are effected in the form of special slugs mounted like those of standard cells. Their inspection require systematic observations outside the pile; therefore, there are more frequent shutoffs of the pile. We try to have the shutoffs and recharging periods coincide. Moreover, the study on a true scale of the behavior of a certain type of fuel may result in ruptures of the jackets which may have grave consequences on the operation of the pile.

Simple irradiations in the horizontal or vertical channels utilize the normal cooling of the pile. This is a difficult problem because the heating of the sample is not well known. The unloading of these samples frequently causes surprises due to an activity which is greater than foreseen (impurities, errors in calculation, etc.).

- In the case of the radioelements which become charged and uncharged while the pile is in operation, their handling requires a delicate, heavy and complex apparatus because one must assure the protection of the personnel, the continuation of cooling and proceed without making any errors.

On the other hand, it is necessary not to produce excessive jumps in reactivity so that the control can follow: any displacement of radioelements must be effected in connection with the control board.

The loops constitute complex units which, from the standpoint of safety, are piles embedded in a breeder pile and for that reason present the same inconvenience and lead to the same requirements, particularly with respect to the continuity of the cooling system. Any damage to any one of these loops may have serious consequences for the operation of the reactor.

Certain points are to be remembered:

- Any limit imposed on a parameter must be connected to a safety alarm channel or a sound and visual signal channel (pre-alarm).
- The safety devices are the organs connecting the pile and the loop. It is therefore necessary that they should operate as little as possible and the shutoff of the pile be of the shortest duration.
- A loop should be capable of being taken out easily and rapidly in view of the existing means; unfortunately, this is not always true, certain loops do not provide the necessary means.

- The experimenter must not consider his loop as the only important one in the pile, or even the only one in the pile. Every experimenter is included in a definite program and must assume only his proper place.

- The supervision of a loop cannot be done by the personnel using the pile; it should be done by a specialized personnel.

However, the user of the pile should be able to ensure the shutoff of a loop by a simple handling from the control board; this operation may possibly cause a shutoff of the reactor.

Any experiment must be studied in keeping with the user of the pile in its details and tested with him; its site must be bounded in common for it is the user who will most frequently and most rapidly intervene in any part of faulty circuits in order to assure the safety of the reactor.

It should be borne in mind that every loop circulating inside the pile may be active and contaminated. It therefore represents a danger for the personnel working it or approaching it. The safety controls must therefore be at an adequate distance and be protected. A device for rapid decontamination should be provided.

The following examples will give an idea of the need for working in teams and for well-established orders.

- Badly set slug, causing the contamination of the premises.

- Badly transmitted orders and safety devices not in place, causing the drop of a sample in a vertical channel. The extraction of this sample was very delicate and required the shutoff of the pile for several days.

- Material obstructing the track of the unloading carriage.

- Lead columns handled by the experimenters without notice to the user and causing the contamination of the pile hall.

Therefore, the following loading procedure should be imperative:

- No loading without submitting to the head of the hall in charge of such loading an authorization countersigned by the head of the experimentation group, the head of the pile, and the individual responsible for local radiation. Such authorization provides, among other things, for the means of unloading.

A well equipped hot cell next to the pile is indispensable, as well as an area for storing the radiated products.

II-4.

The pile operates under the best conditions of safety.

Such safety includes the protection of the personnel and of the pile.

II-4.1.

The safety devices of the pile are numerous; they are classified into two categories: pre-alarm and alarm.

The pre-alarm is connected with the failure of a supervision apparatus or of a secondary operation control or when a parameter of operation exceeds the so-called pre-alarm value which, if continued, would bring about an alarm causing the shutoff of the pile.

In the latter case, it is the duty of the head of the shift to localize and analyze the phenomenon, and on the basis of such analysis, either to change or to proceed with the established working program.

An automatic device is not provided on the pre-alarm systems, for the following reasons:

- Innumerable causes of pre-alarm requiring a complexity of apparatus.

- Desire to have the least possible number of failures and the least possible number of errors in notices and, above all, appeal to the intelligence and understanding of the phenomena by the personnel of the shift. For this reason the orders are as accurate as possible and this manner of planning the work keeps the personnel in a good position to understand all the phenomena and leaves to them the initiative of acting within a well-defined frame; the personnel is thus not merely a push button obeying blindly the orders of a machine.

As a consequence, the employee who reacts well is able to avoid failures.

The failure of a sector together with the drop of the safety rods causes the unbalasting of a certain number of machines and the starting of emergency groups and the resumption of the main apparatus required for the continuity of the cooling and the supervision of the pile and experiments.

It should be noted that the power supply of all the registering devices which can bring about an alarm (including experiments) should be provided with special power sources.

The table included in the appendix shows the frequency of rod drops for EL-2 and EL-3.

Any movement in the pile which may cause a change in the reactivity should be done with the safety rods raised and the power and health channels should be supplied with power through the pile but in the shut off position.

The removal of the cells is done with the pile shut off, their cooling is mandatory if they are taken out shortly after the shut off of the pile. But it must be noted that if there are periods of time without cooling it is necessary to take them into consideration for the safety of the personnel and the pile. Therefore, one should provide a lapse of time before the discharge of the pile greater than the theoretical lapse of time.

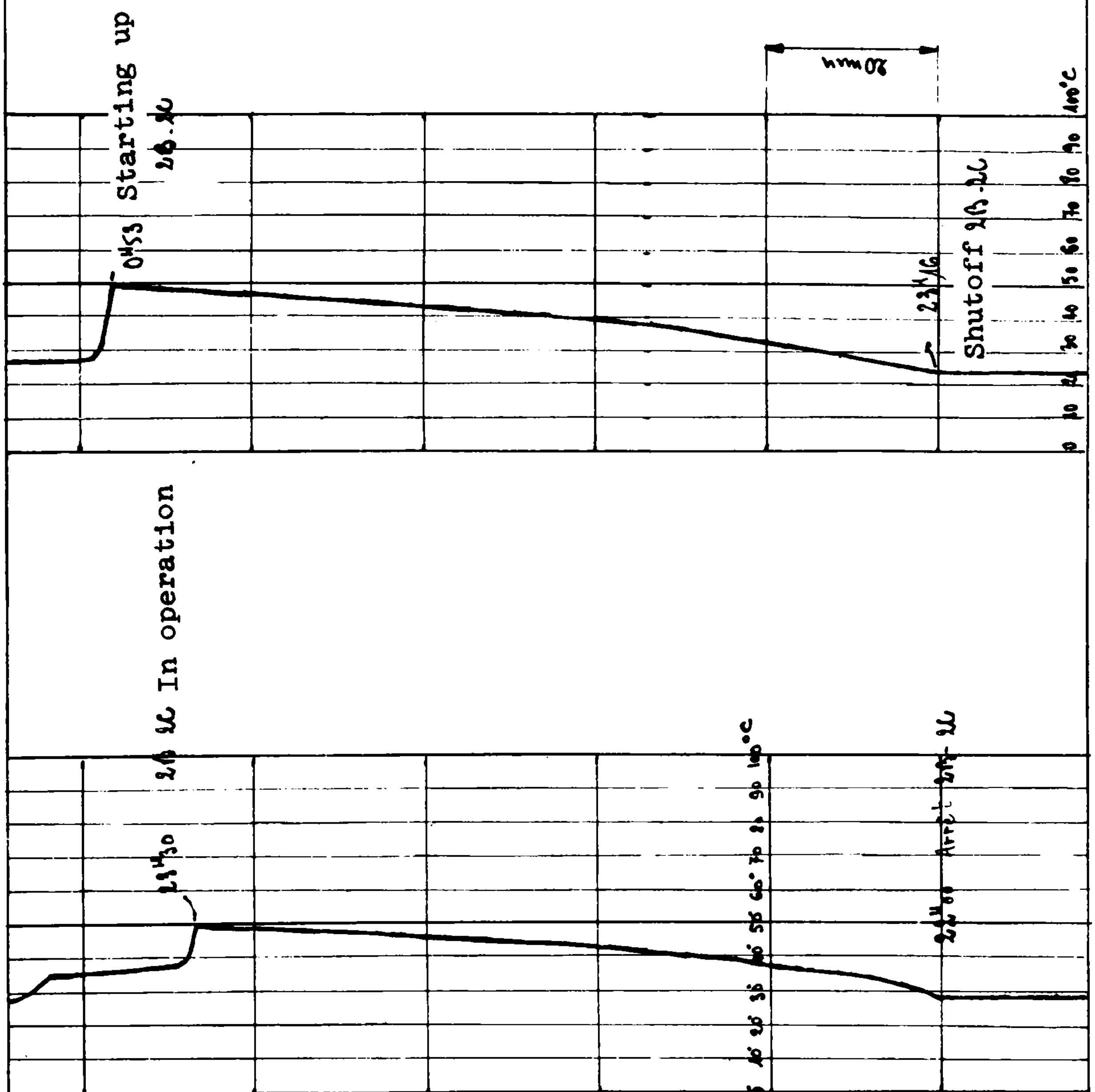
In addition, one should keep in mind the period of time required, to reach, without cooling, the nearest tank in case of failure of the cooling machine.

This relatively long period of time therefore limits the heat power to be extracted, required from the unloading hoods.

Included in the appendix is the heating curve for a pile cell, with water emptied from the tank. More than 200 cell movements were made on the El-3 without any failure.

The safety measures must be reliable. The gravest danger is discontinuity of the cooling; all revolves around this fact (El-2 12 W/cm^2 - EL-3 220 W/cm^2 to be extracted).

EL3. Temperature of UO_2 Cell (A_1)



HEATING IN HEAVY WATER TANK WITH
NITROGEN ATMOSPHERE

II-4.2.

The safety of the personnel is assured by the protection devices, the supervision, orders and training although the basis of protection against the radiation is unknown and vague.

This supervision in the various sites is indispensable; checking of radioactive dusts and aerosols is just as important, if not more so.

The evaluation of the danger varies according to the nature of the aerosol or the gas (strontium, argon, plutonium); it is therefore necessary to analyze the phenomenon with the least delay.

The protective measures should be adequate; the unloading hoods in particular must, for this reason, be of considerable weight and should have no leaks.

One should call upon the intelligence of the personnel and their comprehension of the phenomena so that they may be able to protect themselves from the dangers as efficiently as possible. This is the purpose of the personnel training.

At the same time total safety devices (all or nothing), on certain parameters, mitigate the estimating errors of the personnel.

The orders should be known by the entire personnel of the shift particularly the safety orders; a special instruction booklet is published for this purpose.

Moreover, a pocket meter (fountain pen type) is furnished to each individual operator enabling him to check himself; this has a good psycholocal effect.

The safest protection policy is to admit only a minimum of individuals within the tight enclosure and anything that need not be done within this enclosure should be performed outside: blank assembly, tests, various machine shop operations, computation, management, etc.

DROP OF SAFETY RODS

	Alarm in core supervision		Alarm in radiation supervision		Alarm in thermodynamic supervision		Auxilliary failure alarm			Test-ing	Var-ious	Total
	Core-fail-ure	Elec-tronic cir-cuit	Number	Cause	Excess	Un-timely oper-ation	D_2O Pumps	400 cycles	E.D.F. sector			
EL-2												
1955	0	10			1				13			24
1956	0	1							4			5
1957	0	5			3				9	4		21
1958	1	5							4	8		18
TOTAL	1	21			4				30	12		68
EL-3 (1958)	0	5	1 1	Break in A2 Leak in re-comb-ina-tion cir-cuit	2	8	12	8	9		1 (D_2O leak) 1(un-known)	48
First week 1959	0	13			-	1			9	3	4	30

CHAPTER B

MAINTENANCE

I - PURPOSE

I-1.

Maintenance consists of all the measures and tasks to be performed in order to keep the installations in good shape and enable all the operations scheduled in the program to be carried out.

It consists also of all measures destined to keep the premises in desirable health and sanitation conditions.

I-2.

Maintenance extends from simple "emergency repairs" to "preventive maintenance" allowing intensive utilization (operation) of the apparatuses.

Thus in 1958 EL-2 operated for 6,290 hrs, the shutoffs representing 2,470 hrs of which 1,960 hrs were anticipated (weekly stops, leaves of absence, important changes, etc.) and 510 hrs unforseen (removal of damaged cells, failures in experimental devices and 34 hrs of failures of operating apparatus). The latter represent about 0.5 percent of the operating time.

I-3.

In ordinary industry the basic purpose of maintenance is economy; economy for the purpose of decreasing expenses for repairs as well as in order to achieve an increase in profit from operation. Another purpose is ensuring the safety of the personnel.

In the case of a reactor, the relative importance of these two objectives is reversed: maintenance is first concerned with ensuring safety: the cooling apparatus, the supervision devices must not stop. Economy is next to security; it is, of course, evident that we must likewise take care to avoid putting an apparatus or an aggregate of apparatuses out of service as they are frequently quite expensive. This may also affect the operation and the reputation of the installation.

I-4.

And, at last, in the nuclear industry as in all industries devoted to an intensive operation, maintenance should aim to diminish and even eliminate the work of the operating teams. When everything is in normal condition, the impression given to a uninformed visitor is that the personnel has nothing to do.

II - SCOPE OF MAINTENANCE

II-1.

Almost everything in a reactor needs maintenance:

- the machines, just as in any industry;
- the signalling and supervision equipment; maintenance also consists in eliminating the signalling devices which have become useless either through the elimination of the cause or because past experience has shown that they are not necessary.
- the circuits (mechanical, electrical and electronic), for example, supervision of the exchangers and of the piping. The very delicate problem of joints is far from being solved (tightness, behavior under radiation, etc.).
- the premises. The problem of contamination which has very important psychological repercussions, is an additional restriction as compared to ordinary industries;
- the tools. Under this heading one must include the special clothing of the personnel; the laundry problem is quite difficult.
- the materials peculiar to Nuclear Energy: for example: uranium (problem of storing); heavy water (purification, recombination, drying, centrifugation, etc.);
- Irradiated or contaminated materials: in this case, maintenance goes as far as discarding the material.

II-2.

But there are items which cannot be maintained:

- defective installations due to very short period of construction or unforeseen delays;
- installations made obsolete by technology;
- ill-designed or ill-protected installations;
- installations under flux.

For all of these elements it is not possible to interfere while the pile is working and sometimes it is not possible to intervene at all. This is the case of numerous materials taken out of the reactors and which cannot be re-used because there is no equipment to make work possible on these materials (dismounting, tooling). They become radioactive wastes where as a "hot" workshop would sometimes enable to be reconditioned. The economic aspect of this question should be looked into.

The creation of such workshops leads to various questions:

- the problem of location and of transportation if they are far from the piles;
- the problem of the supply of materials and personnel;

- the financial problem. At any rate it is necessary to make an estimate of expenses as accurately as possible.

A beginning has been made to re-use slugs at EL-2.

III - MEANS TO BE INTRODUCED FOR CARRYING OUT MAINTENANCE (Organization, machines, personnel)

III-1.

The responsibility for maintenance may be assigned at various levels in the hierarchy:

- At the level responsible for the operation: this increases the flexibility and simplifies the use because it is frequently hard to draw a clear line between "operating" and "maintaining"; the data serve the user as well as the one who decides on repair work;
- At the level responsible for several reactors; this allows the centralization of certain important establishments which are not of a permanent use (heavy water workshops, buildings for storing and handling uranium);
- At a level corresponding to a nuclear research center;

This enables the concentration of the establishments used at the same time by the reactors and by the other users (mechanical, workshops, electronic workshops, decontamination workshops, research departments, etc.), this field being of interest particularly to the usual and non-urgent fabrications.

The equipment of a research center, moreover, is centered on the services to be rendered to the majority of researchers at the center; that is, to the laboratories rather than to the large machines which frequently, on one hand, have very special problems to be solved and, on the other hand, because of their design, have engineers specialized in each technique;

- At an administrative level which merely passes out the order to private industry;

This makes it possible to call upon "competent men" without overloading the working force, but requires considerable preparation and is not always possible particularly with active products.

III-2.

This problem arises in all industries and not only in the case of reactors. It seems necessary to reach a compromise.

The permanent responsibility lies with the operation level and in order to achieve the greatest possible flexibility, it is fitting that they should be given greater means than may appear theoretically necessary:

- give each installation the number of teams corresponding to operation under the worst conditions: at Saclay, six teams to make up the shifts where four would normally suffice; experience shows that the other two are necessary to make up for absences (sickness, leaves,...) and ensure current maintenance.
- make up, for each particular job, a small permanent staff ensuring the continuity of the maintenance program.
- make up, for each reactor, small repair shops accessible

to the operating personnel, equipped with standard, elementary tools, specialized tools and spare parts most commonly used.

To make up a common nucleus for a group of reactors, for studies, tooling, storing, and workshops for non-continuous use (heavy water, uranium).

Finally, to group the most general facilities in a central nucleus: management, sanitary and radioactive supervision, decontamination, etc.

III-3.

It should be noted that maintenance always has human and psychological aspects of considerable importance:

- the operating personnel is not eager to see an "outside" team come in to take care of its equipment.

From this standpoint, there is a definite advantage to shift this operating personnel a maximum number of times from the maintenance personnel unless they are used permanently.

There was a case in March 1959 when an accident due to the break of a cell took place. This serious and difficult repair work was done in normal time by the reactor personnel; "outsiders" were used only as specialists (cleaning, machining, dangerous and delicate handling, etc.).

- maintenance work done during the shutoffs provided for the operation is the most important. The operational personnel mostly is used in this work.

This solution is preferred in order to:

- increase the flexibility of the operation and its regularity, thus creating a sort of self-supervision by the operation over the maintenance effected by the personnel proper (See Chapter A).

- keep up the interest of the force for their work and their knowledge of the installations;

- increase social improvements and professional promotion (1)

- necessitates suitable training of personnel, of staffs and strengthening of supervision.

In the case indicated above, a strengthening of supervision in the course of the work has made it possible to eliminate certain tests and gain several days on the time needed for repair work.

- The selection of staffs is very important as well as the improvement of relationships among themselves, and with the members of other services.

The mutual understanding of needs improves cooperation and frequently spares us the trouble to request help from the outside with the corresponding administrative red tape.

- For the operating personnel the maintenance of the installations is in itself an excellent factor of professional advancement.

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(1) Certain employees who were hired as laborers are now pile operators.

IV. GENERAL PRINCIPLES OF MAINTENANCE

IV-1.

For the reactors as for all industrial machines, maintenance was begun with the setting up of the materiel and the repair of defects when they occurred. This method explains the slow rate of operation of the reactors (1) (See the diagrams I and II for EL-2.).

In view of the importance and the number of experiments under way in the reactors, it was necessary to quickly overtake the modern maintenance techniques applied in the standard industries, particularly preventive maintenance.

IV-2.

It was also noted that a large part of the work could not be done while the reactors were in action and that quite frequently it was advantageous to prepare in advance and to wait for the shutoff of a reactor or an organ to effect a simple connection on the circuit. The shutoffs are thus reduced to a minimum possible.

- Thus, on the control panel, the replacement of an amplifier is done by short-circuiting a safety device for less than a minute.
- The emptying of an ion exchanger jar may be done with the pile in operation and merely requires the shutoff of the corresponding circuit for a few hours.
- On the EL-3, for the replacement of a heavy water pump, the pile is shut off only during the loading of the casemate.

IV-3.

For certain important apparatus (blow-pumps, power lines, etc.) this restriction necessitates the doubling and at times the tripling of the equipment in order to effect maintenance for repairs on a piece of equipment stopped while the one, or the other operate, as has already been seen.

The cleaning of the blower motors takes place while the pile is operating, likewise for the replacement of the recombination high-pressure apparatus. One may switch the automatic control from one room to another, thus making it possible to change completely the equipment in a power channel without shutting off the reactor.

IV-4.

In certain cases the design of the installation provides for such changes either without entering confined premises or within a very short time.

Remote-control valves make it possible to replace a heavy-

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(1) One should also emphasize the caution necessary with regard to a device which is entirely new and largely unknown.

water pump without the risk of irradiating any personnel in the basement.

The casing of the automatic valves for detecting breaks in the jackets is designed in such a way that a defective part may be located and replaced within a few minutes. It is noted that in all cases the standardization of the materiel had to precede the creation of the apparatus.

IV-5.

It may happen that certain installations comprise parts whose presence cannot be understood outside of this maintenance idea and the problem of quick repair.

In EL-3 there are three CO_2 blowers when only one would satisfy the required flow.

In EL-2 the thermocouple circuit of the top of the cells comprises three sets of distribution terminals in series; this is to allow a quick change of either one or several thermocouples or of the connection cable or even of the recording apparatus.

IV-6.

Just to indicate the importance of providing detachable circuits which are subject to slight contamination and are easily decontaminated: smooth surfaces or suitable coating (stainless steel or removable paint), with few edges. The choice of paint is important for the nonmechanical circuits (protection of cables, electronic apparatus, etc.).

IV-7.

In many cases, the concern of the operation force is to effect maintenance during the shutoffs provided for the operation proper.

IV-8.

The maintenance personnel must not touch anything without the preliminary approval of the individual in charge of supervision and of the engineer in charge of operations.

For this reason, all work on a circuit, requires the filling out of a work slip which is first presented to the supervision officer (S.C.R.G.R.) and then to the approval by the individual in charge of operation and who is versant with the tasks of the installation. This individual passes on the decision to the operating personnel.

It is in this way that a plumber one day cut off the city water pipe supplying the cooler of a neutron converter.

V. MAINTENANCE METHODS

V-1.

We have seen the great importance of passing from "repair" maintenance to preventive maintenance.

In industry, this shift has taken place in various ways (use of statistics, standard time, machine tickets, setting up preventive maintenance scales, comparison with similar industries, etc.).

Our industry is only in its infancy, and reactor technology is far from being mastered; therefore a certain number of these means are lacking.

In practice, it is necessary to follow through each kind of organ independently of the other ones, issue statistical data (unfortunately, on very shaky figures in many cases) and then deduce the probability of the duration of operation.

Thus the checking of certain parts, such as:

- blowers, heavy-water pumps, fans, etc., should be made every year (6,000 hrs).
- Lift pumps, diesels, etc., every two years.
- Cleaning of electrical and electronic equipment, every six months.

It would be desirable that the checking take place every 6,000 or 8,000 hours corresponding to a period of one year.

These estimates are, to begin with made for the most important components; little by little they will be extended and made more precise.

V-2.

The maintenance and the use of various circuits often lead to devising of important changes. Such changes should be directed toward the following goals:

- Standardization of the constituent parts;
- Simplification;
- Taking out from the flux the parts to be maintained or to be controlled in order to enable action as quickly as possible.

V-3.

In order to make possible standard maintenance, the plans, diagrams, etc., should:

- exist;
- be available to the personnel;
- be up to date.

V-4.

An essential element in the maintenance of the reactors is safety, particularly from the radioactive standpoint.

- There should be an adequate number of supervision personnel not directly tied to the piles;
- they should be versant with the overall installations and of the tasks undertaken; they attend the meetings relating to such tasks; they approve the work slips;
- It is very important that they have the confidence of the personnel and of the staffs; with that in mind we seek for the collaboration between the staffs and the supervisors.

For example, the data on the fountain pen pocket meters, being their task, is actually done in common; this makes it possible for the staff to organize the teams on the basis of the results rather than to prepare a job and then be met with a veto.

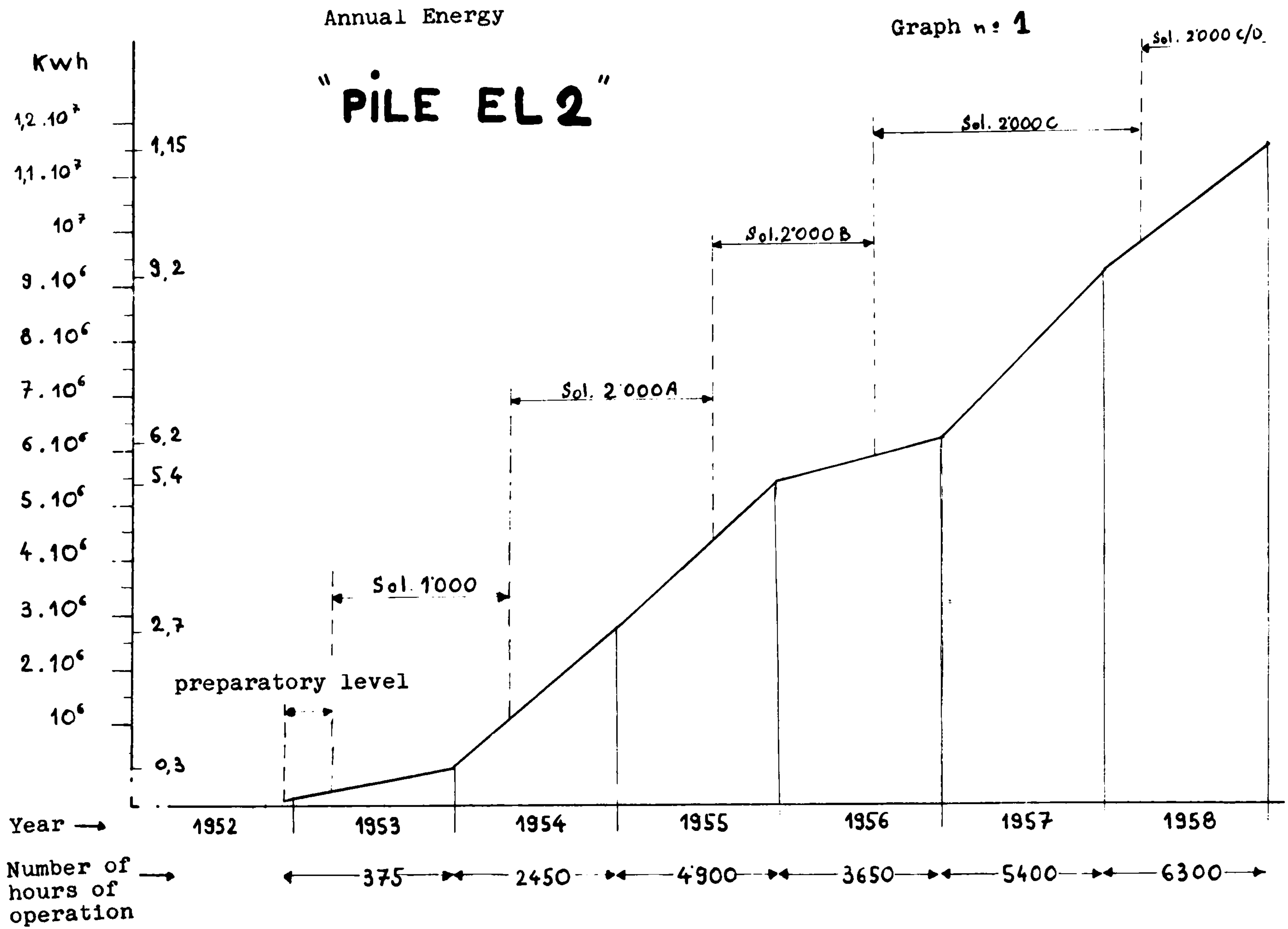
VI. PRICE OF MAINTENANCE

It is always difficult to know the price of maintenance, particularly because it is difficult to separate:

- what is purely operation;
- what is purely maintenance;
- what is purely a change or an addition;

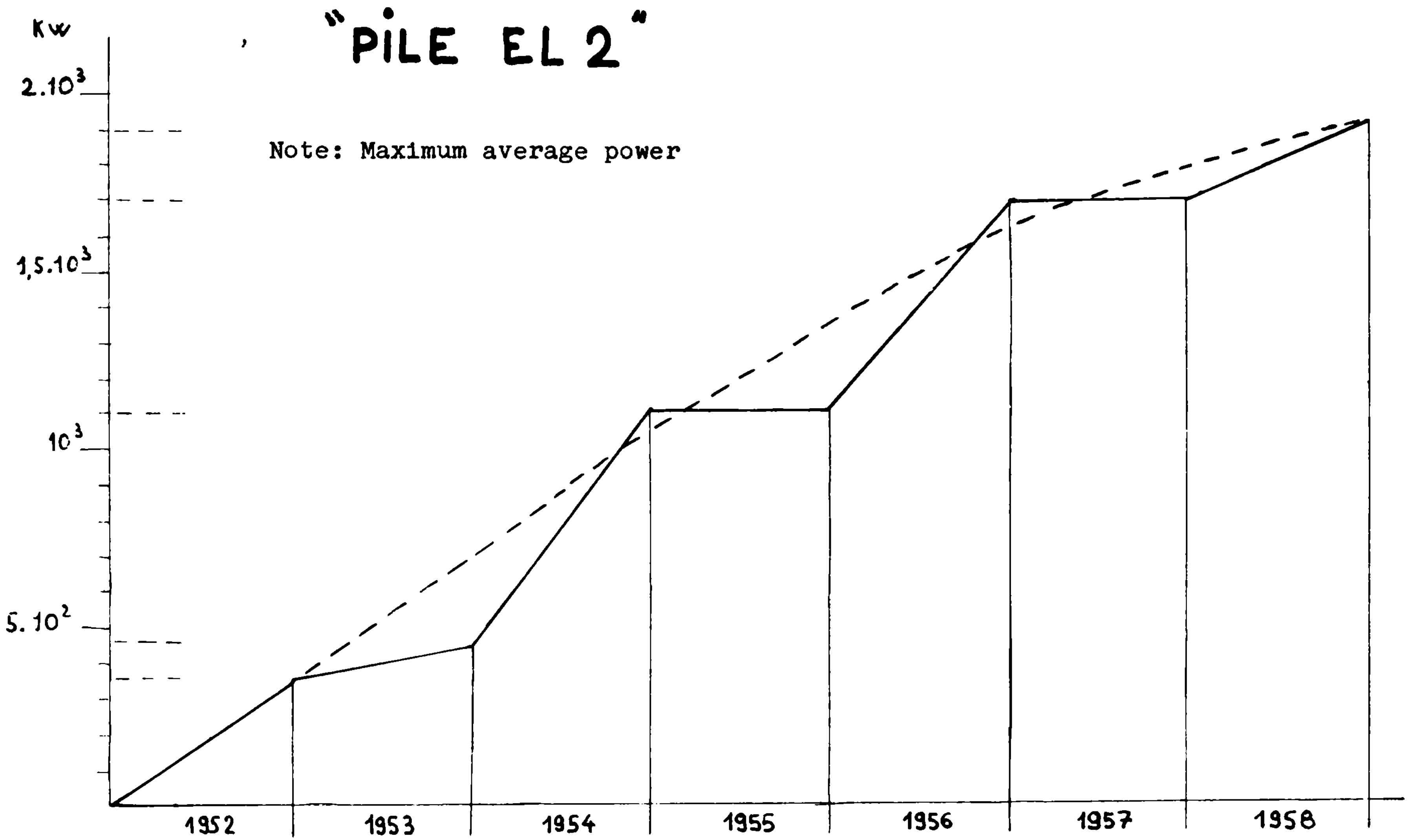
since these three fields react on one another.

Nevertheless, it has been possible to figure on thirty million francs for the gross maintenance annually of the EL-2 reactor.


Annual Energy
Graph n° 1
"PILE EL 2"
Kwh
1,2 . 10^7
1,1 . 10^7
10^7
9 . 10^6
8 . 10^6
7 . 10^6
6 . 10^6
5 . 10^6
4 . 10^6
3 . 10^6
2 . 10^6
10^6
1,15
9,2
6,2
5,4
2,7
0,3
Sol. 2'000 C/D
Sol. 2'000 C
Sol. 2'000 B
Sol. 2'000 A
Sol. 1'000
preparatory level
Year
1952
1953
1954
1955
1956
1957
1958
Number of hours of operation
375
2450
4900
3650
5400
6300

Graph n° 2

Average Power



CHAPTER C

CHANGES AND ADDITIONS

I REASON FOR THESE CHANGES

The dates of the outset of the preparation and of the first divergence of the Saclay reactors are the following:

EL-2: Beginning of study: 1949
First divergence: 27 October 1952

EL-3: Beginning of study: 1955
First divergence: 4 July 1957

I-1.

It is clear that the reactor technique has greatly developed since the outset of the preparation of EL-2 and even of EL-3. It should be noted, moreover, that the French Atomic Energy Commission did not have any element of reference except the pile ZOE which had just been put in operation.

I-2.

Certain ideas which were valid at the time of the preparation may become obsolete and even incorrect at the end of a few years. This is the case of the joints of the EL-2 tank of perbunan under flux.

I-3.

Experience in operation and maintenance may bring about considerable changes in order to facilitate the task of use.

I-4.

One should also note that insufficient time or delays of an unusual nature could possibly bring about the defensive construction of certain components at the time of construction.

I-5.

Finally, if the reactor construction technique has progressed the needs of the users have changed considerably. Certain results which were formerly regarded as essential, have now become established facts and the experiments now under way are clearly different from those of six years ago.

I-6.

It appears from all of these reasons that a research reactor is not like a power reactor which is a definite entity, once and for all; it is, rather, an apparatus in the course of development and adaptation to needs.

II - SCOPE OF CHANGES AND ADDITIONS

II-1.

This scope is very wide because frequently the basic characteristics of the reactor are changed (thus, EL-1, which owed the first letter of its name ZOE to zero power, now operates continuously at 100 KW). However, the changes relating to the pile block are exceptional: they assume work in an active and contaminating medium.

II-2.

The main changes relate to:

II-2.1: cooling circuits:

In EL-2, the primary nitrogen circuit has been replaced by a carbon dioxide circuit;

- The heavy-water circuit which was provided only for the static moderator, has become a cooling and purifying circuit;
- The air circuit has been changed.

II-2.2: The energy distribution:

- The set of electrical circuits has been changed

II-2.3: The control and supervision equipment:

- Progress made in the electronic industry frequently challenges the appropriateness of the present equipment.
- For EL-2 in particular it should be pointed out that at the time of its onstruction, the equipment came largely from "surplus" stock left over after the war, a portion of which was of foreign origin, and the supply of spare parts at the present time requires the gradual change of what is left of the original control panel.

II-2.4: The devices for loading and unloading and experimental means corresponding to changes in requirements or even in the radioisotope market: (in 1955 the weekly production of EL-2 was a few containers, but now it is 50; the sources of radio-iridium were rare, but now they are rather large).

II-2.5: Of course the buildings and their equipment follow such changes: building of annexes, connecting platforms, even basements; office equipment, laboratories, warehouses, etc....

III - MEANS AND GENERAL PRINCIPLES

The observations made on the subject of maintenance are altogether valid:

III-1.

- Maximum utilization of operating personnel;
- Recourse to common specialized means for several piles or the entire Nuclear Study Center.
- Possible recourse to private industry as far as work on new material is concerned.

III-2.

- Avoid as much as possible, provisional changes and, on the contrary, make a precise and complete estimate of the work to be effected.
- Planning the work in order to prepare as much as possible before the shutoff of the reactor and to have only very reduced adjustment installants to be made during such shutoff.
- Estimate of the possible failures and breakdowns on the new instrumentation; in case such incidents should be serious, the duplication of the installation must be foreseen.

In any case, the design of such new contrivances should be such that intervention by the operating or maintenance personnel should be easy and safe.
- All equipment, before adjusting should be tested in blank to avoid trouble in final mounting effected under difficult conditions.

IV - METHODS

IV-1.

We have seen that differences in use and operation may bring changes with them. The new apparatus should be designed to be as definitive as possible.

IV-2.

One should especially stress:
- standardization of the constituent parts;
- simplification of circuits;
- accessibility of parts to be maintained;
- easy decontamination.

IV-3.

The preparatory study should bear in mind general safety, and particularly from the radioactive standpoint.

At the time of execution, the instrumentation which has become useless should, as far as possible, be entirely eliminated in order to avoid overloading the installations.

IV-4.

The changes or additions to be made should be the subject of plans, diagrams, orders, etc., communicated to the responsible personnel and drawn up in accordance with them.

In this connection, it is advisable to provide detachable units enabling a practical and rapid modernization.

IV-5.

Finally, the additions requested by the experimenters should be precise and clear and the requests filed well in advance, to avoid useless, dangerous and troublesome haste.

FAILURES IN OPERATION OF EL 2 AND EL 3 BETWEEN 1 JANUARY 1957 AND 1 JULY 1959

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Commissariat à l'Énergie Atomique, Paris, France

I. GENERAL REMARKS

It is not pleasant to produce, for a symposium, a report on failures in the use of reactors during a given period of time, because subsequent analysis of the causes of the failures most often clearly reveals that these failures could have been averted, either through better supervision of the installations or by taking certain preliminary precautionary measures which also appear obvious after the failure.

Therefore, what is required of us is a sort of public confession of our sins, which we shall make sincerely and completely for each nuclear pile EL-2 and EL-3. However, as our report would be far too long and tedious for our readers if we went into the details of each failure, we will mention only the more interesting failures.

II. CHRONOLOGICAL LIST OF FAILURES OF PILE EL-2

I will recall briefly that EL-2 is a pile moderated by heavy water in which are immersed aluminum plungers containing natural uranium rods cooled by carbon dioxide by a pin circuit (1 view).

18 April 1957 Failure in the Piping System (Jump in Reactivity)

During the starting operation of a blowing engine of CO_2, a jump in positive reactivity causing a fall of rods by the maximum of reactivity. This failure is worth mentioning only because it influenced us to modify operational instructions.

In order to prevent a temperature cycling of rods when the blowing engines were changed and thus caused a decrease in cooling the temperature of the cells was controlled manually. It is now preferred to allow a slight temperature cycling and to proceed by stages of predetermined power levels, either manually or automatically.

12 June 1957 When operation was stopped, water leakage in the cooling circuit of a neutron converter in an experimental canal was discovered. Following a breakdown in water supply in the city, when water circulation was restored, water hammering brought about water leakage in the experimental canal which caused slight contamination. In order to prevent such a failure from occurring again, an independent circuit of demineralized water for cooling of converters was installed.

6 October 1957 Fracture of a Jacket

A rod in the pile shows signs of activity deviating from D. R. G. indications. After observing the evolution of the fracture of the jacket for several days, the rod is extracted and the slightly contaminated CO_2 flows to the chimney after filtration at such a concentration that the activity is inferior to the tolerance limit of 4 millicuries/cubic meters.

After this date, similar failures should repeat themselves several times. It seems that a solution would be a change in the action of the bars, but as we wish to obtain information indispensable for the Metallurgical Department, we shall increase the irradiation of the rods to its maximum extent.

26 November 1957 Serious Failure in Converters

This failure is similar to the previous converter failure but this time the failure occurred when the pile was in operation. Safety measures worked well and the operation of the pile was stopped. However, as a result of the delay between the stop of circulation and the effective stop of the pile, and also due to residual flux, the uranium in the converters had enough time to become heated and to melt.

Moreover, water infiltrated into the graphite of the reflector creating such a degree of antireactivity (400 pcm) that the pile stopped by itself. The air coolant of the graphite brought about a certain contamination by fission products through the chimney of the pile (sixteen times the normal value) for a short period of time.

The water which flowed into the lower chamber of the pile gave this chamber an activity of 2 roentgens/8 hours.

Up to that point the consequences have been very insignificant. Unfortunately, due to insufficient precautionary measures, the removal of the damaged converter caused a slight but general contamination of the hall pile which necessitated three weeks of tedious cleaning.

The pile was started again before the cleaning was completed but the presence in the pile hall of a certain number of experimenters was made impossible as a result of decontamination work.

After this incident severe measures have been taken as regards the safety of the converter cooling system and protection by removable paint and polyvinyl panels, when seriously contaminated material has to be removed.

23 March 1958 Failure in Automatic Control

The automatic control continued to operate in a chamber which went out of order. A pre-alarm system indicated to the pile superviser that the power was too high, but he misinterpreted this indication, relying rather on his first measuring channel. A cell temperature pre-alarm causes the pile superviser to bring down the rod manually. The pile rose to $\frac{3P}{2}$ of the forseen power.

The measures taken consisted in reinforcing operation regulations and in adjusting more strictly the conditions for maximum power alarms. An alarm warning about discrepancies between power channel indications could avert such a failure, but such a system is too costly and does appear indispensable.

28 September 1958 A failure in the automatic control due to a drop in power caused by a defect in automatic switching brings about the fall of a rod due to apparent excess of reactivity.

This incident is not particularly noteworthy except for the fact that it led to the recalibration of different control and operating devices during which certain irregularities were discovered.

It was decided to have periodical inspections of reactimeters.

27 December 1958 A classical fracture of a jacket, but this time the supervisor of the shift, bearing in mind the slowness of the previous jacket fractures, is too slow in stopping the pile. Thus, the contamination of the lattice by carbon dioxide is slightly greater than usual (20 times the normal limit) and certain points of the lattice are contaminated, 4 LMA at the paddle-wheels of the blowers; nevertheless, the damaged rod is extracted within the normal period of time (48 hours). The carbon dioxide is replaced by fresh CO_2 and a pre-alarm system with a klaxon installed in the chamber measuring the activity of the lattice.

16 February 1959 There occurs a sudden and violent fracture of a jacket. During the increase in power of the pile, a violent increase in activity in the carbon dioxide in the lattice causes an alarm, the shift supervisor allows the rods to drop, the D. R. G. did not have sufficient time to indicate which rod is producing the increase in activity, while the cell temperatures do not give any abnormal indications either.

The contaminated CO_2 was evacuated in the normal manner through filters, and it was decided to introduce clean gas into the nuclear pile and to raise the pile by steps to 5 KW in order to find the responsible rod. This rod is actually found after two hours of operation (time required for a complete exploration).

The usual steps for removal of the damaged rod are taken but at the moment of removal a slight entry of air into the network, which is according to regulations of a slightly reduced pressure, caused a violent and localized increase in the activity of the inlet and outlet CO_2 collectors on the roof of the pile. It is almost certain that the air entering the lattice brought about in combination with the bare uranium a phenomenon of pyrophoricity which ignited a small portion of powdered uranium causing intensive contamination of the entire lattice in the proximity of the damaged cell. Nevertheless, the cell is extracted, but only part of the uranium comes out with the plug. It is then decided to fill the plunger remaining in the pile with oil and to extract the plunger and the rod at the same time using a hood specially designed for the purpose. The excessive contamination forces us to empty (discharge) the pile completely and to dismantle the collectors which gives rise to serious difficulties due to the high level of activity on the roof of the pile.

This gives the opportunity to change all the plungers and to examine the tank after removing the heavy water. In order to find the cause of the violent accident which brought about the fusion of uranium fuel elements, we perform an autopsy of several other plungers; we discover deposits of polymerized oil which seem to be the cause of the accident. After a certain time, the solid deposits of oil form a stopper which obstructs the passage of CO_2 into a cell. However, this does not explain the violent nature of the phenomenon. The work of dismantling, reassembly and changing cells lasts a whole month. Thanks to very severe precautionary measures not the slightest trace of contamination is spilled in the hall of the pile. Certain sections of pipe connections serving for the intake and outflow of carbon dioxide, which are accessible with only great difficulty, force us to work on material showing an activity of 2.5 RH on contact.

The Safety Department authorizes us to work at a radiation intensity up to 7.5 rem/h, provided that one rem per person is not exceeded. The available service personnel is more than adequate for dismantling the equipment.

Conclusions Regarding Pile EL-2

Summarizing the failures which occurred in pile EL-2, we see that the most frequent failures were due to jacket fractures. These were intentional to a certain extent since we tried to make the fuel elements last as long as possible. The repair work involved in putting the plant back into operation has been very instructive. It was possible for us to perfect the process of handling contaminated material. I can now state that in close agreement with the S.C.R.G.R., we can now dismantle such material without difficulty. Moreover, the several instances of jacket-fractures which occurred in pile EL-2 gave the departments responsible for the development of the D.R.G. systems the opportunity to prove continually the accuracy and precision of its instrumentation which has now been perfected.

III. FAILURES IN PILE EL-3

During the period starting and increasing the power of pile EL-3, we have had, of course, a certain number of failures due to defects in the materials but none of these failures is particularly noteworthy.

They were cases of breakdowns in the control panel or in the electric power supply. The safety devices provided in advance for such emergencies operated normally and the reactor was started again in the normal way after the circuits were repaired.

On the other hand, the heavy-water pumps of the radiation-proof type with graphite bearings required very delicate adjustment of the rotor; before this adjustment was made, the graphite bearings deteriorated rapidly and forced us to dismantle frequently.

With the exception of one serious failure of a fuel rod, the principal failures worth mentioning have been usually characterized by leaks of helium or of gas containing radio active dust emanating from canals (see list).

<u>7 February 1958</u> We have a minor contamination incident at the <u>top of the pile</u> due to radio active dust escaping from a special vertical device dipping into the lattice.

<u>9 March 1958</u> There was a slight contamination of the atmosphere <u>of the hall</u> of the pile during a rush of nitrogen in the circuit of the gasometers which maintain the pressure of the atmosphere in the tank. The nuclear pile was operating at 8 MW in a nigrogen atmosphere, when a defective connection in the exhaust pipe allowed the radio-active gas to escape in the enclosure instead of conducting it to the extraction jacket. The detection system for gases and radio active dust immediately set off the alarm. The workers put on breathing masks in order to continue the operation of the pile. The leak was rapidly detected and repaired. The activity originated principally from argon 41; the tritium content, 2×10^{-7} c/cubic meter, was very small, 1/100 of the LMA.

<u>13 April 1958</u> <u>Accident With a Fuel Cell</u>

We now come to the only important failure encountered so far in the use of pile EL-3.

This incident took place towards the end of the power tests on the pile. During the night of Saturday, 13 April to Sunday, 14 April, the pile EL-3 had been brought to the power of 15 MW for a flux of 10^{14} n/cm^2/sec, the temperature of the uranium of the rods read on a special cell with thermocouple is 400°C, all operating conditions are normal. At 2:00 a. m. several alarm klaxons sounded at the same time indicating over activity in different sections. We immediately proceeded to ascertain by means of a portable detector that these signals corresponded to abnormal activity in the hall and were not due to a breakdown in supply in the detectors. We then allowed the safety rods to drop. The time elapsed was about 45 seconds.

The shift workers put on individual masks while workers not indispensible for the operation of the reactor were evacuated from the radiationproof enclosure. We then proceeded to a rapid inspection of the entire building. The activity obviously emanated from the recombination reaction and from the helium pipe-system. We took samples of gas before completely evacuating the cap, then instructed the pile conductors to keep watch before the auxiliary board located in the engine room outside of the radiationproof enclosure. The analysis of the samples of the gas in the enclosure as well as personnel checkups disclosed that there was no contamination in the hall. From this moment on, we were convinced that we faced a jacket fracture. Moreover remote heavy-water activity recorders showed that the main circuit had been considerably polluted. (The extremely violent evolution of the activity level at various sections of the pile can be seen in the photographs taken.)

Measures Taken

On Sunday morning, 500 cubic centimeters of helium were withdrawn from the tank atmosphere, spectrometric analysis revealed the presence of Xe 133 and Xe 135 and weak traces of Iodine 133. The activity of Xenon 133 was evaluated at 100 curies for the entire lattice of about 25 cubic meters of helium. This data enabled us to evaluate the weight of oxidized uranium as 1.6 kilogram; this estimate was confirmed quite accurately afterwards.

Specimens of heavy water show upon examination a strong proportion of fission radioiodines I 131, I 133 and the characteristic lines of Xenon 131, 133 and 135. Microscopic examination of filtration remnants discloses the presence of dust of metallic appearance and of grains of uranium oxide.

Purification tests of the heavy water made on these samples show that simply passing over ion-exchange resins completely purifies the heavy water while passing over active coal leaves about 2/3 of the activity unchanged.

In order to find the damaged rod it was decided to divert the pile from its normal function after replacing the atmosphere of the tank by clean nitrogen.

The active helium was evacuated through a filter system installed for the occasion (see the figure) and through the hot filters of the plant. The permanent stations around Saclay and a control jeep with radio connection enabled us to follow at each moment the activity of the air in the vicinity of the chimney, which remained always lower than the normal activity with the pile in operation.

The divergence of the pile which was brought to 50 kilowatts by steps did not disclose the damaged rod, the general activity of the network increased regularly. We gave up the search by this method and proceeded to extract the rods and observe them through X-rays. In the extraction of the eighth rod, only the plug was removed by the unloading machine but the remainder of the cell remained in the tank.

Observation of the tank water with a periscope gave no results. We continued to extract the neighboring rods while the water was purified by passing over resin pots. Then the heavy water was emptied through the resin pots and sent to the stock room in basement 10.

Proper observation in the tank was then made possible. It was established that one cell was broken at the edge of its plug end and fell across the tank. An exact diagram of the position of the cell was made using reference points in the tank and the position of the cell recorded on a blueprint using descriptive geometry. The position of the cell was thus determined to within a few millimeters. The tools to be utilized in the recovery of the cell were designed to the exact scale and immediately manufactured; a mobile arm was to descend through an adjacent hole, seize the cell, put it in a vertical position, while another arm with a special clamp was to grasp the cell from the vertical position so that it could be removed with the normal unloading hood (see the views).

The longest delay was in the manufacture of the tools. Meanwhile, an identical model of the tank and of the damaged rod were made in the machine room and personnel involved in the recovery of the cell was trained with the tools manufactured for the occasion. On 16 May, the cell was extracted in less than an hour at the first attempt; the heavy water was then decontaminated, the nuclear pile recharged in 48 hours and once more ready for operation.

When the power was increased, everything was normal except the activity of helium in the atmosphere of the tank by fission products in the stable mode of operation, which increased by a factor of 30.

Actually the activity of helium two hours after stopping the pile was estimated at 0.2 curie per cubic meter, which according to calculations would correspond to a value of one gram of free uranium in uncovered state in the tank. This uranium has probably settled on the interior surface of the aluminum in the tank.

The causes of the accident have been most definitely established through examination of the remnants of the rod. This fuel rod, identical in all respects with the other rods in the pile, had been placed in a cell whose bottom had been shortened by a few millimeters, in order to introduce it in place of the central canal during measuring experiments on the physical characteristics of the pile at low power. Thereafter, the cell had been placed in a normal position in the lattice through an oversight. Taking into account the slight difference between this cell and a normal cell, the output of water in the cell had not changed noticeably and, therefore, its operating temperature was normal. This cell, therefore, functioned well during the two months when the power was increased, but having a bad resting position on the bottom of the tank, it vibrated under the influence of water streams until the welding of the cell on the plug was broken and the cell then fell across the tank. The cooling value subsequently dropped to practically zero resulting in the fusion of the fuel rod in a very short period of time.

This unfortunate incident gave us valuable information about the consequences which may occur in case of the violent fracture of a jacket in a lattice with heavy water. It revealed that only active rare gases passed into the tank atmosphere and that the other fission products, particularly the iodines, remained in the water from which they could be extracted easily with ion-exchange resins. This incident also caused us to modify the system of detection of fractures of jackets, to introduce a delay in helium injection into the counters and to install a warning system, that is, a signal causing the rods to drop depending on the activity of helium in the recombination circuit.

The other incidents in pile EL-3 since the failure of 13 April are uninteresting in comparison, so that we will mention them only for the record.

14 November 1958 A contamination failure caused by a rush of air in a radial canal had no serious consequences.

20 November 1958 There occurred an incident which might have been troublesome to the heavy-water lifting system emanating from the D.R.G. The automatic system of level adjustment went out of order and caused important variations in the level of the heavy water in the tank and as a consequence of this reactivity, the shift supervisor stopped the reactor. However, it was restarted almost immediately, the adjustment of the water level was done manually until the automatic system could be repaired.

15 December 1958 There occurred contamination of the atmosphere of the hall by helium leakage from the recombination circuit which was not serious but which led us through a most difficult search for leaks. This work lasted several days. The pile continued to operate; personnel wore masks whenever necessary.

19 December 1958 Failure on starting an emergency group during a sector failure. A drop of short duration in the sector voltage created an unstable situation which threw off the relay system, switching on the emergency installation, and disconnecting one of the main cooling pumps and almost stopped the second pump, which might have brought about abnormal heating of the uranium rods even after the safety rods were dropped. This incident caused us to modify the method of emergency supply slightly.

Finally 29 April 1959 An initially inexplicable failure took place during an experiment in the pile. Graphite blocks suspended from aluminum devices were irradiated at a fixed temperature (about 180°C) in the central canal of pile EL-3. During the night the safety rods dropped due to low reactivity. When verifying the temperature recordings of different experiments in the pile, it was discovered that the graphite block in the central canal probably collapsed tearing away its thermocouples, after having experienced a strong increase in temperature. After a 48 hour search into the causes of this failure, it was discovered that the filters installed in the CO_2 circuit which cools the central canal were clogged up by paraffin coming from the protection system of the plug in the central canal. As a result of this, it was decided to put a ban on the use of paraffin in safety devices whenever the danger of its temperature rise and subsequent melting existed. The experiment was dismantled and then placed again in the pile with another plug.

Conclusions for Pile EL-3

The damage incurred by the cells in the EL-3 is due to very special causes. In contrast to pile EL-2 we have been unable to discover any jacket fracture although some of the rods have recorded 5000 MW-days per ton. The troublesome incidents involved contamination due to helium leaks or dust emanating from experimental canals. All the leaks in the helium circuit have now been repaired. Activity detectors have proved to be excellent helium leak detectors.

However, during replacement of the experiment in the central canal we were again surprised by contaminating dust which escaped from the central canal. We were forced to go through a long and tedious process of decontaminating the top of the pile during the annual closedown of the pile. It appears that in a pile of considerable flux as EL-3 the absolute cleanliness of the experimental canals and their cooling circuit is imperative. Extra safety precautions require that all handling of irradiated material or samples emerging from the pile be effected with ample means of protection (polyvinyl, pressure reduction, removable paint), and strict discipline for moving personnel whenever some section is contaminated. The use of mobile showers at the exit of the contaminated zone has greatly helped to limit the dangerous spreading of contamination during dismantling operations.

Despite the knowledge we have acquired we are well aware of the fact that we shall experience more failures in the piles or on the loops at Saclay. But in particular we are relying a great deal on the knowledge to be gathered from the experience of the other participants in this symposium in order to reduce the number of these failures to a strict minimum, and we thank them in advance.

INSTRUMENTATION AND CONTROL OF THE NETHERLANDS MATERIALS TESTING REACTOR

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ABSTRACT

Control rods

The reactor is controlled by 6 identical rods with a total worth of around 30%.
By de-energizing a magnet, the control rod is disconnected from drive mechanism, and falls down by gravity, and the hydraulic force of the coolant flow. The release time is 14 msec and the total drop time 0,6 - 1,0 sec. depending on the flowrate. The drive mechanism of the regulating rod has additional limit switches, which limit the reactivity that can be turned over to the automatic control system to 0,5% by electrical interlocks.

Nuclear flux instrumentation

The nuclear flux is measured by 9 channels. There are two count-rate channels, two log nv channels, two linear channels of which one is used for automatic control and three safety channels.
The safety channels are fail-safe and give scram at an adjustable level. Period scrams are obtained from the count-rate channels and log nv channels at periods of <3 sec..
The safety and period amplifiers have fast relays which are de-energized with scram. These relays interrupt the magnet current of the control rods directly. The drop-time of the relays is less than 10 msec.

Automatic control system

The neutron flux is controlled by an on-off controller, which keeps the reactor power within 0,5% of demanded power.
No provisions are made for automatic start-up of the reactor. Automatic change in power level over a range of 0,1% - 100% of full power is possible by a motor driven exponential power demand potentiometer. This system is also used for automatic power reductions under certain abnormal conditions measured in the reactor plant.

Safety and interlocking system

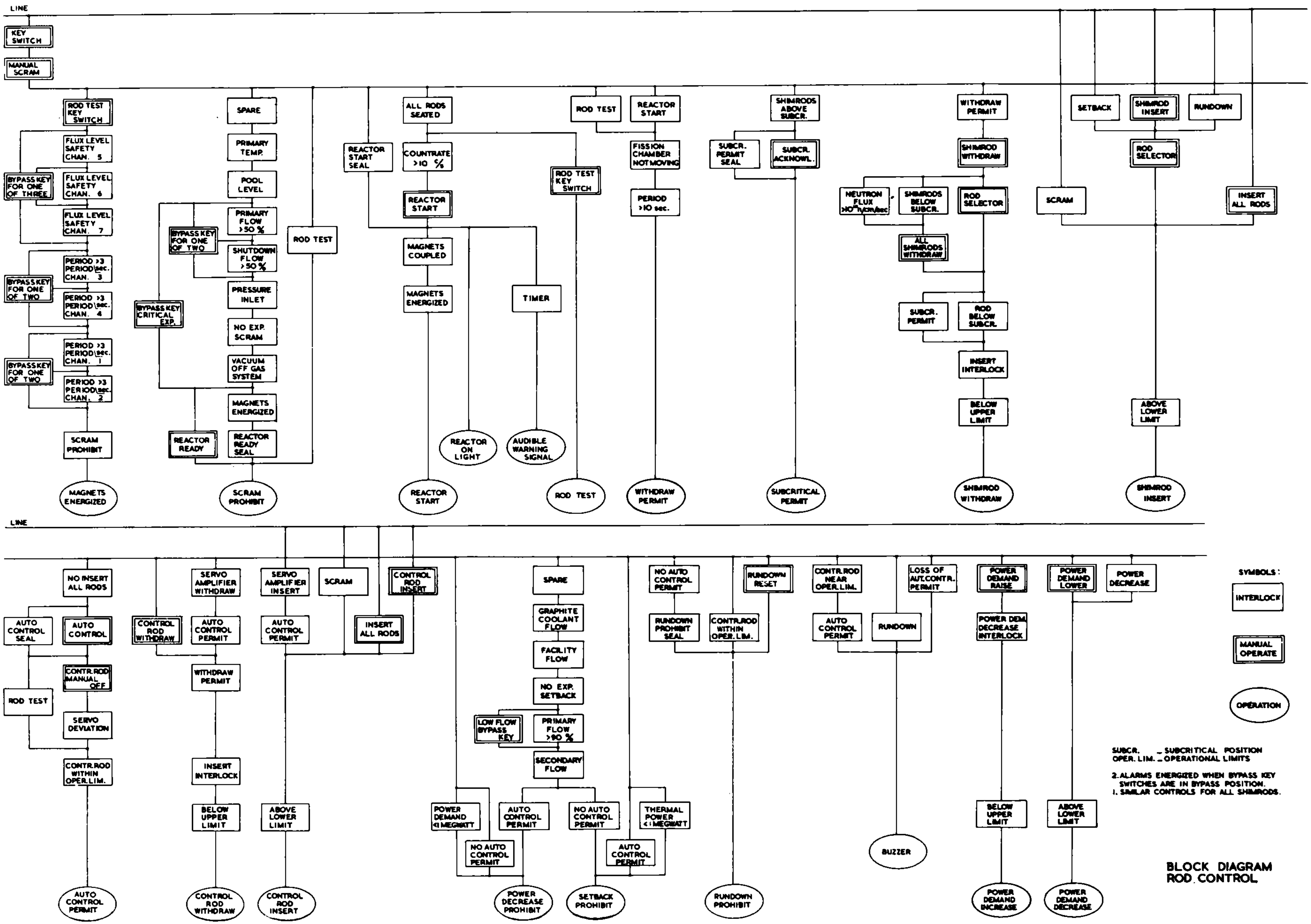
The safety and interlocking circuits serve the following purposes:

1. Safeguard reactor and experiments against destruction
2. Assure the correct procedure of start-up and operation
3. Assure more safe operation
4. Give automatic follow-up

There are three types of actions resulting from abnormal conditions measured in the reactor plant:
alarm, decrease power/setback and scram.
Alarm is used in those cases where the operator has time to take a corrective action. When an automatic reduction in power to 1 MW is sufficient "Decrease Power" will be used. If the reactor is not on automatic control, one of the shim rods will be driven in. Scram is reserved for those cases, where immediate or complete shut-down is necessary.
Backing up of two actions on the same parameter is generally not used. In a few cases alarm will be given before the reactor is scrammed.
The velocity of the control rods has such a value, that the safety amplifiers can safely handle a start-up accident. Interlocking circuits make such an accident very improbable. No automatic shimming has been provided. It is felt as a potential hazard if too much reactivity is controlled by an automatic system.
The number of interlocks is kept to a minimum. This has several advantages:

1. increased reliability
2. greater flexibility
3. less reason for by-passing of interlocks in case of malfunctioning

A description of the interlock system is given on the hand of the Blockdiagram Rod Control.



BLOCK DIAGRAM ROD CONTROL

THE ELECTRICAL POWER SUPPLY SYSTEM OF THE R.C.N. HIGH FLUX REACTOR

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ABSTRACT

The electrical power supply system has been designed according to the following basic principles:

1. The power supply to the various parts of the reactor installation does not depend on the proper functioning of one transformer, one coupling switch.
2. A break-down of the high voltage supply to the transformer stations at the site will not result in a dangerous situation; control over the reactor installation will be maintained up to a certain extend.

In accordance with principle 1 there are two 380 V main switch boards with several independent bus bar systems. The bus bars are fed directly from the transformers while interconnecting of the systems is possible via a coupling bus.
A dieselgenerator (100 kVA) and a battery fed motor-generator (6 kVA) are provided.
The dieselgenerator starts automatically supplying power within 10 sec. after a break-down of the normal mainsvoltage. The motorgenerator delivers uninterrupted power to the main reactor instrumentation and can be used for at least one hour and a half.

The electrical power is directed to the pump and valve motors via relay systems which are in some instances interlocked.
The basic principles of those relay and interlock systems are:

1. The different motors of the primary reactor cooling system are operated remotely, independent of each other. Only the relay system of the shut down pump is interlocked with that of the main cooling pumps.
2. The motors of the secondary cooling circuit and reactorhall ventilation system are operated

automatically in the correct sequence after a remote start command.

3. Pumps, valves, filters and compressors controlling water levels and air pressures, are operated automatically by limit switches.

4. Only the valves after the main primary and secondary cooling pumps and the emergency valve in the supply system of the secondary pumps are motor operated.

5. The relay system of the reactor cooling circuits is not interlocked with that of the reactor control rod system.

   Interlocking of the latter with the former is indirectly via process variables.
   Only the reactorhall ventilation system is interlocked with a manual scram and furthermore with the stack gas activity monitor and the ventilation air inlet temperature.

THE SECONDARY COOLING SYSTEM OF THE R.C.N. MATERIALS TESTING AND RESEARCH REACTOR

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ABSTRACT

The main requirement of the reactor cooling system is to dispose of 20 MW heat. The design data for the inlet and outlet temperatures on the primary side of the heat exchangers are around 49°C and 55°C; the total primary flow amounts to about 46 m^3/hr.

In principle there were three possibilities for cooling the primary circuit water, viz.:

air cooling,
water cooling in a closed circuit,
water cooling in an open circuit.

The open secondary cooling system was chosen because of the availability of water from the North Holland Canal, which can be disposed of into the North Sea.

The canal is an essential part of the North Holland polder system; the water near the site can be considered as diluted sea water, the dilution factor being 10 to 40.

For the reactor power of 20 MW a nominal secondary flow of 2000 m^3/hr is needed, the reserve capacity of the system being 1000 m^3/hr.
The secondary cooling circuit consists of a 1500 m long inlet line through the polderland, the secondary pump-house, a 430 m long pressure line to the primary pump-house and a 810 m long outlet line to the sea, where it terminates in a breakwater.
The pressure line runs through two rows of dunes which are sea defence lines; the Public Water Works of the Government therefore requested several safety provisions.
The basins in the secondary pumphouse have been dimensioned to contain the oscillations of the water level when starting or stopping the secondary pumps.
An automatic chlorine gas dosage installation is provided to minimize the growth of bryozoa.
Because of the two high points in the pressure and outlet line syphoning will take place. A vacuum pump is connected to the last point to prevent breaking of the syphoning action, which is necessary to maintain the full flow in the pipe line.

The heat exchangers in the primary pumpbuilding are made of aluminum; moreover the material in contact with the brackish canal water is Al-clad. All the secondary piping in the primary pumpbuilding is also Al-clad aluminum. The remaining pipes of the secondary cooling system are made of concrete, the pressure line of re-inforced concrete.

The Netherlands Government demands that no activity is disposed of into the sea via the secondary cooling water pipe line. Leakages from the primary to the secondary circuit have to be detected under all circumstances. A special design feature of the heat exchanger minimizes the possibility of leakages into the secondary circuit; for the case leakages do oocur however, several techniques are available to detect these.

CONTROL EQUIPMENT OF THE BR-2 REACTOR AND ITS NUCLEAR MODEL

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1. CONTROL OF THE BR-2 REACTOR

1.1. Control Rods

The BR-2 reactor uses for its control at the most 16 safety rods and one regulating rod. Each rod can be placed in any one of the 64 channels which are created in the core of the reactor. The rods are mechanically independent and are designed to operate from the lid of the tank on which they are fixed.

For each new load, the distribution and the number of control rods to be used are determined by a critical test with a core model, or "Core Mock-up Facility" (CMF). We shall briefly describe later the control rods and the nuclear counting channels of this model.

1.1.1 Safety Rods

These 16 rods have the double feature of being safety rods as well as compensating rods. The control rod has two sections. The lower section having a beryllium rod, and the upper section made of a cadmium tube with aluminum jacket. When the rod is released, it rests at the bottom of a guiding tube, so that the cadmium section is completely inserted in the core.

The complete mechanism includes in its lower section a guiding tube of aluminum with six longitudinal grooves, of 83 mm maximum diameter. The inside diameter of this guiding tube in which slides the control rod is 69 mm. The upper section, which includes the driving mechanism, placed above the tank lid, has a diameter of 117 mm and protrudes by about 875 mm beyond this lid. The total length of the mechanism is 7.437 mm.

The mechanism is immersed in water, and its guiding tube, as well as the extension tubes forming the channel, are perforated to allow for the cooling of the rod. The holes are distributed so that the flow of the water does not hinder the fall of the rod. Similarly, no other mechanical device should interfere with this fall.

The rod itself includes, as we have seen, a beryllium rod which is topped by a jacketed cadmium tube. The whole is inserted in an aluminum tube 2.025 mm long. The beryllium rod has two coaxial cylinders, so as to minimize any stress of thermal origin which is due to the slowing down of the neutrons and so as to allow for the cooling of the beryllium, which is thus done in three parallel ways. Appropriate devices are provided, which permit the correct centering of the various cadmium and beryllium tubes.

This whole apparatus extends in its lower section into a shock absorber. This absorber permits the transformation of the kinetic energy of the rod, at the time of its fall, into potential energy of deformation, and limits the strain which is transmitted to the lid of the tank through the aluminum jacket. The distance of braking for the rod depends both on the height from which the rod is dropped and on the rate of flow of the cooling water. It may vary from 65 to 200 mm. Tests have shown that the strain transmitted to the lid was of the order of 165 kg per released rod.

In its upper section, the rod includes a sleeve, with a fastening tray and a resetting device.

The rod is guided in its motions by three rollers, at its upper and lower sections.

The rod is fixed to the control mechanism by a connecting device.

Before describing this device, it is necessary for us to say a few words of the performance required for the fall of the rod.

The stroke of the rod is 960 mm. As we shall see later on, the connecting apparatus comprises an electro-magnet. The interruption of the current supply to the magnet permits the rod to fall. A period of 45 ms elapses between the interruption of this supply and the complete freeing of the rod; but it is 20 ms after this interruption that the fall of the rod begins.

The fall due to gravity of the rod would be slowed down by water. However, it is necessary that the rod travels 305 mm in less than 250 ms.

To obtain this rate of fall, it is necessary to stretch a spring, the force of which is added to that of gravity so as to produce a sufficient acceleration. This releasing spring, when it is stressed, produces a force of 57 kg. The rod itself weighs 11 kg in water. But one must add a force of 39 kg which is due to the flow of the cooling water of the rod, so that the total supporting force for the rod is 107 kg.

It is not possible to provide in a maximum diameter of 58 mm an electro-magnet possessing such a supporting force, the required current being 0.1 a. The electro-magnet used possesses a force of 61 kg. It is therefore necessary to provide the supporting apparatus with a force dividing mechanism. The allowed reduction ratio divides the force by four. It is this reduced force, therefore, that is applied to the electro-magnet, the difference being directly absorbed by the body of the electro-magnet. The weight of the connecting mechanism, however, must be added to this reduced force, so that the electromagnet must now support 51 kg.

The electromagnet itself constitutes the lower part of the driving mechanism of the rod, and will therefore have a translational motion of 960 mm.

The driving mechanism of the rod changes the rotational motion (20 revolutions/minute), produced by a reduction gear, into a translational movement (100 mm/min.). This transformation is effected by means of a ball bearing device so as to decrease the losses of power due to friction.

To obtain this rotational movement of 20 revolutions/min., a reduction gear is used, which is set in motion by a three-phase motor. The rotating speed of the motor is 1,400 revolutions per minute, the reduction ratio being 70.

The motor and the reduction gear are enclosed in a compartment containing compressed air. This permits the use of materials and techniques common with this type of unit.

It is, nevertheless, necessary to prevent any leaks of primary radioactive water towards the reservoir which holds the tank, and to prevent any leaks of water from the reservoir to the compressed air compartment, which is reserved for the motor and the reduction gear.

These requirements lead to the adoption of a double rotary joint. Pure water is injected in the middle section of the joint at a pressure higher than that of the tank: the water in the tank is subjected to a pressure of 13 kg/cm^2 and the pure water is injected in the joint under a pressure of 15 kg/cm^2. A slight leak of pure water may therefore occur towards the tank and, on the other hand, there may be a slight leak towards the compressed air compartment. A valve is set at the lower part of this compartment, so as to permit the evacuation of this water which is subjected to the pressure of the compressed air, that is 1.5 kg/cm^2. This pressure is slightly higher than the hydrostatic pressure of the reservoir at this level. Two devices are provided, to prevent any accidental penetration of this water through the enclosure of the control mechanism. Any abnormal flow of water which is injected in the joint is reported, as well as any rise of water in the compartment.

As far as the electromagnet is concerned, many problems are raised. We have seen that its supporting force is 61 kg, and that the lag, after the end of the excitation of the coil, is of 20 ms. To obtain these characteristics, it is necessary to make a judicious choice of the magnetic material, of its surface treatment, of the wire used for the coiling of the coil's insulation and winding, as well as of the wiring outlets. These problems are important because of the presence of the magnet in the water and in a flux of gamma rays.

Several position signalization systems are provided for the control rod. Two pairs of selsyns are connected to the control mechanism by kinematic chains, with no back-play, thus permitting the determination of the position of the rod when it is connected to its control mechanism. A multiturn potentiometer records the position. The hooking on of the rod is indicated, as well as its release. These last two signals are given by proximity contacts, which by the closing of their magnetic circuit stop the exitation of auxiliary relays located outside the control rod. The use of these magnetic contacts is indispensable in water which is under pressure, and with the limited space available.

The reduction gear, which contains the connections with no back-play transmitting the motion to the selsyns, can also rotate low and high end-of-run cams.

The electromagnet and the magnetic contacts being subjected to the action of gamma rays, they will have a rather limited life-span, and their dismounting must be as easy as possible. To this effect, a six peg connector has been provided. From this connector six wires branch out, which go up three by three along a tube, and then group themselves on the ball bearing device, forming a flexible, helical connection. The length of this connection varies from 250 mm to 1,250 mm when the rod is moved. These wires then go through a double stuffing-box, reach the compressed air compartment and with the other conductors, then reach the radiation proof connector receiving the supply cable of the rod.

1.1.2 Regulatory Rod

The regulatory rod cannot be released, and constitutes an element of the control channel which is intended to maintain a constant flux in the reactor, when the existing perturbations are weak. The channel can also modify the power of the reactor within a period of time varying from 20 to 90 seconds.

The rod is always fixed to its driving mechanism, and its speed varies from 150 to 500 mm/s. It is set in motion by means of a diphase auxiliary motor and by a suitable reduction gear.

Since the whole mechanism is reversible, it is necessary to provide an electromagnetic brake for the motor shaft, for the rod to be controlled manually. A tachymetric generator is also coupled to the motor.

The stroke of the rod is 550 mm long, but it may be displaced inside the core by means of a telescopic coupling which may be maneuvered under water by means of an appropriate device.

The mechanism transforming the rotational movement from the reduction gear, into a translational movement transmitted to the rod, comprises a double rack and a shaft mounted on a ball bearing device with balls moving longitudinally, in order to decrease the friction.

A central end stop provided with two springs limits the stroke of the rod. The control mechanism is also equipped with end-of-run cams, and with two other cams which permit the automatic action of the safety rods, when the control rod reaches the limits of its stroke.

The indication of the position of the rod is made possible, the same as with the safety rods, by two pairs of selsyns and by a multiturn potentiometer which records the position.

Also, the same as with the safety rods, a rotatory joint prevents the leaks of primary water toward the reservoir. A compressed air compartment contains the electrical control and signalizing equipment and the reduction gear.

1.2. Nuclear Measurements

The nuclear equipment comprises 10 measurement channels.

Two channels function at the source level, that is for fluxes between 8.3 and $8.3.10^5$ neutrons/cm^2.s.

Two other channels give logarithmic indications of usage for fluxes going from 1.10^4 to $2.5 \cdot 10^{10}$ neutrons/cm^2.s.

Three linear safety channels operate between the following fluxes: 1.10^8 and $1.8 \cdot 10^{10}$ neutrons/cm^2.s.

All these fluxes are those which must be measured at the location of the detectors.

The nuclear equipment also comprises a regulating channel and two channels measuring power by the activity of the nitrogen 16 which is produced.

The detectors of the first 8 channels are located in a radiation proof casing, immersed in the water of the reservoir.

For the starting channels, fission chambers are used as detectors. The operation and the regulating channels are equipped with ionization chambers which are compensated with regard to the gamma flux. The detectors of the two nitrogen 16 channels will be ionization chambers which are not compensated, and which are therefore sensitive to gamma rays.

The function of the channels is to give an indication and a record of the measurement of the flux levels, from the source level of the reactor before the critical point up to the highest levels obtainable under normal operation, and even above that level before the action of the safety devices takes place. The channels also give the same indications during periods of reduction of the power or of stoppage of the reactor.

Under the same conditions, the channels provide measurements and records of the increase or decrease periods of the neutronic flux.

The measuring channels also permit the realization of various locking possibilities in the control operations of the safety rods and of other mechanisms. These lockings are performed mainly by the two fission chamber channels (starting channels).

The safety channels also supply direct current to the electromagnets of the safety rods, and control this current.

The measuring channels are also used to transmit signals of power reduction, of emergency insertion of the safety rods and of emergency stoppage when the neutronic flux or its positive period exceed certain adjustable predetermined values.

The assembly of the safety channels offers, moreover, the possibility of switching any one of the 15 electromagnetic circuits of the safety rods on a supply unit called "Junior Scram," to insure, under some experimental conditions, the release of only one rod. This is the "minor release" operation.

The continuity of the indications which are given requires a recovery of at least a factor of ten between the fluxes measured by the starting channels and the operating channels.

The detectors of the starting channels will be so located as to provide a lag of a factor of ten between the fluxes measured by the two detectors, taking into account the displacement lockings of these detectors.

Mechanisms insure the displacement of the detectors. They are motorized in the fission chambers, and remote controlled from the control panel.

The other probes are moveable manually. Moreover, the probes of the two logarithmic period channels are placed in a lead shielding.

1.3. Rods, Lockings and Defect Signalization Command Circuits

When the flux is less than 10^{-5} of the nominal flux, only the intermittent raising of the safety rods is allowed. This intermittent raising consists in a period of raising, followed by a period of rest, which time may be adjusted for each particular case.

It is then possible to control the rods in continuous motion. This control is made by groups. The selection of a group is made from a keyboard located at the control panel. Individual controls also permit the correction of the rod positions.

The control rod can function automatically, as soon as the flux has reached one percent of the nominal value. Moreover, as soon as the control rod reaches the lower section of its stroke, it engages to cause the fall of the group of safety rods which has been selected, until the control rod has again reached the midpoint of its stroke.

In the other direction, when the control rod reaches the top of its stroke, the intervention of the safety rods is not automatic.

With an automatic control, the operator can determine the new power level which he wants to reach. A motorized logarithmic rheostat permits the comparison of one signal with that coming from the nuclear measuring channel. Two speeds can be adjusted on the reheostat, so that it is possible to reach the new power level by periods of 20 or of 90 seconds. The comparison of error signal is correctly amplified, so that the auxiliary motor of the control rod can be set in motion.

Let us also point out that it is possible to verify the correct operation of the driving mechanisms of the safety rods without being forced to lift the rods up. A special circuit permits this test, when the rods are not attached ("raise test").

1.4. Equipment of the Control Room

The control room will include a control panel permitting the control of the nuclear section of the reactor.

Facing the operator, behind the control panel, is a control switchboard housing the electronic equipment, the recording apparatus and the defect signalizing devices.

Under the control room, another room will be equipped with the auxiliary apparatus which is necessary to the operation of the control circuits.

2. CONTROL OF THE CORE MODEL (Core Mock-up Facility: CMF)

The CMF of the BR-2 possesses a core which is identical to that of the reactor. It also uses control rods and it possesses nuclear instrumentation. This combination, which permits the control of the CMF, has many points in common with the control installation of the reactor.

2.1. Control Rods

The control rods are nevertheless highly simplified. Since the CMF has a power of only about 50 kw, it is not necessary to provide a forced circulation to cool it; by this very fact, the core does not have to be located in a tank. Besides, fluxes being weak, a reduced water protection is adequate.

The mechanisms of the control rods can therefore be shorter, and the double rotatory joints are unnecessary.

The CMF possesses the same number of control rods as the reactor. They are not any longer all motor driven. Eleven of them are controlled manually, or by means of a detachable motorized tool: these are the poison rods. Five compensatory and safety rods permit the starting of the reactor. Finally, a regulating rod has been provided.

As in the case of the reactor, the mechanisms are entirely immersed. The control rods, strictly speaking the poison rods, of the safety rods and of the compensatory rods are identical to the safety rods of the BR-2. The same identity can be found between the regulating rods of the reactor and of the CMF.

A special arrangement comprising a sleeve allows the fixing of the mechanism to a channel extension tube. This sleeve is operated from the platform by means of a special device.

2.1.1. Poison Rods

The rod launching mechanism is very simple since it solely comprises a ball bearing device whose nut is fixed on the rod. The screw extends into a shaft which can be directly controlled by means of an appropriate device.

The only complication resides in the position indicator for the rod, which surrounds the control shaft and which must indicate the correct position within 1 mm. This requirement needs the use of three concentric needles as well as of a reduction unit by means of gears and a Maltese cross. This miniature transmission gear case operates entirely in water.

A motor driven device has been developed which is adaptable to the control shaft while keeping the reading of the position indicator possible. This tool is controlled from a platform located above the water level. Revolution indicators which can be set at zero allow the raising or the lowering of a rod by a definite amount.

2.1.2. Safety and Compensatory Rods

These rods are releaseable and therefore possess an electromagnet. Miniature circuit breakers indicate whether the rod is attached to its launching mechanism, or whether it is released. As in the reactor, these electrical devices, which are in water, are connected to the driving head of the mechanism by a helicoidal connection, the length of which must also go from 250 to 1,250 mm when the rod moves. The problem, nevertheless, is easier to solve here, because of the absence of radiation damage and of high pressures.

One difficulty arises however. The shortening of the mechanism is such that it is necessary to develop an electromagnet allowing the passing through of the ball bearing device. It is therefore necessary to study this electromagnet carefully, so that the presence of the ball bearing device may not disturb the magnetic circuit.

The lag for the electromagnet is here also less than 50 milliseconds. Also, the rod must travel 0.305 m in less than 0.250 s. These requirements, as well as the limited space available for the electromagnet, lead to the utilization of a releasing mechanism which comprises a reduction of the effort applied to the electromagnet, thus permitting at the same time the utilization of a releasing spring.

The same as with the reactor, the launching mechanism which comprises the motor, the reduction gear, the selsyns, the potentiometer and the end-of-run circuit breakers, is enclosed in a casing which is supplied with compressed air.

2.1.3. Regulating Rod

It comprises the same control mechanism, but it is not releaseable. There is therefore no electromagnet.

2.2 Nuclear Instrumentation

The nuclear measuring channels here are seven in number.

Two channels function at the source level for starting. During operation, two logarithmic period channels as well as two safety channels are used. One last channel is used for the regulation of the reactor.

The starting channels cover a range of 2.5 to $2.5 \cdot 10^5$ neutrons/cm^2.s. and utilize a fission chamber as a detector.

The logarithmic period channels cover a range of $3.3 \cdot 10^4$ to $6.7 \cdot 10^{10}$ neutrons/cm^2.s. The detectors are compensated ionization chambers.

The safety channels operate within the same range, and utilize the same detectors.

The same considerations hold true for the regulating channel.

The extreme end of the range of the logarithmic period channels corresponds to 300 percent of the nominal power. The safety channels permit a measurement of up to 150 percent of this same power.

The launching channels are completely conventional. Let us note that the preamplifier is associated with the fission chamber in a radiation proof probe which is plunged in the reservoir water. The motorization of these probes is projected. One of the channels indicates linearly the rate of counting. The second furnishes a logarithmic indication and is also equipped with an audible indicator.

The two logarithmic period channels are identical and furnish a logarithmic indication of the flux level and a periodic indication.

These channels are equipped with recording devices, and also give alert signals or emergency stoppage signals for various levels of power and period.

The two safety channels are not identical. One of them has two types of scales, which are in a ratio of 1 to 100. They also comprise two units which maintain in position the rods supplying the electromagnets of the control rods. Naturally, the safety amplifiers send out emergency stoppage signals to the rod holders, which may moreover receive signals from the logarithmic period rods.

The regulatory channel acts completely or not at all on the control rod, which possesses only one displacement speed. This channel keeps the power of the CMF constant. To arrive at that result, the reference level is compared with the actual level by means of an installation which is incorporated in the recording device. The error signal which results from it is amplified and stabilized by a counter reaction from the control rod. The signal then engages the rod in the direction which is appropriate for its regulatory function.

2.3 Control of the Rods

The rod control circuits are quite comparable to those of the BR-2. The locking requirements are nevertheless less stringent.

Let us point out that while the five safety and compensatory rods are identical, they are not all associated to identical control circuits.

One or two of these rods are treated as safety rods. The others, four or three, are compensatory rods. It is these last which will insure the starting of the reactor. But their operation will not be permitted until the rod or the two safety rods have arrived at the highest point of their travel.

2.4. Control Panel

The panel is built in one piece, so as to be moved near the reservoir, if needed.

This panel comprises a central section and two lateral wings. These last are slightly inclined in relation to the central panel, so as to facilitate supervision. The operator has at his disposal a small console desk which is attached to the central panel section and on which the indications of rod positions and all the controls and signalizations which are necessary to the operation have been set.

The electronic apparatus is laid out over the whole panel. The upper part of the panel comprises the defect signalization equipment. All the electronic devices are extracted from the fore section. The most important of them can be swung on their slide bars so as to facilitate access to the wiring. This panel is equipped with adequate ventilation.

3. CONCLUSION

The present pamphlet constitutes only a short survey of the installations of nuclear control of BR-2. The description is incomplete. The reader will be benefited if he completes his documentation by reading the studies published by N.D.A. "Nuclear Development Corporation of America" and by the C.E.N. "Centre d'Etudes Nucléaires".

We take advantage of this opportunity to praise the spirit of collaboration which we have encountered with the engineers of the C.E.N. and of the N.D.A., thanks to whom an important portion of the work which is described above has already been performed successfully.

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CONTROL EQUIPMENT OF THE BR-2 REACTOR: CONTROL OF RADIATION AND RADIOHYGIENE

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The Br-2 reactor, of very high neutron flux, is intended for materials testing.

In order to guarantee complete safety in the various experiments provided for, it was indispensable to provide a very complete radiation control equipment.

Large use was made of scintillation counters with sodium iodide crystals, activated with thallium, permitting selective measurement of gamma radiation as a function of energy, and at the same time ensuring a high yield.

The types of control carried out are as follows:

- detection of leaks of fission products in the primary water circuit;
- detection of N 16 in the secondary water circuit at the outlet of the exchanger beds;
- measurement of level of radiation of water in the pools;
- measurement of rate of Argon 41 in the ventilation air;
- measurement of the activity of dust in the ventilation air;
- measurement of contamination of hands and feet of personnel who stayed in the hot premises;
- measurement of level of gamma radiation surrounding the working premises;
- measurement of rapid and slow neutrons at the level of the reactor core.

In the following we are giving a brief description of the measuring apparatus designed for such measurements:

MEASUREMENT OF FISSION PRODUCTS

A defect in the grain of the fuel elements is expressed by the presence of fission products in the water in the primary coolant circuit.

A sample of this water is taken and passed through an ion exchange system which separates certain fission products (particularly iodine 135).

The unit comprises two distinct parts:

- the chemical concentration apparatus is mounted on its own frame. After a pressure reducer brings the water back to atmospheric level, the latter goes successively through a filter, an anion exchanger, then a cation exchanger, where the period of storage is about 2 minutes.

In the latter exchanger, a photomultiplier with NaI scintillating crystal observes the exchange column.

- The electronic apparatus comprises:

a gain amplifier of 1000

an amplitude selector which makes it possible to measure only the lines of Iodine 135

a linear logarithmic flow meter

an alarm system

a sweeping unit

a galvanometric recorder

As soon as an alarm indicates that the set level of activity has been surpassed, the sweeping unit is automatically started, recording the spectrum of the fission products during the entire period of the alarm.

MEASUREMENT OF THE NITROGEN 16 ACTIVITY IN THE WATER IN THE SECONDARY COOLANT CIRCUIT AND OF THE ACTIVITY OF THE GAMMA RADIATION IN THE POOLS

The measurement of the activity of Nitrogen 16 makes it possible to detect a leak between the primary and secondary coolant circuits. Nitrogen 16 is produced in the nucleus of the reactor by the action of the neutrons on the O^{16} (reaction: O^{16} (n,p) N^{16}) and is conveyed to the primary circuit while undergoing a rapid decay (half-life: 7 sec.).

Any leak between the primary and secondary circuit results in the appearance of Nitrogen 16 in the secondary waters. These waters, withdrawn by tapping with an auxiliary pump, are sent along in a container provided with an NaI scintillator having walls that can sustain 12 atmospheres. The latter is connected to two standard control units located near the hatch.

As Nitrogen 16 emits mainly a gamma ray of 7 MeV, it will be possible to make a very specific measurement by selecting the impulses of the scintillator corresponding to an incident radiation energy of 6 to 7 MeV, that is, the impulses caused by photoelectrical effect and the production of electron-positon pairs.

The measurements on the pool waters is done in a similar fashion but in the energy range above 100 KeV.

Eight pool measurements and four exchanger measurements constitute the water circuit observation network of the BR-2.

DETECTION OF ARGON 41 IN THE VENTILATION CIRCUITS

Argon 41 is produced by the capture of neutrons by natural argon in the air serving to ventilate the containment building and the water circuits building and the hot laboratories process building. In particular, the air cooling the biological shielding as well as the "beam ports," is subjected to an intense neutron flux.

The air, whether that in ventilation system pipes or merely in the premises (for example, in the hermetically sealed access doors), is sent by an auxilliary pump to a shielded gas tank located outside of the buildings and buried underground. Such installation makes it possible to considerably diminish the background radiation. A scintillator with NaI crystal of 3"/3" is mounted on the upper cover of the housing and is connected to the standard units as above. The amplitude selector is controlled in such a way that only the 1.4 MeV Argon line is measured.

Eight measurement circuits for Argon 41 are provided in the assembly of the ventilation system.

MEASUREMENT OF THE AIR DUST ACTIVITY

This measurement is important in the detection of leaks of fission products and particularly of plutonium in the form of microscopic particles. Such leaks may take place in connection with errors in handling in the hot laboratories or in the handling of fuel elements. The air is sucked in through a filter paper contained in a housing with very thick shielding (10 cm of lead) and is observed continuously by a Geiger counter with thin wall (simultaneous alpha and beta measurements).

The mechanical assembly of the paper drive, pump, standard THT flow meter as well as control alarm panel are mounted on a very handy mobile carriage. An air flow meter is incorporated in the installation.

The control panel makes it possible to check the working condition of the various parts of the machine as well as of the alarm systems.

Inside the shielded housing is mounted a calibration source moved by Ledex type magnetic pull motor. This source, either remote or local-controlled, is placed opposite an opening permitting the irradiation of the GM counter.

The air flow is 8 m^3 per hour. The control of the background radiation is obtained by the rapid feed of the paper, placing opposite the detector a non-charged portion of dusts. Such control may be effected by remote or local means.

MEASUREMENT OF SLOW AND FAST NEUTRONS

This measurement is done in the premises which are on the same level as the reactor core and in the axis of the access ports whose opening produces a strong irradiation of the environment.

The measurement of rapid neutrons is made by means of polyethylene-wall proportional counters working on the principle of recoil protons. The measurement of the slow neutrons is carried out by BF-3 proportional counters.

Standard electronic units are used for amplifying and integrating the measurement. The unit formed by the chassis, including the detector and the recorder, are mounted on a mobile carriage.

STANDARD ELECTRONIC UNITS

These units were built for adaptation to all possible uses, such as measurements of gamma rays from N^{16}, A^{41}, I^{135} by scintillation, rapid and slow neutrons as well as the Geiger-Müller counters of thin wall in the dust machines. They comprise the following:

1) One gain preamplifier, 1 to 75 ohms output impedance.

2) One chassis amplifier and amplitude selector. This chassis includes a 100-gain amplifier with a 5-position attenuator mounted with a printed circuit, an amplitude analyzer with lower and higher thresholds separately adjustable by accurate potentiometers, mounted with a second printed circuit. The adjustment range of the two thresholds is from 0.4 to 10.4 volts.

The total number of these installations for the BR-2 is about 25.

3) A flow-meter ("Counting-Ratemeter") Lin-Log, with regulated high voltage. The apparatus comprises the following three principal parts:

- An integrating flow meter, with 12 linear scales, covering the ranges from 0.5 to 10,000 cps as basic scales, and 1 logarithmic range with 4 1/2 decades from 0.3 to 10,000 cps. The stability of the reading is better than one percent (on linear). Four time constants are chosen by a switch for the linear ranges. The input sensitivity threshold is 200 mV; the deadtime 2.5 μsec. and maximum integration time (2RC) is 360 sec. on the lowest range. The apparatus is provided with an outlet for a galvanometric (1 ma) or electronic (100 mV) recorder.

- A THT stabilized for 1/1000 and adjustable by potentiometer; the range of adjustment is from 500 to 2000 V.

- An alarm circuit with adjustable triggering threshold is incorporated.

MEASUREMENT OF SURROUNDING GAMMA RADIATION

This measurement is specifically intended for the protection of the personnel; it is made in more than 50 different places.

The detectors are made up of an ionization chamber of about one liter in volume filled with argon under pressure (10 atm.) and an incorporated electrometer. The electronic circuits utilize transistors.

The current of the chamber constitutes the grid current of the electrometer.

In addition to the detector, the detecting unit comprises an audio-frequency modulation amplifier circuit capable of furnishing several volts at a low impedance.

The high-voltage supply of the chamber is also obtained from this circuit.

The energy response is constant to within 15 percent between 100 KeV and 2 MeV. The calibration of the chamber may be accomplished from a distance by a source of strontium in connection with a Ledex type magnetic-pull motor and appearing opposite a very thin stainless steel window (20 μ), located on the upper cover of the chamber.

The measurement range extends over 6 decades in logarithmic response, generally only 3 of these decades are used; that is:

measurement range from 0.01 mR/h to 10 mR/h
" " " 0.1 mR/h to 100 mR/h
" " " 1. mR/h to 1000 mR/h.

A special option is provided for the range between 1 mR and 1000 R.

The temperature compensation of the unit is made up in a way to give a deviation lower than 2 percent for a variation of 30°C.

Junction Boxes

The connection of the detector units to a central measuring plant is done through junction boxes collecting the outputs of several detectors in a single lead to the central plant. Moreover, these junction boxes have a high-voltage supply for triggering calibration circuits.

Units of Measurement and Alarm

These units comprise a detector circuit with an output adapted to a milliamperemeter and a recorder measurement resistance; an alarm circuit triggers an electromagnetic relay when the gamma activity surpasses a predetermined level. The assembly of these two circuits is mounted in a small chassis.

These units are assembled in groups of five.

The following adjustments are provided: measurement or calibration switch;

potentiometer for gain adjustment;
potentiometer for zero adjustment;
potentiometer for adjustment of alarm calibration signal;
potentiometer for alarm threshold adjustment.

If two alarm levels are desired, the milliammeter for measurement includes a contact whose position is adjustable and which triggers an electromagnetic relay when the needle reaches its height.

General Control Unit

This unit comprises the luminous signs for alarm and indication of calibration. "Reset-alarm" and relay set switches designed for ensuring the control and safety of the operation are also included.

Stabilized Low Voltage Supply

This low-voltage supply furnishes the required 10 volts D.C. for the operation of all the circuits using transistors. The variation in output voltage is less than 0.1 percent when the line voltage varies by ± 20 percent and the temperature is 30°C.

Measurement Hatch

The hatch comprises, in two rows, the 10 units of measurement and alarm, the control panel, the low voltage power supply as well as the safety circuits.

All of these circuits are completely transistorized.

APPARATUS FOR CHECKING HANDS AND FEET

This apparatus is intended for the sanitary checking of the personnel working in the laboratories or premises with danger of deposit of radioactive substances in solid form. It comprises a metal stand with base for placing feet, surrounded with a 5 cm lead shield up to mid-height of leg; two openings at a person's mid-height permitting the placement of the hands, with 5 cm lead protection.

The measurement of the beta activity is by means of long GM counters. The electronic circuits include a stabilized THT and integrating circuits with very long time constant permitting the accumulation without loss, of the signalling of several minutes of counting; a timer on the control board permitting the adjustment of counting time; the reading is done on a milliammeter.

EXPERIMENTAL FACILITIES OF THE R.C.N. MATERIALS TESTING AND RESEARCH REACTOR

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1. Introduction:

The RCN materials testing and research reactor is of the ORR type (Oak Ridge National Laboratory research reactor).*

Many facilities are available which give access to the core region. In these facilities use can be made of thermal and fast neutron fluxes up to about 10^{14} n/cm^{2}sec at a power level of 20 MW.
Compared to the ORR the number of horizontal beamtubes has been increased from six to ten, by omission of one of the large engineering facilities.
Special attention has been directed towards convenience in setting up experiments and provision of ample space for equipment, associated with experiments.

A discussion of the experimental facilities is presented in this report.

The experimental facilities may be divided into two main groups, which will be discussed in the following chapters.

Chapter 2. The in-core and reflector facilities

They consist of:

2.1. In-core through tube
2.2. U-tube
2.3. Holes in the beryllium reflector and aluminum elements
2.4. Fuel and reflector element positions, (re-entrant loops, instrumented capsules, etc.)

Chapter 3. The facilities outside the core box

They mainly consist of:

3.1. Horizontal beamtubes (ten in total)
3.2. Large horizontal facility (thermal column)

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* A general description of the facility was presented at the EAES enlarged symposium, Noordwijk 1957: Report, RCN-Ext.-1059, "Description of Dutch Materials Testing and Research Reactor".

(Text continues on p. 234.)

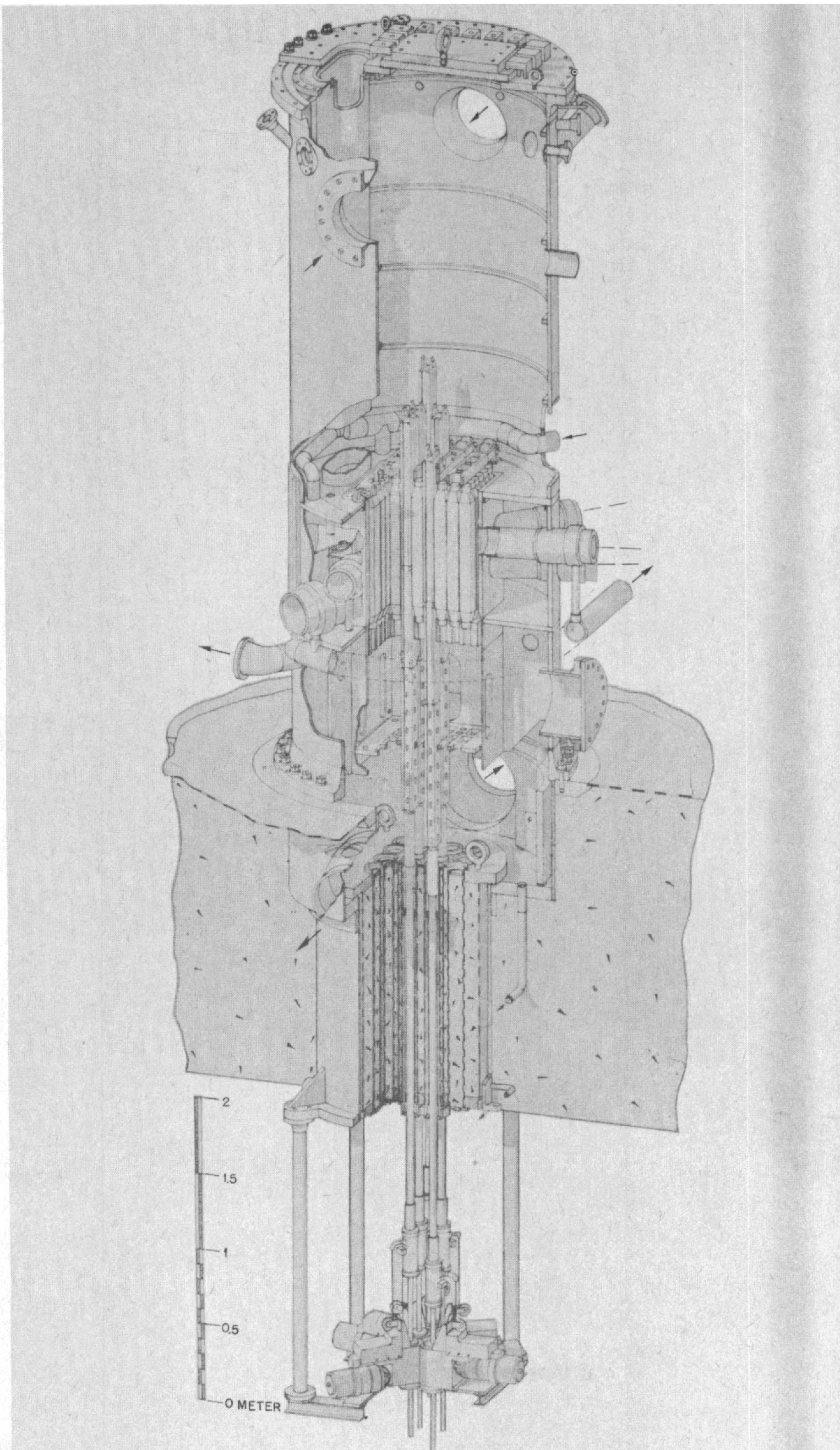


Fig. 1—R.C.N. reactor vessel.

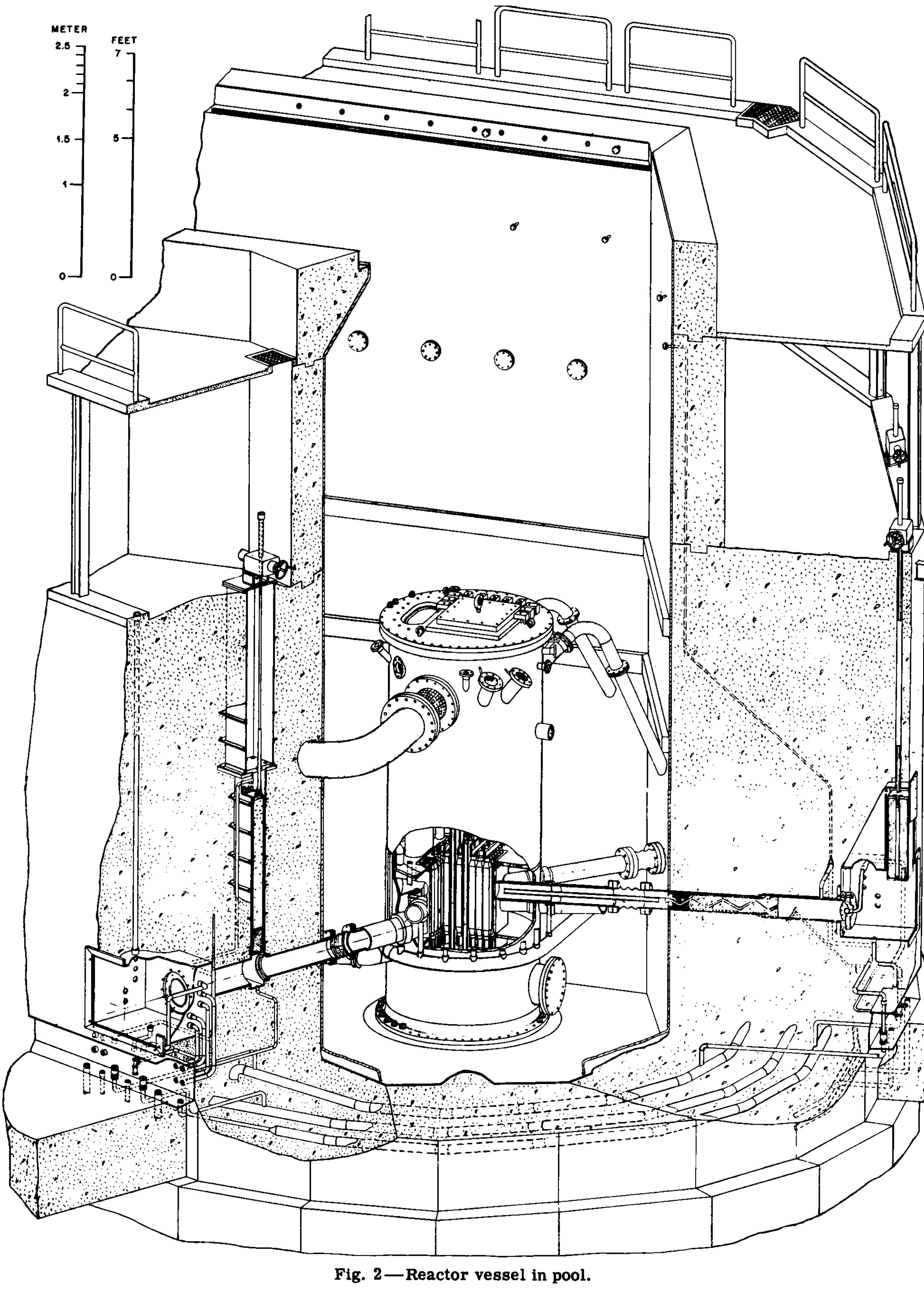


Fig. 2—Reactor vessel in pool.

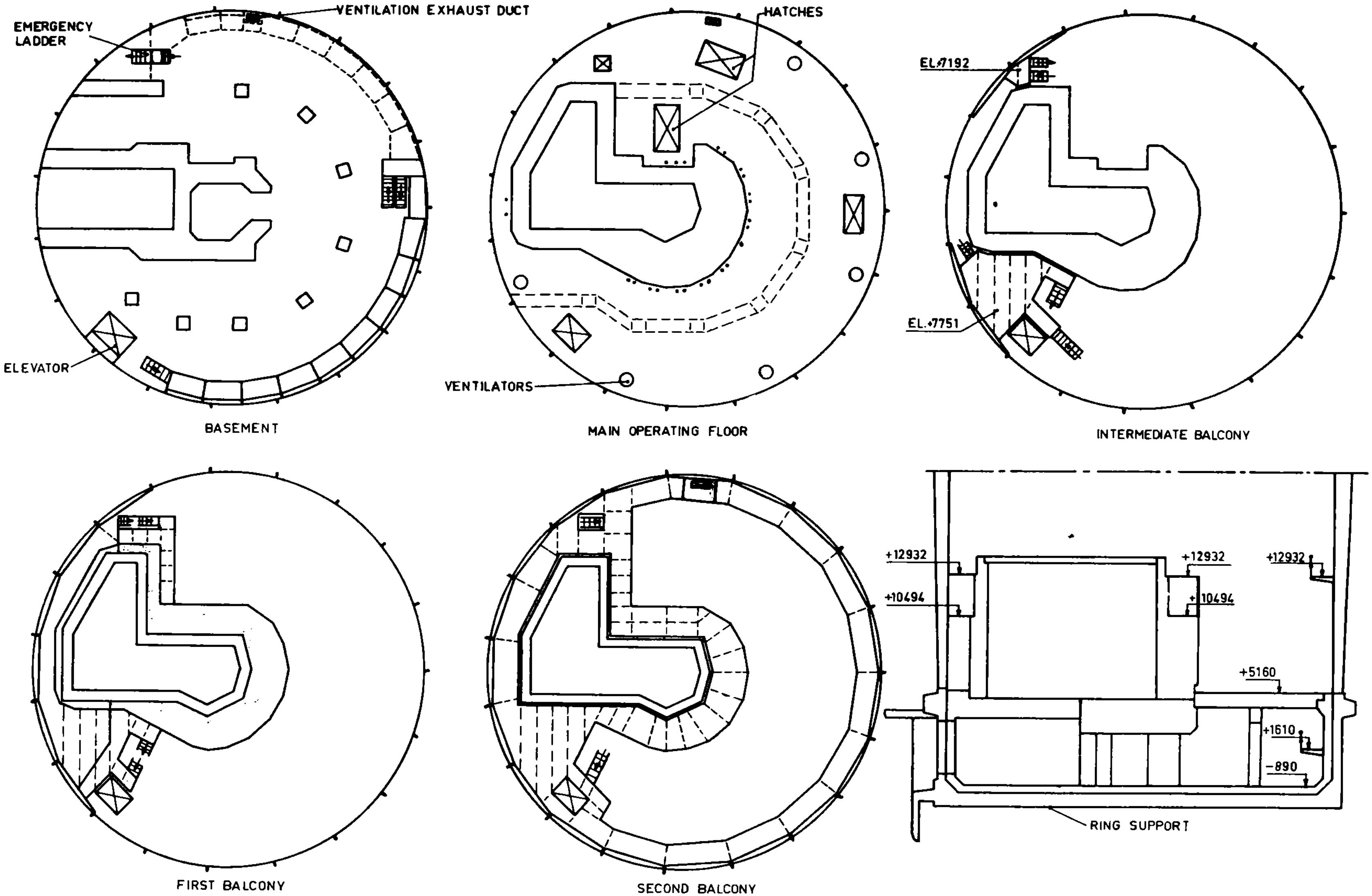


Fig. 3 — Working levels.

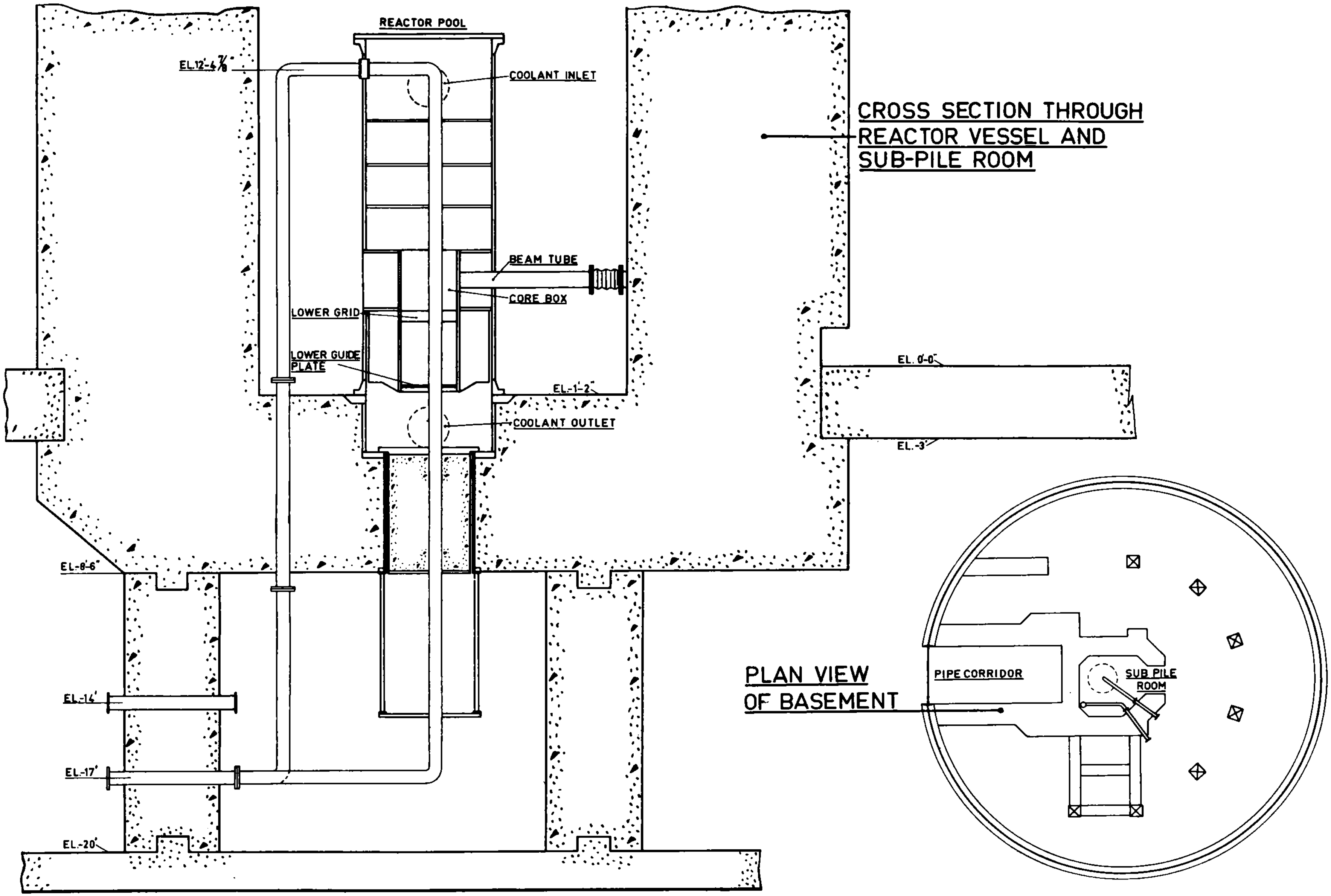


Fig. 4—Schematic layout of in-core through loop.

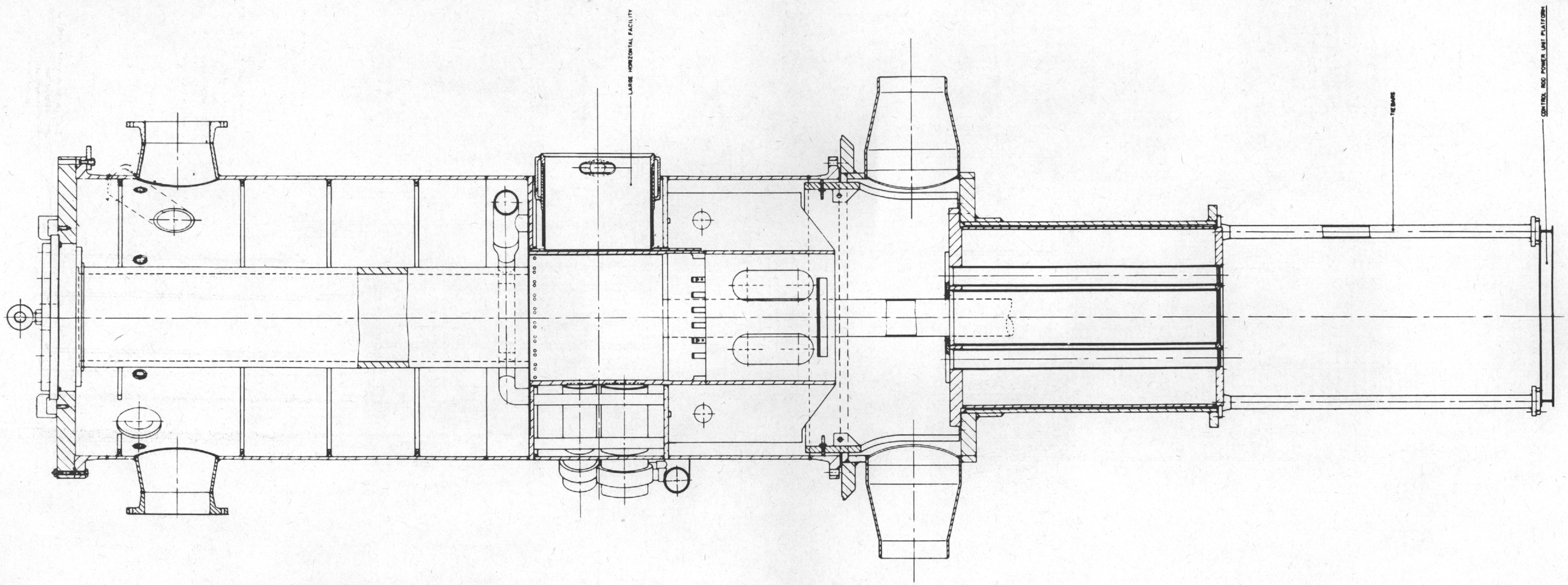


Fig. 5A—Cross section through vessel.

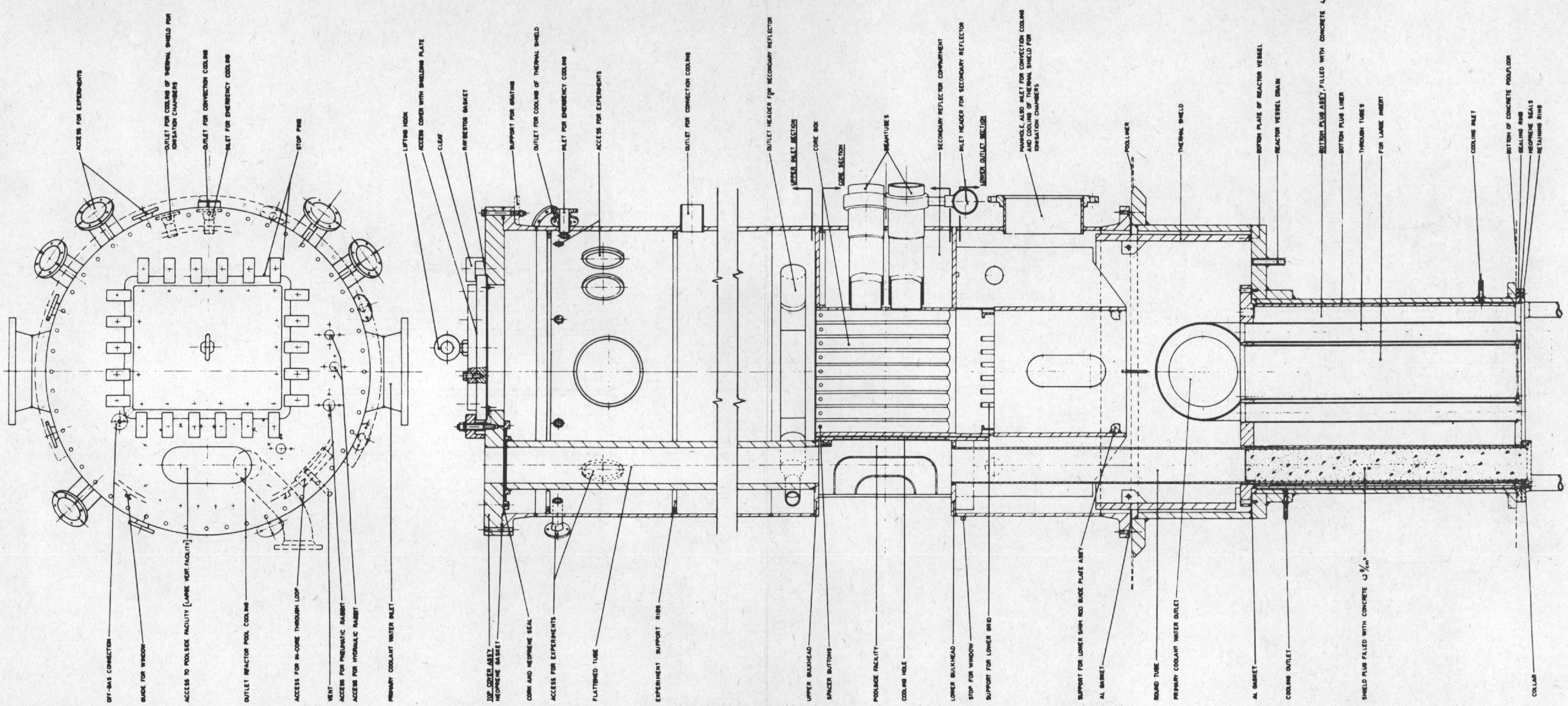


Fig. 5B—Cross section through vessel.

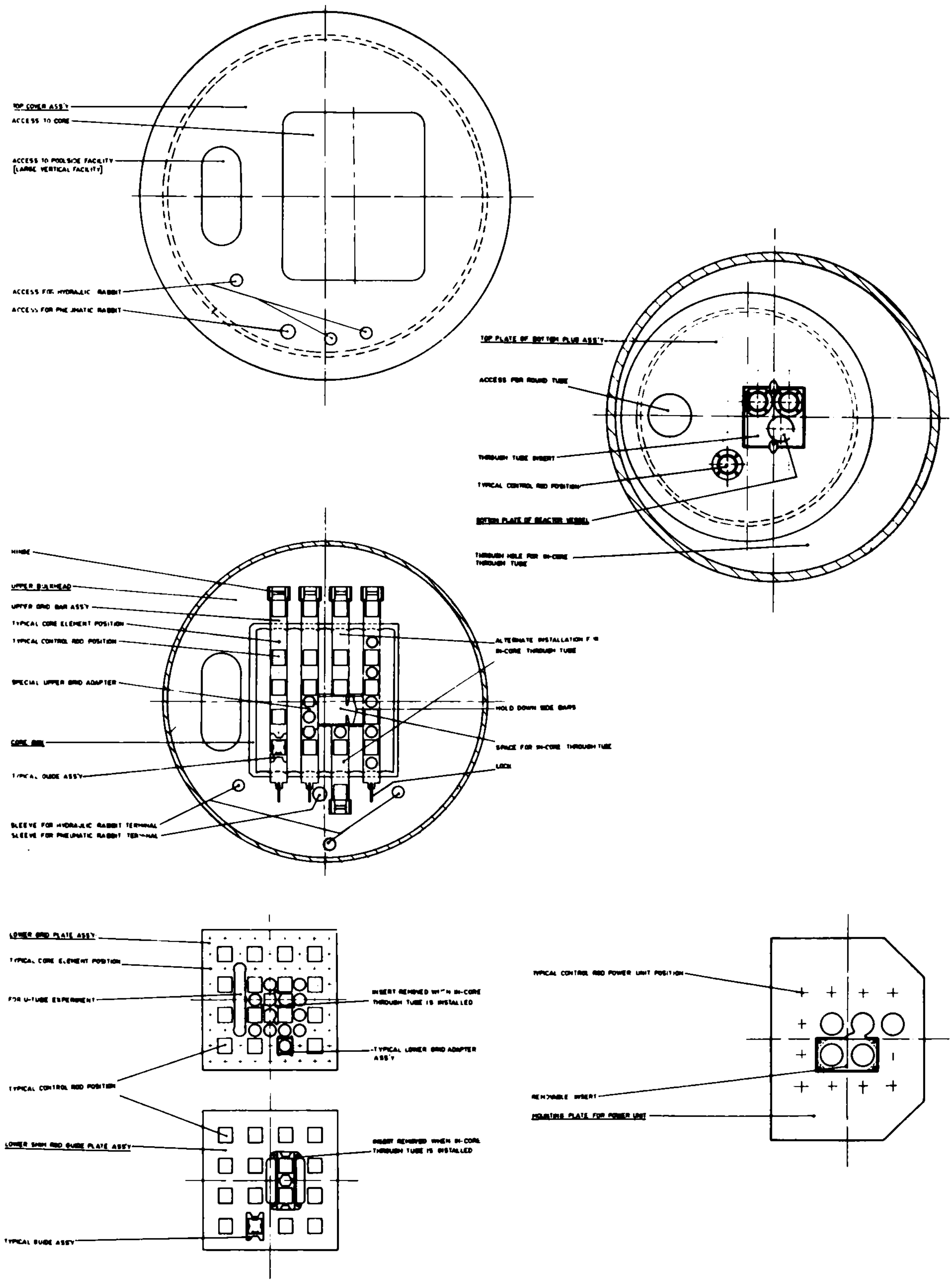


Fig. 5c—Cross sections through vessel.

3.3. Pool side facility
3.4. Rabbit facilities (one pneumatic and three hydraulic rabbits)

A general impression of the reactorvessel is given in figure 1, whereas figure 2 shows the reactor vessel in its position in the reactor pool.
Figure 3 presents a survey of the working levels in the reactor building: the basement, the main operating floor and the balconies. It also gives an idea of the available space for the erection of equipment.

2. The in-core and reflector facilities

2.1. The through-tube facility

If desired it is possible to install a through tube (diameter around 15 cm) in the vessel.
This tube will enter the vessel through an access in the upper tank part. From there on it will run downwards through the entire vessel, leaving the vessel via the bottom plug. The tube will then run from the sub-pile room into the main basement, where the necessary experimental equipment will be installed.
The return line will run through a hole in the bottom of the reactor pool.
A schematic diagram of the through-tube circuit is given in figure 4.

The access to the vessel is shown on the middle part of figure 5, top view of the vessel.

Certain design features will facilitate the installation of a through tube, although this will still require a major operation.
These provisions may best be seen on the right-hand part of figure 5, where several cross sections through the vessel are presented.

One of the normal upper grid bars must be replaced by an alternate installation. This special assembly consists of two parts. (fig. 5, second cross section from top).

The lower grid plate and shim rod guide plate assemblies have been provided with special inserts. When the through tube is installed these inserts should be removed by lifting them out of their positions. (figure 5, third and fourth cross section from top).

The normal insert in the bottom plug assembly having 4 shim rod access possibilities, must be lowered from its position and replaced by a through tube insert, having a through hole besides 2 normal shim rod holes. (figure 5, second cross section from bottom)

Finally, the insert in the control rod power units mounting plate should be removed. (figure 5, bottom cross section).

It may be mentioned at this point that there are 16 shim rod positions available for installation of the 6 rods, required in normal operation. This provides maximum flexibility for arrangement of the core components. The available positions can for example be seen on the

right-hand part of figure 5, cross sections of lower grid and shim rod guide plate.
The holes in the bottom plug that are not used, are provided with plug inserts. (see e.g. figure 1).
It is also possible that the spare holes are used for small size in-core through loops.

It will be clear that installation of a through tube will require removal of the fuel elements and complete drainage of the vessel, so that exchange of the bottom plug inserts will be possible.

In the walls of the sub-pile room several through pipes have been provided to facilitate connections with the remaining part of the basement.

2.2. The U-tube facility

A second in-core irradiation facility consists of a U-tube. For this purpose the side-walls between five adjacent holes in the lower grid plate have been omitted, so that a slit of 5 x 1 grid positions became available. (see figure 5, right-hand part, third cross section from top). No special provisions in the upper grid assembly are necessary, as the tube will run between 2 adjacent bars.

The slit in the lower grid plate will make it possible that both legs of the tube can run through the core region, the return bend being below the lower grid. The tube can have a maximum diameter of about 7 cm. Access to the vessel can be made through 2 of the 5 entrant pipes, 15 cm in diameter and 45° inclined, located on the upper tank part. (see figure 5, middle part, top view of the vessel).

For connections with loop equipment use can be made of 4 inclined pipes, connecting the reactor pool with the basement. (see 2.4.)

2.3. Irradiations in the reflector elements

The beryllium reflector elements are provided with plug inserts. The plug body is 47.5 mm in diameter, whereas the hole in the element is 52.4 mm.

There are 2 types of aluminum filler elements, completely solid elements, and elements with a plug insert. The latter are of a similar design as the beryllium elements, except that, where the beryllium plugs are solid, the aluminum plugs are provided with a through-hole, 22 mm in diameter.
For illustrative purposes the aluminum elements with plug are shown in figure 6.

If the holes in the reflector elements are required for irradiation purposes, the plugs can simply be lifted out of the elements with a handling tool. Smaller irradiation capsules may be positioned in the aluminum elements without removal of the plug.

An example of an irradiation tool in the 52.4 mm diameter holes is a rod, equipped with several trays along the length of the rod, each tray being loaded with a certain number of standard capsules.

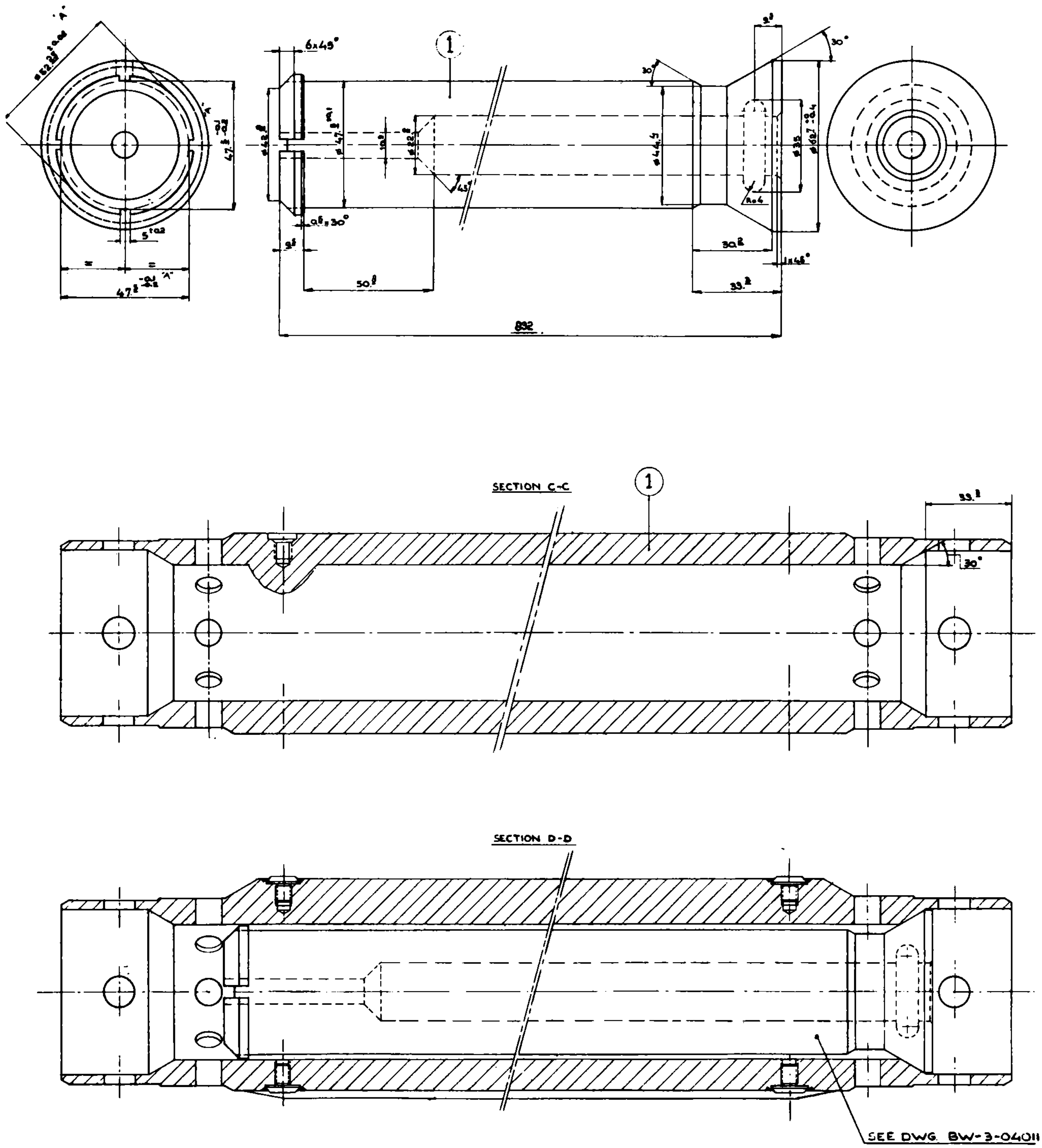


Fig. 6—Aluminum filler elements with plug.

2.4. Fuel and reflector element positions

It will be clear that in principle it is possible to use any position in the core box for irradiation purposes, by omission of a fuel or reflector element.

If a closed circuit is required for an experiment in a core element position, use can be made of re-entrant tubes.

A schematic diagram of such a circuit is presented in figure 7.

Connections to the outside of the vessel are made via one of the 5 flanged access pipes, 15 cm in diameter and 45° inclined, located around the inlet section of the vessel. (see also 2.2.).

Four of these correspond with 15 cm diameter pipes, embedded in the east wall of the reactor pool, and connecting said pool with the basement, where bulky experimental equipment may be erected. (see also 2.2.).

At this point, the attention may be directed towards the provisions for the installation of instrument leads.

Eight access pipes, 5 cm in diameter and equally spaced around the perimeter of the upper tank will mainly be used for instrument leads. Four of these pipes, on the south side of the vessel, are 45° inclined, the remaining four, located on the north side, are perpendicular to the tank wall. (see figure 5, middle part, top view of the vessel).

Four circumferential ribs, 25 x 37.5 mm, welded to the inside of the upper tank, and each provided with 10 holes, 19 mm in diameter, can be used to support the instrument leads inside the vessel. They may also be used for clamping experimental tubing etc. to the tank wall. (see figure 1)

The instrument leads can leave the reactor pool either through the 7.5 cm electrical conduit outlets above overflow level, or through the 10 cm access tubes on the first balcony level. (see figure 2).

In figure 8 a possible schematic lay-out of leads to an instrumented capsule is shown.

3. The facilities outside the core box

3.1. The horizontal beamtubes

The horizontal beamtubes terminate on the east and south side of the core box; eight of the tubes are 175 mm in diameter, the remaining two have a diameter of 250 mm. Five of the 175 mm tubes are situated approximately 115 mm above the horizontal centreline of the reactor core, the remaining tubes 115 - 150 mm below the centreline.

A general impression of the lay-out of the beam-tubes may be had from figure 9, which gives a schematic

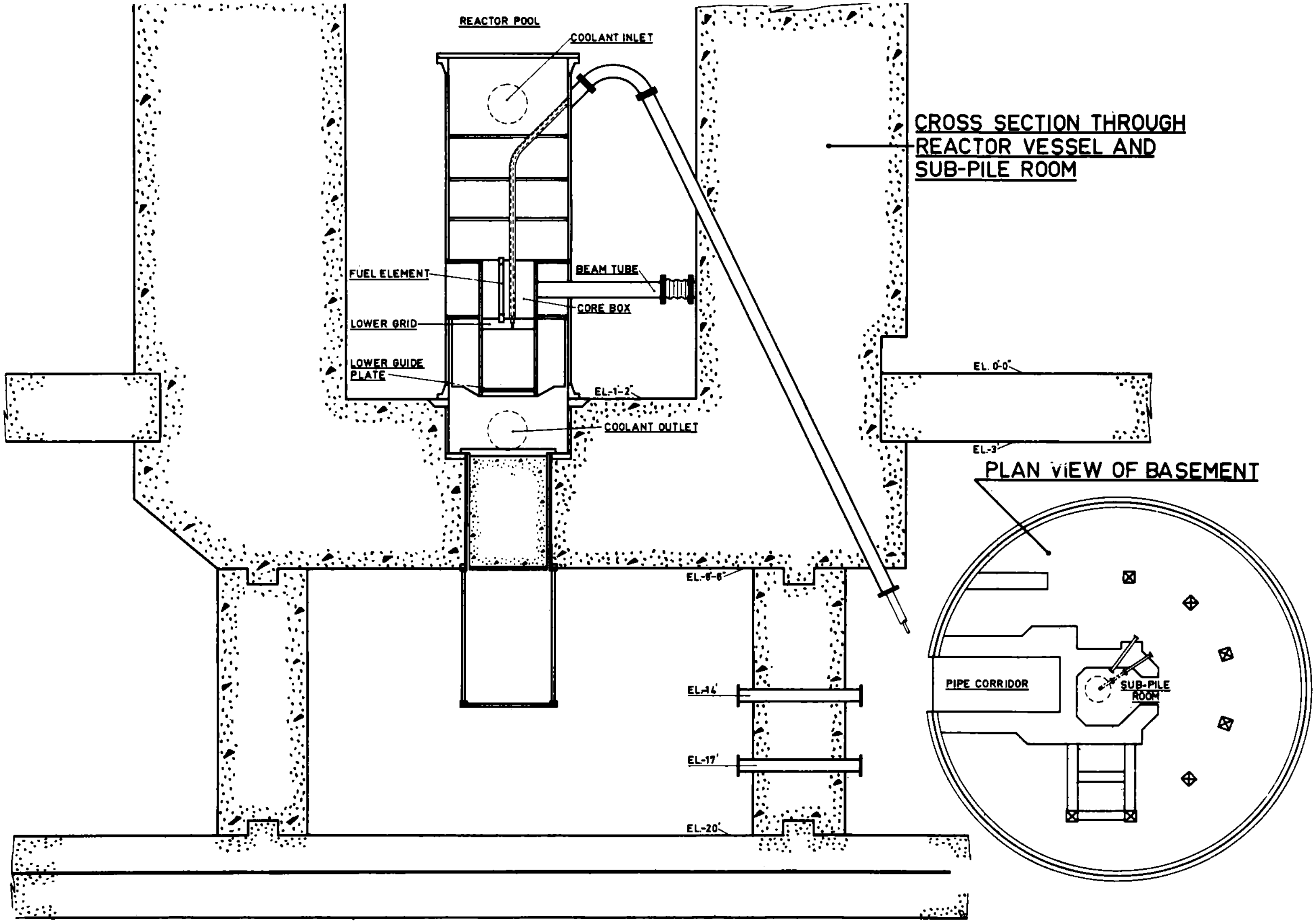


Fig. 7—Schematic layout of in-core re-entrant loop.

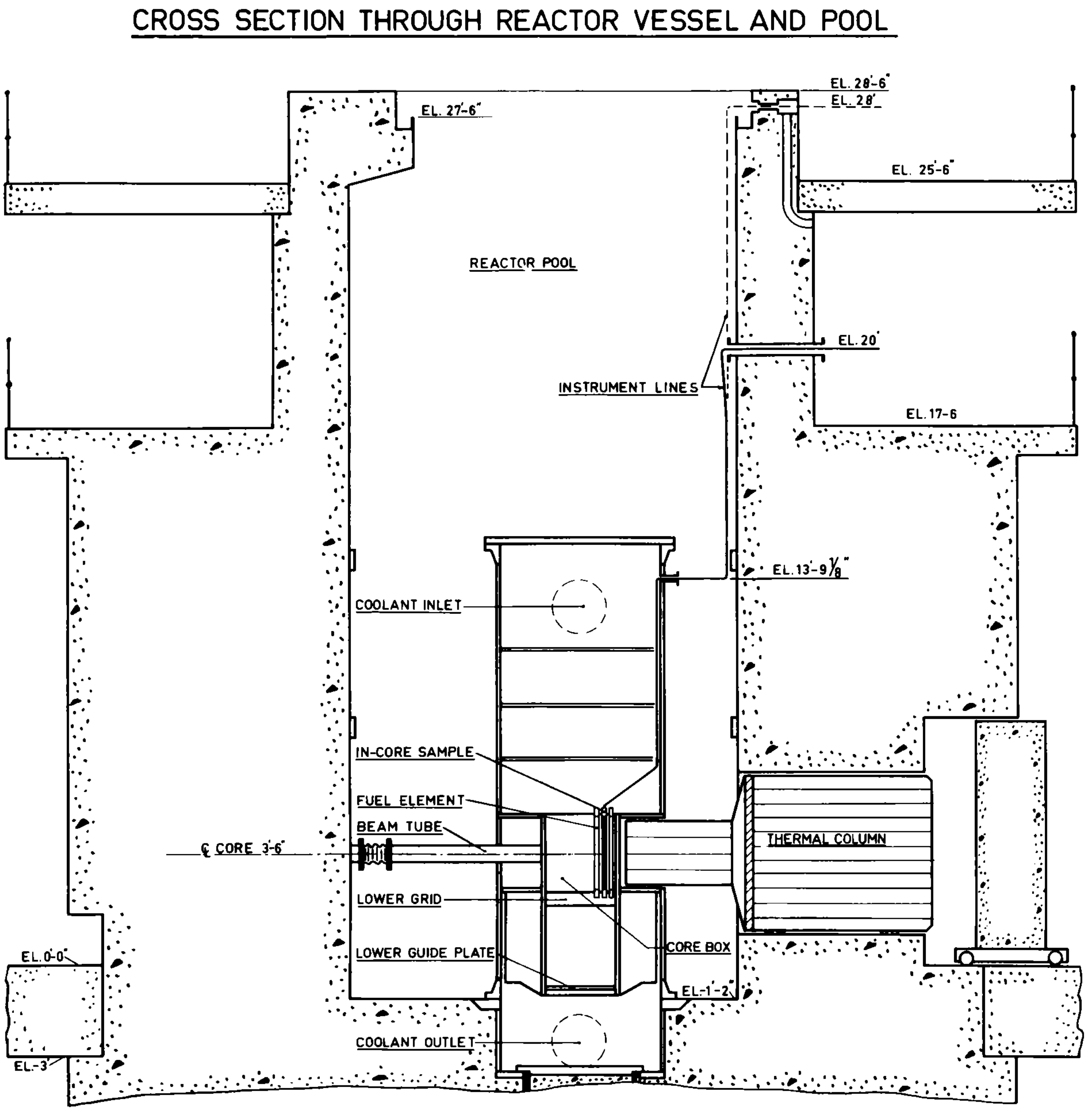


Fig. 8—Schematic layout of in-core instrumented sample.

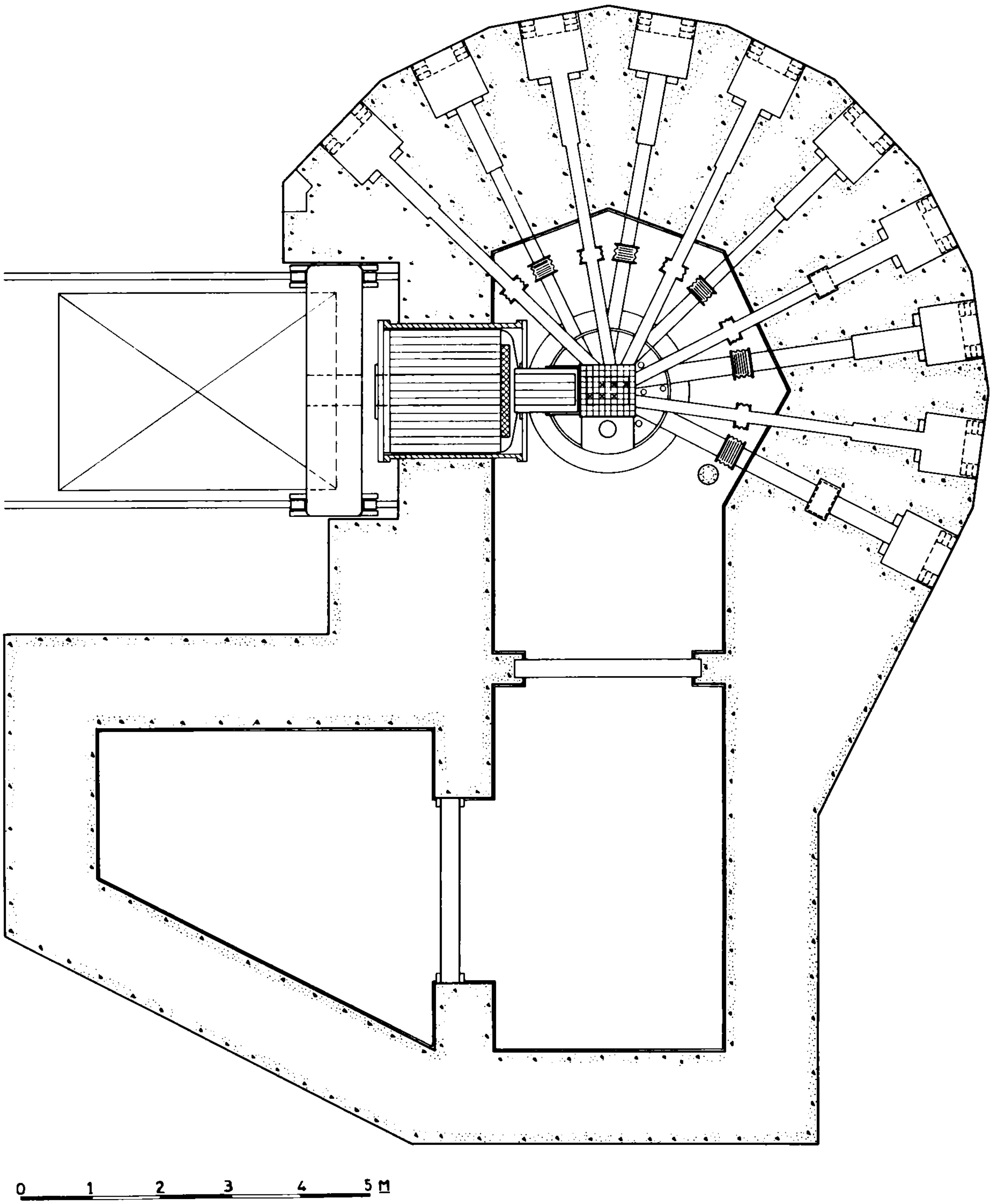


Fig. 9—Schematic horizontal cross section through reactor vessel and pool structure.

horizontal cross-section through the reactor vessel and pool, and from figure 1.
At the outer pool wall the beamtubes end in cubicles. Eight of the tubes are equipped with a lead shutter, housed in an extension of the cubicle at the outer face of the concrete shield. The remaining two tubes, one of 175 mm and one of 250 mm diameter, have an internal shutter of comparable design. The latter arrangement will make it possible that during a shut-down period collimators or other equipment are rearranged, without extensive provisions of external shielding being necessary.

The drive units of the shutters are located on the first balcony and are operated either manually or by means of an electric motor.

Figure 2 gives a more detailed picture of 2 beamtubes -including cubicle and chase- with an internal and external shutter respectively.
The beamtubes are made up of two aluminum sections, one welded to the reactor vessel and the other embedded in the concrete of the biological shield. The sections are connected to each other by a stainless steel bellow expansion joint, located just outside the shield, in the pool.
The section that is welded to the vessel is partly surrounded by a concentric sleeve. Cooling water from a horizontal inlet header enters the secondary reflector compartment of the vessel through the annulus thus formed. (see fig. 5, middle part).
The section embedded in the concrete is of a stepped design. The section is flanged to the back plate of the port box assembly. A cooling water return and drain connection are provided on this part of the tube. In the part with the larger diameter a stainless steel track is bolted to the bottom part of the tube. It acts as a guide, when inserting plugs and equipment.

The port box assembly, embedded in the concrete on the outside of the biological shield mainly consists of a cubicle, and a housing for the shutter for 8 of the beamtubes.
Below the cubicle at floor level there is a steel lined recess, the so-called chase. It extends between all beamtubes, and serves as a connecting station between outside facilities and the cubicles and beam-tubes.
The lay-out of the service connections is given in figure 10, and is also shown on figure 2. Two vertical tubes for experimental leads connect the cubicle to the first balcony.
Two sets of 4 tubes each, connect the cubicle with the chase, and can be used for any purpose. Two other connections lead to the off-gas system and to the facilities cooling water supply respectively.

In the chase, connections are available to the facilities cooling water supply and return header, the hot and warm drains, and to the off-gas system (one connection per pair of cubicles).
In front of the chase two vertical 25 cm diameter stepped pipes provide for connections to the basement.

The shield plug assembly for insertion in the beamtubes is aluminum lined and concrete filled. At the core side a boral disk, thickness 6 mm, is bolted to the plug end. It provides for thermal shielding of the plug concrete. The plug is of a stepped design. Its length is about 2.30 m and extends from the inner face of the concrete shield to the port box assembly. By means of a steel flange it is bolted to the flange of the beamtube.

The part of the plug with the smaller diameter is supported and kept concentric with the beamtube by means of a double roller assembly, 60° apart. The part with the larger diameter is supported by means of the above mentioned steel track.

The annulus between plug and tube is about 6 mm wide and provides sufficient clearance for the beamplug cooling water return. The beamplug is pierced by three helicoidally wound channels, 25 mm in diameter. They can be shut off at both ends by stainless steel plugs. One of these channels is used for the cooling water supply. This channel is extended by means of a 1.25 m, long aluminum tube. The cooling water therefore leaves the tube near the core face. (see figure 2).

It will be clear that in case a tube is used for an experiment, a different plug arrangement may be necessary.

In conclusion, it may be mentioned, that a storage facility for radioactive beamtube plugs is available in the wall of the gastight building. The facility is shielded with sand. There is room for thirteen tubes in total. The storage tubes are on the same elevation as the beamtubes. Each of them is connected to the off-gas system.

3.2. The large horizontal facility

3.2.1. General description

The large horizontal facility consists of a plug liner, a plate and a nose weldment. The plug liner is an aluminum cylinder, 183 cm long and 165 cm inner diameter. The cylinder is anchored in the 1.2 m thick biological shield and extends about 41 cm into the pool. The poolliner is welded to the cylinder. Coolant supply and return connections are provided on the bottom and top side respectively, both connections being in the concrete shield.

A 304 sst plate, 82 mm thick, is bolted to the plug liner flange at the pool side. A circular hole, concentric with the cylinder is used for mounting of the nose weldment.

The aluminum nose weldment is flanged to the plate. The nose weldment extends into the large circular tube, 63 cm in diameter, which is welded to the reactor vessel on the north side.
This tube is partly surrounded by a concentric sleeve. This sleeve has the same function and receives cooling

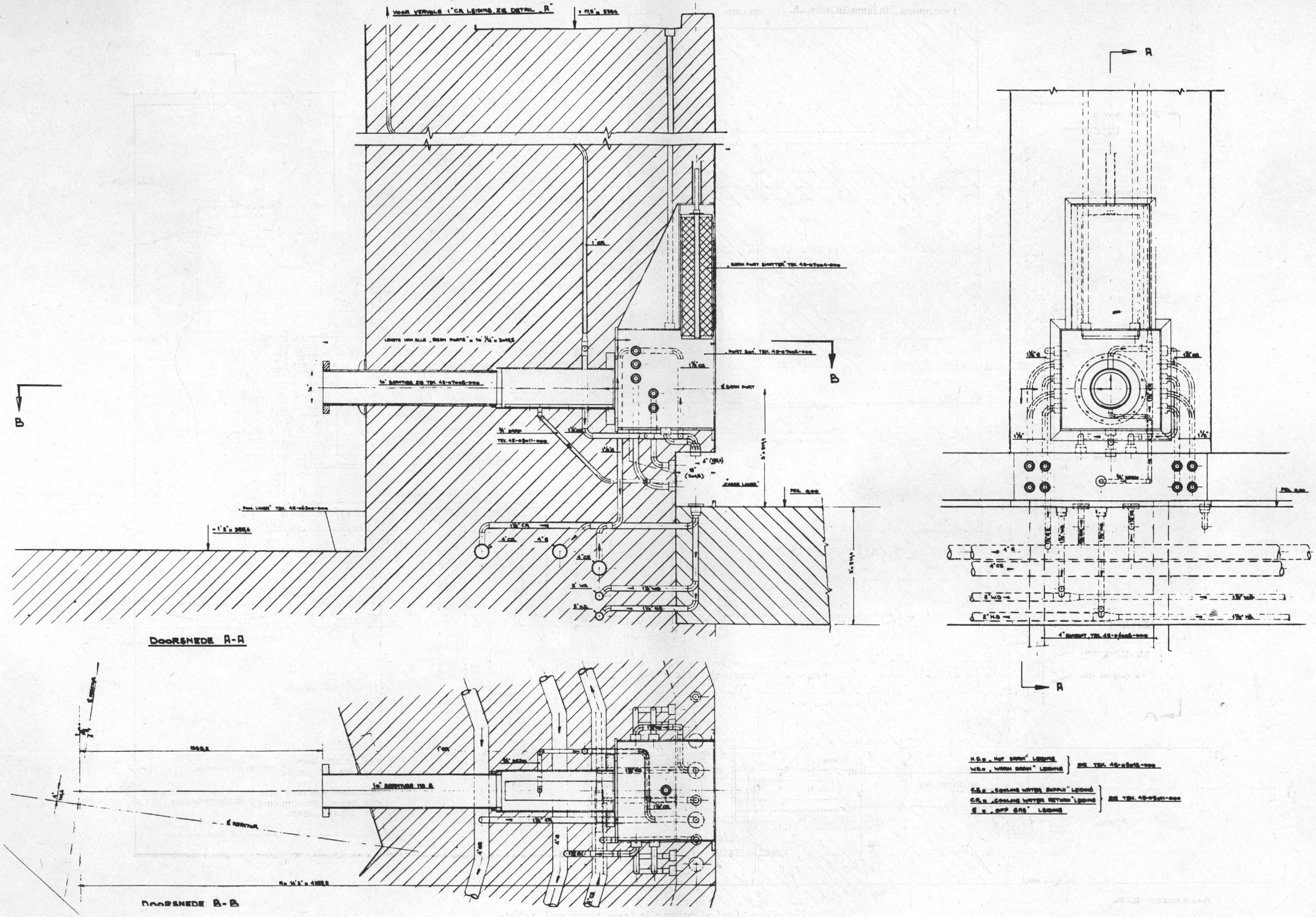


Fig. 10—Service connections for beam tubes and cubicles.

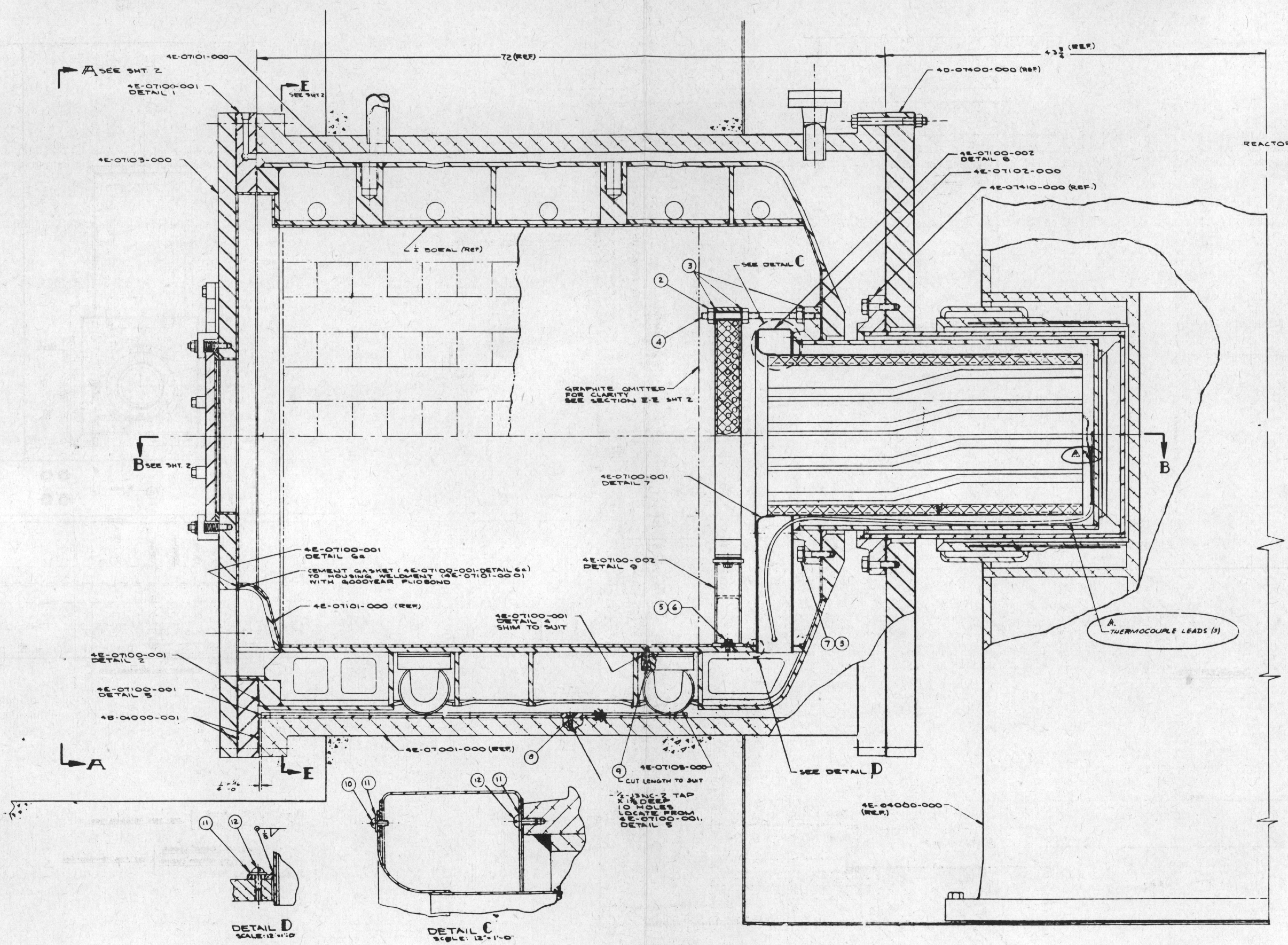


Fig. 11—Thermal column in large horizontal facility.

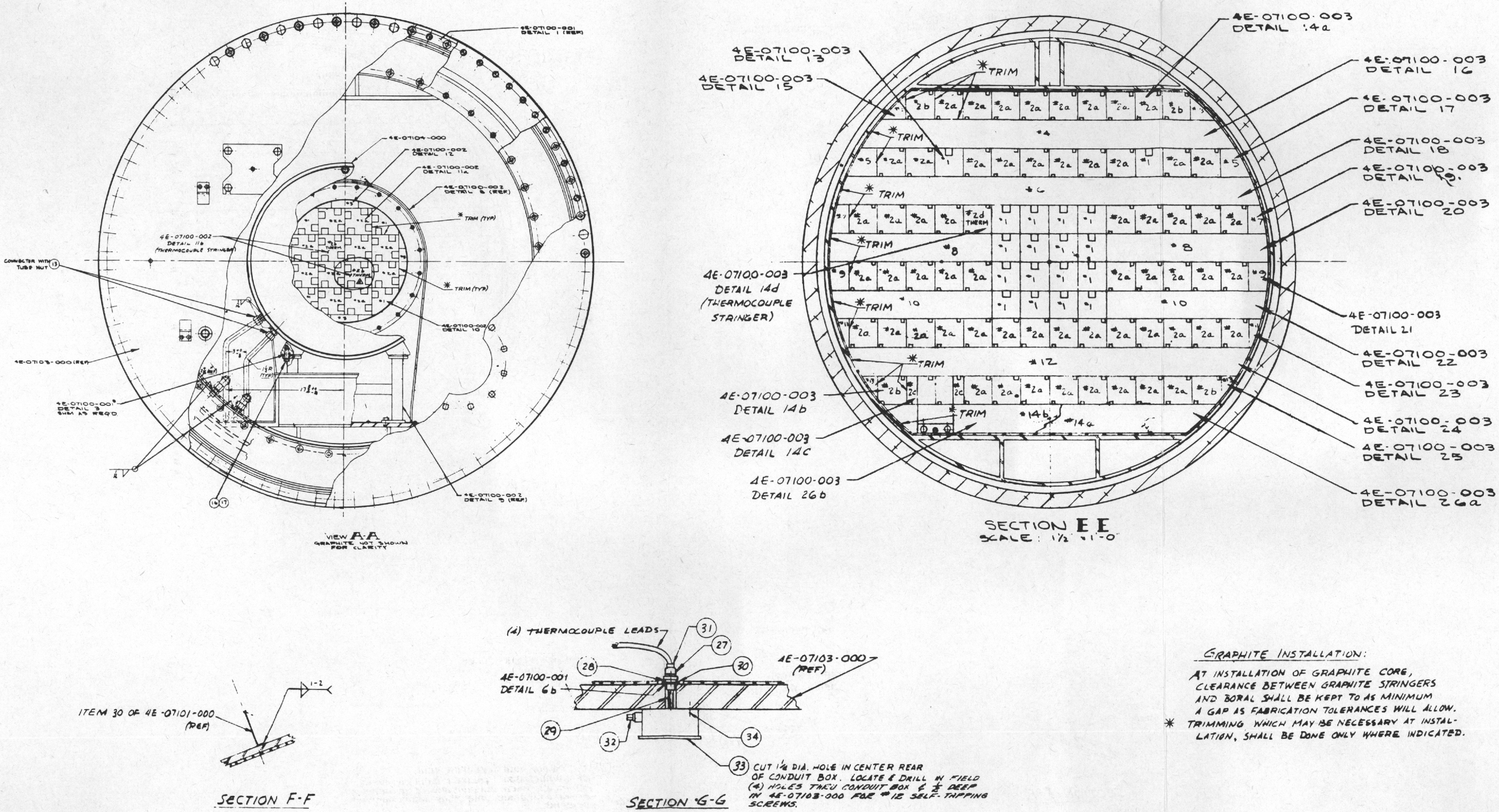


Fig. 11B—Thermal column in large horizontal facility.

water from the same header as in the case of the beamtubes. Figure 5, left-hand part gives an impression of this circular hole.

At the outer surface of the biological shield there is no port box assembly as compared to the beamtubes, but instead there is a recess in the biological shield, 90 cm deep, 2.45 m wide and 3.35 m high.

The service connections to the large facility are provided along the surface of this recess.
Two 5 cm diameter steel pipes connect the ceiling of the recess with the first balcony via two elbows.
Moreover, there are four vertical through-holes, steel pipes stepped from 20 to 15 cm; plugs can be inserted for shielding purposes. All six through-holes are sealed at the top end by gaskets and covers.
Three vertical stepped pipes, 25 cm in diameter, lead to the basement.
Inlet and outlet connections for pool water, facilities cooling water and CO_2 coolant are provided.
Finally, connections to the hot and warm drain, and off-gas system are available.
An impression of the service connections may be had from the flow diagram, given in figure 12.

In front of the large facility a door can be moved by means of caster assemblies, supported by a steel track. The door is partly filled with concrete, partly with lead blocks. In this door 3 holes are provided with plug inserts. They correspond to the access holes in the thermal column cover plate.

In front of the large facility a hatch for lowering plugs, experiments, etc. into the basement is available. Normally it will be closed by concrete blocks.

A schematic impression of the facility, including door and hatch, is given in figure 9.

The facility is especially meant for large engineering experiments. The facility line may also contain a shielding plug provided with holes, which can be used for smaller experiments. These holes, in turn, will be equipped with plugs similar to the ones for the beamtubes.

No engineering experiments for the large facility have been planned yet. In the beginning a removable thermal column will be installed. This installation will therefore be discussed in the next part of this chapter.

3.2.2. The thermal column

In figure 11 several cross-sections through the thermal column are shown.

Except for the cover plate the whole installation is a self-contained unit, that is supported on the bottom of the large facility liner by means of two caster assemblies, so that the facility is movable over a stainless steel track, which is bolted to the liner.

The thermal column unit consists of the housing including the support structure, the thermal lead shield, and the graphite blocks in the nose and body.

The housing is made of 304 stainless steel and lined with 6 mm boral sheet, to capture neutrons escaping from the column, except for the part extending into the nose. This part is made of 6061 aluminum alloy, and is bolted to the stainless steel shell.

The lead shield is mounted just behind the nose part. It consists of an aluminum alloy (6061) hollow disk, filled with lead. Its diameter is 72 cm and its effective lead thickness about 50 mm. Due to the high gamma absorption rate in the lead, cooling of the shield is necessary. Facilities cooling water is therefore conducted through spirally wound tubing, 12 mm o.d., embedded in the lead. The coolant supply and return lines run through the stainless steel seal ring, which is bolted between the column cover plate and the plug liner flange.

Both the column nose compartment and body or shell compartment are stacked with graphite blocks. The length of the graphite stacked sections is 92 and 122 cm respectively. The sides of most of the blocks are provided with channels serving as passages for the CO_2 coolant.

The left-hand views on figure 11 give an impression of the graphite stacking.

Eighteen of the graphite stringers, dimensions 10 x 10 x 122 cm, in the shell compartment are of a special design. They are provided with 14 holes each, diameter 3.8 cm, for insertion of irradiation capsules, maximum length 8.9 cm.
This type of stringer is shown in figure 13.
Sixteen of these stringers, in a 4 x 4 pattern, are symmetrical to the centreline of the column, and correspond to the large access hole in the cover plate. The remaining two correspond to two small access holes in that plate.

The stainless steel cover plate seals the CO_2 compartment from the outside atmosphere. Two holes in this plate, 25 cm in diameter, are for the inlet and outlet of the CO_2 coolant. Three square access holes are provided in the plate. They can be closed off by cover plates.

The graphite is cooled by CO_2 gas. The direction of the flow is from the cover plate to the nose. At the core side the flow is directed backwards via the annulus between graphite and aluminum housing. At the end of this annulus the flow is collected in a ring compartment, and from there directed to the outlet.

The outside CO_2-cooling circuit is shown on the flow diagram of figure 12. Two blowers are available for pumping the coolant around. In the return line from the column the gas passes a coarse and a fine filter. The CO_2 heat exchanger is cooled by water, taken from the pool cooling system. Venting to the off-gas line is possible. A CO_2 supply system is also provided for.

Pool cooling water, taken from the 7.5 cm line to the CO_2 heat exchanger is used for cooling the space between the column housing and facility liner, and the annulus between the graphite nose compartment and facility nose weldment.
The water is returned via a 5 cm line, ending in the reactor pool. (see figure 12).

The seal ring between cover plate and liner flange, that was already referred to above has the function to separate the water and CO_2 compartments, which are outside and inside the column housing respectively.

Temperatures at several crucial points in the column will be measured by means of five thermocouples.

3.3. The pool side facility

The pool side facility gives access from the pool to the west face of the core box. The vessel wall is interrupted at this point, to yield a compartment rectangular in shape and extending between the upper and lower bulkhead.

The compartment can also be reached from the top of the vessel through a flattened tube, 50 x 20 cm, that extends between the upper bulkhead and the vessel top cover; this is an open connection.

The facility can best be seen on figure 5, middle part. The flattened tube is also shown in figure 1.

A round tube, 20 cm in diameter, connects the facility to the basement via a hole in the bottom plug. The tube is welded to the lower bulkhead and extends to the lower end of the bottom plug in the sub-pile room. Sealing of the pool and reactor water is done at this point. When not in use the tube will be provided with a concrete filled plug. (see figure 5, middle part).

Vertical guides, one on each side of the facility can be used for mounting an experiment, or for installation of a window to separate the facility from the pool.

In the facility either a U-tube experiment or a single tube experiment, running from the top cover to the bottom plug, may be installed.
It is also possible to install an experiment directly from the pool into the compartment.

No reactor components or in-vessel facilities are obstructed by an experiment in this facility, thereby preserving one of the elegant features of the swimming pool type reactor.

3.4. The rabbit facilities

A schematic lay-out of the hydraulic and pneumatic rabbit systems is presented in figure 14.

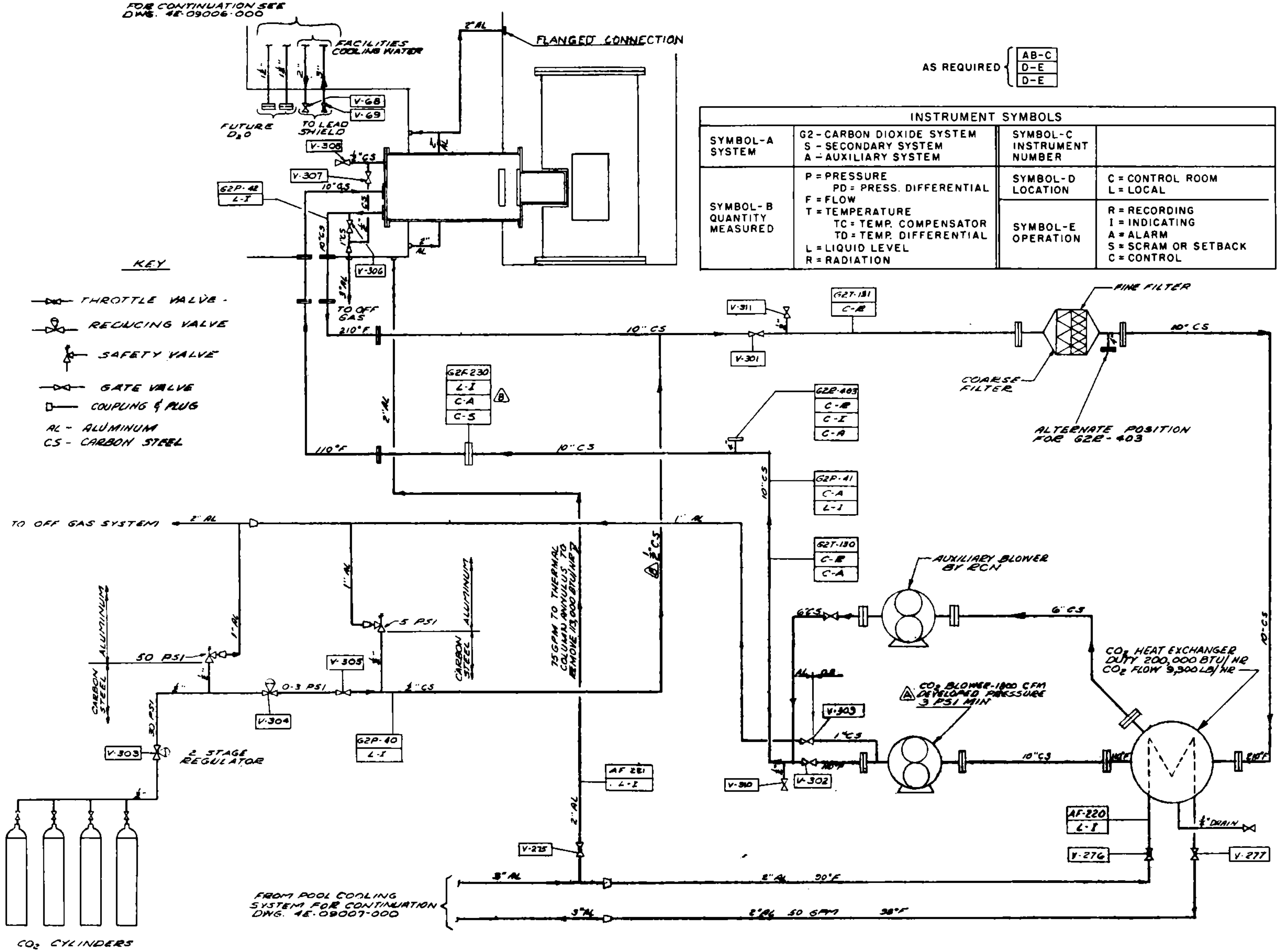
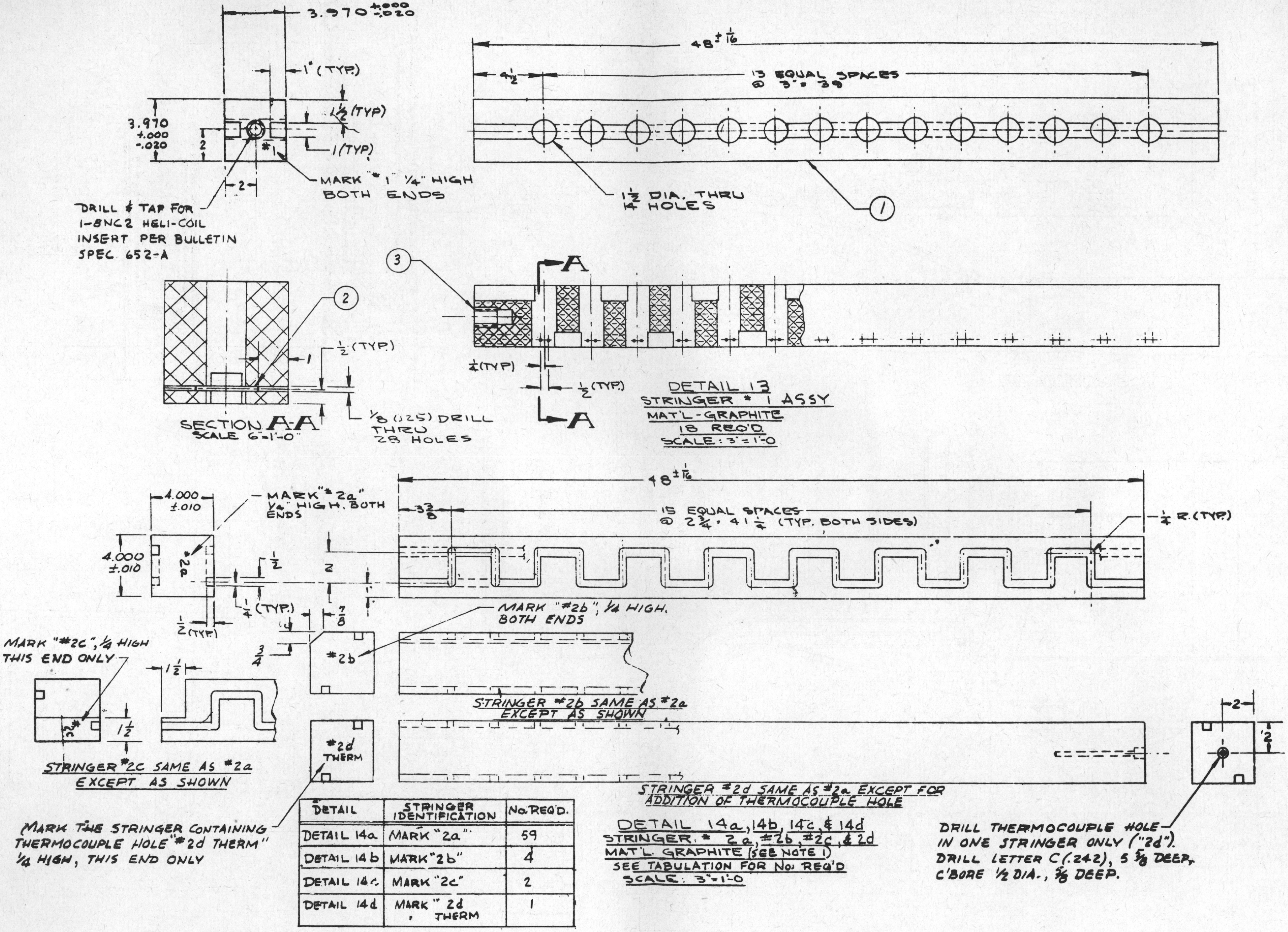


Fig. 12—Thermal column graphite cooling flow diagram.



DETAIL	STRINGER IDENTIFICATION	No. REQ'D.
DETAIL 14a	MARK "2a"	59
DETAIL 14b	MARK "2b"	4
DETAIL 14c	MARK "2c"	2
DETAIL 14d	MARK "2d THERM"	1

Fig. 13 — Thermal column graphite stringer for irradiation purposes.

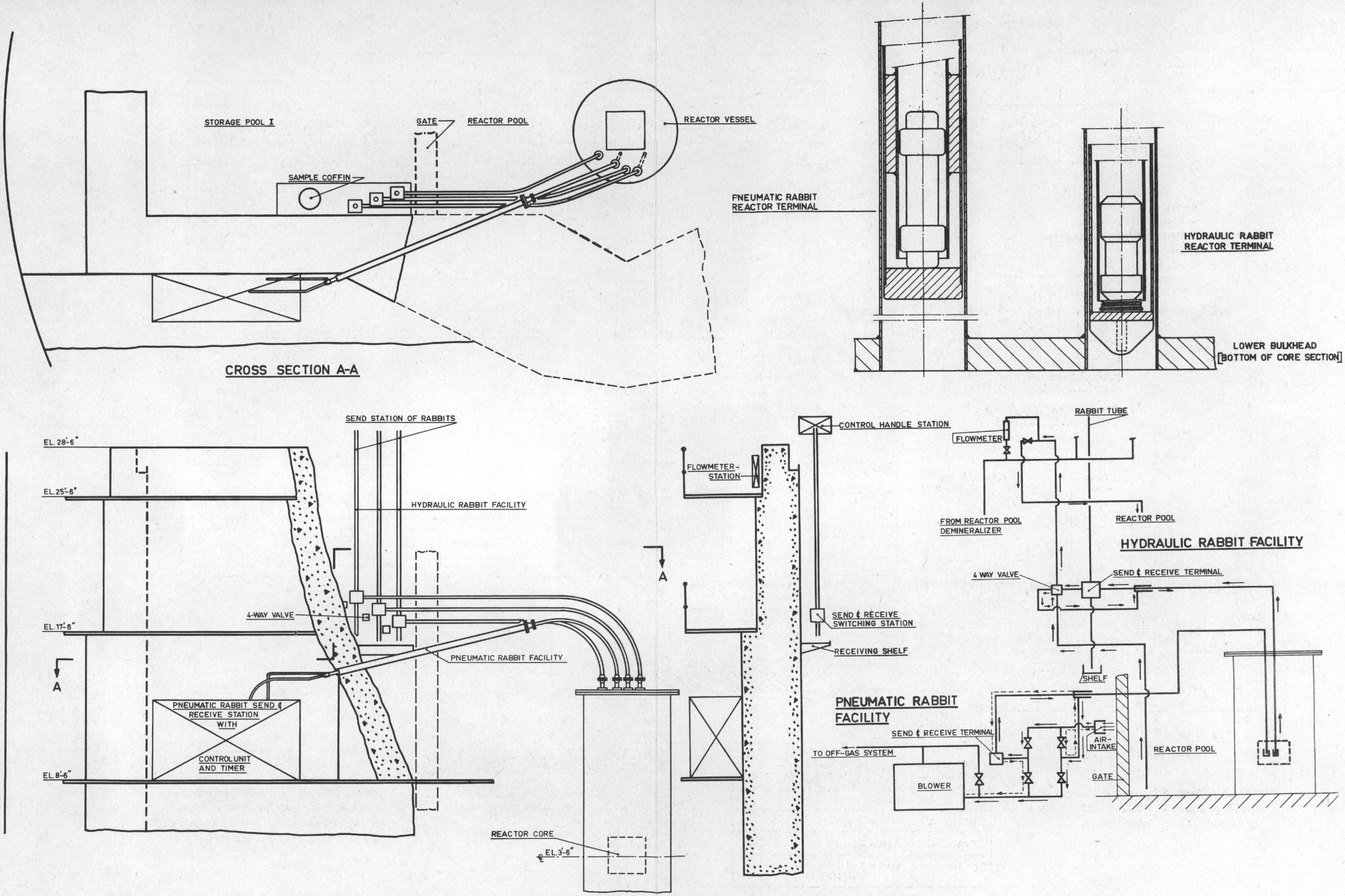


Fig. 14—Layout and flow diagram of pneumatic and hydraulic rabbit facilities.

3.4.1. The hydraulic rabbit systems.

Three hydraulic rabbit systems will be installed. They permit the irradiation of capsules in the high neutron flux field just outside the core box for periods, ranging from a few minutes upwards.

Each systems consists of a rabbit send tube, a send and receive station, a flow control station with associated piping, and the rabbit terminal in the reactor vessel.

The rabbits have an outside diameter of 2.5 cm and a length of 7.5 cm.

The concentric tubes, leading from the send and receive station to the reactor vessel, pass through the pool gate jam, are flanged to the vessel top cover and end in a 7.5 cm sleeve, extending between upper and lower bulkhead in the vessel. The location of these sleeves can be seen on the right-hand part of figure 5, second cross section from top.
The inner send tube has an I.D. of 3 cm, the O.D. of the outer tube is 5 cm.

A capsule is introduced into the rabbit tube, which rises above pool level, near the pool wall. It drops by gravity into the send and receive station, where it is brought in the right position for sending.

The capsule is driven into the reactor vessel by means of a water flow. This water is taken from behind the pool water demineralizers.
Metering and regulation of this flow is done on the second balcony.
The direction of the flow is taken care of by the flow control station, which is in fact a four-way valve. The control of both the send and receive station, and the four-way valve, is done from the top of the pool, by means of extended handles.
At the rabbit terminal in the reactor vessel, the capsule is stopped in its fall by means of a spring. After return of the capsule in the send and receive station, it can be brought to drop onto the receiving shelf, by way of an exit tube. This shelf provides enough space to support a container for the activated samples.

The operation of the hydraulic system may best be studied from the flow diagram given in figure 14.

It must be mentioned that the exit of the water exhaust pipe is near the bottom of the reactor pool.

When the system is not used, natural convection may be maintained in the system by opening a valve located in the by-pass of the water inlet line, that also ends into the reactor pool; this flow path is shown in the flow diagram on figure 14.

3.4.2. The pneumatic rabbit system.

One pneumatically operated rabbit facility will be installed. It is meant for irradiations near the core for periods, ranging from a few seconds upwards.

The period of irradiation can be controlled manually or by means of an automatic timer, operating in the range between two seconds and twenty minutes.
A micro-switch, located approximately 6 m from the terminal in the vessel, will indicate that the rabbit has left for the high flux region.

The driving force of the rabbit is derived from an underpressure in the direction of the movement, developed by a blower located on the intermediate balcony. The inlet air is taken from the building, the outlet of the blower is connected to the off-gas line.

The system will accommodate a rabbit with an outside diameter of 3.5 cm and a length of 11.3 cm; the rabbit contains the sample to be irradiated.

The installation of the concentric tubes (send tube I.D. = 3.6 cm, outer tube O.D. = 5.7 cm) is identical to those of the hydraulic system. (see 3.4.1.) The position of the rabbit sleeve in the vessel is shown on figure 5, right hand part, second cross section from top.

The operation of the system may best be seen on the flow diagram of figure 14.

When the system is not in operation a valve in the blower by-pass line to the off-gas system may be opened, thereby maintaining an air flow in the rabbit piping system.

It is the intention to enlarge the facility at a later date with a change station, from where the irradiated samples are pneumatically transported to the chemistry laboratory.

Acknowledgment:

Many of the drawings used to illustrate this write-up were capably prepared by Mr. W. Brieko on a short time notice.
His help is greatfully acknowledged.

CONTRIBUTION TO THE IMPROVEMENT OF PNEUMATIC SYSTEMS FOR THE TRANSPORT OF SAMPLES IRRADIATED IN NUCLEAR REACTORS

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INTRODUCTION

In the course of studying the experimental devices of a reactor, we have had occasion to compare the qualities of conventional pneumatic transport systems with those required in the elaboration of a system which is analogous but designed this time to send samples into or within the confines of the core.

A number of factors come into play in determining the adequate parameters, that is, those permitting a safe and rational use of the system. As these factors have a primary importance from the standpoint of the flexibility and the safety of the system itself as well as the standpoint of its effect on the control of the reactor, it was essential, therefore, to remove all the unknowns of the problem.

It was for these reasons as well as the inability of finding on the market any economical systems corresponding to the required operating characteristics, that the BelgoNucléaire was led to perfect a system which satisfied a maximum of the requirements in that field.

The studies and the executions of the prototypes could be carried out only through the cooperation of the firms of "Mercantile Marine Engineering and Graving Docks Company" and especially "Electro Navale et Industrielle" in Antwerp.

We should like here to thank the management and personnel of these companies for their precious advice and the efficient help which they were kind enough to give us.

PART ONE

GENERAL CHARACTERISTICS AND EFFECTS ON THE CONTROL OF A REACTOR

1.1. GENERAL REMARKS

Before beginning the technical description of the solution proposed, it appears important to us to summarize the general operating characteristics of a pneumatic transport system of samples to be irradiated in a nuclear reactor. On the whole, such a system should be designed to be able to send samples from a principal station, located near the reactor towards one or several secondary stations, situated within the confines of the core. The positions of these secondary stations are selected so as to respond to the chosen conditions of irradiation (predominance of rapid or thermal neutron flux, predominance of gamma rays, etc...).

It follows that, in principle, it is thus only the principal station that is accessible, the secondary stations being inside of the reactor's shielding.

One may state that the system chosen will correspond to the intended purposes if, on one hand, it allows easy and safe handling of the radioactive samples and, on the other hand, it does not interfere with the basic data set down at the time of establishing the criteria governing the control of the reactor.

As far as high-flux reactors are concerned, the problem appears to us all the more interesting as the radioactivity of the irradiated elements is generally very high and the precautions taken with respect to controls are very elaborate.

1.2. EFFECT OF A SAMPLE ON THE DYNAMIC BEHAVIOR OF THE REACTOR

It is clear that the samples sent into the irradiation stations may, according to their nature, represent either a poison for the core (this is the case of a neutron absorber), or, on the contrary, represent an excess of reactivity.

In this report we shall limit ourselves to the case of absorbent materials.

Under these circumstances, the effect of introducing samples in the core is the same as introducing a control rod; during extraction, on the contrary, the reactivity increases. Moreover, it is clear that the latter increases even more rapidly when the transport rabbit leaves faster from the zone in which the samples exercises a perceptible influence on the core. In addition, the latter influence being greatest at the beginning of the travel, one will understand the advantage of reducing as much as possible the initial speed of extraction of the rabbits.

If we examine the possible effect of a rapid transit on the operation of the reactor, the following events are likely to take place:

a) The rapid extraction of a sample causes a sudden increase in the reactivity which give rise to transients of very short period and have the effect of releasing the emergency shut-off of the reactor.

b) In certain cases, particularly when the resolution time of the safety counting channel is too long so that an emergency stop does not take place because the period is temporarily too short, or even when the regulating rod does not compensate the reactivity sufficiently rapidly, it may happen that the power rapidly reaches a value sufficient to release the emergency stop, but this time through the power counting channel.

In conclusion, we can state that the rapid extraction of a rabbit containing an absorbent element would have as a general effect the release of the emergency stopping devices in a manner which we may regard as improper. In order to remedy this situation the possibilities of the device described in this report have been explored as thoroughly as possible.

1.3. PRACTICAL CASE AND SIGNIFICANT FIGURES

In order to fix ideas and to give the order of magnitude of the effects of the speed of extraction of the rabbits on the reactivity left available and, therefore, on the kinetics of the system, we have analyzed the transient phenomena liable to take place under various experimental conditions.

Simplified equations of the kinetics have made it possible for us to trace the curves giving the variations in power corresponding to the insertion of two values of the reactivity, that is, excesses of 0.10 dollars and 0.40 dollars, respectively (Figs. 7 and 8).

For each one of the two values we have studied the growth of the reactor power for three determined modes of reactivity insertion, namely:

- instantaneous insertion (step function) (curve A);
- progressive insertion in 0.1 seconds (curve B);
- slow progressive insertion in 1 second (curve C).

One will immediately note the clear difference between the values of the initial periods as well as between the rates of increase in power with time. These curves enable us to meet the observations made in paragraphs 1.2 a and b above, and the conditions causing an improper emergency stop.

For each one of the Figs. 7 and 8, it will be seen that the period is greater when the reactivity increases slowly (that is, that the rabbit is extracted more slowly). It may therefore be deduced that as far as the difficulty mentioned first is concerned, a slow starting speed brings about the desired remedy.

As for the second difficulty mentioned, a comparison of the curves A, B, and C given in the different figures shows, on one hand, that the final power attained increases with the value of the reactivity introduced (compare the scales of the ordinates of the figures in question), on the other hand, that a certain value of the power is attained most rapidly when the rabbit is extracted rapidly. Moreover, a comparison of curves A, B, and C on the same drawing shows that the action of the regulating rods will have a greater chance of preventing the reactor from reaching the power limit fixed on the corresponding counter channel when the rate of increase of the power is small (that is, when the rabbit is withdrawn more slowly).

We may likewise deduce that the solution proposed brings an efficient remedy to the second difficulty.

REMARK

The extraction duration times chosen are actual values and correspond to the results of the measurements made on various models of secondary stations. These results, moreover, are given below in this report.

PART TWO

DESCRIPTION AND OPERATION OF THE PROPOSED SYSTEM

Bearing in mind the requirements of safety and simplicity of construction, we believe that the system which we are presenting offers a number of appreciable advantages and among them that of being extremely economical. We will try to describe it briefly, stressing particularly its difference from known systems.

2.1. GENERAL DESCRIPTION

The system consists, in principle, of several transport pipes inter-connecting the principal stations and the secondary stations, and inflow conduits providing the necessary flow of air. We have had the opportunity to apply the study of such a circuit on a pool type reactor and this installation appears in the diagram in Fig. 1.

This installation comprises in particular:

a) A Principal Station

This station is made up of a reinforced concrete platform in which the ends of the conveyor pipes terminate.

On the platform runs a roller track (the latter can, by the way, be circular (Fig. 4) or straight) on which moves a shielded container (Fig. 2) capable of stopping nuclear radiations. The air intake takes place through the carriage and a return intake conduit connecting the latter with a fan, after flowing through the main valve. The inflow conduit has the special feature of being in the form of a system of flexible piping which facilitates the displacement of the carriage and its positioning above any one of the pipe ends.

b) Secondary Stations

The latter are located near the reactor and consist of the lower end of the transport pipes. The end of each of the latter is arranged as follows:

- on one hand, the pipe is provided on its lower side with a valve which is normally closed under the action of gravity and which covers a series of openings;
- and, on the other hand, permanent holes pierced in the vertical wall directly connect the transport pipe with the dispatch inflow conduit surrounding it at the height of the secondary station (Fig. 5).

It will be noted that each of the dispatch inflow conduits is connected to the fan by means of a collector placed further off, making it possible to reduce the main valve to a single unit.

2.2 OPERATION OF SYSTEM AND PRINCIPLE OF THE PROCEDURE

The rabbit travels in the transport pipe alternately in one direction and in the other under the effect of the decreased pressure created in one or the other of the inflow conduits connecting the terminal stations to the fan.

The control of the unit is governed by the positions of a multi-channel, motorized, principal valve and the positions of slide-valves (Fig. 3) located in each of the transport pipes.

a) The Case of Dispatching a Rabbit into the Reactor (That is, from the main station to the secondary station)

The main operating valve is placed in a position so that the fan, once it is started, causes a reduction of pressure in the dispatch inflow conduit connecting the secondary stations to the said valve. In this way and by means of the short circuit openings, a reduction of pressure is also created in the transport pipes.

It is then sufficient to place a rabbit on the slide valve of a transport pipe and to open the latter for the cartridge to be sucked through the pipe and sent to the secondary station selected.

It should be noted that the fact that suction is created in all transit pipes at the same time presents no inconvenience because, in this way the rabbits already situated in the secondary stations are not subjected to the suction, as they are below the level of the short-circuit openings. Moreover, when the selected slide-valve is opened, the main flow of air takes place through the appropriate conduit.

b) The Case of Withdrawing a Rabbit From the Reactor (That is, from a secondary station to the main station)

After having brought the mobile carriage over the opening of the specific pipe, a sealing ring is lowered to ensure the tightness of the joint between the fixed piping and the mobile container (Fig. 2).

The main valve having been placed in the appropriate position and the fan having been started, the latter sucks in air from the piping through the carriage and the flexible pipe.

It is then sufficient to open the slide valve of the circuit involved for the flow of air to increase rapidly in the transport pipe selected. The valve of the corresponding secondary station then rises by the action of the suction created above it; this opens the lower openings of the transport pipe and creates an air current sufficient to start the transport rabbit.

The latter, as it is being displaced, leaves behind it successive rows of short-circuit openings and the effective air flow and the speed of the cartridge consequently increases gradually. Once it is beyond the last row of these holes, the rabbit rapidly acquires a speed sufficient to reach and pass through the no-return (flap) valve placed in the mobile carriage. At the end of the travel the rabbit is thus trapped between the end stop of the carriage and the no-return valve. It will therefore remain there when one closes the slide valve which started the process we have just described.

It is then easy to imagine the operations that follow; the carriage containing the irradiated rabbit is displaced on the panel and brought over a discharge pipe; at this moment it suffices to retract the no-return valve for the element to drop by gravity into a shielded transport container.

c) Dispatch Toward the Reactor of an Already Radioactive Sample

Let us remark, by the way, that an interesting alternative is immediately workable when it is a matter of redispatching into the reactor a cartridge containing a radioactive element.

It suffices to bring the lower transport carriage under the withdrawal conduit, make the necessary connections, and to place the upper mobile carriage in a withdrawal position. When the main valve is properly placed and the fan started, the air is sucked through the two carriages and the withdrawal conduit, thus bringing the rabbit in the upper carriage and beyond the no-return valve.

The upper carriage is then brought above the opening of the selected transport pipe and the redispatching towards the reactor takes place in the manner described in a), with the difference that the rabbit does not start when the slide valve is opened but only when the no-return valve is also retracted.

2.3 ADVANTAGES OF THE SYSTEM

- The withdrawal of a sample from a reactor is made in a gradual manner with all the advantages that this involves with respect to the control of the reactor.
- At any time, even in the case of complex operations, such as the re-injection of a radioactive rabbit, the operating personnel is protected from radiation.
- Thanks to the upper mobile carriage, no errors in withdrawal will lead to an accident because all rabbits withdrawn must necessarily pass through the carriage.
- The use of a single carriage allows only one sample to be withdrawn at a time; this is a supplementary advantage from the standpoint of the excess of available reactivity.
- With the collector placed in the inflow channel connecting the secondary stattons to the fan or the free air, it is possible, in addition to diminishing the number of pipes used, to reduce the control of the system to a single main valve.
- As there is only a single principal valve, it is impossible to carry out simultaneously the two operations of injection and withdrawal in two separate transport pipes. Indeed, the valves can be in only one of the two positions corresponding to a reduced pressure in only one of the inflow conduits (dispatch or return).

- A favorable effect of secondary importance is obtained in that a rabbit injected into the reactor creates as it reaches the secondary station a compression chamber between the closed lower valve and the lower end of the rabbit; this has the effect of slowing the arrival at the station and of damping shocks.

PART THREE

TESTS ON A PROTOTYPE AND RESULTS OF MEASUREMENTS

3.1. EXECUTION OF PROTOTYPE CIRCUIT

In view of the complexity of the problem and the numerous unknowns of a technological nature inevitably arising in the course of studying this type of problem, we decided to develop a pilot installation. For this purpose we built a principal station and a secondary station connected by an aluminum transport pipe, having an interior diameter of 40 mm and a total length of 8 m. The difference in level between the two stations is about 6.5 m, and the piping comprises 4 elbows of 90° and radial dimension of 1 m.

Moreover, the terminal stations were built of transparent material so as to be able to observe directly the operation of the various mechanisms. This arrangement also allows the departure and arrival of the rabbits to be filmed, and thus allows the speeds and transit times of the transfer rabbits to be determined exactly. The fan used developed a maximum static manometrical height equal to 700 mm of water. Adjustable valves located in the inflow conduits made it possible to carry out measurements over a large range of pressures.

3.2. POSSIBILITIES OF EXECUTION AND TECHNOLOGICAL CONCLUSIONS

In addition to the various curves deduced from the filmings which we shall discuss in the following paragraph, we arrived at the following conclusions:

- Possibility of achieving gradual departures and damped arrivals.
- Possibility of mechanical execution satisfactory from the standpoint of safety of operation.
- In particular, we met with no major difficulty in the performance of the mobile carriage and the slide valve. The ring serving as a sealing joint and its control mechanism, worked perfectly well while the no-return valve worked several hundred times without any hitch. Moreover, the handling of the slide valve proved to be convenient and accurate.

3.3. RESULTS OF MEASUREMENTS

The results of measurements on the prototype installation have been plotted as graphs:

a) Figure 9 shows the results for a pressure difference of 400 mm of water and rabbits of different weights when the short-circuit holes are obstructed. The distance covered by the element is given in the ordinate and the corresponding time in the abscissa. From this graph one may deduce that the speed of departure of the rabbit is about 130 cm per second and that the time required to travel 25 cm (that is, the distance required to leave the zone of influence of a given core) is of the order of 0.1 s.

b) Fig. 10 gives the results for the same pressure difference of 400 mm of water and rabbits of different weights when the entire system proposed by us is used. It will be seen that the speed is almost independent of the weight of the rabbit and that the time required for the latter to travel the initial 25 cm, measured in the secondary station, is here of the order of one second.

c) Fig. 11 gives the values of the total transit time taken by the rabbits to travel the distance separating the secondary station from the principal station, that is, 8 m. of piping.

One will notice the clear difference between the average initial speed and the average total speed of the rabbit.

PART FOUR

APPLICATION OF THE SYSTEM TO THE CASE OF A SPECIAL REACTOR

4.1 SERVICE CONDITIONS

As the Société BelgoNucléaire obtained the control of the reactor destined to equip the nuclear chemistry laboratories of the State University of Ghent, we had the opportunity to study in a more practical manner the system of pneumatic transport whose principle we have just discussed.

Below we shall emphasize particularly some of the essential characteristics imposed by the operator and to which our system has responded perfectly.

The operator wanted the reactor in question to constitute primarily a source of neutrons destined essentially for the production of radioisotopes of short and even very short half-life. The number of irradiation stations had to be very high while the quality of the available neutron flux had to cover a wide range as far as cadmium ratio values are concerned. Moreover, because the handling was to be done principally by students, it had to be both extremely simple and give the best possible guarantees of safety.

The reactor used is a pool type reactor having a nominal power of 15 kW. Its core consists of stainless steel pipes filled with uranium oxide and is surrounded by a thick graphite reflector with an aluminum shield so as to have available a high thermal flux in the largest possible volume.

We solved the problem of irradiation stations by providing for the installation a veritable battery of 16 sample transport pipes, the battery resting essentially on the principles which we have advanced, and whose secondary stations are located in the reflector of the reactor.

Fig. 6 shows an assembly view of the pool of the reactor and makes it possible to get an idea of the layout of the pneumatic pipes in the pool.

4.2. PROPOSED SYSTEM OF PNEUMATIC TRANSPORT

The transport pipes are of aluminum of 2 S quality, having an interior diameter of 40 mm. They run from a single starting point, a dismantling joint provided above the pool, to the secondary station. The secondary station itself consists of two concentric pipes; on one hand, an interior transport pipe with a lower valve and adequate short-circuit holes and, on the other hand, an exterior pipe forming the return intake conduit; the latter is closed by welding at its lower end and is connected to an air return pipe which leads up to the surface of the pool. The assembly was made by welding and was water-tight and, as the only dismantling joints are above the water level of the tank, the irradiations are effected in a strictly dry manner.

Let us note, by the way, that each one of the circuits can be dismantled separately and may be withdrawn in bulk from the pool without hindering the operation of the circuits remaining in place.

4.3 SUPPLEMENTARY MECHANICAL IMPROVEMENTS

Aside from the direct and known advantages resulting in part from the gradual start of the elements outside of the reflectors, and in part from the permanent shielding of the capsule until its arrival in the transport carriage located under the arrival platform, the system was provided with the following supplementary improvements:

a) Rapid electrical positioning of the mobile carriage above the outlet of the pre-selected transport pipe.

b) Luminous signs indicating the occupancy situation of the secondary stations.

c) Pilot lights indicating that an element has arrived at the mobile carriage.

d) Control of main valve by a push-button excluding all intermediate positions.

e) Automatic shut-off of fans.

f) Automatic control of certain circuits by incorporating "timer" circuits of predetermined function.

We shall mention here only the essential accessories of this installation, which, in our opinion, will contribute to make it an extremely practical educational tool.

We will also mention that the system is completed by mechanical and electrical interlocks preventing any improper operation and any handling which is not authorized by the foreman in charge of the reactor. For that purpose, the latter has available indicator signs necessary to supervise the proper operation of the system and switches to prevent any use not in conformity with the safety regulations.

CONCLUSIONS

We think we can conclude by stating that the system which we have studied and whose practical application has made it possible to solve numerous questions pertaining to the operation of an educational reactor, can be used in other reactors, for example, reactors used for the study of the behavior of materials under the effect of radiations.

In fact, the latter frequently put at the disposal of experimenters relatively high neutron fluxes and the samples subjected to irradiation generally show high activities. It is advisable to keep the latter under constant and adequate shielding.

Moreover, the control of such reactors is not an easy matter and the violent withdrawal of a strongly absorbent sample may give rise to disagreeable perturbations.

It is for these two main reasons that we have perfected the device which we have just discussed.

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10 September 1959

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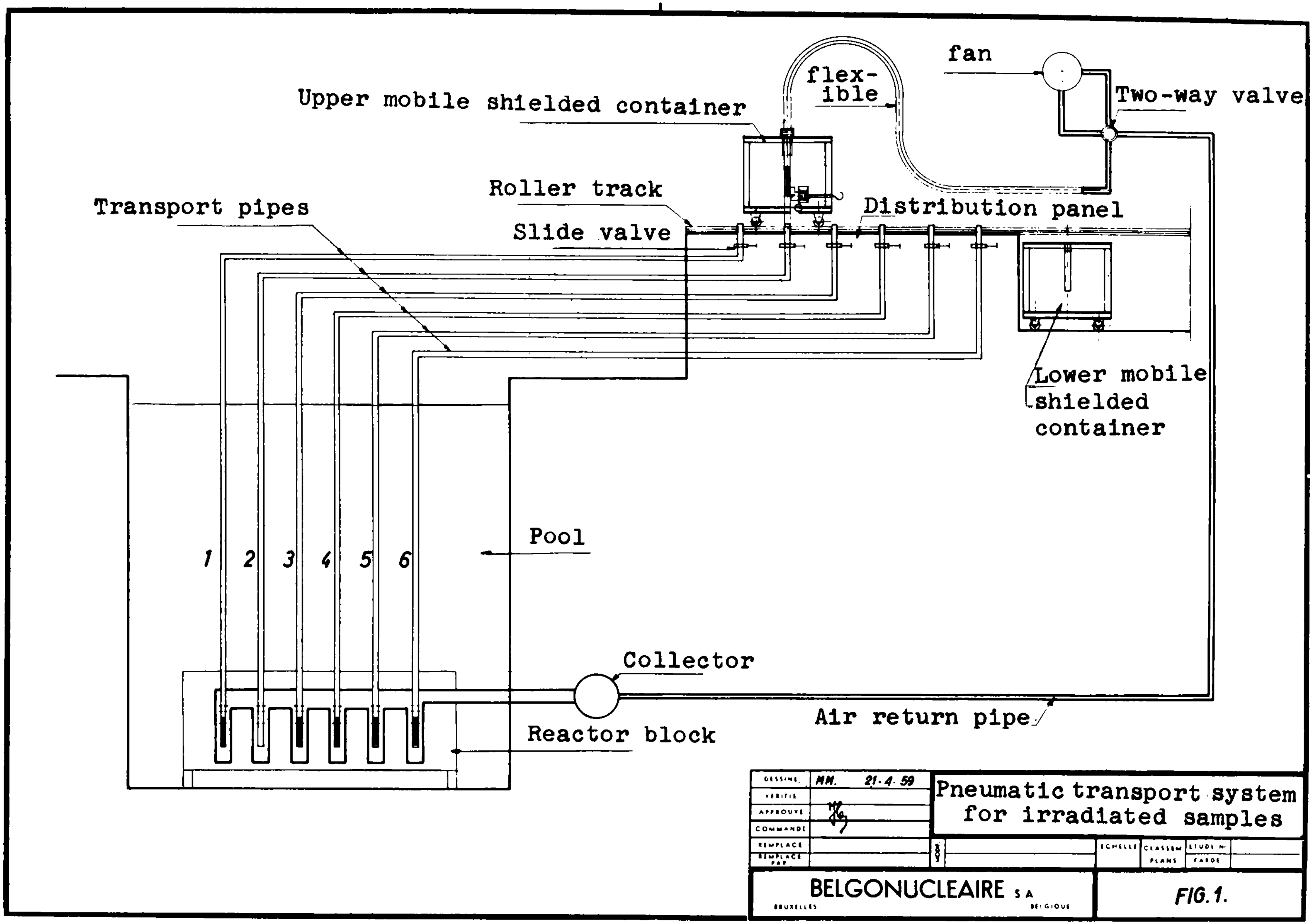
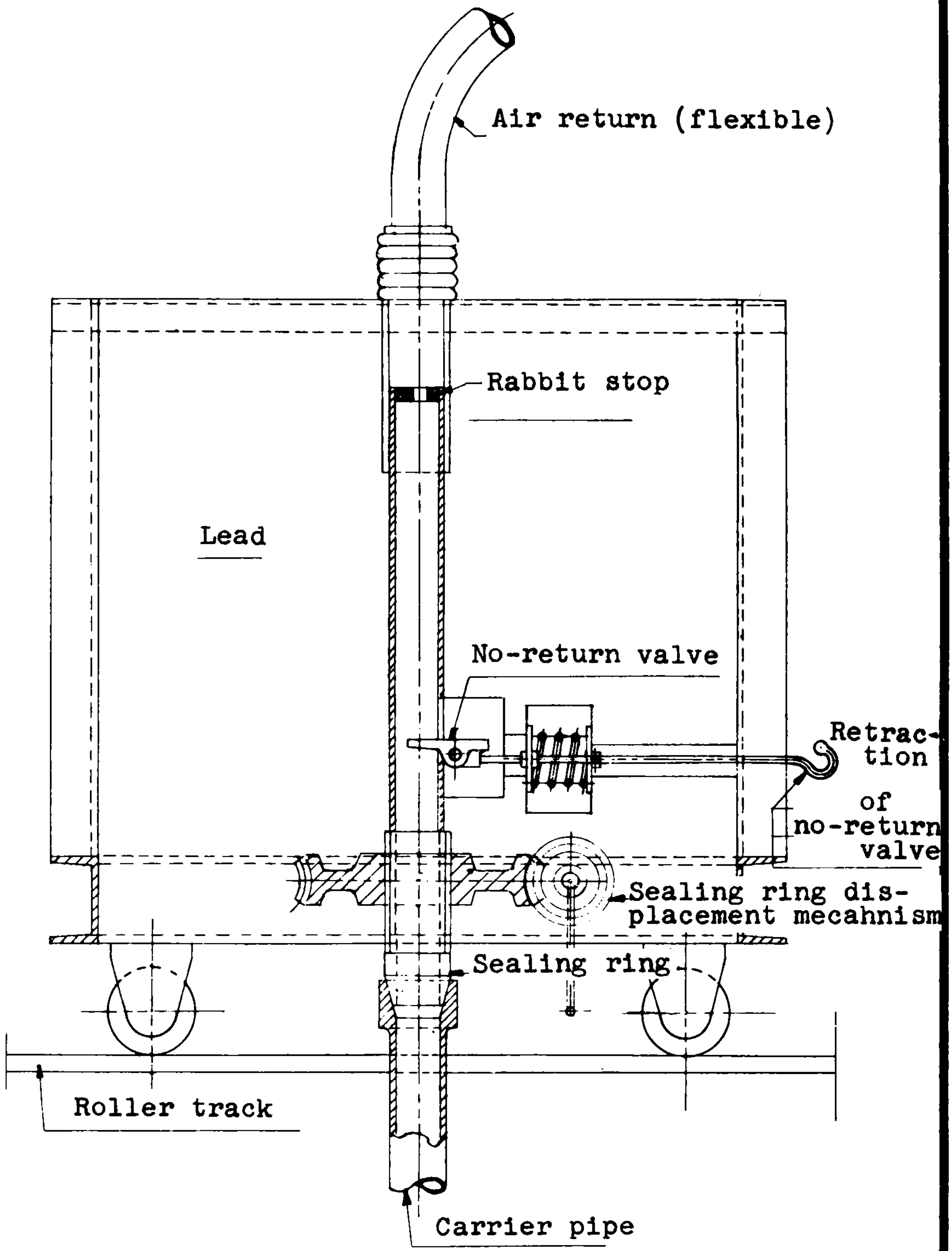

fan
flex-
ible
Upper mobile shielded container
Two-way valve
Roller track
Transport pipes
Distribution panel
Slide valve
Lower mobile
shielded
container
Pool
1
2
3
4
5
6
Collector
Air return pipe
Reactor block
DESSINE
MM.
21-4-59
VERIFIE
APPROUVE
COMMANDE
REMPLACE
REMPLACE PAR
Pneumatic transport system
for irradiated samples
ECHELLE
CLASSEM
PLANS
ETUDE N°
FARDE
BELGONUCLEAIRE S.A.
BRUXELLES
BELGIQUE

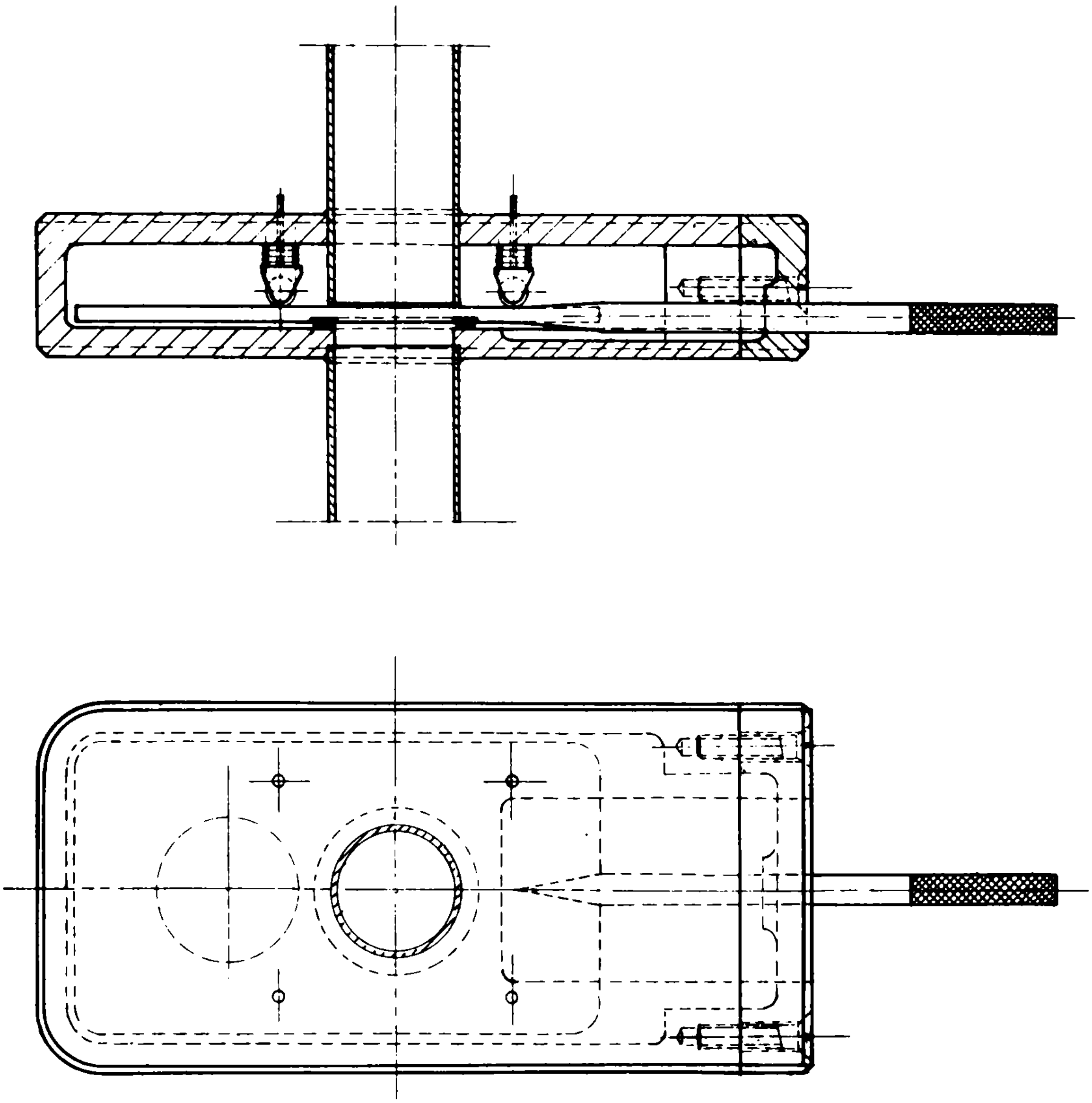
FIG. 1.



DESSINE	M.M. le 20.4.59	Mobile shielded container	
VERIFIE			
APPROUVE			
COMMANDE			
REMPLACE		MODIF.	ECHELLE / CLASSEM. PLANS / ETUDE n° 107 / FARDE
REMPLACE PAR			

BELGONUCLEAIRE S.A.
BRUXELLES BELGIQUE

FIG. 2.



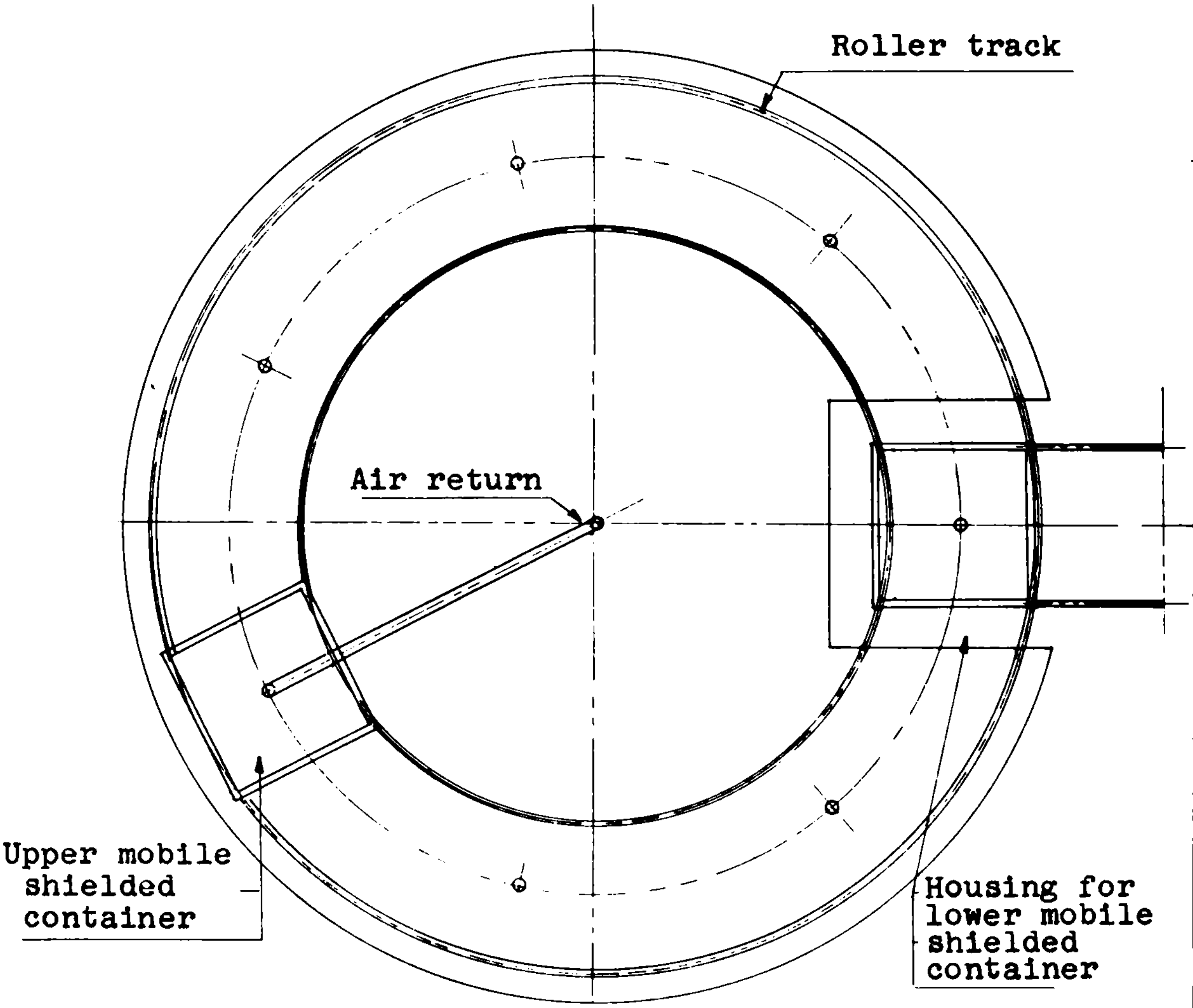
DESSINE	21. 4. 59
VERIFIE	
APPROUVE	
COMMANDE	
REMPLACE	
REMPLACE PAR	

Slide valve

MODIF.	ECHELLE	CLASSEM. PLANS	ETUDE N°	107
			FARDE	

BELGONUCLEAIRE S.A.
BRUXELLES BELGIQUE

FIG. 3.

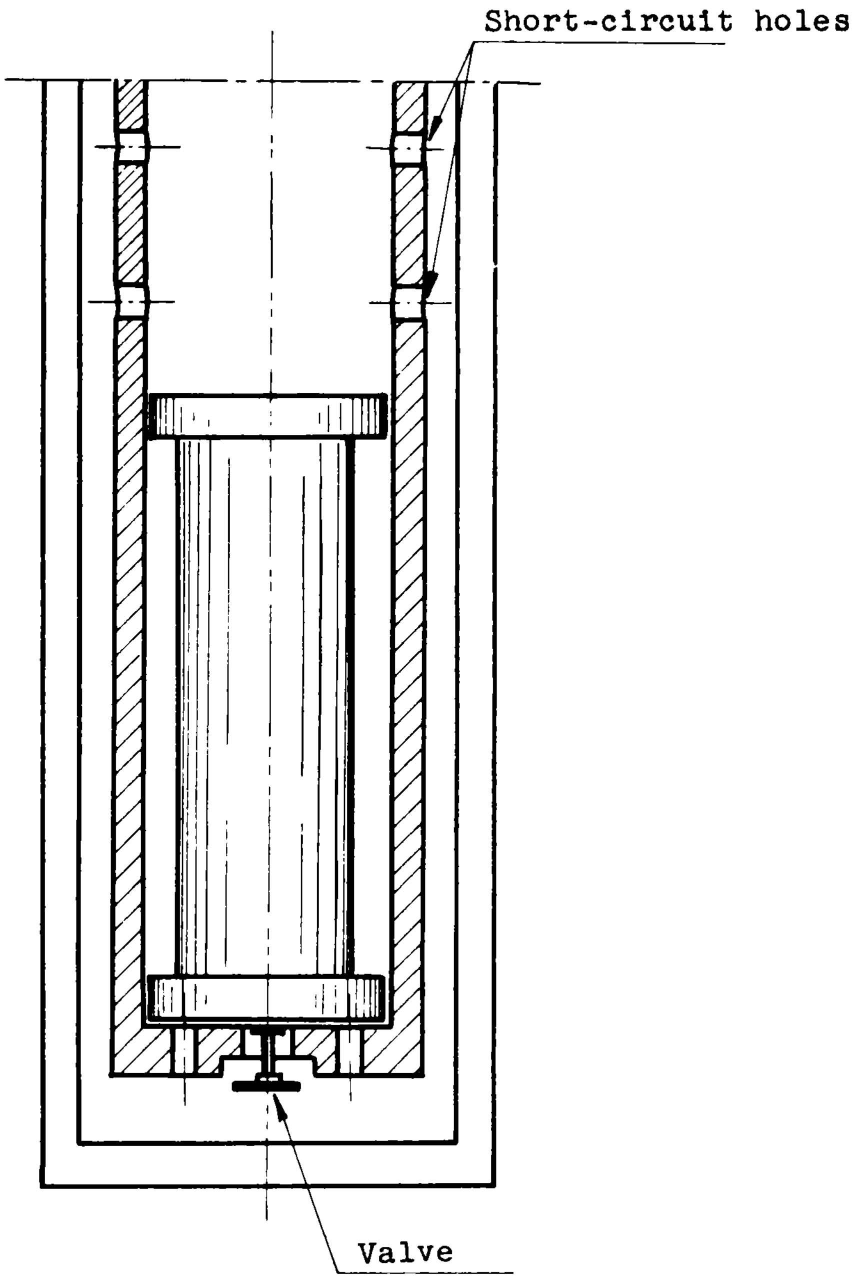


Plan view

DESSINE	M.M. 21.4.59	Distribution panel Case of a circular roller track			
VERIFIE					
APPROUVE	[signature]				
COMMANDE					
REMPLACE		MODIF	ECHELLE	CLASSEM PLANS	ETUDE N° FARDE
REMPLACE PAR					

BELGONUCLEAIRE S A
BRUXELLES BELGIQUE

FIG. 4.

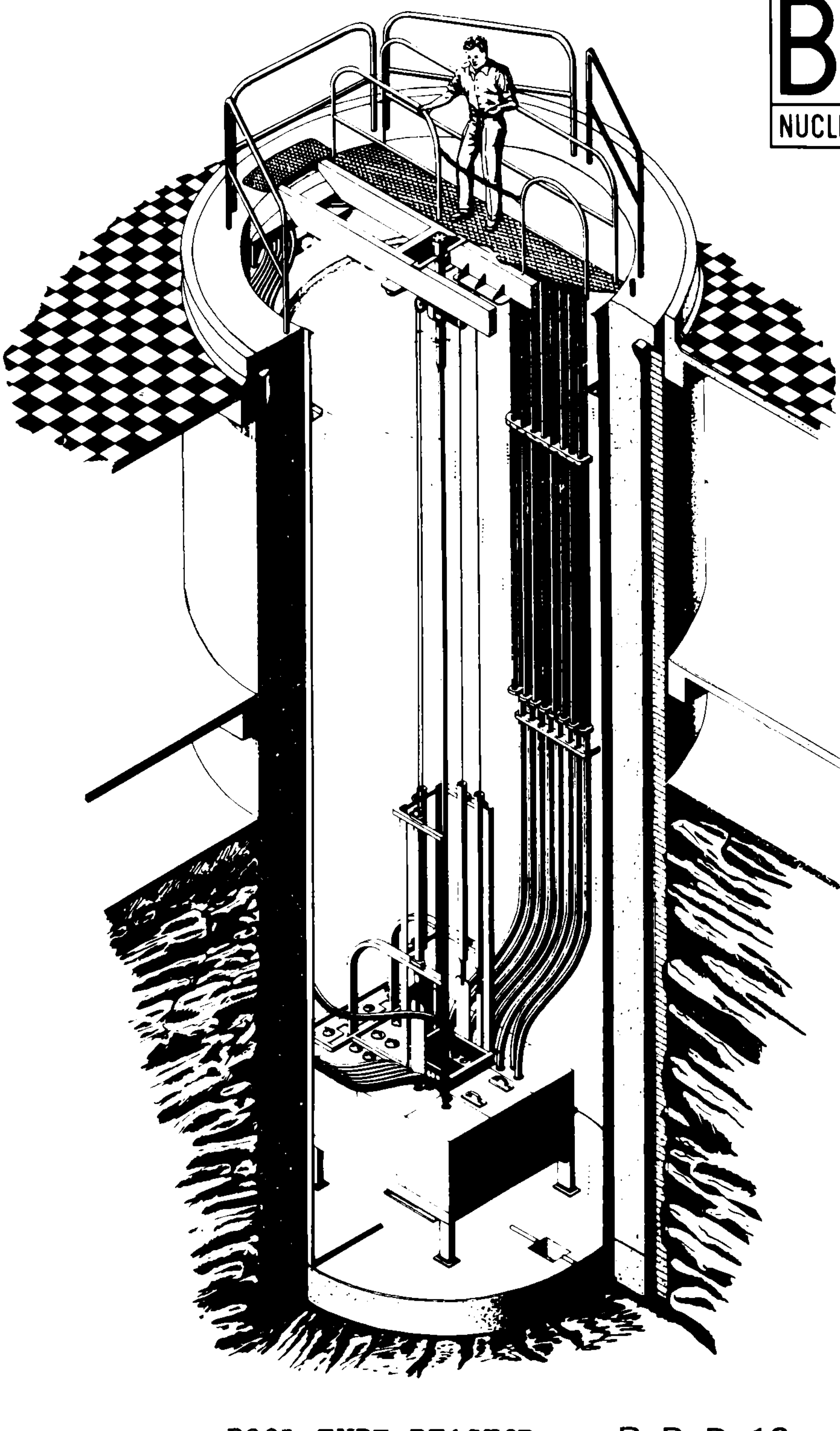


DESSINE		Damper and gradual departure of element to irradiation position
VERIFIE		
APPROUVE		
COMMANDE		
REMPLACE	MODIF.	ECHELLE / CLASSEM. PLANS / ETUDE N° 107 / FARDE
REMPLACE PAR		

BELGONUCLEAIRE S.A.
BRUXELLES BELGIQUE

FIG. 5.

BELGO BN NUCLÉAIRE



POOL TYPE REACTOR R.P.D.10

FIG. 6.

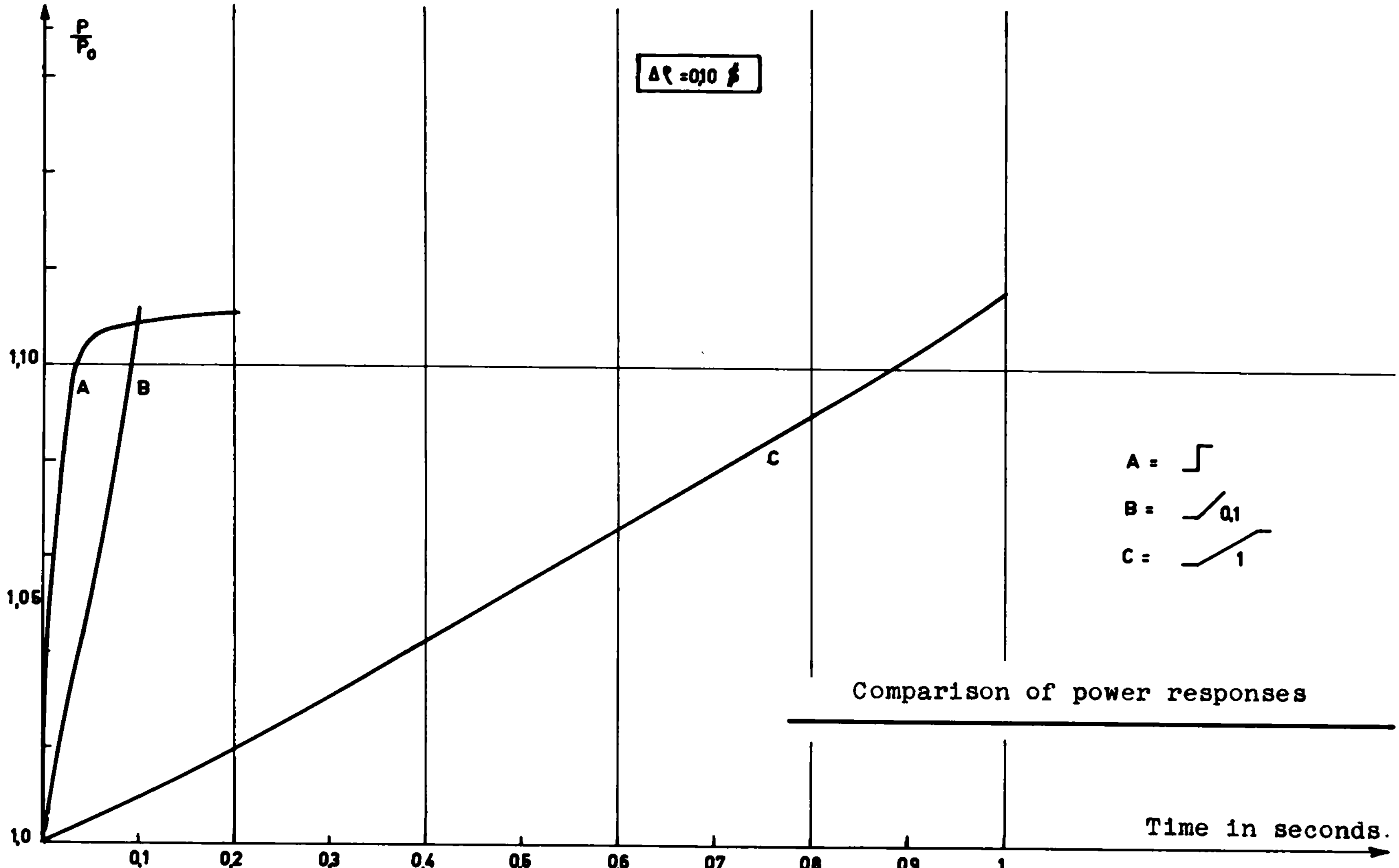

P/Po
Δρ = 0,10 $
1,10
1,05
1,0
A
B
C
A =
B = 0,1
C = 1
Comparison of power responses
Time in seconds.
0,1
0,2
0,3
0,4
0,5
0,6
0,7
0,8
0,9
1

FIG. 7.

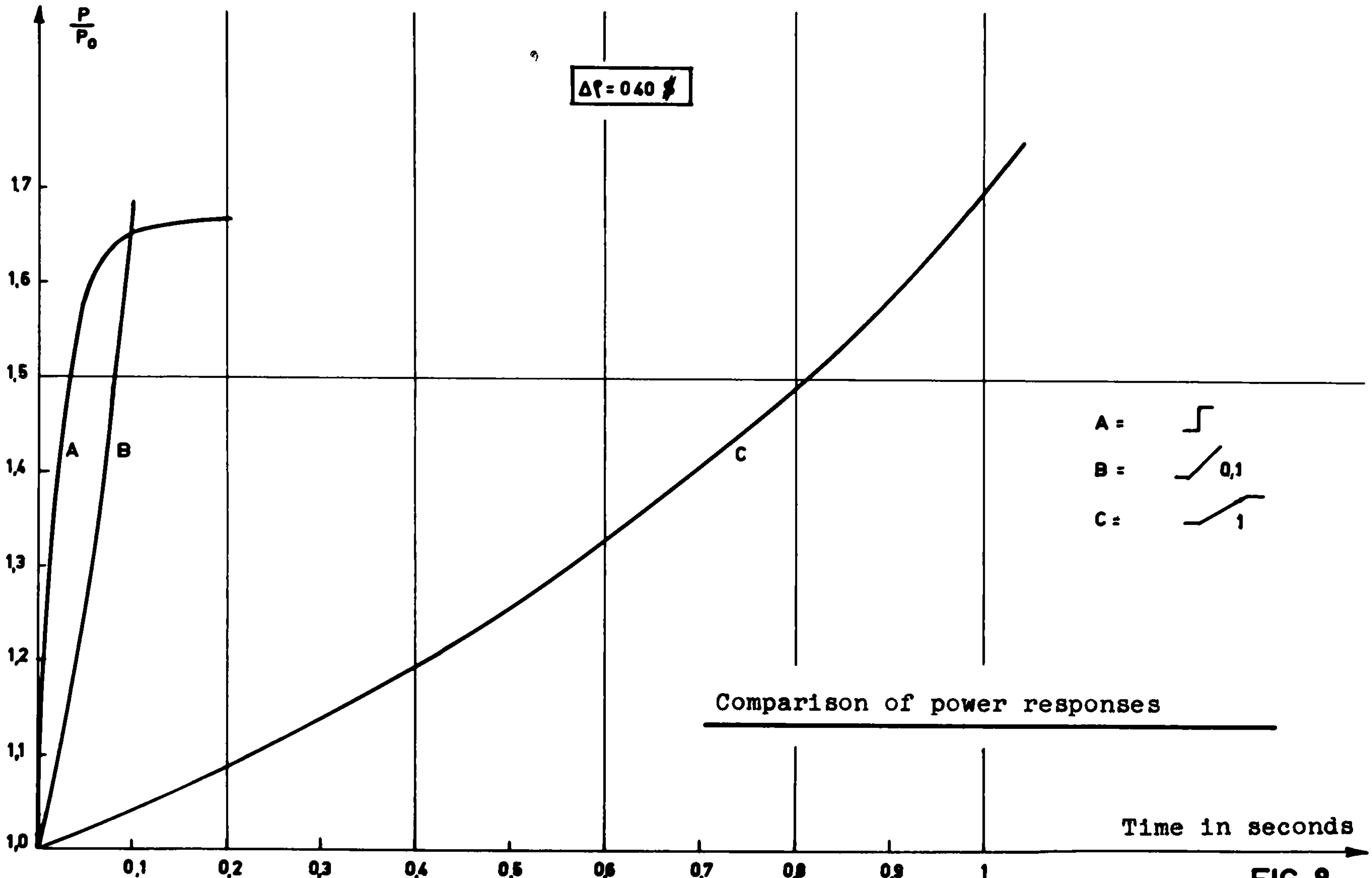

P/Po
Δρ = 040 $
1,7
1,6
1,5
1,4
1,3
1,2
1,1
1,0
A
B
C
A =
B = 0,1
C = 1
Comparison of power responses
Time in seconds
0,1
0,2
0,3
0,4
0,5
0,6
0,7
0,8
0,9
1

FIG. 8.

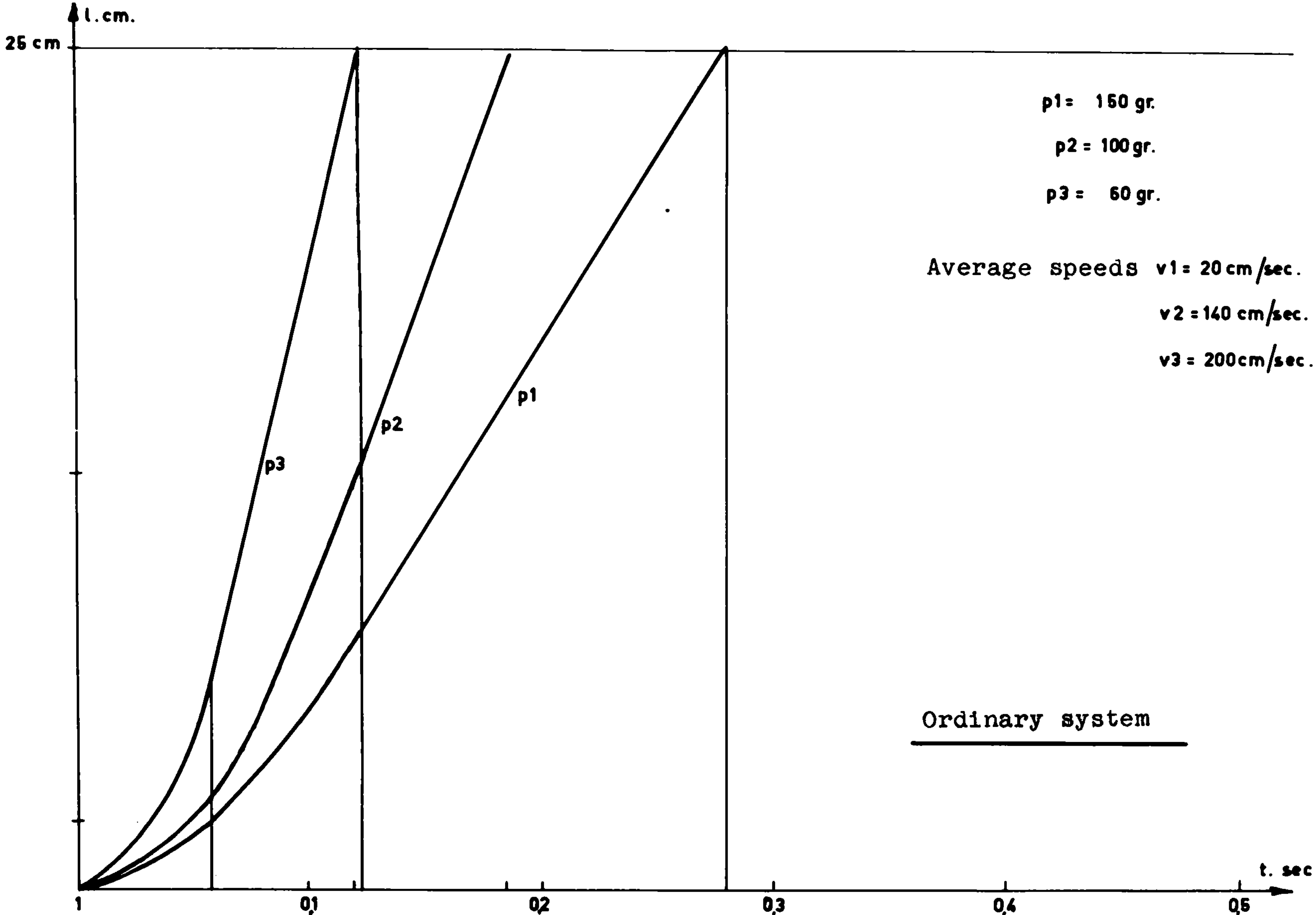

l.cm.
25 cm
p1= 150 gr.
p2 = 100 gr.
p3 = 60 gr.
Average speeds v1 = 20 cm/sec.
v2 = 140 cm/sec.
v3 = 200 cm/sec.
p1
p2
p3
Ordinary system
t. sec.
1
0,1
0,2
0,3
0,4
0,5

FIG. 9.

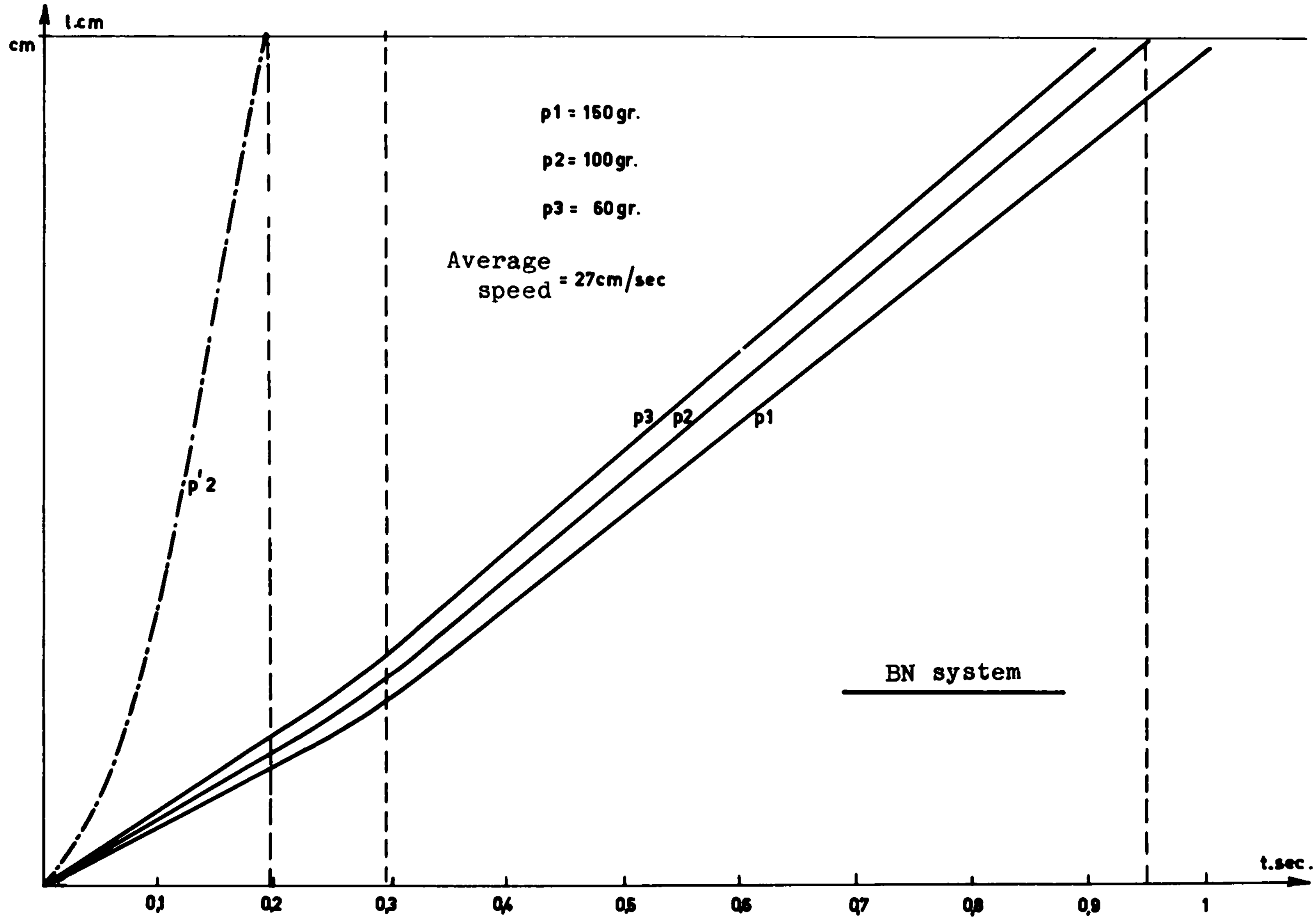

l.cm
cm
p1 = 150 gr.
p2 = 100 gr.
p3 = 60 gr.
Average speed = 27cm/sec
p3
p2
p1
p'2
BN system
t.sec.
0,1
0,2
0,3
0,4
0,5
0,6
0,7
0,8
0,9
1

FIG. 10.

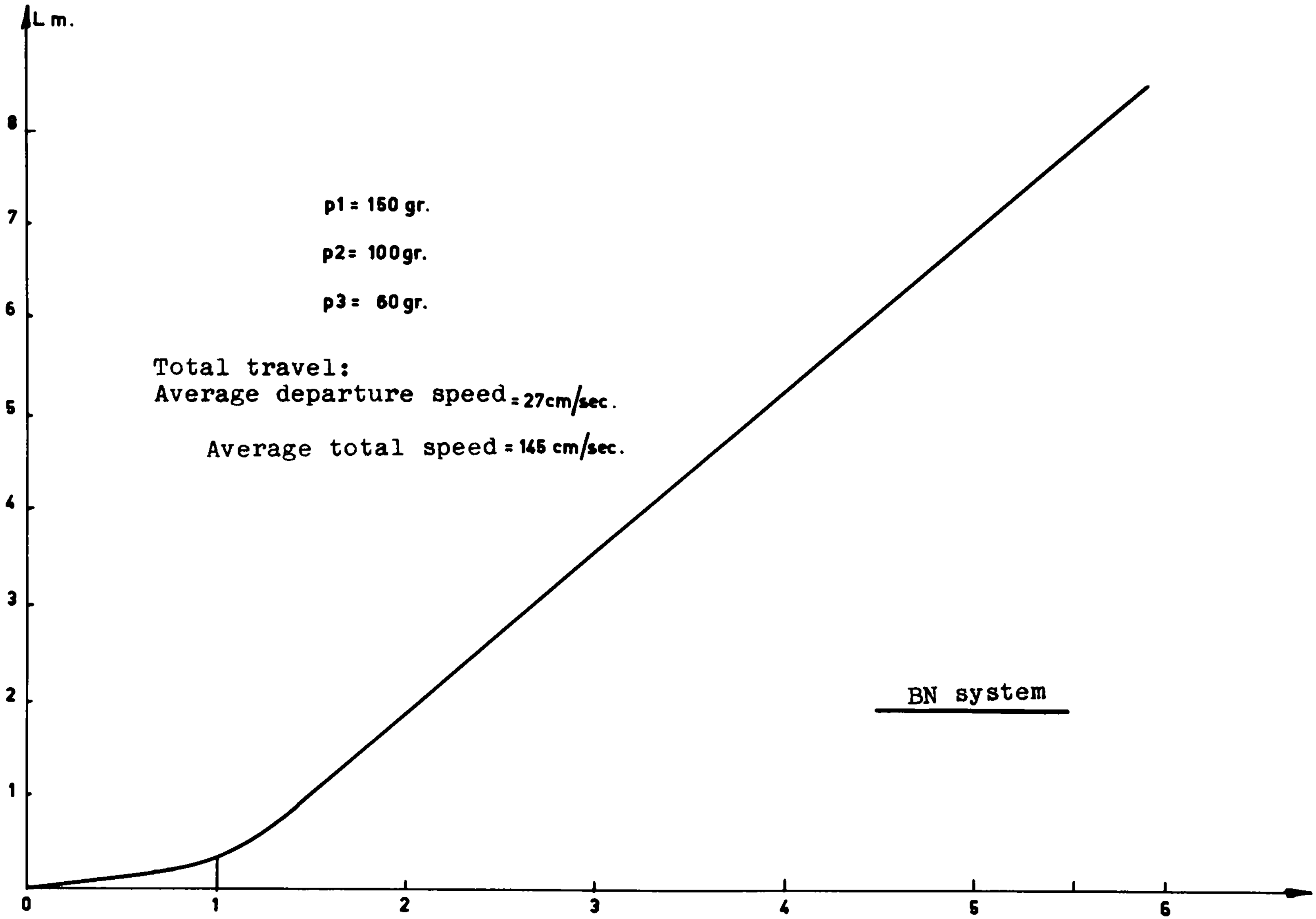

L m.
8
7
6
5
4
3
2
1
0
1
2
3
4
5
6
p1 = 150 gr.
p2= 100gr.
p3 = 60 gr.
Total travel:
Average departure speed = 27cm/sec.
Average total speed = 145 cm/sec.
BN system

FIG. 11.

DESIGN OF LOOPS FOR HIGH FLUX REACTORS

D. C. KAULITZ
Hanford Atomic Products Operation, Richland, Wash.

ABSTRACT

Both single pass and recirculating low and high pressure systems have been designed and operated in high flux reactors. A large number of experimental irradiations have been successfully carried out in these facilities.

Successful operation of these loops, with the careful design of individual experiments, has provided vital information in fuel element service life and has permitted evaluation of parameters critical in fuel element design.

* * * *

The most meaningful fuel testing method has been found to be the actual irradiation of specimens under controlled or measured conditions. It has not been found possible, in ex-reactor testing, to duplicate all of the conditions imposed upon a fuel element in service. In the design of loops for high flux test reactors it is important that the loops are constructed so one can carry out failure mechanism studies under real or simulated service conditions. In the associated research and development programs aimed at design improvements which will result in improved service life, a similar need exists for equipment in which the testing of new designs may be carried out.

In discussing the design of loops for a fuel evaluation program we will discuss some of the testing facilities we have constructed in the Materials Testing Reactor and the Engineering Test Reactor at Arco, Idaho. These reactors were constructed for engineering test programs and Hanford uses these reactors to study new fuel concepts. The fuel specimens can be irradiated to study failure mechanisms and can be operated at high specific powers, fluxes and temperatures so that operational difficulties at higher powers can be investigated before an increased operating limit is established.

A brief description of the Materials Testing Reactor and Engineering Test Reactor is necessary in order to discuss the loops we have designed and operated in high-flux reactors. Both of the test reactors are high-flux hetrogeneous enriched-fuel reactors. The active part of the reactors consists of closely packed vertical-plate assemblies, the individual plates being made of aluminum clad uranium-aluminum alloy. The plates are spaced to allow water flow between them, the water serving as both coolant and moderator.

Immediately surrounding the small enriched lattice is a water cooled primary reflector of beryllium and aluminum. This whole assembly of active lattice and reflector is mounted in a tank system through which the water flows and which contains the control rods and their bearings. Outside the tank system are a secondary reflector of graphite, a thermal shield, and a biological shield, the whole forming an approximate cube of about 34 feet to a side.

The first Hanford off-site facility constructed was a single pass system designed for use in the Materials Testing Reactor. The first unit was designed to handle the testing program at Hanford to evaluate fuels for low temperature, water cooled reactors. The unit circulates pressurized, demineralized water in a one-pass system in the reactor. The single pass system was designed to satisfy three criteria:

1. Sufficient pressurization to suppress boiling on fuel surfaces.
2. A closed system which could contain the fission products if a rupture did occur and thus prevent contamination of the normal MTR cooling water.
3. Accurate control of flow and other variables.

There are several features of fuel element testing that are not found in other industrial testing operations which introduce unique complications. The specimens become highly radioactive as a result of irradiation and must be handled remotely to avoid injury to personnel. When a failure does occur, the resulting spread of contamination can constitute a serious safety problem and be expensive to clean up. Further, the level of understanding of the processes occurring in fuel elements during irradiation is not, at present, high. One of the most important parts of a testing procedure has been found to be the careful examination of fuel specimens after irradiation. However, useful information cannot usually be obtained from failed or damaged specimens that have undergone excessive corrosion after failure or that have been mechanically damaged in handling operations or during forcible discharge from fuel element channels in which they may have become stuck.

The in-reactor portion of the loop in the MTR was designed to: (1) permit easy removal of deformed or failed specimens; (2) permit the recharging of irradiated (and hence highly radioactive) specimens; (3) be quickly and simply connected and disconnected; (4) fit in a cross-sectional area approximately three inches by three inches; (5) not leak under pressure; (6) be relatively inexpensive to fabricate and maintain.

These design criteria are required by the working conditions in the reactor tank and by the demand for fuel measurements truly representative of the element in the "as-irradiated" condition. The first condition is important in a fuel element development program because of the wealth of information that is potentially available from the careful examination of failed specimens. Damage caused by handling could compromise the results of the post-irradiation examination. The ability to recharge irradiated specimens is an invaluable feature. Progressive changes in dimensions, appearance, and properties which can be determined non-destructively could be followed by removing the specimens from the irradiation facility, making the examination and measurements, and then recharging the specimens for further irradiation. The need for a system that can be operated quickly and easily arises because handling conditions in the MTR Tank must be done either by remotely operated tools or by an operator working in a radiation zone. The size restriction is necessary to avoid interference with adjacent experimental assemblies. The system must not leak; if a specimen should fail the radioactive contamination must be kept from reaching the reactor process water which surrounds the assembly during operation.

It was found that the following arrangement satisfied the requirements listed above. The specimens are contained in a thin-walled tube (or "basket" in MTR terminology) which has been ribbed at the lower end as shown in Figure 1. The block that fits in the MTR reflector has one large diameter hole drilled in it and a length of heavy walled pipe is inserted in the block as shown in Figure 2. The basket is inserted in the pipe assembly with lugs on the bottom of the basket which serves to center the lower end of the basket in the pressure pipe. The upper assembly is coupled to the pipe assembly with a large "quick-disconnect" as shown in Figure 3. The top of the basket, which is fitted with a stiffner ring, can be seen protruding from the pipe assembly. Water enters and leaves the upper assembly in parallel streams through two commercial quick-disconnect couplers. Inside the upper assembly as shown in Figure 3, the inlet water tube is bent over the axis of the assembly just below the point of entry into the upper assembly. The tube extends down

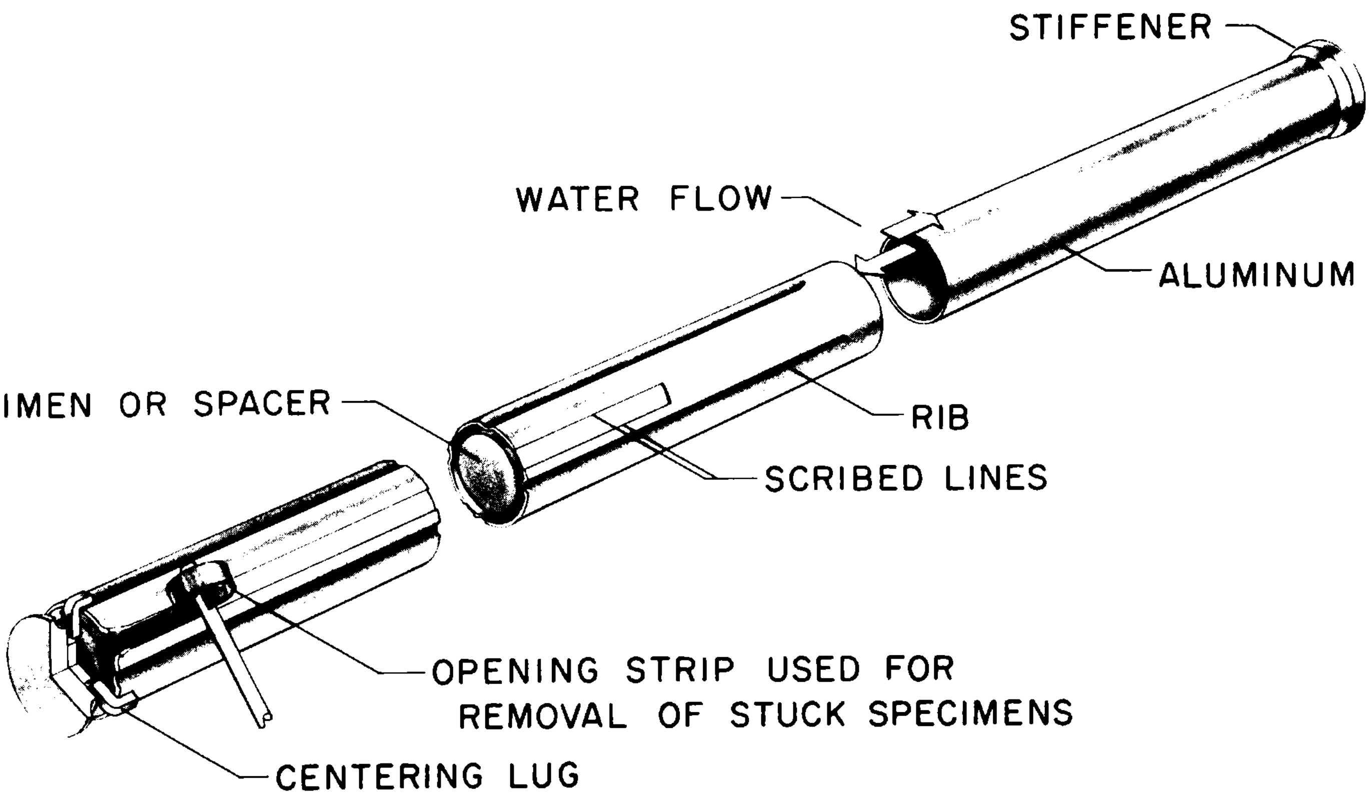


Fig. 1—The basket.

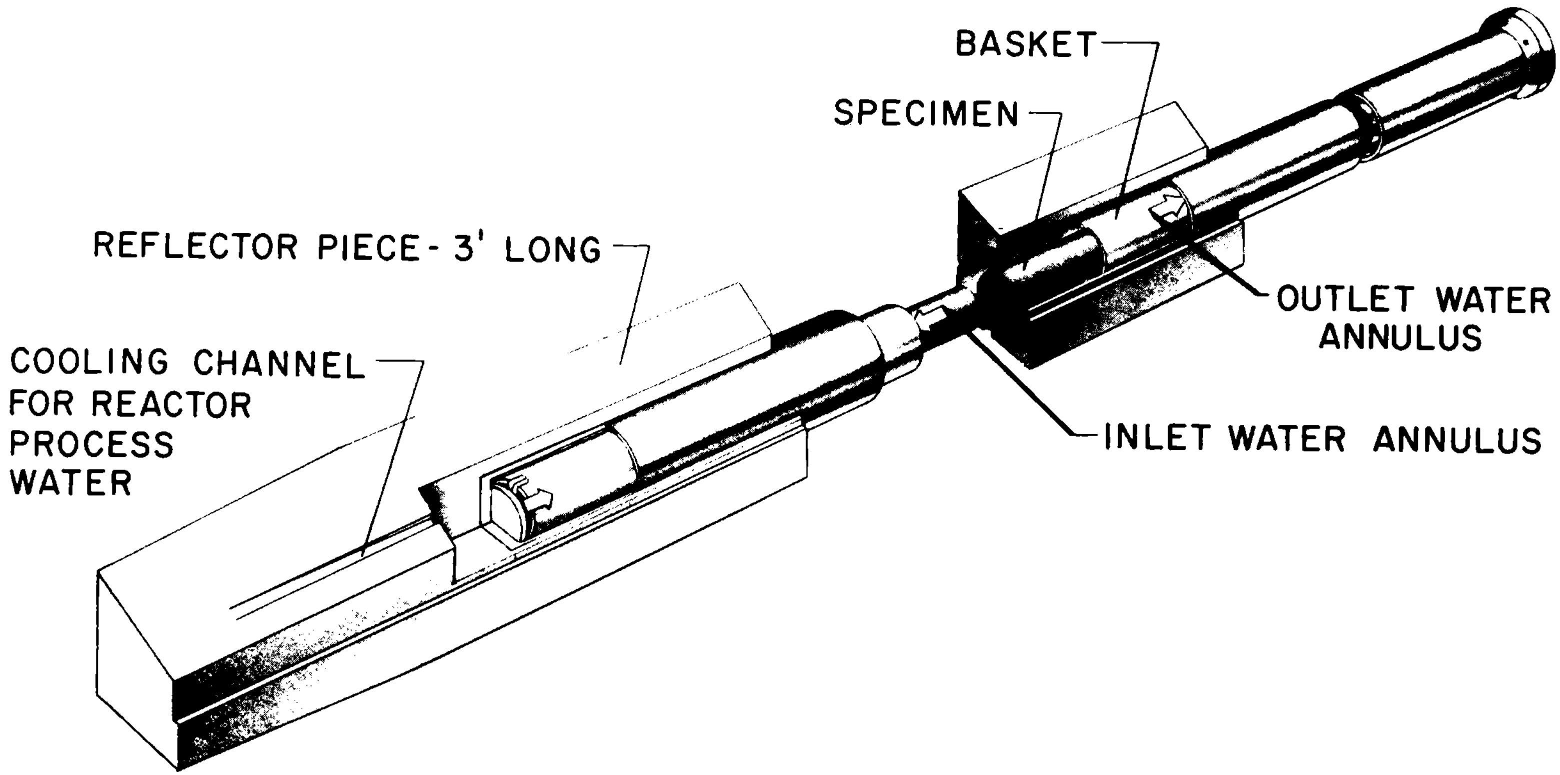


Fig. 2—The block assembly.

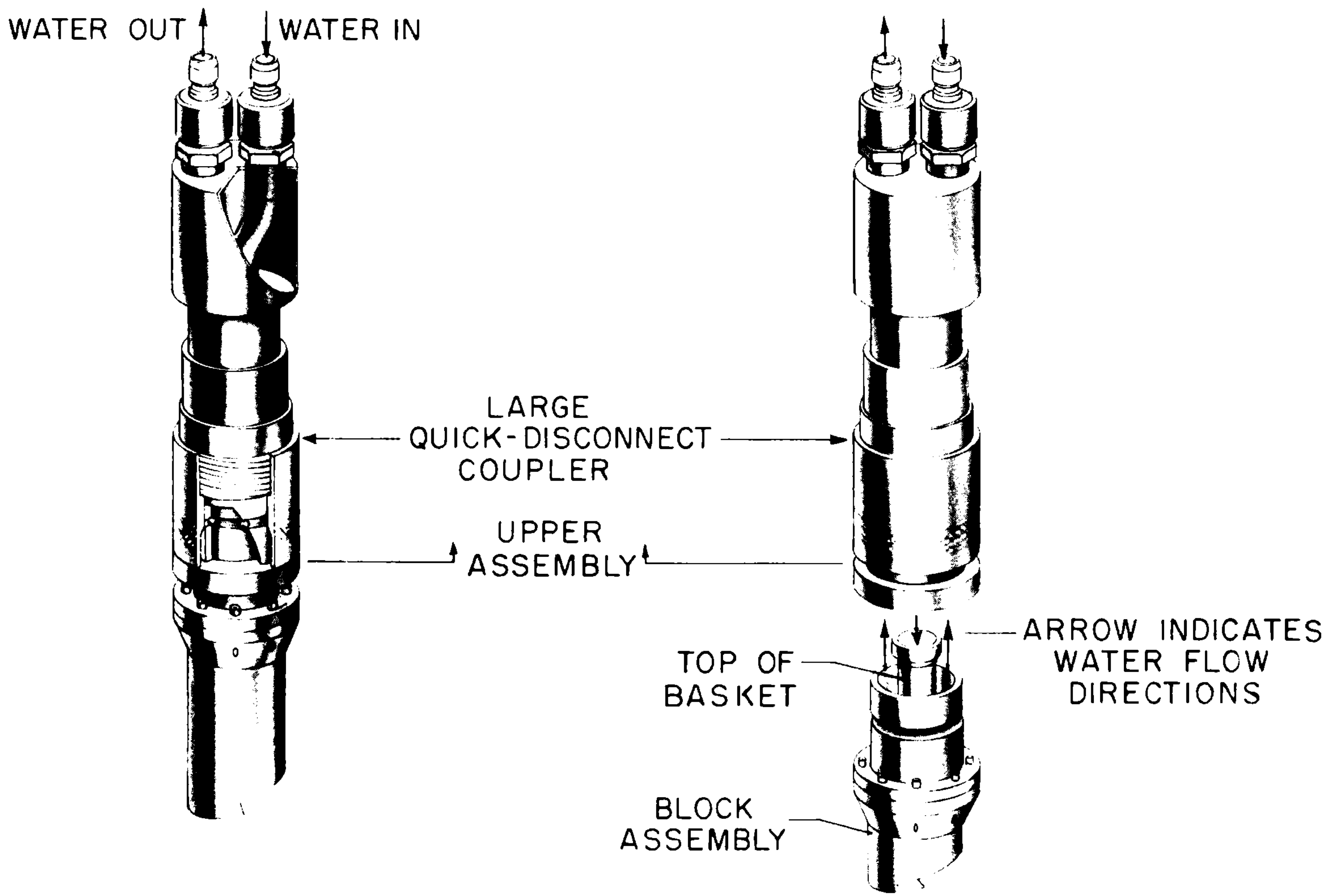


Fig. 3—The upper assembly.

through the upper assembly to a length such that when the upper assembly is coupled to the block assembly the tube slips inside the basket. An "O" ring on the inlet water tube affects the seal. Thus, in operation, the pressurized inlet water enters the upper assembly, flows down the basket, through the annulus around the specimens and out the bottom of the basket. The water then flows up the annular space between the basket and pipe assembly and out the quick-disconnect coupler on the left of Figure 3. A schematic drawing of the whole assembly (Figure 4) illustrates how the water flows through the system.

The use of the thin-walled tubing for the basket has proven to be an important feature of the design. The tubing makes it easy to remove stuck specimens from the basket with no mechanical damage to the specimens. To simplify removal even more, the baskets are scribed so that a coffee can type, longitudinal operation can be carried out (Figure 1). The facility design requires little strength of the basket.

The rest of the system is composed of standard centrifugal pumps and instrumentation to protect the reactor in case of sudden temperature or pressure changes. It is a one-pass system as shown in Figure 5; consequently, it was not necessary to use canned rotor pumps because any leakage on the inlet side would not be hot or radioactive. In the discharge of the process water the normal flow is to cold storage; in case of fuel failure a gamma spectrometer sends a signal to the three way valve and discharges the water to a 500 gallon tank; after all fission gas has been removed the water is dumped to hot storage. The system operates at 25 gpm and 500 psi with cold water.

Nearly five years of operation have been completed, during which failures and stuck elements have been removed with minimum damage to the specimens, and the importance of the rechargeability feature to a fuel element program has been demonstrated by experiments carried out to date.

Of principal interest to this paper is the operating experience of our facility in the MTR. The one-pass system is simple and versatile, and involves minimum construction costs. In addition, the single pass coolant system used materially simplified other requirements, such as contamination containment and clean up. Versatility was incorporated by designing the pressure tube, which houses the test samples, to serve as both a flow channel and discharge container capable of fission product containment. In addition the pressure tube can be recharged, that is, test samples can be removed after irradiation and a new (or the same) test assembled in the same tube and recharged in the reactor. However, by having several pressure tubes, test assemblies are normally prepared for reactor insertion prior to the required time, thus causing a minimum delay to reactor operations during scheduled shutdown periods.

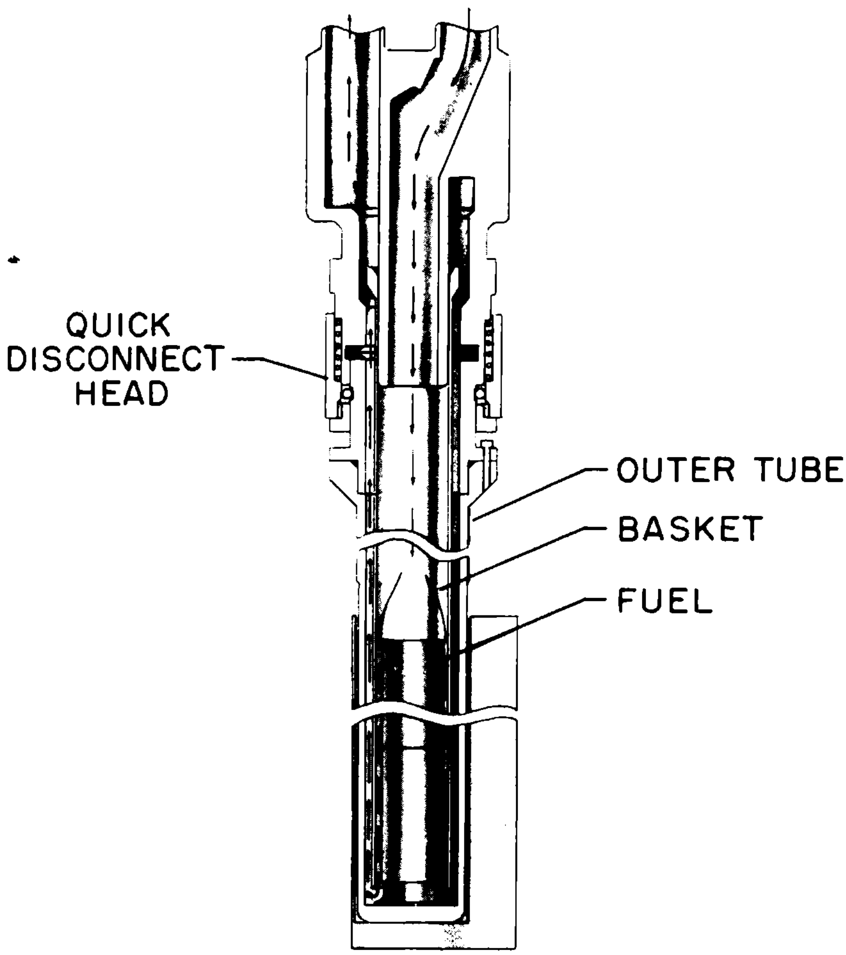


Fig. 4—Schematic drawing of the whole assembly.

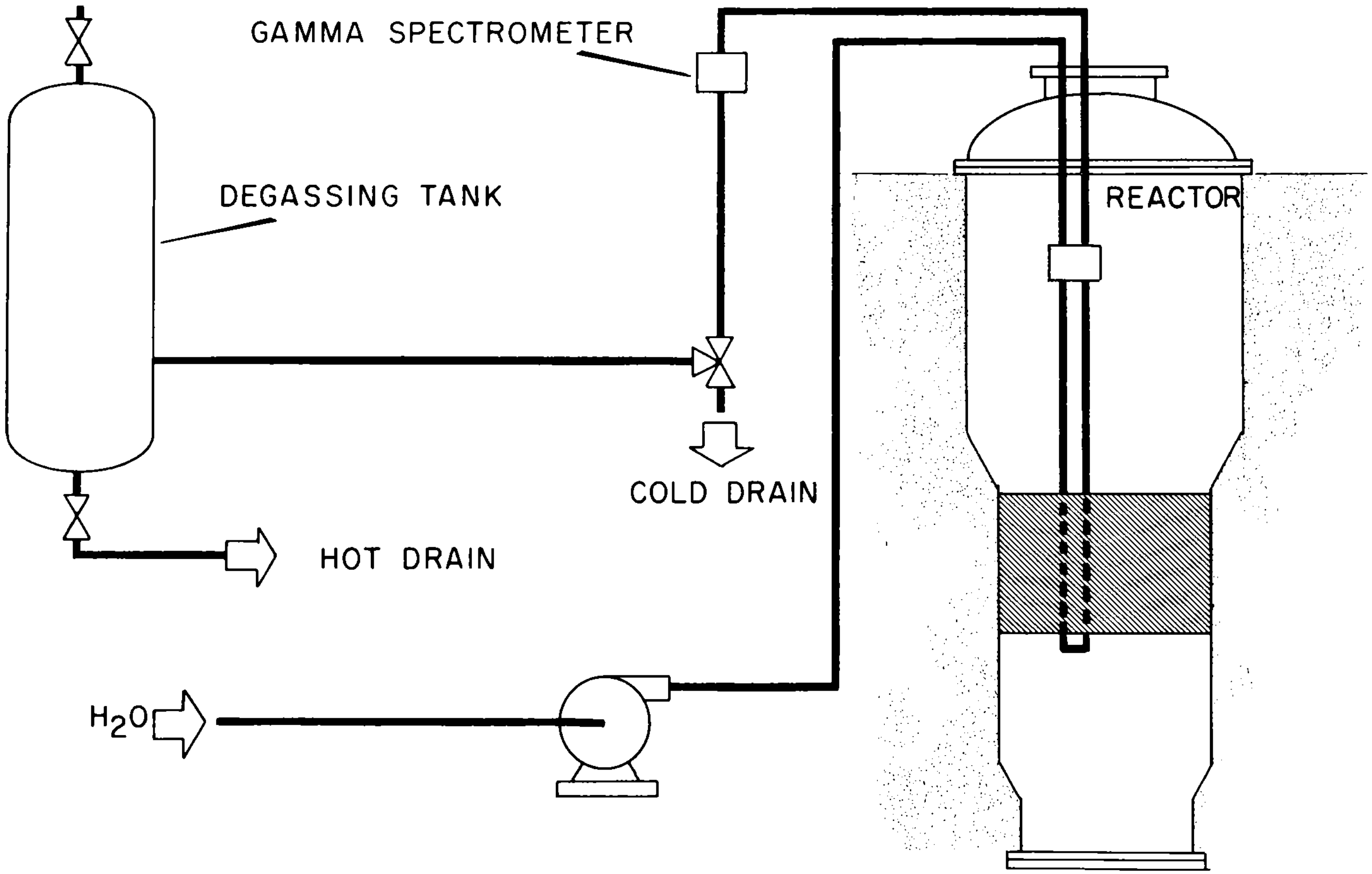


Fig. 5—Schematic flow diagram—GEH-4 MTR.

Knowledge gained from the operation of the low pressure system in the MTR aided in determining design criteria for the construction of high pressure loops in the ETR. A vertical cross section of the reactor showing the major components and general arrangement of one of the loops is shown on Figure 6. The ETR has the control rod drive located at the lower head leaving the upper portion of the reactor free for loop piping. The power generation exclusive of experimental facilities is 175 MW and the approximate thermal flux is of the order of 3 to 5 x 10^{14}, depending on experimental loading.

The lower head of the reactor vessel has penetrations for nine loop facilities as follows:

one 9-by-9 inch through facility
one 6-by-9 inch through facility
three 6-by-6 inch through facilities
four 3-by-3 inch facilities

In addition experimental holes are provided in the reflector.

Hanford occupies two of these positions as shown on Figure 7. The core position occupied is 6 x 9 inches in cross section and the reflector position 3 x 3 inches. The loop we installed in the 6 x 9 position is designed for high flow and high power service while the loop in the 3 x 3 position is of moderate flow but also designed for high pressure service.

The primary loop components and schematic flow diagram are shown in Figure 8. Some of the auxiliary loop components, such as the make-up system, are common to both systems. Each loop is composed of the primary components such as the reactor facility tube, heat exchangers, coolant circulation pumps, flow, temperature and pressure controls and instrumentation, surge tank and pressurizer with heaters and loop piping. The auxiliary components include the clean-up system, make-up pumps and tank, discharge systems, shielding and miscellaneous instrumentation.

Operating conditions shown in Figure 9 give us a reasonable degree of flexibility to accommodate our fuel element development program.

The control valves are of the pneumatic type designed to fail in the direction to provide the greatest protection to the reactor and loops in case of air or instrument failure. Use of air operated valves which require 3 to 15 psi greatly simplifies installation and maintenance.

All instruments and components are of standard manufacturer's design where possible, and have been arranged for simple and easy servicing.

The make-up system for each loop contains two positive displacement

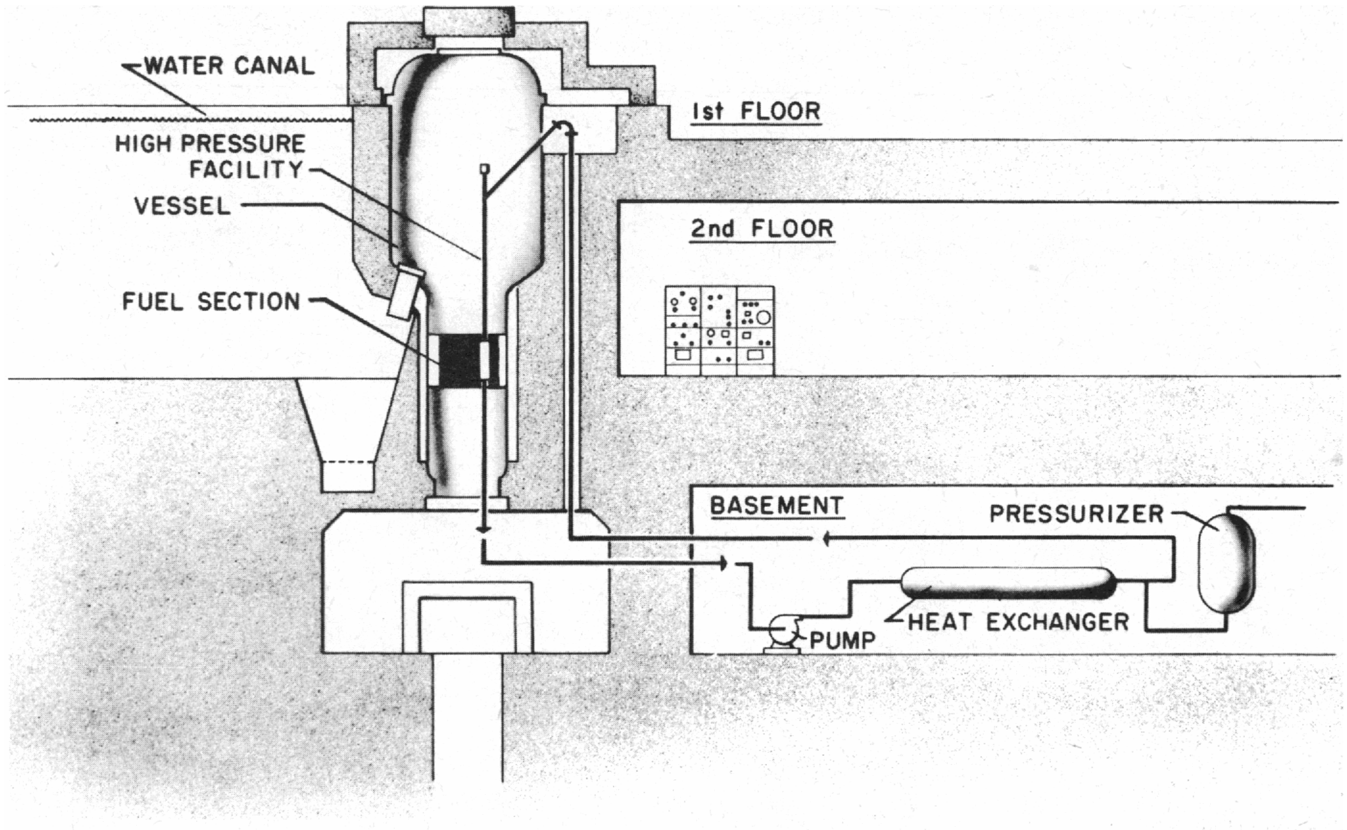


Fig. 6—Engineering Test Reactor.

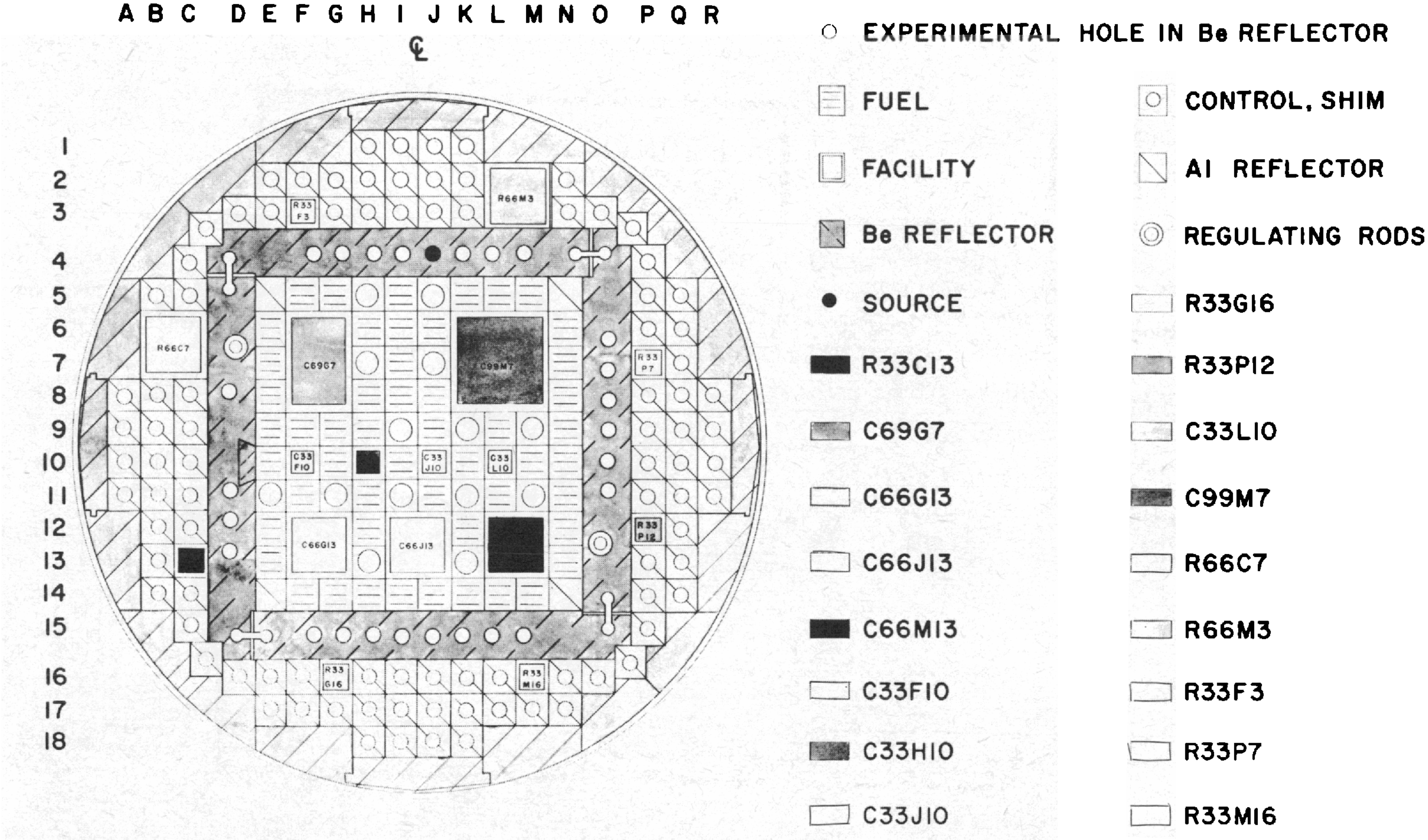


Fig. 7—ETR core arrangement.

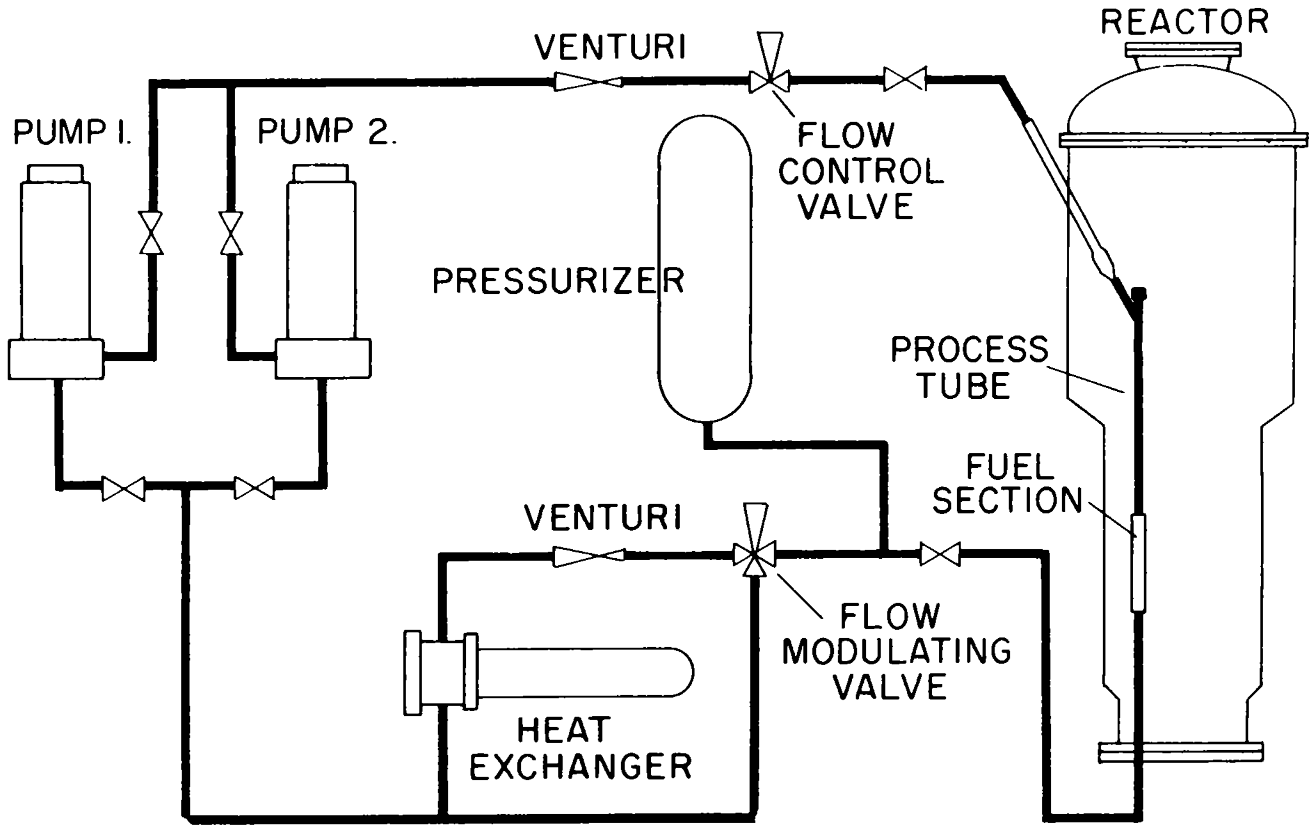


Fig. 8—R 3X3-P7 loop flow diagram.

FACILITY	C-9X9-M	R-3X3-P7
Total power output	2250 kw	500 kw
Operating pressure	2000 psi	2000 psi
Temperature–inlet	500°F	500°F
–outlet	600°F	550°F
Flow–primary loop	150 gpm	50 gpm
–cooling water	300 gpm	100 gpm
–cleanup bypass	2-4 gpm	1-2 gpm
Water purity	>½ megohm	>½ megohm
	<1 ppm solids	1 ppm solids
	pH 5.5–6.5	pH 5.5–6.5

Figure 9

pumps which are rated at 3750 psig, piped in parallel to furnish demineralized water to the loops to compensate for losses by leakage, and to provide higher pressures for hydrostatic testing.

The main high pressure piping system and components are designed for 2500 psig and 600 F. Piping is Type 347 S. S., schedule 80, 1-1/2 inch nominal size in the 3 x 3 and 2 inch nominal size in the 6 x 9 loop. All pressure piping is designed in accordance with ASA and ASME codes. Each loop has a primary heat exchanger for removing reactor heat. The 3 x 3 and 6 x 9 loops are rated at 1.5×10^6 Btu/hr and 8×10^6 Btu/hr, respectively, at the rated temperature and pressure, although greater amounts of heat can be removed to permit operation at low coolant temperature and pressure. Temperature control is effected by regulating the portion of the primary coolant which is allowed to bypass the heat exchangers.

Two pumps are normally operated in both loops with one pump capable of providing shutdown flow in case of failure of the other pump. The 6 x 9 loop is provided with a third pump for operational flexibility. Pumps in both loops are canned rotor, zero leakage type. The pumps in the 3 x 3 loops are piped in parallel and in the 6 x 9 are piped in series.

Although many ex-reactor loop components have become more or less standardized, the design of in-reactor sections varies widely, and continued improvement is required to achieve designs that will enhance operational flexibility. A view of the 3 x 3 in-reactor tube is shown in Figure 10. The pressure piping within the reactor is 347 S. S. An air annulus is provided by an aluminum shroud tube surrounding the pressure tube so that only a controlled amount of heat may flow into the reactor cooling water from the test facility. This amount of heat is small enough so that surface boiling does not occur on the outside of the assembly. The upper end of each facility tube is equipped with a quick closing plug to facilitate

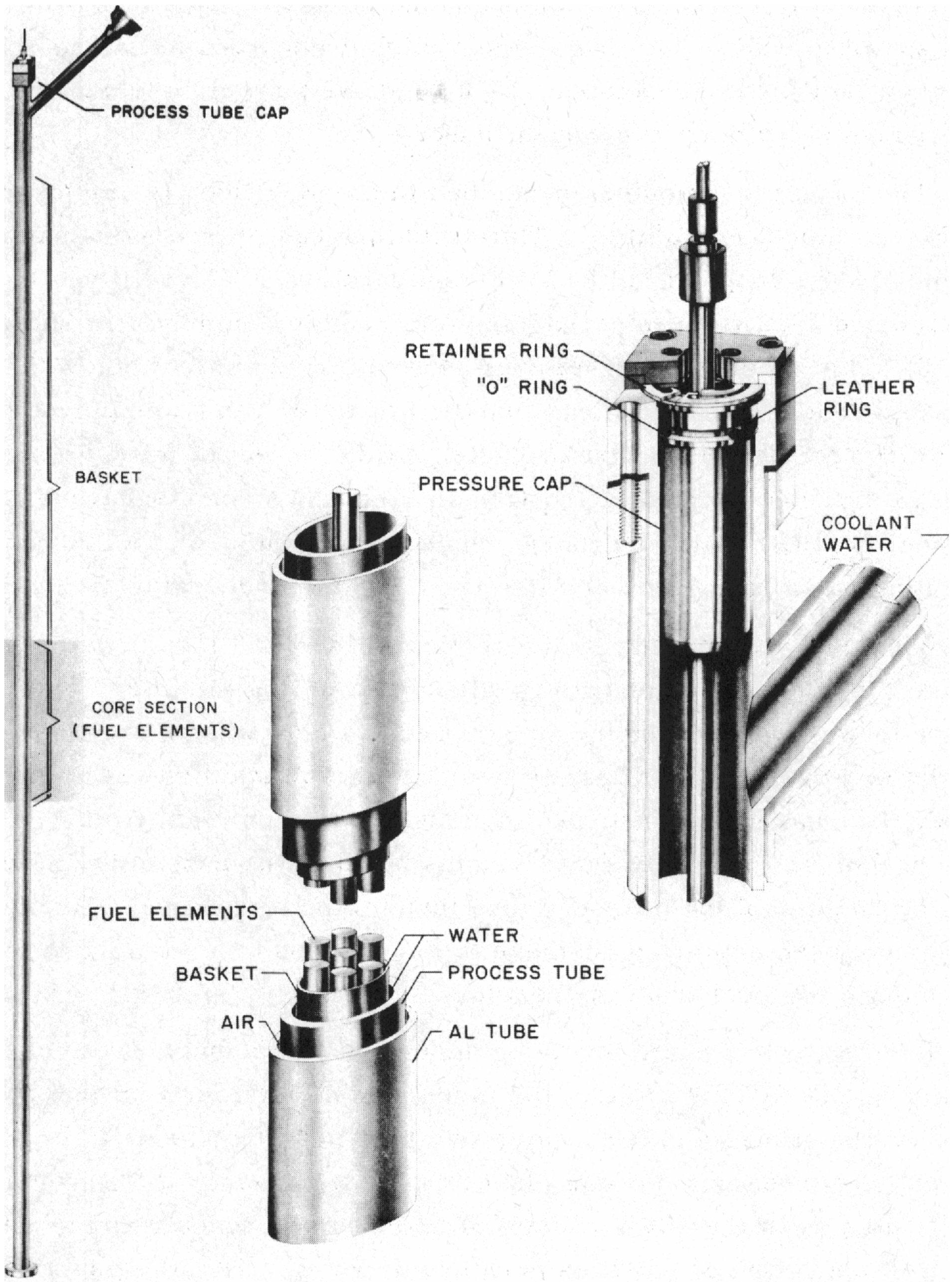


Fig. 10—ETR 3X3 in-reactor section.

charging and discharging of the test section. As shown in Figure 10, the pressure cap of the plug is held in place with a snap ring, and the 2000 psi water in the system is held by a silicone O ring backed by a Teflon ring. The normal reactor cooling water temperature is about 110 F which is sufficient to cool the cap assembly and protect the seal during reactor operation. The cap can be readily removed by compressing the snap ring with a remote control tool as shown in Figure 11. This type of high pressure seal has been in operation in the 3 x 3 facility since October, 1958, and has proved to be capable of rapid remote assembly without subsequent leakage.

The basket technique as described in the MTR loop is employed in both the 6 x 9 and 3 x 3 facility. This technique has proven as valuable in high temperature loops as in the lower temperature MTR facility. The basket method also permits recharging of irradiated elements as shown in Figure 8. The top of the process tube is terminated six feet below the top of the reactor vessel. An additional four feet of water can be employed by using a spool piece fabricated by the ANP site. With ten feet of water to serve as shielding, the tube or "basket" containing the fuel can be discharged into the ETR canal via the reactor discharge chute without the use of a discharge cask. It can also be loaded back into the facility by reversing the procedures.

The ability of the in-reactor section of the 3 x 3 loop to withstand abnormal fuel surface temperatures without impairing the integrity of the pressure tube was shown during an early experiment when a stainless steel specimen was overheated. Despite the fact that the stainless specimen was cracked, the annulus between specimen and basket completely plugged by oxide, and the basket wall partially destroyed by melting, the assembly was discharged without difficulty and the pressure tube found to be undamaged. Had the basket technique not been employed, damage to the process tube would have required its immediate removal from the reactor.

Several experiments involving unplanned and planned ruptures have been carried out to date. One of the more complicated irradiations was a planned rupture which admitted water to a high temperature metallic uranium specimen. This experiment was planned in order to study the consequences of reaction between the water coolant of an operating reactor and uranium heated well above the critical temperature of water. The high metal temperature was achieved by introducing insulation between an unbonded aluminum can and the metallic uranium. Intentional rupture of this can then permitted observation of the deformation of the can, and extent of corrosion of uranium. The conditions were such that the time between clad failure and reactor shutdown could be accurately measured.

The test fuel element as shown in Figure 12, was fabricated from

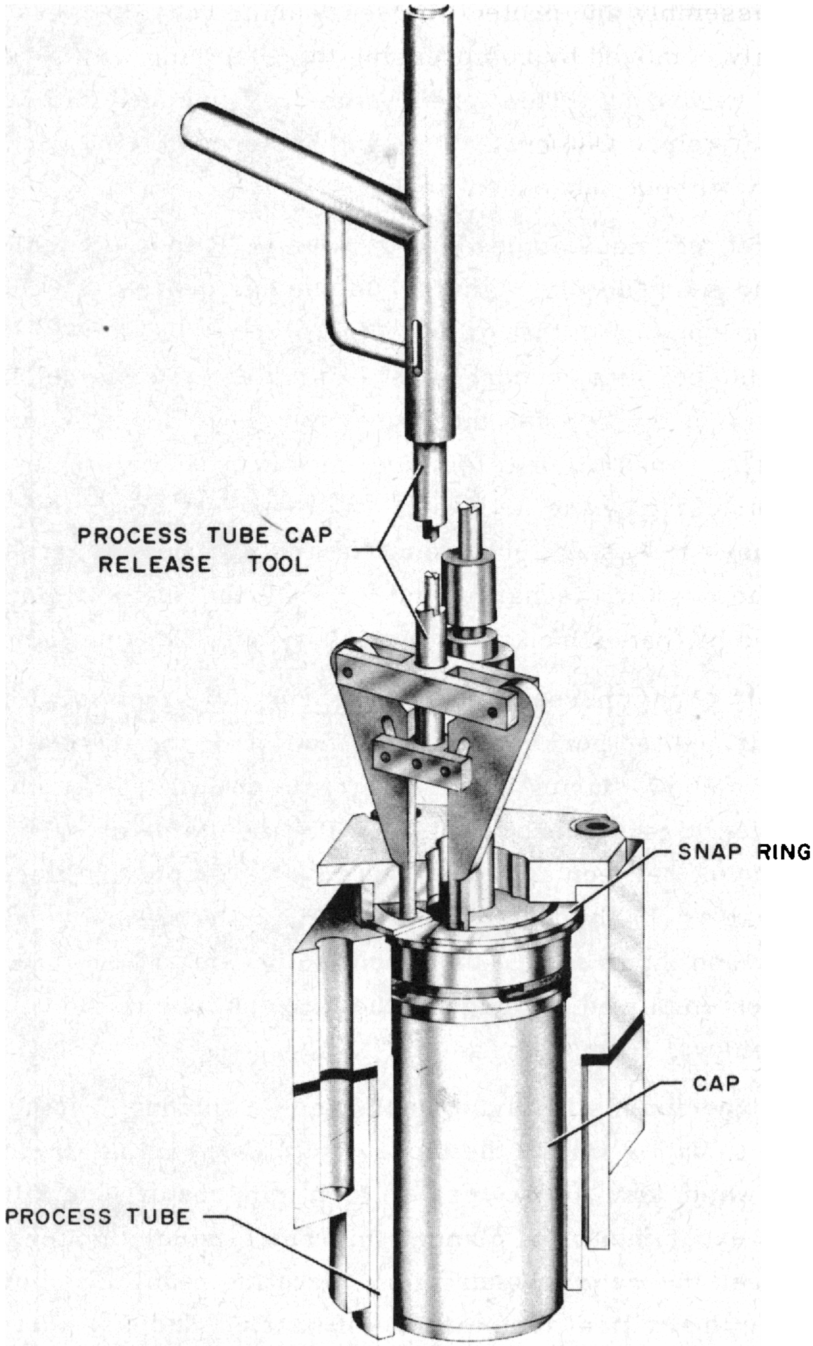


Fig. 11 — Pressure cap and removal tool.

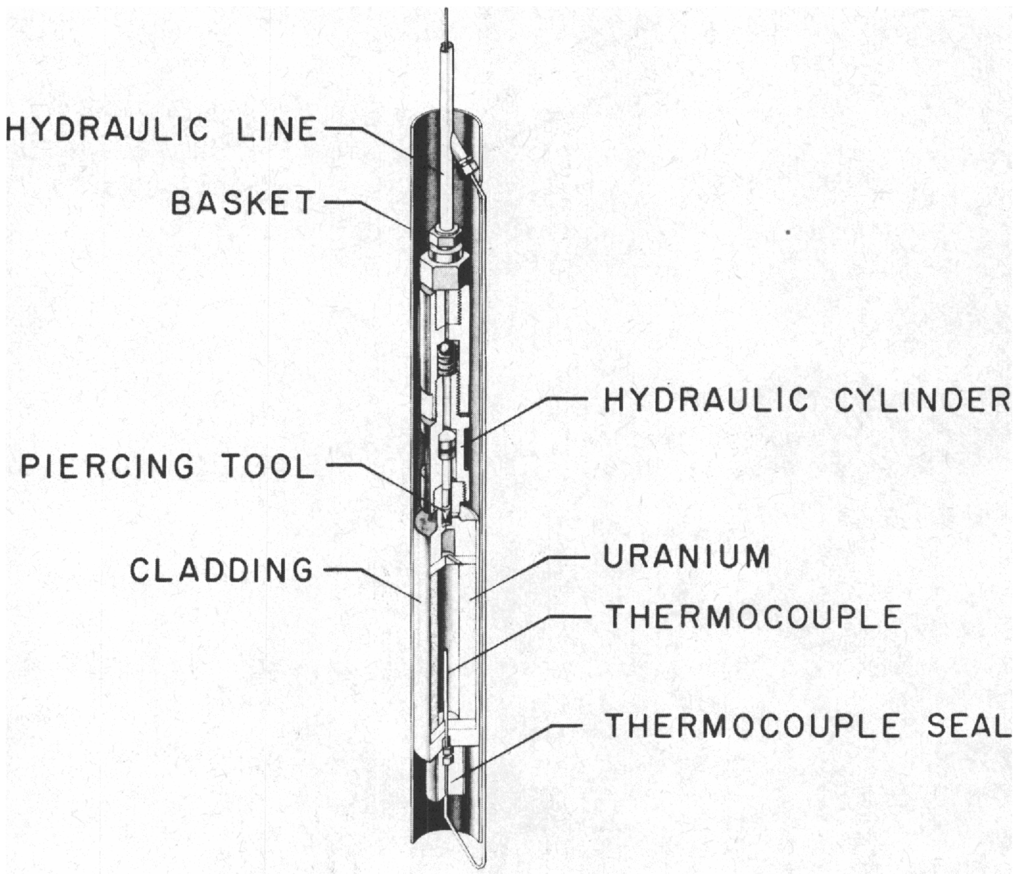


Fig. 12—Intentional rupture fuel element.

natural uranium. A hydraulic mechanism, with water as the hydraulic fluid, was attached to the top end cap of the fuel specimen. After the experiment was charged and the reactor brought to full power, the mechanism was actuated from outside the reactor with a small hydraulic pump. The clad specimen was ruptured in the loop by piercing an aluminum diaphragm in the end cap. Penetration of the aluminum diaphragm allowed cooling water to enter the uranium core. The maximum metal temperature was measured by an axial thermocouple fitted into the bottom end with a high pressure tube fitting and a metallic seal. Rupture of the cladding was confirmed about 60 seconds after the hand hydraulic pump was actuated. The contaminated water passed through the loop's fission break monitoring system, which automatically scrammed the reactor.

Other types of fuel irradiations have been performed in the MTR and ETR, some ending by unexpected fission breaks, some operating with molten UO_2 cores, and others having unique fuel element configurations. However, because of the design of the testing loops it has never been required or necessary to provide special casks or other equipment to discharge these irradiated specimens from the reactor.

One other major factor in the design of an in-reactor pressure tube for a fuel testing program is the ability to remove and replace the tube, should this be required. The feasibility of this removal procedure was demonstrated when the 3 x 3 tube in the ETR required removal following the failure of a weld in the aluminum shroud tube. The successful tube removal technique involved the following equipment and procedures. The special equipment required was a 3500 pound split cask, 50 inches in length, and containing 5 inches of effective lead shielding; a platform to support the cask over the reactor, and three cable slings. To perform the discharge a two-ton building crane was used.

To prepare the tube for removal required disconnecting the inlet piping at the flange provided in the top of the reactor tank, and cutting the outlet piping in the sub-reactor room a few inches below the reactor's bottom head. The platform and cask were positioned across the reactor top above the loop tube. One half of the cask was set in the final lifting position, as shown in Figure 13. The other half was separated to allow the "Y" shaped top of the tube to be lifted above the top of the positioned cask half and anchored to it. The separated half was then lifted into position and the entire cask bolted together around the tube, as shown in Figure 14. One end of each of the three slings was attached to the crane hook, and the other ends were attached to the top of the tube and each half of the casks. The two slings attached to the cask were of the same length and were longer than the tube sling. The

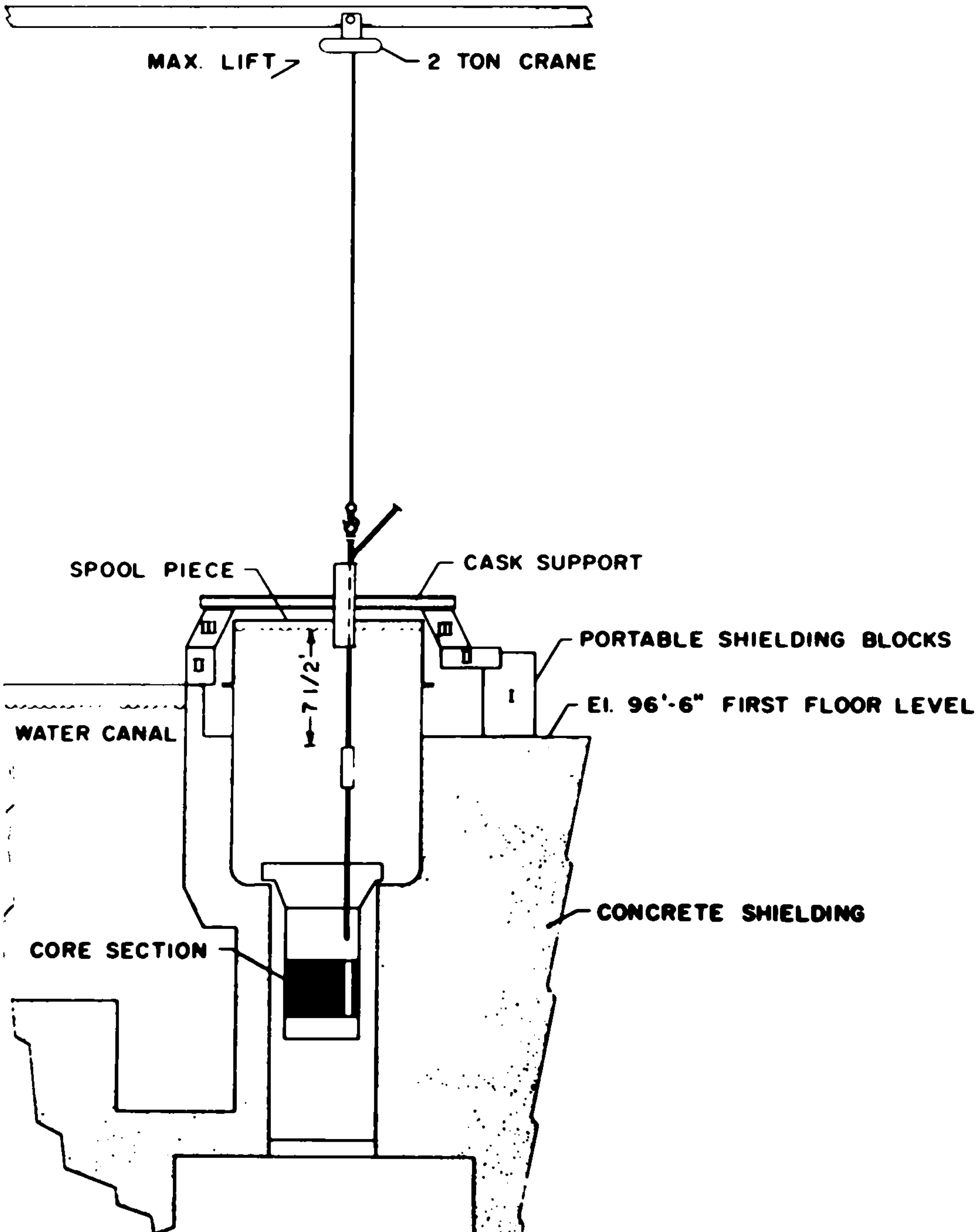


Fig. 13—Tube removal procedure.

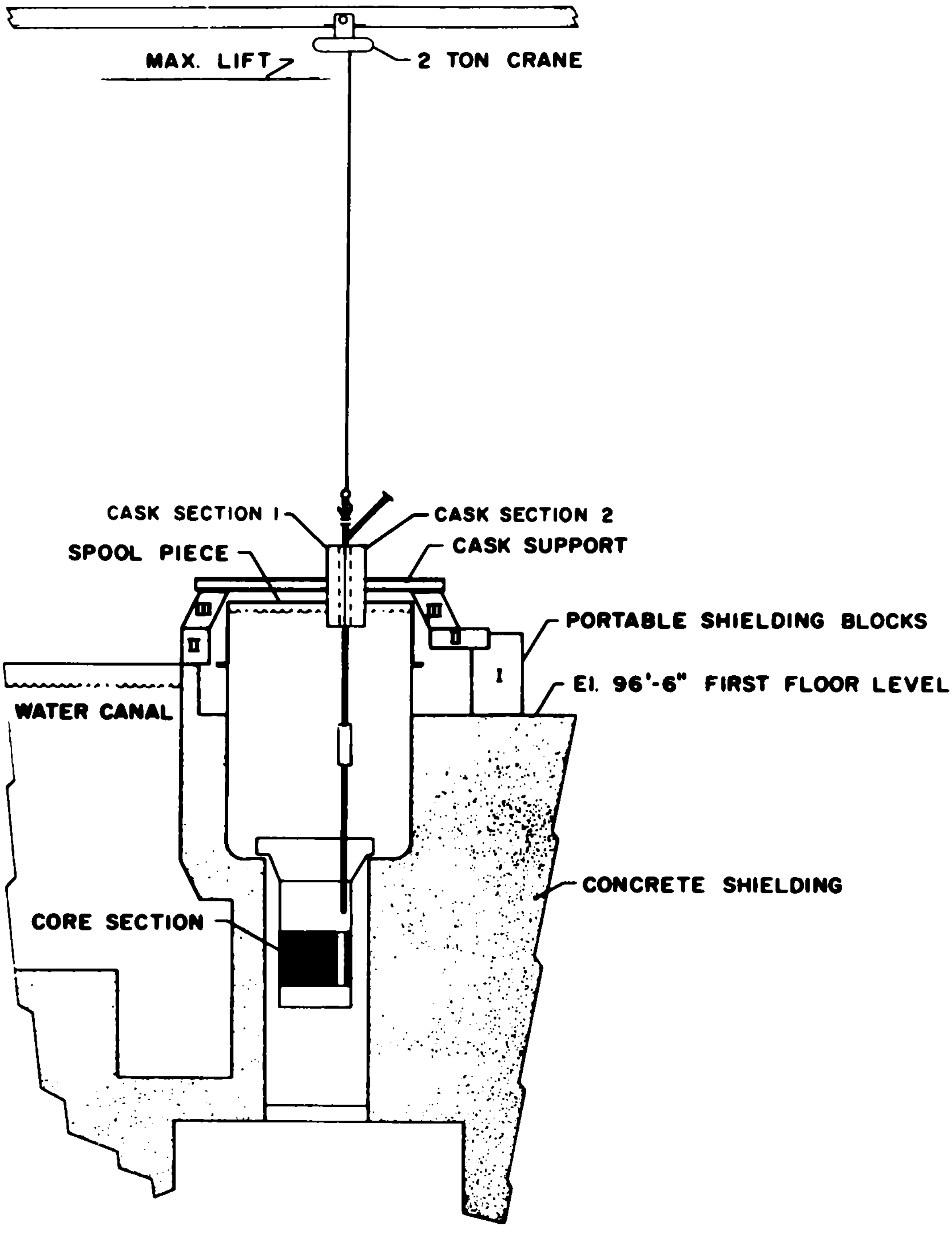


Fig. 14—Tube removal procedure.

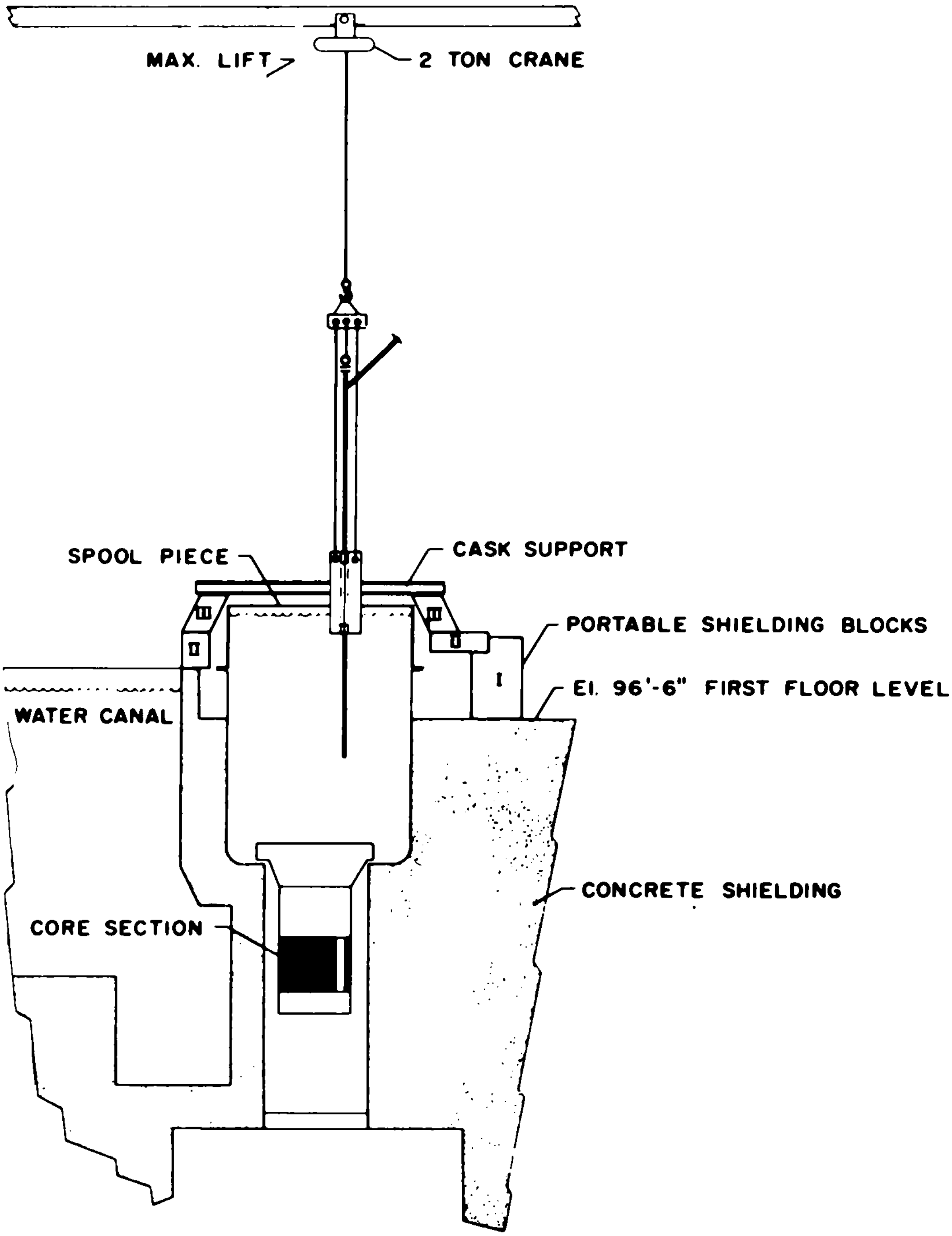


Fig. 15—Tube removal procedure.

length of the slings were established so that when the slings were taut, the reflector portion (highly radioactive section) of the tube was centered inside the cask, as shown in Figure 15.

The actual removal process required about three minutes. The tube and cask were lifted out of the reactor tank and moved over to the canal and lowered in. The tube was vertically lowered onto the canal bottom until the tube sling became slack and could be removed. The cask was then raised above the water, unbolted, and the two halves separated from around the tube.

The surface activity of the reflector section of the tube was measured to be about 15,000 r/hr. During the removal process the highest measured radiation field was about 200 mr/hr approximately 18 feet away, at an angle of about 45° below the bottom of the cask.

Although it would be difficult to place a monetary value on the data and experience gained from irradiations performed in high flux test reactors, the absolute necessity of the test reactor irradiations cannot be questioned because of their invaluable contributions to the advancement of reactor fuels technology and design. The flow loops used as a principal tool in these programs can contribute most effectively if simplicity of design and instrumentation are utilized to minimize equipment difficulties and reduce loop down-time to a minimum.

USE OF IRRADIATION ZONES IN THE FUEL ELEMENT LATTICE OF RESEARCH PILES

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Commissariat à l'Énergie Atomique, Paris, France

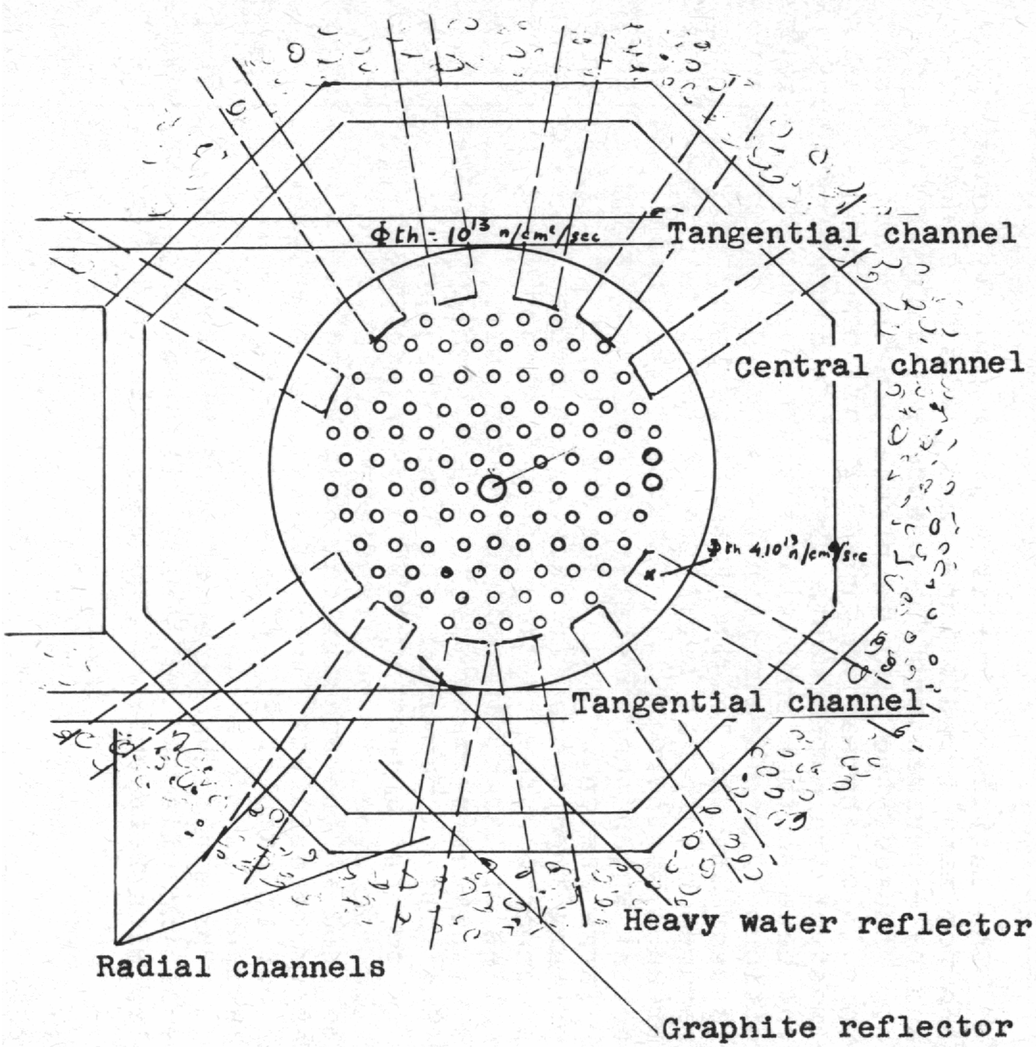
I. INTRODUCTION

The lattice of research piles usually constitutes a choice region for many irradiations, and there is always a very great demand by experimenters for channels located in the core of the pile. Indeed, the flux is highest in the fuel element lattice whether it be thermal neutrons or fast neutrons or gamma rays. For most of the experiments these high intensity radiations are sought because they enable us either to improve the measuring conditions if one deals with the study of an instantaneous effect, or to decrease the length of the experiment if one investigates a phenomenon which is sensitive to the total dose of irradiation received. Furthermore, it is usually the only case one can find places at which the intensity of the flux is sufficiently uniform over large regions. This flux uniformity over large regions is of primary importance for many experiments; one can mentiôn in this connection the testing of fuel elements which are intended for piles of large dimensions (power piles) in which the element will be placed permanently in a practically constant neutron flux. For the test to be valid, the power dissipation along the element was to be kept a constant as closely as possible. One can also mention the case of the irradiation of a large number of samples to be used in the statistical measurement of some parameter. In order to be accurate, this statistical measurement has to be carried out on samples which have received essentially the same irradiation.

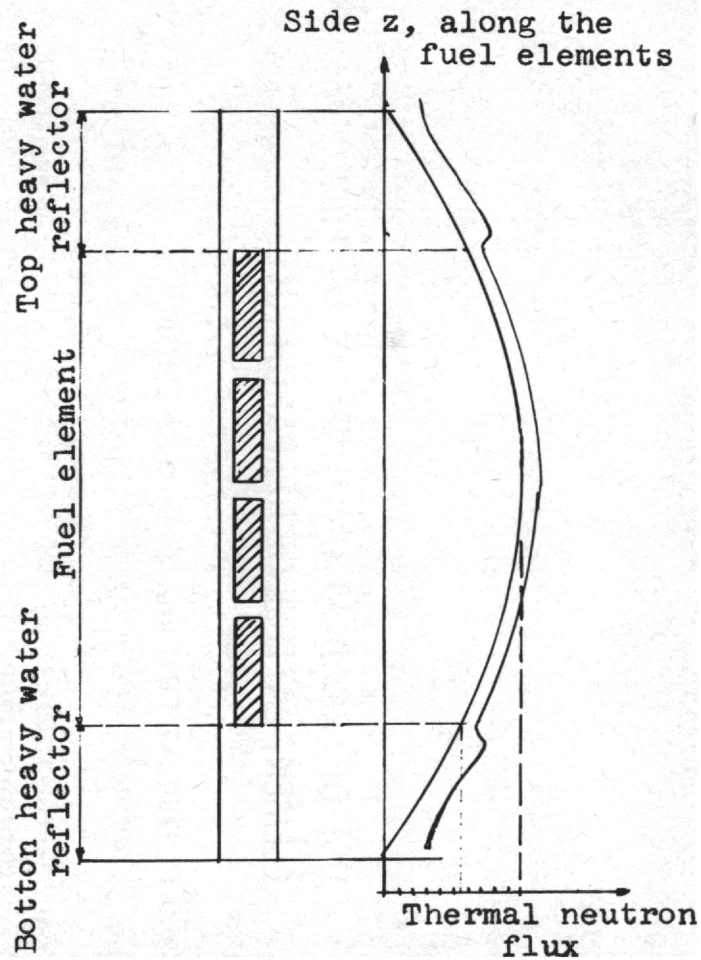
The many advantages of experimental sites located in the core of the pile can be easily illustrated by some numerical examples with reference to the EL-3 pile (see Figure 1).

Fig. 1

Schematic Section of Pile EL-3



EL-3 Distribution of Thermal Neutron Flux Along the Axis of a Fuel Cell



In this pile, which is equipped with top and bottom heavy water reflectors, the axial distribution of neutron flux, parallel to the fuel elements, is such that it is possible to obtain along a distance of 1300 mm a thermal neutron flux whose extreme values are in a ratio 1.5.

On the other hand, in a radial channel of the same pile the thermal neutron flux decreases by about a factor of 7 in a distance of 700 mm. The only channels that can be compared from the point of view of flux uniformity with the channels located in the core of the pile, are those that are referred to as tangential. In these channels the thermal neutron flux is in fact constant $\pm$ 15 percent for a distance of 1000 mm; however, the absolute value of this flux is very much lower than it is in the center of the pile (by about a factor of 10); the fast neutron flux is practically negligible.

In spite of their advantages, one has to admit that the irradiation regions located in the fuel element lattice have also certain drawbacks. Though one of their dimensions is usually rather large since it corresponds to the height of the fuel elements, the total volume is often small. Furthermore the connections between the parts of the experimental setup in and outside the pile are usually difficult to arrange because of lack of space which is mainly due to the cluttering of numerous control and operation apparatus of the reactor itself, located above the pile. Finally, the last objection, though it is not the least, results from the effect of experimental setups on the reactivity of the pile. This effect is naturally greatest in the regions of high flux. The charging of the core by various experimental materials is limited by the available reactivity, and the portion of the reactivity available for the experiments cannot exceed certain limits for reasons of safety.

However, the irradiation regions located in the core of research piles have, in general, many more advantages than drawbacks. But it does not follow that, when the lattice of a pile consists of a large number of channels, it has the maximum number of advantages. Each type of experiment gives rise to particular conditions, and the geometry and the location of a channel make it more suited for certain experiments than for others. It is this aspect that we will consider now, and we will attempt to draw certain standards which will enable us to determine for each type of experiment the type of channel that is best suited for it. The examples which we will use will be related to heterogeneous piles such as EL-2 and EL-3, but the conclusions which will be drawn from it will be applicable on the whole to other types of piles.

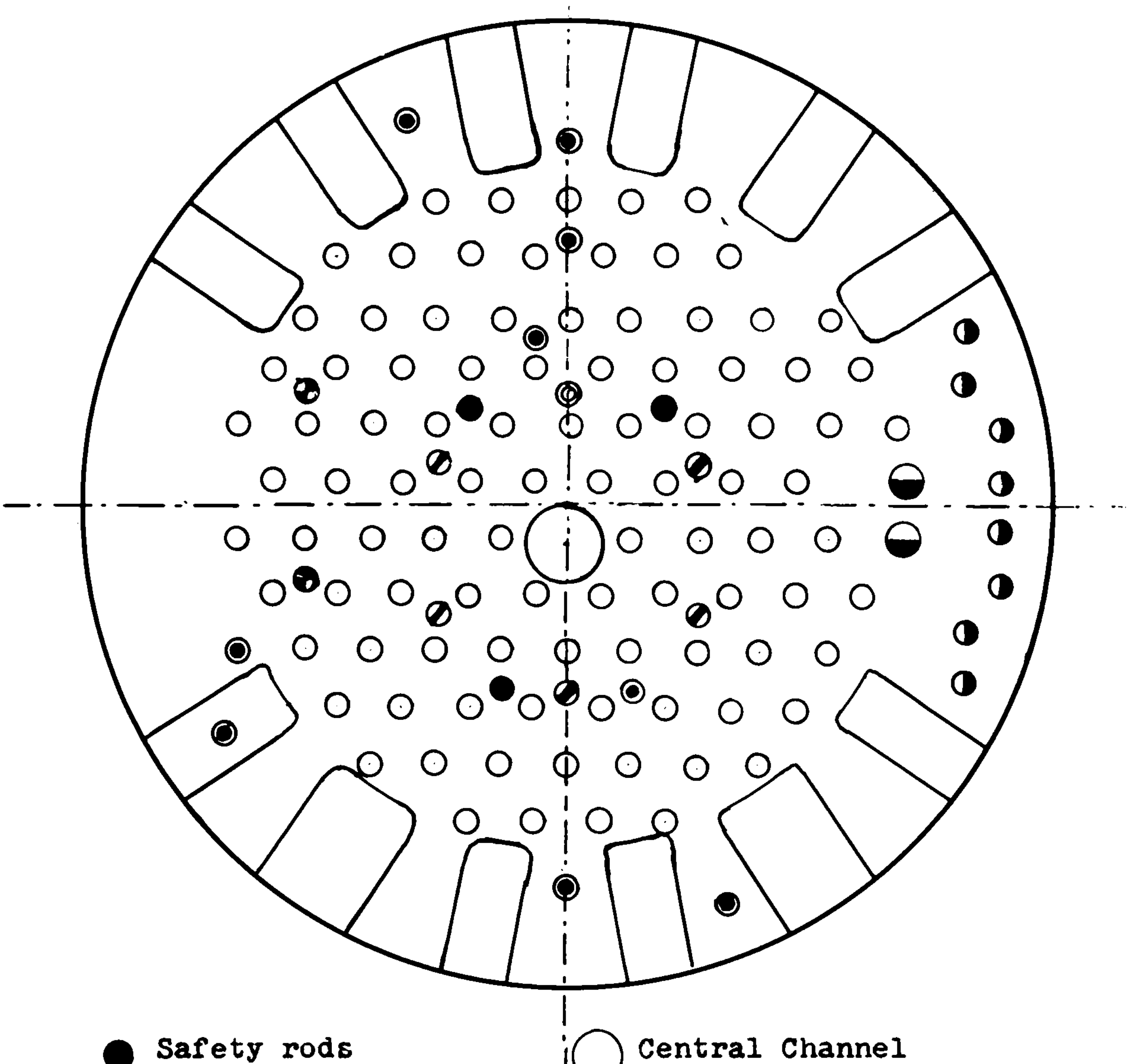
II. POSSIBLE IRRADIATION REGIONS IN A FUEL ELEMENT LATTICE

a. Channels Located Between the Fuel Elements or in Their Place

Let us consider the case of the EL-3 pile whose lattice is shown in Figure 2. This lattice, which is of the centered hexagonal type, contains for experimental purposes:

Fig. 2

Lattice of Pile EL-3



● Safety rods

⊘ Compensation rods

Control rods

○ Uranium cell

Central Channel

Irradiation channels (DS-1 and DS-9)

Vertical convertors

Isotope channels in heavy water (There are also 22 isotope channels in the graphite reflector)

- a channel with an inside diameter of 180 mm at the position of the most centrally located uranium cell.

- Three channels (designated by DS-1 to DS-3) with a 48 mm diameter, and located in the center of a triangle formed by three uranium cells.

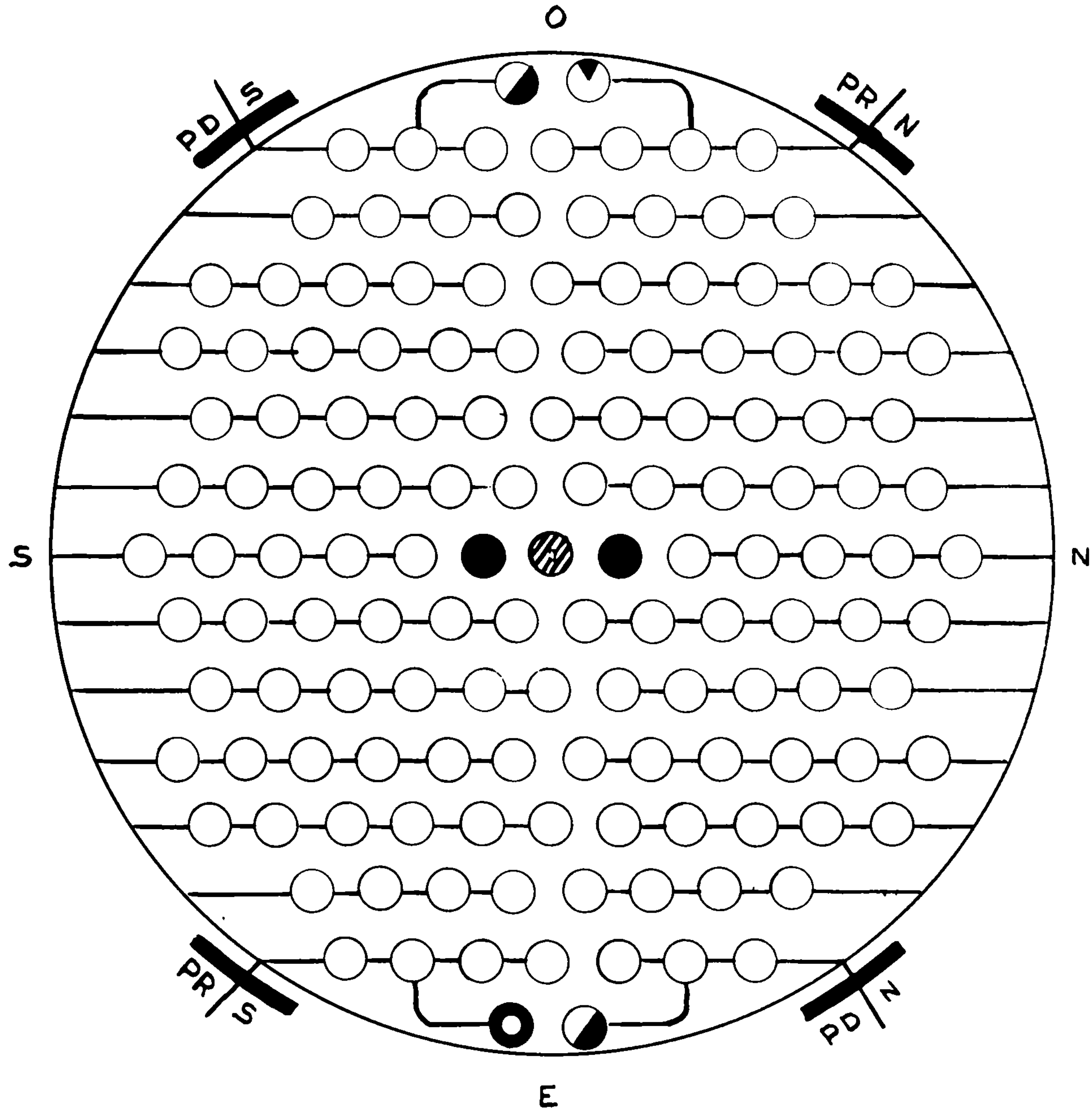
Furthermore, six other channels (designated DS-4 to DS-9), identical with the previous ones, are located either at the border of the core, or a little further out in the heavy water reflector. In the case of pile EL-2, whose lattic is shown in Figure 3, only one channel located between the fuel elements is available. It has a diameter of 50 mm and is at the position of the most centrally located uranium cell.

The usefulness of such channels depends very much on their geometry and what one intends to do with them. For the irradiation of fuel materials one tries at first to obtain a high flux of thermal neutrons; furthermore this type of experiment usually requires the installation of rather voluminous loops, and the channel must therefore be of relatively large dimensions. On the other hand, if one deals with the irradiation of structural materials or moderator materials, a rather small irradiation volume with a high flux of fast neutrons is usually sufficient. However, the flux variations of thermal and fast neutrons as a function of the dimensions of the channel are actually opposite to those stated above. The larger the channel, the smaller the thickness of the moderator which separates it from the neighboring uranium cells, and as a consequence the lower the flux of thermal neutrons. To this effect is added the increase of neutron leakage along the axis of the channel, the two phenomena acting in the same direction. Again in the case of large channels, the lack of moderator favors the increase of flux of fast neutrons. The same reasoning could be applied in the opposite sense for the case of a channel with a small diameter, in which, consequently, one will have an increased thermal neutron flux, and a decreased fast neutron flux, though such channels can only be used for the irradiation of small samples which most frequently (structural or moderating materials) require a large fast neutron flux. We shall see in Chapter III what conclusions have been drawn from these considerations.

As an illustration, the thermal and fast neutron flux as a function of the radius of a channel in the position of the most centrally located uranium cell of the EL-3 pile (channel designated as "Central Channel") has been plotted in Figure 4. In each case the flux of fast neutrons has been calculated by the method of collision probabilities starting from sources which consist of the uranium cells surrounding the channel. A summary and bibliographic references for this method is given in the report of R. ECKERT, "Fast Neutron Dosimetry in Research Reactors". The calculation of the thermal neutron flux has been carried out in the following manner: the central channel is replaced by a diffusing medium such that the flow to the surface of the channel is equal to the one calculated by Davison's theory (CRT 319); the distribution of the flux in the pile with several media constituted in this manner has been calculated theoretically in two groups with the Ferranti computer of the Mathematical Physics Department.

Fig. 3

Lattice of Pile EL-2



Normal Uranium cells

Safety rods

Central channel

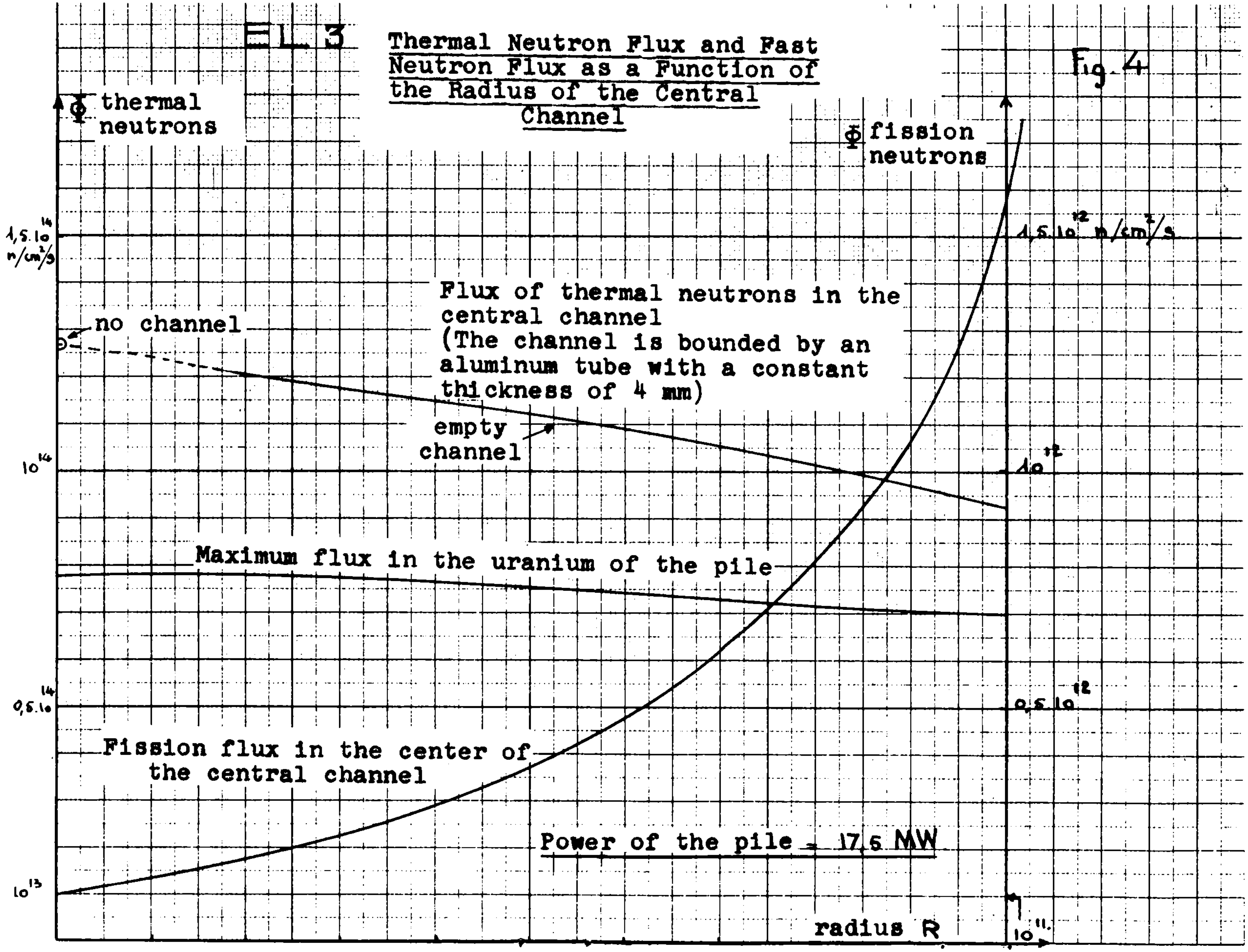
Vertical converters

PD Starting panel

PR Adjustment panel

Independent cell

Heavy water feed in


EL 3
Thermal Neutron Flux and Fast Neutron Flux as a Function of the Radius of the Central Channel
Fig. 4
Φ thermal neutrons
Φ fission neutrons
1,5.10^14 n/cm²/s
1,5.10^12 n/cm²/s
10^14
10^12
0,5.10^14
0,5.10^12
10^13
10^11
Flux of thermal neutrons in the central channel
(The channel is bounded by an aluminum tube with a constant thickness of 4 mm)
no channel
empty channel
Maximum flux in the uranium of the pile
Fission flux in the center of the central channel
Power of the pile = 17,5 MW
radius R

Apart from the questions related to the intensity of the neutron flux, one has to consider the following advantages of channels located between the fuel elements:

One can cool the samples by a circuit which is completely independent from the main circuit of the pile; this independence enables one to choose both the nature and the pressure of the fluid. This results in a large choice of experiments that can be carried out. This independence can also be used to carry out experiments at temperatures differing markedly from those that would be imposed by the requirements of the main circuit of the pile. One has however to add that in this case it is often necessary to achieve very good thermal insulations between the channel and the moderator of the pile. This necessity is evident for very low temperatures, if one takes into consideration the numerous precautions one has to take in order to avoid a too rapid evaporation of liquefied gases. For high temperatures the necessity of thermal insulation is closely related to the properties of the moderator of the pile; thus, in the case of piles with water not under pressure one has to limit the heat exchange in the direction channel $\longrightarrow$ moderator to a maximum valve of the order of 1 watt/ cm^2 in order to avoid any local boiling.

b. Use of False Cells for Experimental Purposes

It is always possible to replace a fuel cell by a false cell which contains the sample that is to be irradiated. In this instance one usually conforms to the external geometry of the normal cell and one uses the main circuit of the pile for the cooling of the samples.

As regards the intensity of the thermal and fast neutron flux such an arrangement is basically analogous to the ones that have been considered in the previous paragraph. It is however placed in an extreme position which favors to a maximum an increase of the thermal neutron flux and reduces considerably the fast neutron flux. Such an experimental setup is thus particularly suited for the irradiation of fuel samples, and perhaps also for the preparation of radioisotopes obtained by thermal neutron reactions. However, for fuel tests such a setup has the important drawback that it can only operate under the very restricted conditions imposed by the cooling circuit of the pile: nature of the fluid, pressure, mechanical constraints, temperature of the jacket. In general, one cannot consider using such a setup for the full-scale testing of actual slugs intended for an arbitrary type of pile. There is however a special field of application reserved for them, which is the initial testing of new fuel alloy on which no information on its behavior under radiation is available. In EL-2 and EL-3 many cells have been and are still being used for this purpose; this enables engineers to carry out preliminary tests to determine whether a certain type of fuel alloy is worth investigating in greater detail. Fuel elements which have been studied, or are being studied by this method range from plutonium-aluminum alloys to uranium alloys which contain a small proportion of stabilizing metals such as molybdenum, aluminum, zirconium, etc.

To give a definite idea, and to show the importance of such experimental setups let us point out that the thermal neutron flux before the introduction of the samples is 25 percent higher than the one which exists in a channel of the same diameter located in the center of a triangle whose three corners are occupied by normal fuel cells of the pile.

c. Independent Cells

These are cells that are identical with the normal cells of the pile but whose cooling circuit is entirely separated from the main circuit of the pile.

This case is thus completely identical to the previous one as far as the intensity of the thermal neutron flux is concerned. It has however the great advantage that one can carry out tests in a much wider range of thermodynamic characteristics.

Here, the nature of the coolant, its pressure and its temperature are no longer imposed by the characteristics of the research pile itself. The independent cells are thus an intermediary stage between the false cells which were considered in the previous paragraph and the specialized loops intended for final tests on fuel elements for a particular type of pile.

In the EL-2 pile, an independent gas cell is installed at the periphery of the lattice; the fluid actually used is carbon dioxide, whose pressure and flow rate can be adjusted and can reach 15 kg/cm^2 and 1200 kg/hour respectively. The same cell can be used with nitrogen or helium according to experimental needs by making only slight modifications which can be carried out rapidly.

A heavy water independent cell is being constructed for EL-3; in the pile, it will be located in a region of high flux ($\sim 10^{14}$ n/cm^2/sec) not far from the central channel. Two locations are provided for the installation of two independent cells (see Figure 2). The first cell to be installed in the pile will not be under pressure, only the flow rate of heavy water will be variable from 0 to 12 m^3/hour. The second cell will perhaps be designed to function with water under pressure or even with organic liquids if this would be necessitated by the experimental program.

In general, one can say that independent cells are very interesting for many technical and thermodynamic investigations, but is is not possible to install a large number of them in the same research pile. This limitation arises mainly from the fact that these circuits are usually rather bulky, and that one rapidly attains a prohibitive cluttering on top of the pile or in the chambers adjacent to the pile block.

d. Converter Channels

These channels, intended for irradiations with fast neutrons, consist of hollow fuel elements. The fast neutron flux which exists in the region in which the samples are placed can be easily calculated by the method presented by V. RAIEVSKI at the Noordwyck Colloquium in 1955; one can also find some information on this method in the report of R. ECKERT "Fast Neutron Dosimetry in Research Reactors."

These converter channels usually use the main circuit of the pile for the cooling of the fuel element. The cooling of the samples can usually be carried out either by the same main circuit or by an independent circuit. In EL-2, for instance, converters and samples are cooled by the main CO_2 circuit of the pile, (10 kg/cm^2 pressure). In EL-3, the converters are cooled by the main heavy water network, and the samples by a special circuit of CO_2 at 3 kg/cm^2. This operational procedure arises from the necessity to avoid the presence of moderator fluid between the fuel and the samples if one wants to avoid a decrease of the intensity of the fast neutron flux. This precaution is actually really useful only in the case of light water where each millimeter of water causes a decrease of about 13 percent in the flux of neutrons of energy exceeding 1 Mev; in the case of heavy water this loss per millimeter of water is already only 2.5 percent.

The converters installed in the lattice of piles usually have a small diameter. There are two reasons for the choice of such small dimensions. First of all, since one wants to use the main circuit of the pile for the cooling of the converter it follows that the dimensions of the latter should be rather close to those of the normal fuel elements of the pile, if one wants to avoid making important mechanical modifications. Furthermore, a reduced diameter of the fuel favors a high fast neutron flux. This is clearly shown by the curves of Figure 5, which were obtained from the report by V. RAIEVSKI, mentioned above.

The small available volume of such converters limits their use to the irradiation of small samples, without outside connections; for instance, they can be used for samples which are to be used for physical chemistry studies or for the preparation of radioisotopes resulting from a fast neutron reaction.

Converters of this type are used in the Saclay piles: two, and soon four in EL-2, two in EL-3. The effective dimensions and the fast neutron flux are as follows:

EL-2 - Effective diameter: 28 mm - effective length: 300 mm. Flux of fast neutrons with energy above 1 Mev: 0.8×10^{12} n/cm^2/sec. (Pile being operated at 2000 KW).

EL-3 - Effective diameter: 38 mm - effective length: 1300 mm. Flux of fast neutrons with energy above 1 Mev: 9.10^{12}n/cm^2/sec. (Pile being operated at 18.5 MW).

e. Inside of the Standard Fuel Cells

In piles in which the fuel elements are hollow, the spaces available in the center of these elements can naturally be used for irradiation by fast neutrons. This is the case in a pile like EL-3 where the uranium rods consist of hollow cylinders with diameters of 22 x 29 mm; it is also the case for most piles which use considerably enriched uranium-aluminum alloys. One has then exactly the same conditions as in the case of the converters which were discussed in the previous paragraph. In EL-3, for instance, the small inside diameter (22 mm) favors obtaining a high fast neutron flux; thus the flux of fast neutrons with energies above 1 Mev reaches 3.10^{13} n/cm^2/sec. in a fuel slug located at the point of highest charge of the pile when the latter operates at maximum power.

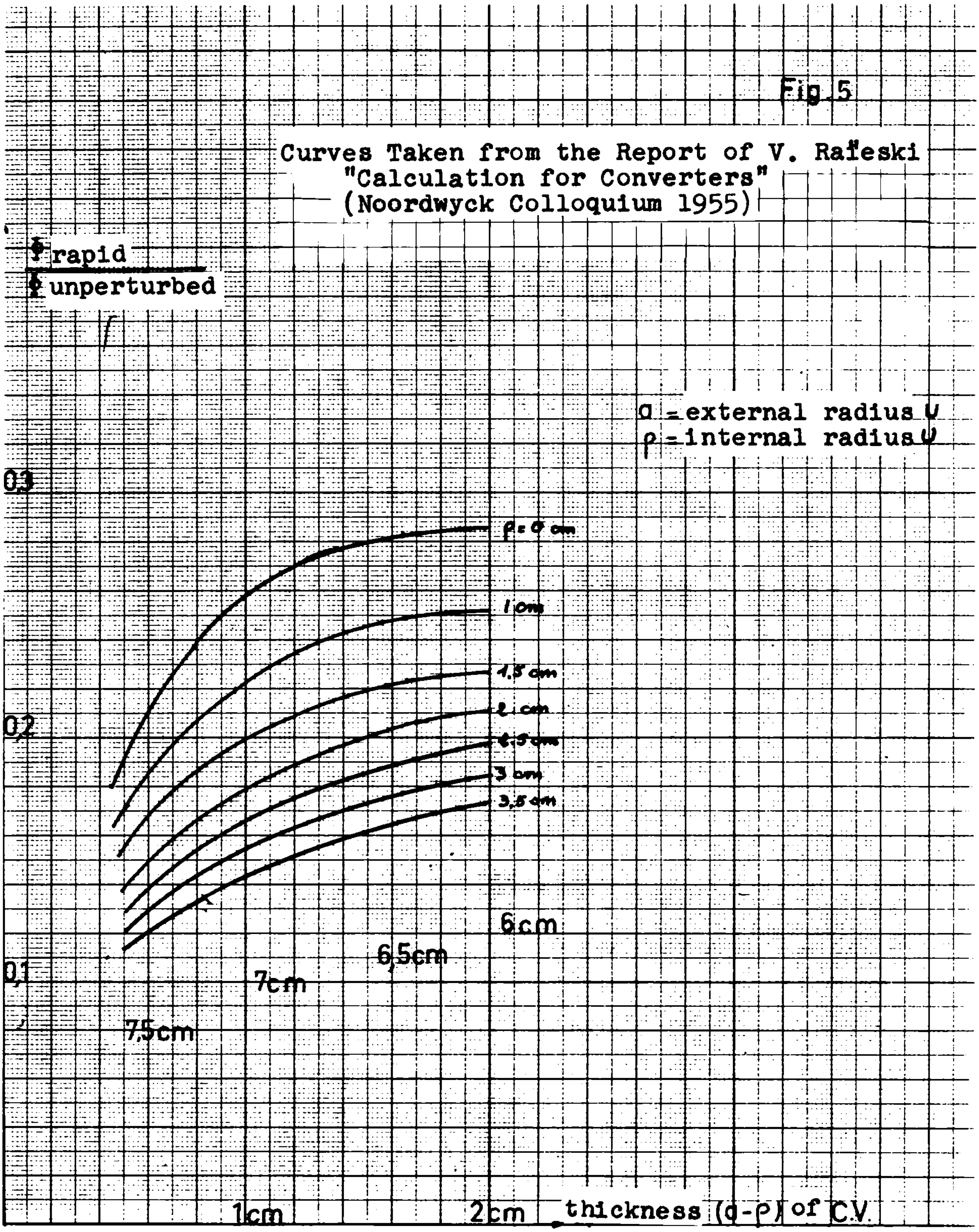


Fig.5
Curves Taken from the Report of V. Rafeski
"Calculation for Converters"
(Noordwyck Colloquium 1955)
Φ rapid / Φ unperturbed
a = external radius
ρ = internal radius
0,3
0,2
0,1
ρ = 0 cm
1 cm
1,5 cm
2 cm
2,5 cm
3 cm
3,5 cm
6cm
6,5cm
7cm
7,5cm
1cm
2cm
thickness (a-ρ) of C.V.

If the hollow cell is made of one piece and jacketed on the inside, the cooling of the converter itself and of the samples is usually performed by the main circuit of the pile; it may then be necessary to take into consideration the remark made in a preceding paragraph about the weakening of the fast neutron flux caused by the presence of the moderator between the fuel and the samples. This problem does not occur in the case of the EL-3; however another difficulty appears. Each fuel cell is made in four parts as can be seen from Figure 1, and each slug has the schematic form indicated on Figure 6. The slugs are closed at their ends; it is thus impossible to cool the samples which are enclosed within. These conditions naturally restrict the field of application, since it is impossible to impose a definite temperature on the samples, which is an important factor in many experiments. The temperature of the sample is determined by both the temperature reached by the fuel at the operational power chosen, and by the temperature gradient that exists between the samples and the fuel. This temperature gradient itself is dependent on the thermal power dissipated in the samples, as a consequence of the intense irradiations to which they are subjected, and on the unavoidable thermal resistance between the samples and the fuel. This thermal resistance can be reduced to very low values of the order of 1.5°C per watt dissipated in the sample by filling with helium, or even better, by using, for instance, aluminum foil or zirconium shavings as stuffing. The amount of thermal power dissipated by the samples depends on their nature. The following table gives as an illustration several typical values which correspond to the flux conditions of the most highly charged slug of the pile.

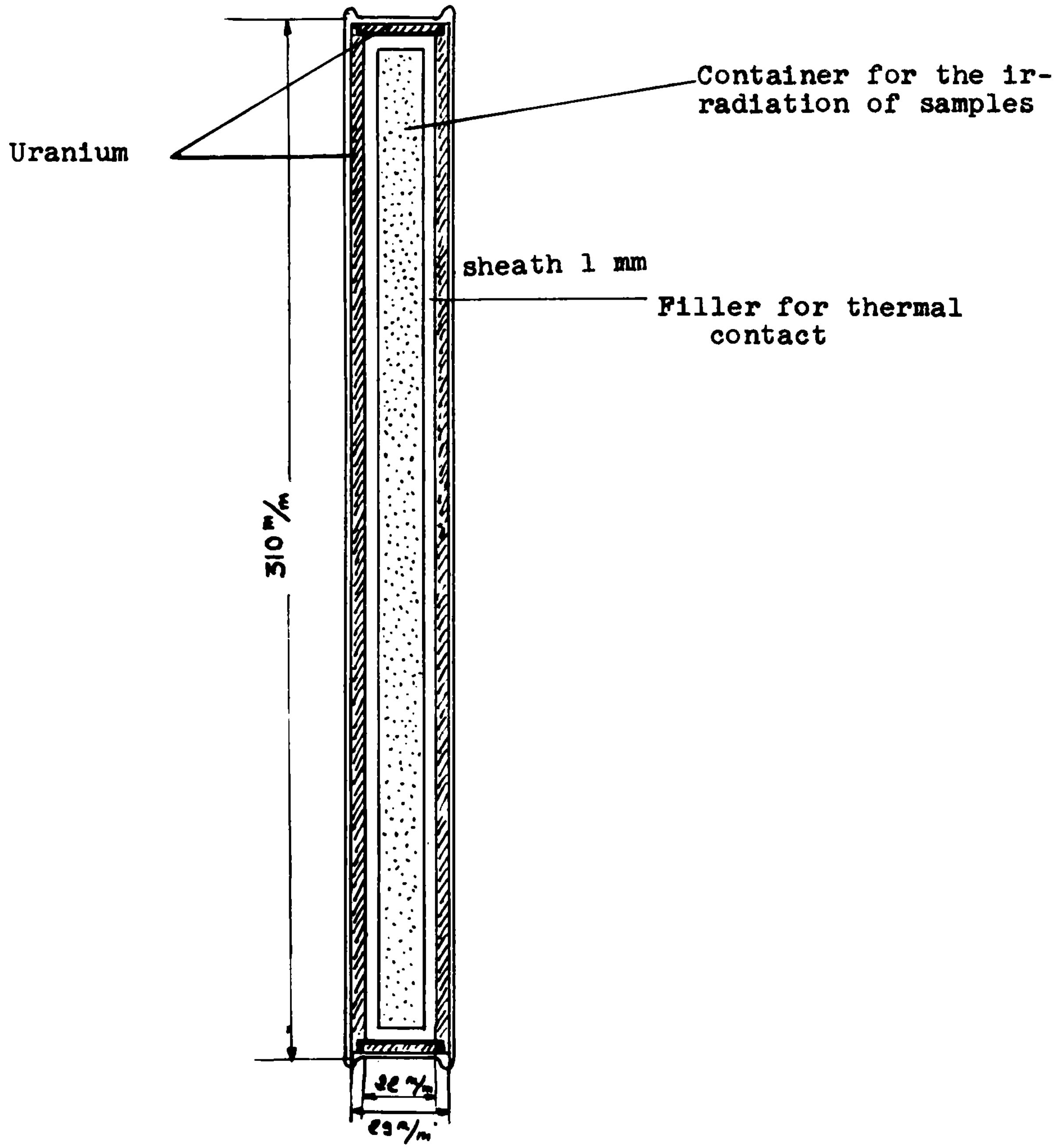
(Fast flux: 3×10^{13} - thermal flux: 7×10^{13} - gamma flux: 2×10^{8} r/h)

Nature of the substance	Al	Mg	Zr	Inox Steel 18/8	Ni	Cr	Be	BeO	Mn
Thermal power (watt/gram)	0.7	0.5	0.5	1.4	1.9	1.5	0.6	0.5	9

Summarizing one can say that the temperature of the samples irradiated inside the most highly charged slug is between 400 and 500°C when the pile operates at maximum power. In spite of this inconvenience, numerous structural materials have already been irradiated in this fashion with EL-3. Let us add that this inconvenience sometimes changes into an advantage, namely when one wants to study structural materials destined to operate at high temperatures. One can even design special slugs whose fuel has been calculated to reach the desired temperatures. In this manner one can avoid the construction of complicated loops requiring electrical heating.

Fig. 6

Schematic Representation of a Fuel Slug EL-3 Used for the Irradiation of Samples



Another inconvenience of irradiations carried out inside fuel slugs arises from the fact that the samples are tied to the life of the slug and that they can only be recovered after dismantling in a hot laboratory and eventually decontaminating.

The fuel elements of EL-3 have been constructed in this manner because at the time of their manufacture the internal jacketing of hollow cylinders presented difficult technological problems. Today, the advances made in this field will enable us to consider in the future cells made of one piece jacketed on the inside and on the outside; it will then be possible to use these cells in the same manner as the vertical converters discussed in the preceeding paragraph.

f. Channels Located at the Boundary of the Lattice and the Reflector

The interest of such channels lies mainly in the use of the high level of thermal neutron flux which exists in this region. The region is not suited for irradiations by fast neutrons, in spite of the proximity of fuel cells; channels located in the lattice are much better suited for this purpose. On the other hand all irradiations necessitating large thermal neutron fluxes can be carried out in this location: study of fuels, manufacture of radioisotopes, etc... Consideration of available space can also play a certain role; certain experiments which are too cumbersome to be carried out inside the lattice of the fuel elements, especially if the latter is compact, can be carried out much more easily in this peripheral zone.

The amount of increase of the thermal neutron flux depends on the leakage from the core and on the deceleration and diffusion properties of the reflector. In order to increase the leakage of thermal and fast neutrons, the size of the core should be small and the migration area unit should be big; the reflector should have a moderating power $(\frac{\xi\Sigma_s}{\Sigma_a})$ as large as possible.

One can conclude from the above that piles with light water and having a small volume would be particularly suitable as far as the utilization of the peripheral zones is concerned, provided that the channels are located very close to the core: under good conditions it will then be possible to obtain for the samples thermal neutron fluxes that are close to those that exist in the center of the core. From the point of view that concerns us here, these piles would still be improved by using a reflector with better moderating power than light water. The PEGASE project which is at present investigated by the Mathematical Physics Department is based on this principle.

The advantage is smaller in the case of heavy water piles, however it is great enough to justify the installations of such channels. In EL-3, eight isotope channels and four channels of 48 mm (designated DS-4 to DS-7) are located in the heavy water reflector, close to the boundary of the core; though they are located a little beyond the maximum of the flux increase, a thermal neutron flux of the order of 3×10^{13} n/cm^2/sec. is available when the maximum flux of the lattice is 10^{14} n/cm^2 sec.

III. CONCLUSIONS

After having examined the advantages and disadvantages of the different types of channels which one can encounter in the core of research piles, we shall now attempt to summarize the conclusions which we have reached. We will attempt to define the installation and geometry conditions which each type of channel has to satisfy so that the conditions for the different categories of experiments will be optimal. For this purpose we classify the experiments into three groups:

- Irradiation with fast neutrons
- Irradiation with thermal neutrons
- Gamma irradiation.

a. Irradiations with Fast Neutrons

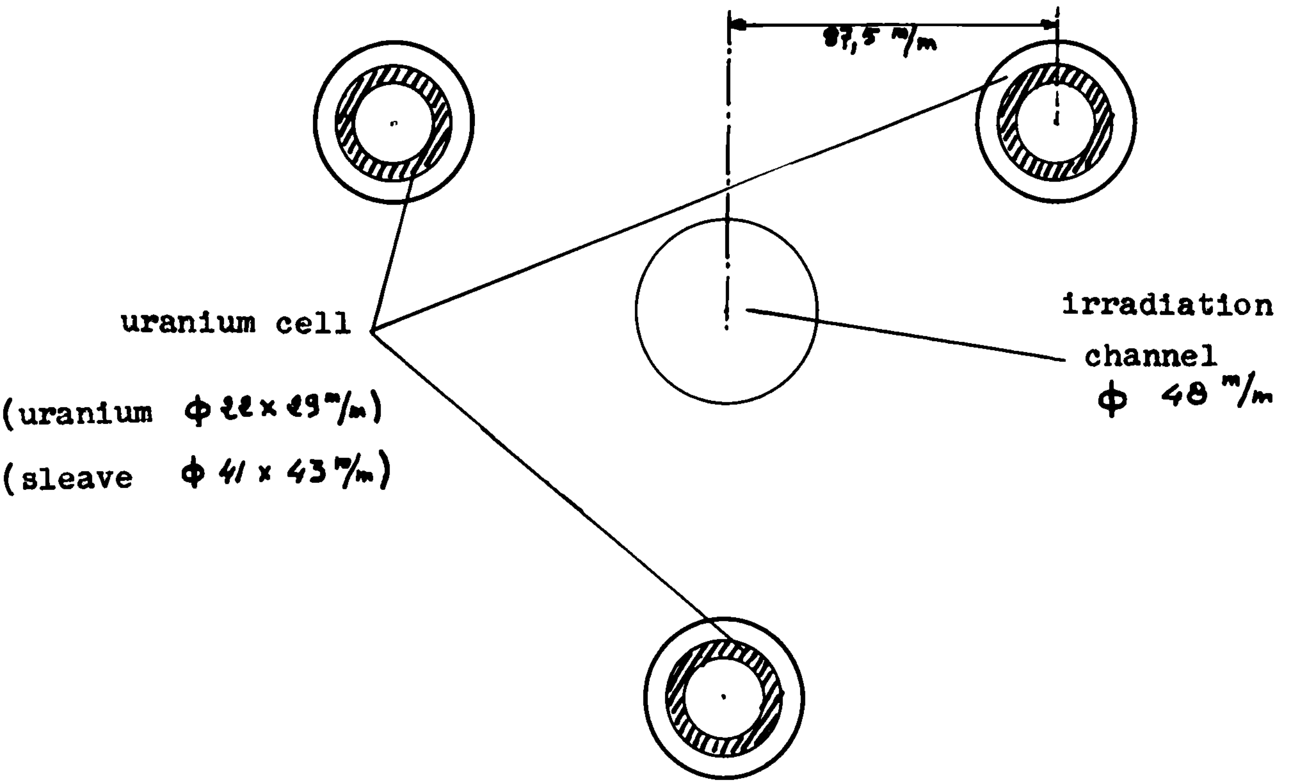
We have seen that channels located in between the fuel elements are poorly suited for this purpose. Converters are much better suited, but those that are installed in the lattice of piles are usually of small dimensions, and this reduces their field of application; however let us recall that it is their small size which makes it possible to obtain a very high fast neutron flux. Such converters could thus be reserved for the irradiation of small samples which requre a high fast neutron flux. For this application the best solution would be to make the fuel elements of the pile in the form of hollow cylinders (see §II-e). If such a use is planned from the beginning of the project of the pile it will be easy to design if necessary, a cooling circuit for the samples which is independent from the main circuit of the pile; in this manner one can eliminate the inconveniences which result from the presence of moderator between the fuel and the samples.

Another solution has to be found for fast neutron irradiations which require a larger volume than the one available in the center of the standard fuel elements. In our opinion this can be done by making some hollow fuel cells whose dimension would be imposed by the experimental needs. Such cells lead to a less favorable use of uranium from the point of view of the reactivity, but they improve considerably the possibilities of irradiation with fast neutrons as compared with the channels located between the fuel elements. As an illustration we shall evaluate the improvement factor in a particular case. The first diagram of Figure 7 represents a channel with a diameter of 48 mm located between three uranium cells; such a channel exists in the EL-3 pile and is designated DS (see Figure 2). The following fluxes exist in this channel when the pile operates at maximum power:

- fast neutrons: 10^{12} n/cm^2/sec.
- thermal neutrons: 10^{14} n/cm^2/sec.

In the second diagram of Figure 7 the DS channel has been left out and one of the standard cells which surrounded it has been replaced by a special cell whose dimension is large enough so that the effective diameter reserved for the irradiations is 48 mm as in the previous case. The definition of the special cell (quantity of uranium) has been carried out in both cases corres-

EL-3 Fast Neutron Irradiation Channel



Substitution of the Previous Channel by a Special Converter-Cell

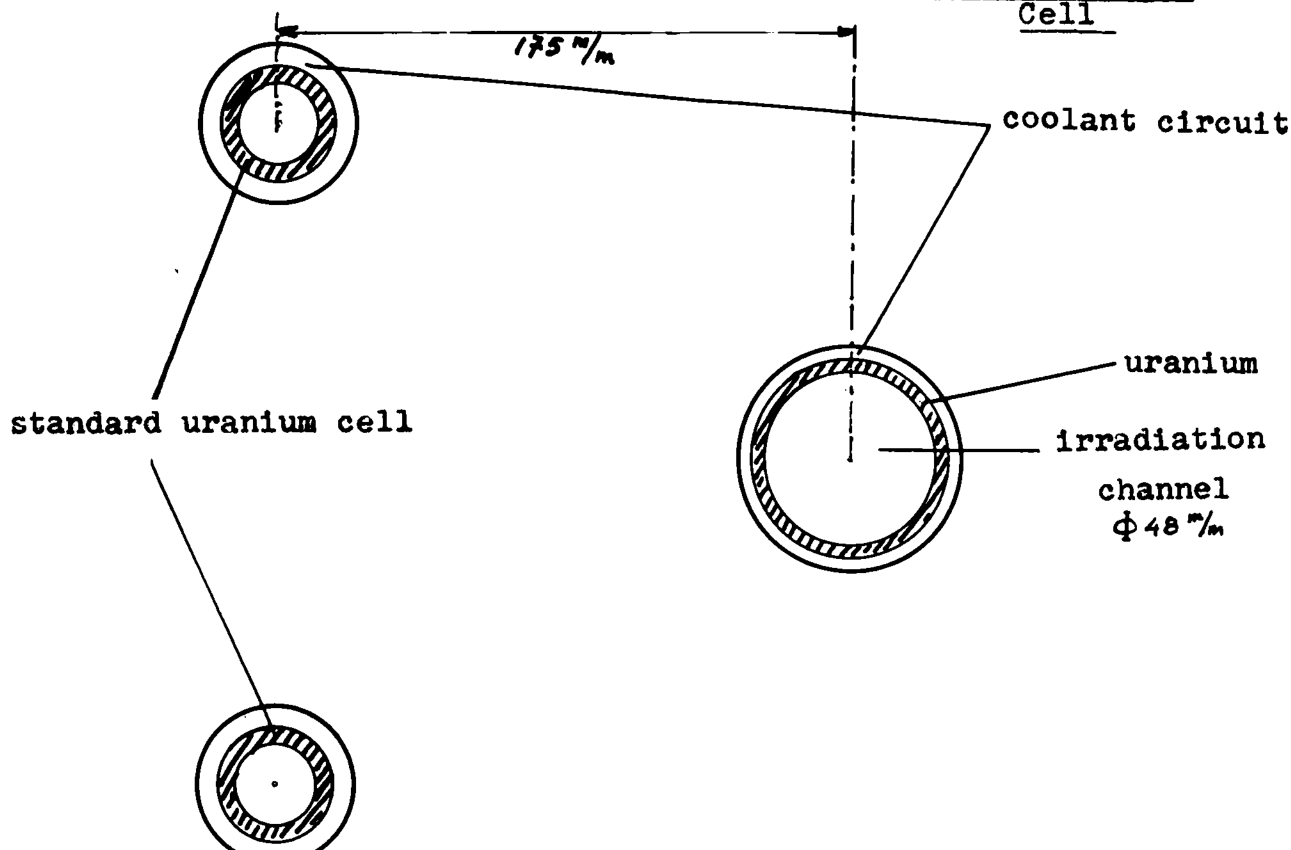


Figure 7

ponding to weak perturbations of the normal lattice of the pile.

- The power liberated in the special cell must be equal to that of the normal cell ($\Delta P = 0$).
- The reactivity introduced by changing the cell must be equal to zero. ($\Delta\rho = 0$).

For $\Delta P = 0$ one has the following characteristics:

- Fast neutron flux: 1.7×10^{13} n/cm^2/sec.
- Thermal neutron flux: 7.3×10^{13} n/cm^2/sec.

Effect on reactivity: $\Delta\rho = -70$ pcm

For $\Delta\rho = 0$ one has:

- Fast neutron flux: 1.8×10^{13} n/cm^2/sec.
- Thermal neutron flux: 6.6×10^{13} n/cm^2/sec.
- Power variation: $\Delta P = +22\%$ (or + 45 KW for a central EL-3 cell).

One can see that by making certain slight modifications of the characteristics of the lattice it is possible to improve considerably the conditions of irradiation with fast neutrons. It is to be noted that the recommended procedure does not only increase the available fast neutron flux, but also results in a smaller action of the thermal neutron flux on the samples. This latter effect can also be considered as an improvement, since in this manner the samples will have a smaller effect on the reactivity of the pile.

In view of the above one of the possible projects for the remodeling of the EL-2 pile has been investigated. In this project, the lattice (shown schematically in Figure 8) has 12 cells of large diameter in its center; the Laplacian of these cells is quite bad (3.65 m^{-2}). As a compensation, the exterior part of the lattice has been set up with slightly enriched uranium (1.35 percent). The cells of the outer region can be made in the form of hollow cylindrical elements so that they can be used for the irradiation of small samples. The macroscopic neutron flux in the entire lattice is shown in Figure 9.

Inside the cells of the central region, the fission neutron flux reaches a value of 4.4×10^{12} n/cm^2/sec. for a total power of the pile of 2,500 KW.

b. Irradiations with Thermal Neutrons

The channels best suited for these irradiations are -- as shown in paragraph II-a -- located in between or in the place of fuel elements. We have also seen -- in paragraph II-f -- that with certain reservations channels located at the periphery of the core are also suited for this use.

Channels for irradiation with thermal neutrons have to be designed differently depending on whether they are intended for small samples (preparation of radioisotopes for instance) or for large loops (investigation of fuels, for instance). In the first case usually small dimensions suffice; these favor a high flux. A rather large volume is always required for loops; one will thus decide on the smallest possible channel which is compatible with the dimensions of the planned loop. In this manner one obtains the highest possible thermal neutron flux. However, another fact has to be taken into consideration here. Most of these loops

Fig. 8

Project of Remodelling EL-2

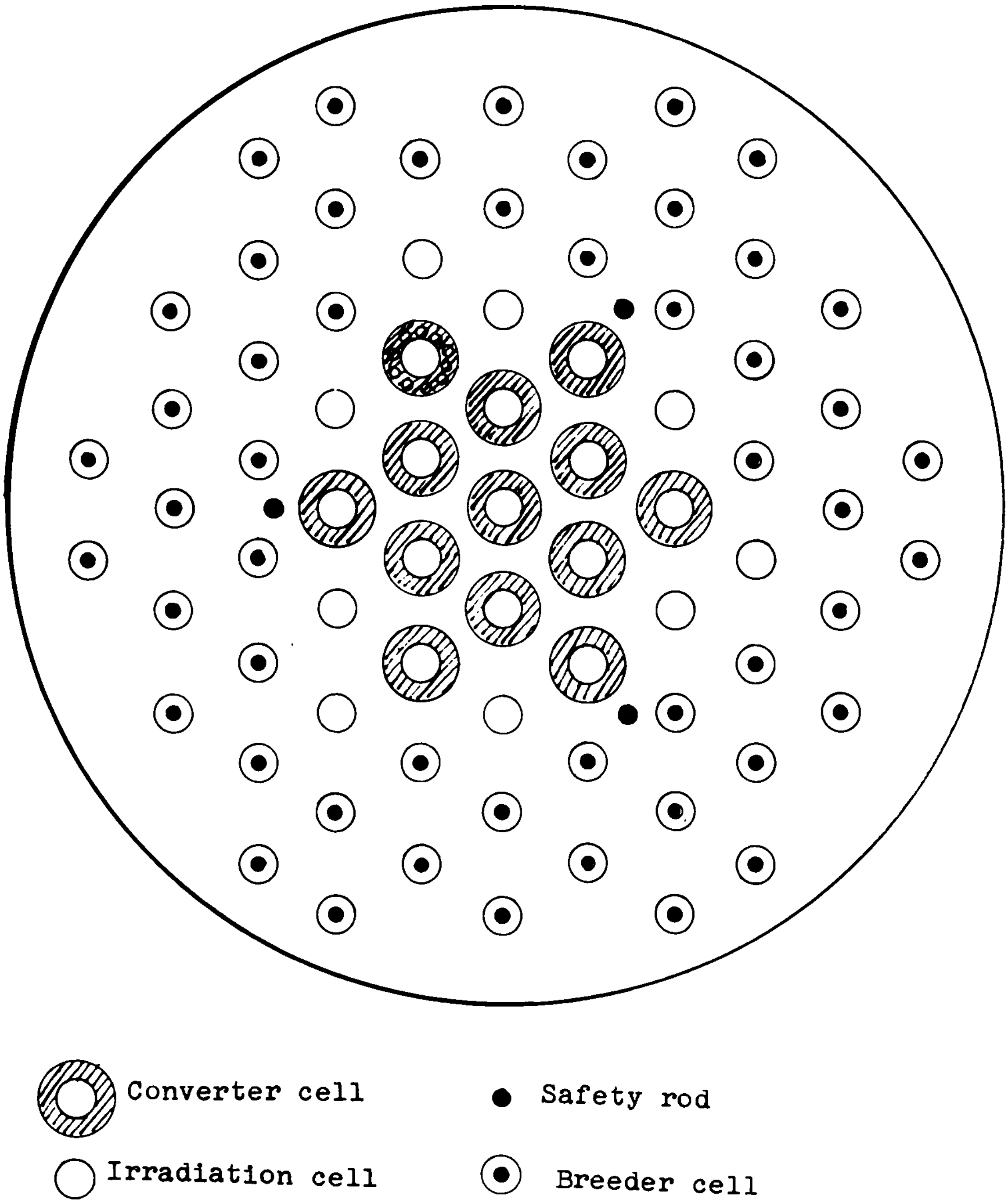
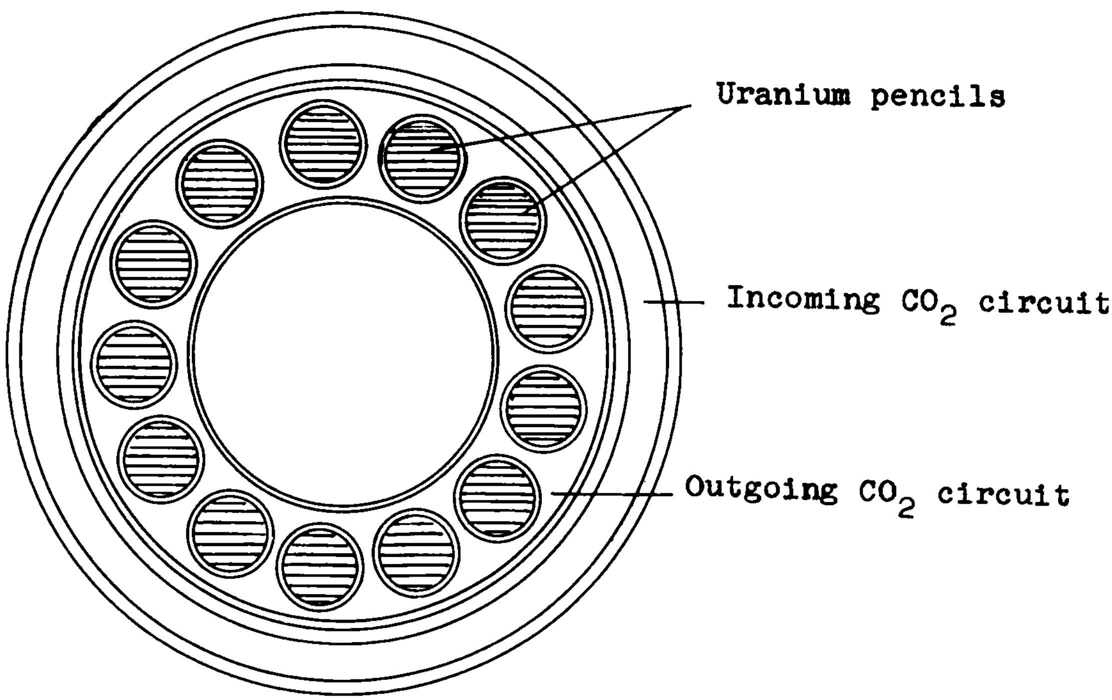
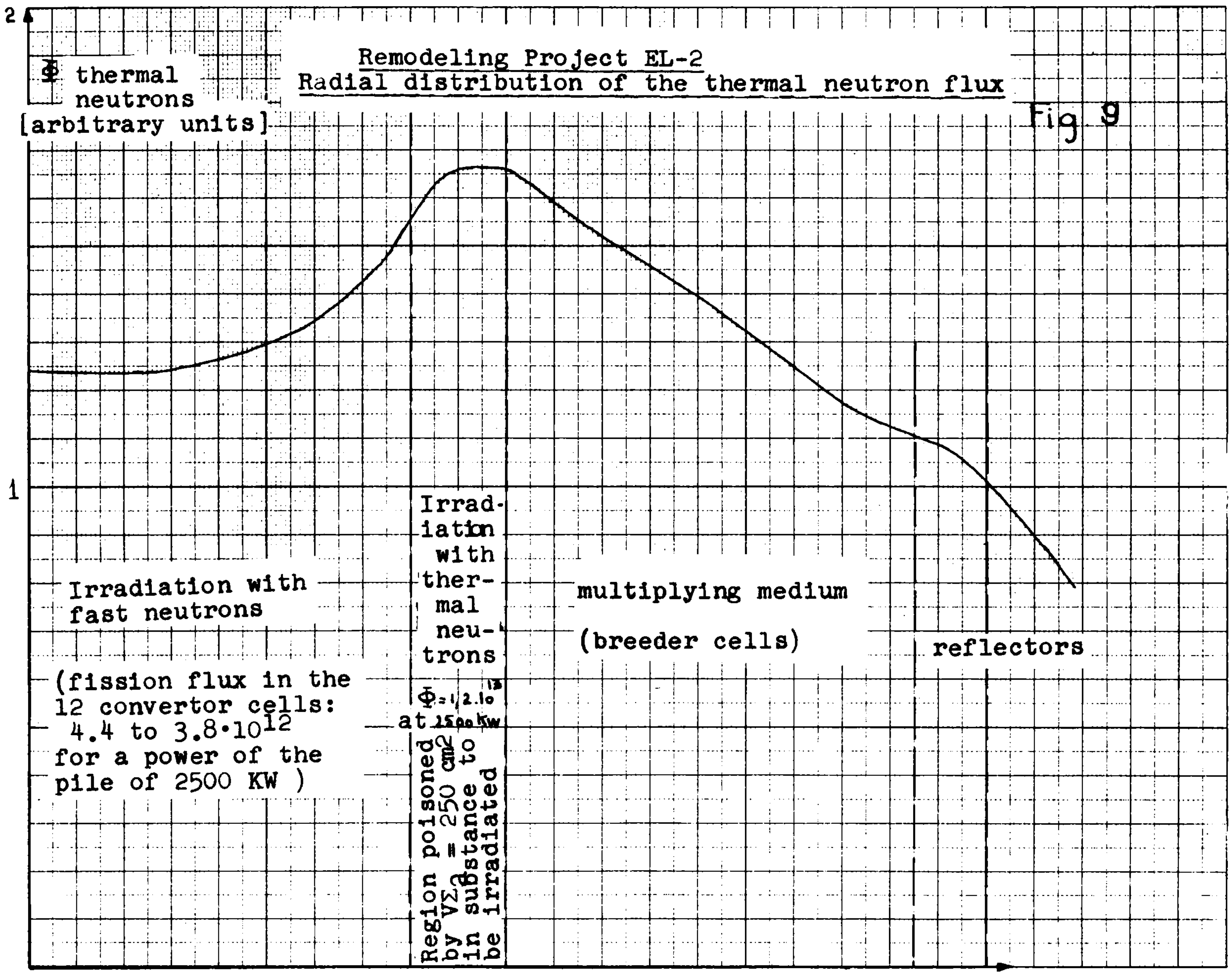


Fig 8 (2)

Converter Cells Remodeled EL-2




Remodeling Project EL-2
Radial distribution of the thermal neutron flux
Fig 9
Φ thermal neutrons [arbitrary units]
2
1
Irradiation with fast neutrons
(fission flux in the 12 convertor cells: 4.4 to 3.8·10^12 for a power of the pile of 2500 KW)
Irradiation with thermal neutrons
at
Region poisoned by VΣa = 250 cm2 in substance to be irradiated
multiplying medium
(breeder cells)
reflectors

have a pressure tube which is intended to withstand the pressure at which one wants to operate, as well as a number of construction tubes which are needed for the circulation of coolants and necessary insulations. All these tubes can be made out of different materials (stainless steel, aluminum, magnesium, zircalloy, etc...) whose effect on the reactivity of the pile and on the absolute value of the flux that is actually available for the samples can vary within wide limits. Materials that are less neutron absorbing usually have mechanical properties that make it necessary to use them with a considerable thickness, which leads to an increase in the size of the irradiation channel. It can be interesting in certain cases to make a channel whose dimensions are somewhat larger, in order to use these weakly absorbing materials rather than to restrict oneself systematically to channels of minimum dimensions which necessitates the use of materials with large mechanical resistance. As an example we will mention the project of the gas loop for the investigation óf fuel elements considered for the central channel of the EL-3 pile. A first version of this loop consisted of a stainless steel pressure tubing with a diameter of 124 x 131 mm; the effect of this structure on the reactivity of the pile would have reached about 6000 pcm and the average thermal neutron flux on the fuel slug would have been of the order of 10^{13} n/cm^2/sec. In another version using a pressure tube of AG3 alloy (aluminum + 3 percent magnesium) with a diameter of 122 x 136 mm, the effect on the reactivity of the pile is now only 700 pcm and the average flux in the fuel carriage reaches 5.6×10^{13} n/cm^2/sec. Let us mention that these values are obtained from calculations based in both cases on the same channel (central channel EL-3 - effective diameter: 180 mm). One could adjust the channel in each case to the outer dimensions of the chosen pressure tube; the numerical results would be slightly changed with an increase of flux and a decrease of antireactivities. The conclusion would however definitely be in favor of an aluminum pressure tube.

c. Gamma Irradiations

In a given volume element, the intensity of the gamma flux is directly related to the density of the fission in the surrounding regions. Consequently, gamma flux will be more intense wherever fast neutrons are also more intense. Channels intended for irradiations with fast neutrons are also suited for gamma irradiations. For different types of piles, the greater the specific power in the fuel elements, the greater will be the maximum gamma intensities. This means that the same power, piles of small volume (which use highly enriched uranium) will make a more intense gamma flux available, just as they provide a high fast neutron flux; however, the experimental volumes will naturally be smaller.

The need for gamma irradiations occurs usually in ionization effect research; thus, these irradiations will usually be not at all affected by the presence of fast neutrons since these increase the ionization density through the presence of recoil nuclei which they produce. On the other hand it might be necessary to carry out certain gamma irradiations in the absence of thermal neutrons if one wants to avoid the activation of the irradiated samples. In this case it is necessary to surround the channel with cadmium for instance. This would be impossible to do in the core of the piles because of the too large antireactivity which such structures would bring about. It will then be necessary to operate in the reflector, at a rather great distance from the core of the pile. In these regions the fast neutron flux is practically zero, and the cadmium jacketing technique allows us to obtain a practically pure gamma flux. One can also consider the use of the capture reactions of the thermal neutrons in substances used for jacketing in order to increase the gamma flux on the samples. This seems rather unrealistic because of the relative values of the gamma flux and the thermal neutron flux which exists in the reflector of piles.

A calculation made for one of the tangential channels of the EL-3 pile gives the following results:

- Gamma dosage without jacketing: about 3.5×10^6 roentgen/hour.
- Gamma dosage with cadmium jacketing (0.7 mm thickness): previous dosage + 4.10^4 roentgen/hour.

This calculation indicates that cadmium is not an interesting substance for the increase of the flux of gamma rays due to capture of thermal neutrons. On the other hand with a stainless steel jacket of sufficient thickness (3 mm) to cut by a half thermal neutron flux, the gamma dosage in the channel reaches 5×10^6 roentgen/hour (increase of about 40 percent); this effect is mainly due to the large energy of the gammas due to capture in the iron, and to the increase of the number of captures. However, if one wants to suppress absolutely the thermal neutrons, the thickness of the steel becomes such that the effective diameter of the channel is considerably reduced. It thus seems impossible to achieve both a considerable increase of gamma flux and an almost total suppression of thermal neutrons without too great a decrease in the effective volume of the channel.

d. For all the types of channels that we have considered, whether for the irradiations with fast neutrons, thermal neutrons or gamma, the design of their geometry and of their total number for the entire pile has to be the subject of a detailed investigation which takes into consideration not only the total available reactivity of the pile but also the effect which these channels - charged or not - can have on the fuel elements or on the control and measuring elements which surround them (flux perturbation). It is also necessary to examine the kinetic behavior of the pile in case of rupture of the channels; for instance the content of heavy water of the central channel of the EL-3 would release about 1500 pcm; this implies that very strict safety precautions must be taken for loops that are installed there.

We thank Mr. DESANDRE-NAVARRE, Mr. ECKERT and Mr. MARFAING for their help in the preparation of this report.

FAST NEUTRON DOSIMETRY IN RESEARCH REACTORS

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I. INTRODUCTION

The experimental program of research reactors includes always a large number of irradiations with fast neutrons.

These irradiations are performed within the framework of varied studies but the majority fit into one of the following three categories:

- Study of the change of the properties of the moderator substances (Wigner effect on graphite, beryllium content, etc...).
- Study of the change of the mechanical properties of the structural materials.
- Study of radiochemical decompositions.

The zones where the irradiations take place, are of two types:

- Converters placed in the moderator or in the breeding medium of a reactor.
- Irradiation tubes placed in the breeding medium.

Whatever the chosen irradiation zone may be, one has to know in every case the intensity and the spectrum of the fast neutrons at that point in order to relate the observed effect to the cause that has produced it.

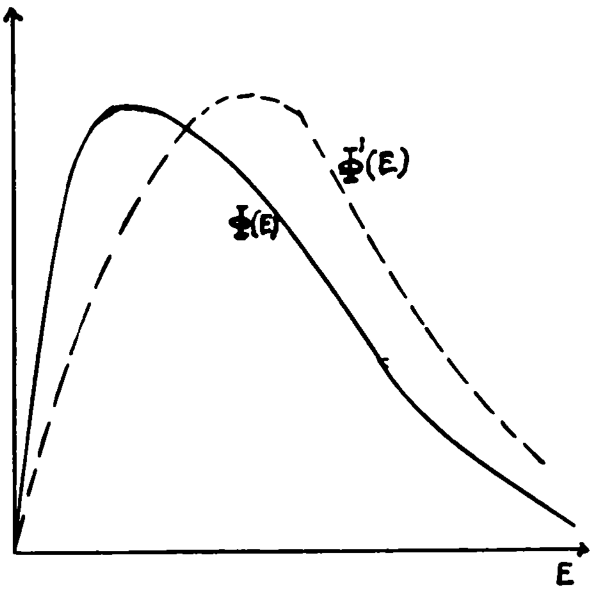
The object of this report is to show the results obtained at the Service of Large Reactors of Saclay, in the dosimetry of fast neutrons.

We will also show how we think it feasible to improve the present methods by determining experimentally the shape of the neutron spectrum.

In the following only the case of threshold detectors will be considered.

The underlying principle of the measuring method with such detectors is well known; we will review it briefly, in order to show what difficulties are met in its practical application.

At the instant of their emission, after the fission of the uranium atoms, the neutrons have a definite energy spectrum:



$$\Phi(E) = A e^{-E} \,\mathrm{sh}\, \sqrt{2E}$$

This spectrum is represented by the heavy-line curve on the diagram. However, before reaching the zone where they are used for the irradiation of samples, these neutrons can collide on the materials that are between their point of origin and the irradiation zone (uranium, structural materials, moderator in some cases). Therefore the actual spectrum in the irradiation zone will differ, in general, from the virgin fission spectrum; it may, for instance, follow the dotted curve denoted by $\Phi'(E)$ in the diagram. If E_s is the threshold of the detector used, it is known that the activity that it will take on can be written:

$$A = K \int_{E_s}^{\infty} \Sigma(E)\, \Phi'(E)\, dE$$

K being a proportionality coefficient depending on the characteristics of the detector.

The flux to be determined equals:

$$\Phi_{total} = \int_0^{\infty} \Phi'(E)\, dE$$

and the flux acting on the detector is:

$$\Phi_2 = \int_{E_s}^{\infty} \Phi'(E)\, dE$$

It is therefore necessary to know the shape of $\Phi'(E)$ between zero and infinity in order to evaluate the ratio

$$R = \frac{\Phi_{total}}{\Phi_2}$$

Φ_2 will be determined from the measurement of the activity A and the average effective cross-section $\bar{\Sigma}$ defined by:

$$A = K\bar{\Sigma}\Phi_2 = K\int_{E_s}^{\infty}\Sigma(E)\Phi'(E)\,dE$$

i.e.

$$\bar{\Sigma} = \frac{\int_{E_s}^{\infty}\Sigma(E)\Phi'(E)\,dE}{\int_{E_s}^{\infty}\Phi'(E)\,dE}$$

The calculation of $\bar{\Sigma}$ requires therefore that the spectrum $\Phi'(E)$ be known.

Summarizing, the determination of the total rapid neutron flux will consist of the following operations:

- Determination of the spectrum between E = 0 and E = ∞
- Calculating $\bar{\Sigma}$
- Measuring the activity A of the detector and determining Φ_2 from the relation:

$$\Phi_2 = \frac{A}{K\bar{\Sigma}}$$

- Calculating the ratio $R = \frac{\Phi_{total}}{\Phi_2}$ from the spectrum $\Phi'(E)$.
- Calculating the total flux: $\Phi_{total} = R\,\Phi_2$

We can see, therefore, that the determination of the rapid neutron flux by means of the threshold detectors, without prior knowledge of the energy spectrum of these neutrons would be unrealistic.

In the first part of the following report, we will show how it is possible to calculate first the neutron spectrum, and subsequently the average effective cross section $\bar{\Sigma}$. We will then investigate the experimental results obtained by this method.

In the second part, we will show how it is possible to determine experimentally the neutron spectrum by means of threshold detectors and resonating detectors.

The outline of this report will therefore be as follows:

II - <u>Dosimetry Methods by Threshold Detectors, Based on a Calculated Spectrum</u>
II - a - Calculation of the Spectrum
II - b - Effective cross-sections to be used in the dosimetry with threshold detectors
II - c - Results
III - <u>Experimental Determination of the Neutron Spectrum</u>
III - a - Spectral distribution of the neutrons after collision

with a light or heavy water moderator.

III - b - Spectrum determination by means of threshold and resonating detectors.

Remark -

The result of the calculation such as it was given above may also be written:

$$\Phi_{total} = R\,\Phi_2 = \frac{\int_0^\infty \Phi'(E)\,dE}{\cancel{\int_{E_s}^\infty \Phi'(E)\,dE}} \cdot \frac{A}{K} \cdot \frac{\cancel{\int_{E_s}^\infty \Phi'(E)\,dE}}{\int_{E_s}^\infty \Sigma(E)\,\Phi'(E)\,dE}$$

$$\Phi_{total} = \frac{A}{K\,\bar{\Sigma}'}$$

with $$\bar{\Sigma}' = \frac{\bar{\Sigma}}{R} = \frac{\int_0^\infty \Sigma(E)\,\Phi'(E)\,dE}{\int_0^\infty \Phi'(E)\,dE}$$

noticing that

$$\int_{E_s}^\infty \Sigma(E)\,\Phi'(E)\,dE = \int_0^\infty \Sigma(E)\,\Phi'(E)\,dE$$

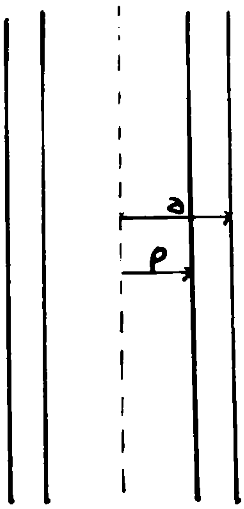
since $\Sigma(E) = 0$ for $E < E_s$.

In the literature on dosimetry with threshold detectors $\bar{\Sigma}'$ is most frequently used. In the following pages we will therefore use $\bar{\Sigma}'$ in order to be able to compare our calculated values with those published so far.

II - DOSIMETRY METHOD BY THRESHOLD DETECTORS, BASED ON A CALCULATED SPECTRUM - [1]*

II - a - Calculation of the Undisturbed Neutron Spectrum 1

1°/ Fast Neutron Flux in a Converter



Let us consider a converter of inside radius ρ, outside radius α, and let us calculate the flux at the converter axis. We will assume that the converter is placed in an infinite medium. The fast neutron flux is given by the expression:

$$\Phi(E)\,dE = S_0 \frac{P_f(E)}{1 - Z_f(E)\,\mathcal{P}_f(E)} N(E)\,dE \quad (1)$$

N(E)dE being the number of fission neutrons emitted per unit time in the interval E, E + dE, and

S_0 the number of these neutrons originating per cm^3 in the uranium, and

$$P_f(E) = \frac{1 - k(a-\rho)\Sigma_t(E)}{\Sigma_t(E)} = \frac{1 - \int_0^{\frac{\pi}{2}} e^{-\frac{a-\rho}{\cos\theta}\Sigma_t(E)} \cos\theta\, d\theta}{\Sigma_t(E)}$$

being the probability that a neutron of energy E will penetrate through a sphere of unit cross-section, placed on the converter axis (2), and

$$\mathcal{P}_f\left[\frac{a^2-\rho^2}{a}\Sigma_t(E)\right] = \mathcal{P}_f(E)$$

the probability of collision for a neutron of energy E inside the converter (3), and

$$Z_f(E) = \frac{\nu\Sigma_f(E) + \Sigma_s(E)}{\Sigma_t(E)}$$

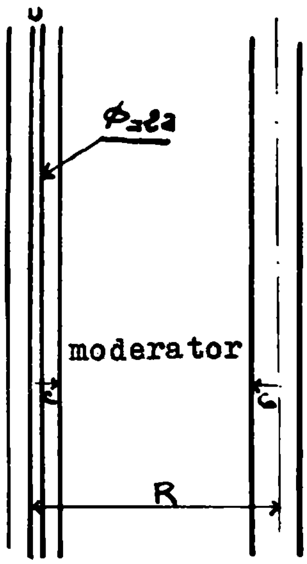
Σf, Σs, Σt being respectively the fission, diffusion, and total macroscopic effective cross-sections.

The calculation of the expression $\frac{P_f(E)}{1 - Z_f(E)\,\mathcal{P}_f(E)}$ shows that it varies only slightly as a function of the energy for the uran-

* Numbers in parentheses refer to the bibliography.

ium thickness commonly used in converters. We can therefore assume that the neutron spectrum in a converter follows the distribution N(E) of fission neutrons.

2°/ Calculation of the Fast Neutron Flux in an Irradiation Tube



Let us consider a uranium rod of radius a placed inside a tube of radius c and let us calculate the flux of undisturbed virgin neutrons at the axis of the tube of radius Co. We will assume these tubes to be infinitely long and without a longitudinal gradient of flux. Let R be the distance between the axes of the two tubes and πf the probability for a fission neutron to escape from the uranium rod. πf is practically independent of the energy of the neutrons as was shown in the previous paragraph. The number of neutrons escaping a section of unit height will be given by:

$$S_f = \pi_f \nu \Sigma_f \bar{\Phi}_{th} \pi a^2$$

$\bar{\Phi}_{th}$ being the mean thermal flux in this section.

We will have now at the axis of the irradiation tube:

$$(2) \qquad \Phi(E)dE = \frac{S_f}{2\pi R} \left\{ \frac{\pi}{2} - Ki_0\left[(R-c-c_0)\Sigma_m(E)\right] \right\} N(E)dE$$

with

$$Ki_0(x) = \int_0^x K_0(t)dt = \frac{\pi}{2} - \int_0^{\frac{\pi}{2}} e^{-\frac{x}{\cos\theta}} d\theta$$

$\Sigma_m(E)$ being the effective collision cross section of the moderator for neutrons of energy E.

The curves 1, 2, 3, 4, 5 give respectively the spectral distributions:

1 - of the fission spectrum according to Watt (4)

$$N(E) = A e^{-E} \operatorname{sh} \sqrt{2E}$$

2 - of the fission spectrum according to R. B. Seachman (5)

$$N(E) = B\sqrt{E}\, e^{-\frac{E}{1,29}}$$

3 - of the spectrum in a converter
4 - of the spectrum in the central channel of reactor EL-3
5 - of the spectrum in the central channel of reactor EL-2

All these curves have been normalized to the same maximum in order to demonstrate the displacement of the maximum along the energy axis.

II - b - Effective Cross Sections to be Used for the Dosimetry Using Threshold Detectors

The detector's activity is expressed by:

$$A = V \bar{\Sigma}' \Phi (1 - e^{-\lambda t})$$

A = activity
V = detector volume
λ = radioactive decay constant of the substance formed
t = irradiation time
$\bar{\Sigma}'$ = activation cross section of the substance used
Φ_{total} = fast neutron flux

In the introduction we saw that $\bar{\Sigma}'$ was defined by the expression:

$$\bar{\Sigma}' = \frac{\int_0^{\infty} \Sigma(E) \Phi'(E) dE}{\int_0^{\infty} \Phi'(E) dE}$$

The effective cross-sections in paragraph II - c - have been calculated by this equation. The spectral distribution of $\Phi'(E)$ used has been determined in each case by the method described in paragraph II - a -.

II - c - Results -

We applied the methods described above to the following three irradiation zones:

- Central channel of EL-3
- Central channel of EL-2
- Vertical converter of EL-2 (situated at the periphery of the grid)

The first two cases correspond to irradiation tubes placed between the fuel elements; the spectrum of the fast neutron flux is therefore expressed by the equations of paragraph II - a - 2 -.

The third case corresponds to a regular converter. The neutron spectrum is thus a pure fission spectrum (II - a - 1).

We have used the reaction S^{32} (n, p) P^{32} for measuring fast neutron flux. In each case we have used the calculated effective cross-section $\bar{\Sigma}'$.

In the table below the values for $\bar{\sigma}'$ ($\bar{\Sigma}' = N\,\bar{\sigma}'$) as well as Φ calculated and Φ measured are given.

Φ measured is determined by the method given in the introduction; it will be recalled that this method includes the measurement of sulfur activity and the calculation of the spectral distribution of the flux.

Φ calculated is also determined from the calculation of the spectral distribution of the flux, but sulfur activity measurement is replaced here by the calculation of the absolute value of the fast neutron flux from the intensity of the fission sources in the uranium, i.e., from the mean value of thermal neutron flux in this uranium.

	Converter EL-2	Central Channel EL-2	Central Channel EL-3
$\bar{\sigma}'$ sulfur	60 mb	124 mb	101 mb
Φ calculated neutrons/cm^2/sec	1.1 10^{12}	2.4 10^{10}	1.6 10^{12}
Φ measured neutrons/cm^2/sec	1.2 10^{12}	?	1.6 10^{12}

The experimental results that we have obtained so far are in excellent agreement with the measurements. However, the results would have been less satisfactory with a detector of threshold energy lower than the threshold energy of sulfur, because a part of the slowing down spectrum would have participated in the activation reaction. This is the reason we are trying to work out a method that will enable us to determine the fast neutron spectrum experimentally (see § III).

<u>Remarks</u>:

It should be stressed that the value $\bar{\sigma}' = 60$ mb that has to be used inside a converter is in perfect agreement with the value 60.3 mb, determined experimentally by R. Richmond (Nucleonics, Vol 17, No 1, January 1959). This is a good verification of our method of calculation, for R. Richmond's measurement was carried out inside a converter, i.e., with a neutron flux having a spectrum of fission neutrons, as we have seen in paragraph II - a - 1.

We are not able to compare our calculations to the other two experimental effective cross sections for the reaction S^{32} (n, p) P^{32} in the same table of Nucleonics, because the method of measurement used are not given. In any case, it is likely that those who made these measurements worked with a spectrum rather different from the fission spectrum, judging by the values they indicate:

$\bar{\sigma} = 154$ mb (reference 16 in Nucleonics)
$\bar{\sigma} = 21$ mb (reference 7 in Nucleonics).

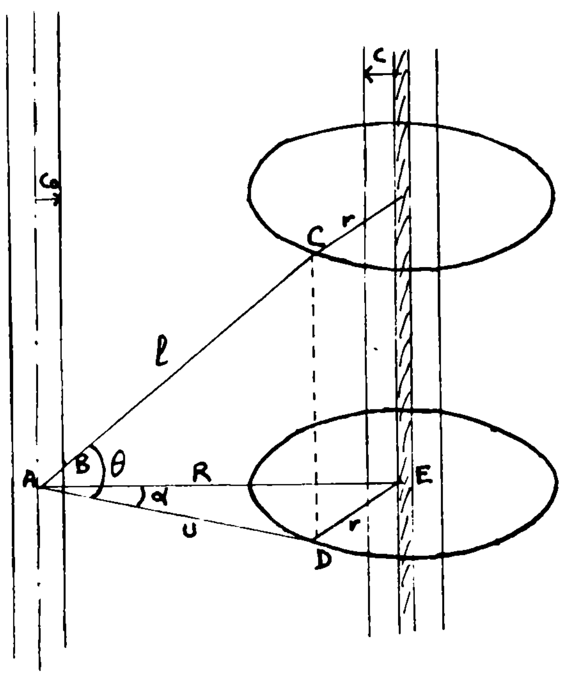
As for the value given by D. J. Hughes (30 mb), it would seem to us that its discrepancy with the value of 60 mb stems from the fact that the effective cross section for the effective threshold (3.5 Mev) was taken as 65 mb, whereas according to the BNL 325, it should be around 150 mb. The calculation of D. J. Hughes carried out on the basis of the latter value of 150 mb would give σ = 75 mb; this brings us nearer to the value that we consider at present the most accurate one when working with a fission spectrum. It is also likely that in the calculation of D. J. Hughes, made before 1953, the value of the effective threshold has been taken too high; at that time the value of this threshold was determined from the calculation of the penetrability factor, whereas today we are able to forego the purely theoretical calculation of the penetrability factor, because we can use the experimental curve of BNL 325 (Second edition, July 1958) for the effective cross-section of sulfur (reaction n, p) as a function of the energy.

III - <u>EXPERIMENTAL DETERMINATION OF THE NEUTRON SPECTRUM</u> -

III - a - <u>Spectral Distribution of Neutrons After Collision with a Moderator Nucleus</u>

We have seen in § II - a - 2 - that the flux of virgin neutrons originating in fuel elements can be put in the form:

$$\Phi(E)dE = \frac{S\ell}{2\pi r}\left\{\frac{\pi}{2} - Kj_0\left[(r-c)\Sigma m\right]\right\} N(E)dE$$



The number of virgin neutrons originating in fuel element that will undergo collision in the element volume of dV around the point C and that after this collision will reach point A, equals:

$$\Sigma_m(E)\,\Phi(E)dE\ \frac{e^{-\Delta\ell\,\Sigma m(E)}}{4\pi\ell^2}\,dV$$

with $\Delta\ell = BC = \frac{u-c_0}{\cos\theta}$ $\qquad \ell = AC = \frac{u}{\cos\theta}$

$dV = u\,du\,d\alpha\,dz \qquad dz = \frac{\ell\,d\theta}{\cos\theta}$

Integrating over the total volume the flux of neutrons of energy E in the interval dE at point A after collision with the moderator equals:

$$④\quad F(E)dE = \frac{S_f\,dE}{2(1-\alpha)\pi^2}\int_E^{\frac{E}{\alpha}}\frac{\Sigma_m(E')N(E')dE'}{E'}\int_{\varphi=0}^{\varphi=\pi}\int_{u=C_0}^{\infty}\frac{\frac{\pi}{2}-Kj_0(r\Sigma_m)}{r}\left[\frac{\pi}{2}-Kj_0(u-C_0)\Sigma_m\right]du\,d\varphi$$

with $\alpha = \dfrac{(A-1)^2}{(A+1)^2}$ and $r^2 = u^2 + R^2 - 2uR\cos\varphi$

In order to simplify the calculation we have taken C = 0; we will do the same in expression (2) obtained in § II - a - 2.

We will now compare expressions (2) and (4); for this purpose we will examine the ratio

$$\frac{F(E)dE}{\Phi(E)dE} = \frac{F(E)}{\Phi(E)}$$

representing the ratio of the neutron fluxes reaching point A after collision with the moderator, and without collision.

$$\frac{F(E)}{\Phi(E)} = \frac{\dfrac{S_f}{2(1-\alpha)\pi^2}\displaystyle\int_E^{\frac{E}{\alpha}}\frac{\Sigma_m(E')N(E')dE'}{E'}\int_0^{\pi}\int_{C_0}^{\infty}\frac{\frac{\pi}{2}-Kj_0(r\Sigma_m)}{r}\left[\frac{\pi}{2}-Kj_0(u-C_0)\Sigma_m\right]du\,d\varphi}{\dfrac{S_f}{2\pi R}\left[\frac{\pi}{2}-Kj_0(R-C_0)\Sigma_m\right]N(E)dE}$$

We find that, starting from a certain value in the neighborhood of 3 Mev, this ratio becomes quite small and thus the fluxes of neutrons whose energy are higher than this value is mainly due to neutrons that have not undergone collision.

The spectrum of neutrons having an energy above approximately 3 Mev is thus an undisturbed spectrum and its form is given by expression (2) of § II - a - 2.

III - b - Spectrum Determination by Means of Threshold and Resonating Detectors

Taking into account the previous remark we can now try to determine the neutron spectrum above the thermal region.

1°/ Epithermal Region (0.4 ev to 1 Mev) -

In this region we assume that we have a slowing-down flux that obeys a 1/E law. This hypothesis may be verified by measuring each time the activity of several resonating detectors irradiated, open, and under cadmium. If we call A_{nu} the activity of the open detector and A_{cd} the activity of the detector with cadmium, irradiated under identical conditions, then the fact that the ratio:

$$\frac{q}{3\Sigma_s} = \frac{\overline{\sigma_{th}}}{\left(\frac{A_{Au}}{A/Cd} - 1\right)\int_{0,3}^{\infty} \sigma(E)\frac{dE}{E}}$$

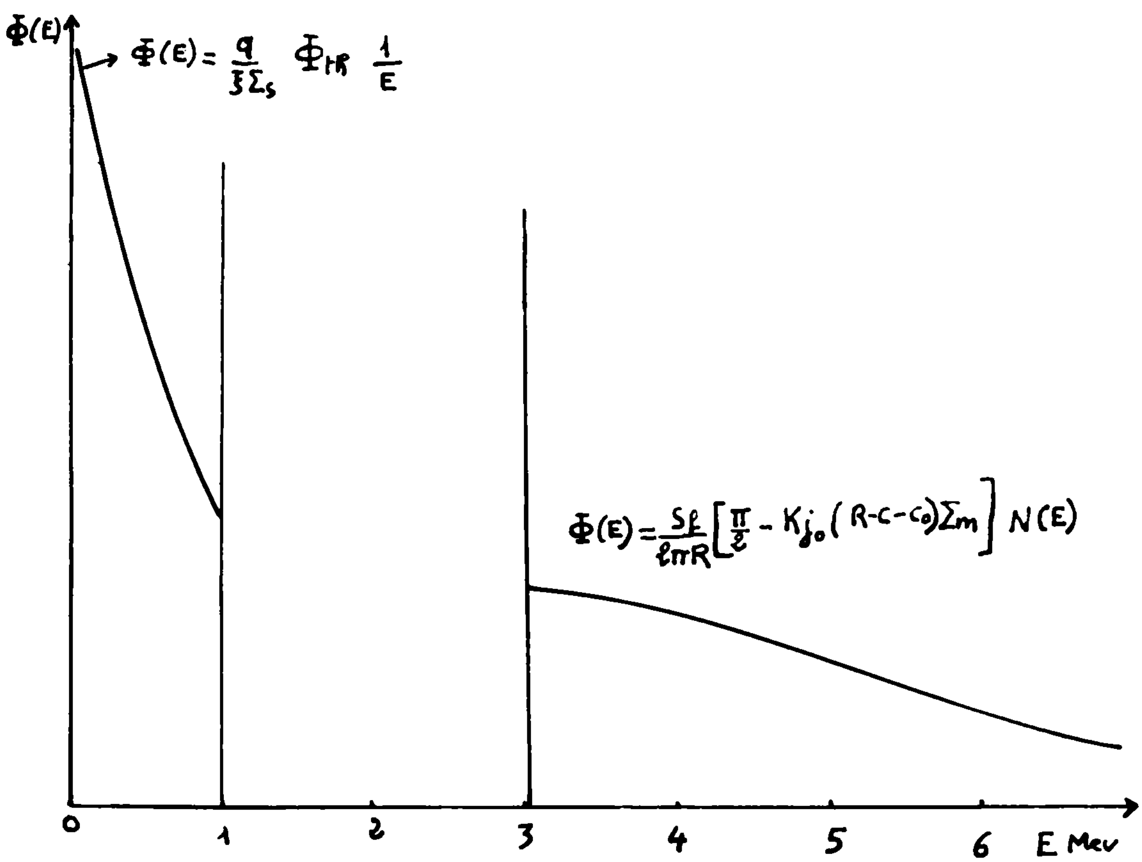
is constant for detectors of gold, In and Mn will be a proof that we do not actually have a slowing down flux:

$$\Phi(E)\,dE = \frac{q}{3\Sigma_s}\,\Phi_{th}\,\frac{dE}{E}$$

where $\overline{\sigma}_{th}$ is the effective activation cross-section for thermal neutrons of a given detector, and $\int_{0,3}^{\infty} \sigma(E)\frac{dE}{E}$ its integral of resonance.

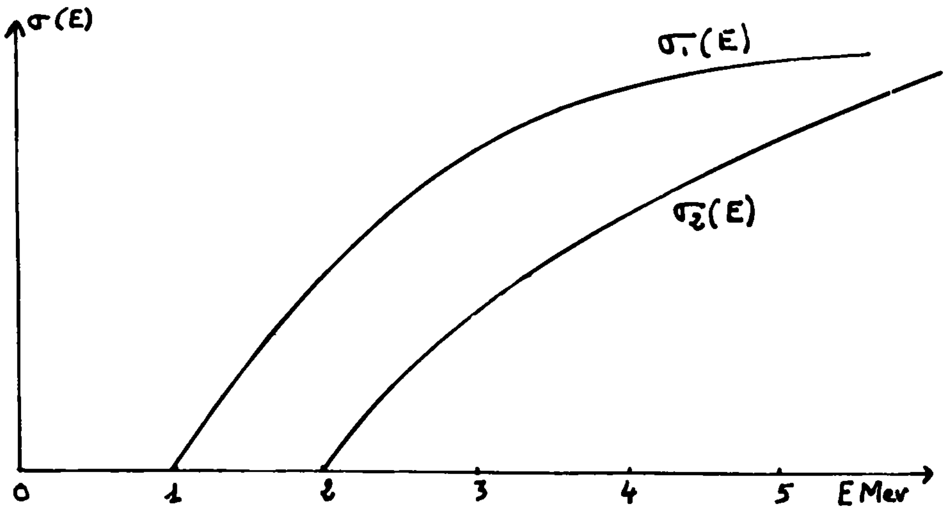
2°/ <u>Fast Region $\geq$ 1 Mev</u> -

Two sets of data are available for determining the spectrum in the rapid region; the spectrum up to 1 Mev and the spectrum calculated from equation (2) that applies above 2 to 3 Mev.



From these two curves we can attempt to determine the spectrum by the following method:

- Above 3 Mev we assume that the spectrum is known (equation (2)).
- Below 1 Mev we proceed then as follows:



Let us consider two detectors having thresholds of 1 Mev and of 2 Mev and let us assume that the effective cross sections $\sigma_1(E)$ and $\sigma_2(E)$ as functions of the energy are well known.

The activities of these detectors after irradiation under identical conditions may be written

$$A_2 = N_2 V_2 \left(1 - e^{-\lambda_2 t}\right) \int_0^\infty \sigma_2(E)\, \Phi(E)\, dE$$

$$A_1 = N_1 V_1 \left(1 - e^{-\lambda_1 t}\right) \int_0^\infty \sigma_1(E)\, \Phi(E)\, dE$$

N_2 and N_1 are the numbers of nuclei per cm^3 for the detectors used and λ_2 and λ_1 are the radioactive decay constants of the substances formed.

These equations will be written as follows:

$$A_2 = B \int_2^3 \sigma_2(E)\, \Phi(E)\, dE + B \int_3^\infty \sigma_2(E)\, \Phi(E)\, dE$$

$$A_1 = C \int_1^2 \sigma_1(E)\, \Phi(E)\, dE + C \int_2^3 \sigma_1(E)\, \Phi(E)\, dE + C \int_3^\infty \sigma_1(E)\, \Phi(E)\, dE$$

The integrals $\int_3^\infty \sigma_2(E)\,\Phi(E)\,dE$ and $\int_3^\infty \sigma_1(E)\,\Phi(E)\,dE$ are well known and therefore the activities can be written

$$A_2 = B\int_2^3 \sigma_2(E)\,\Phi(E)\,dE + K_2$$

$$A_1 = C\int_1^2 \sigma_1(E)\,\Phi(E)\,dE + C\int_2^3 \sigma_1(E)\,\Phi(E)\,dE + K_1$$

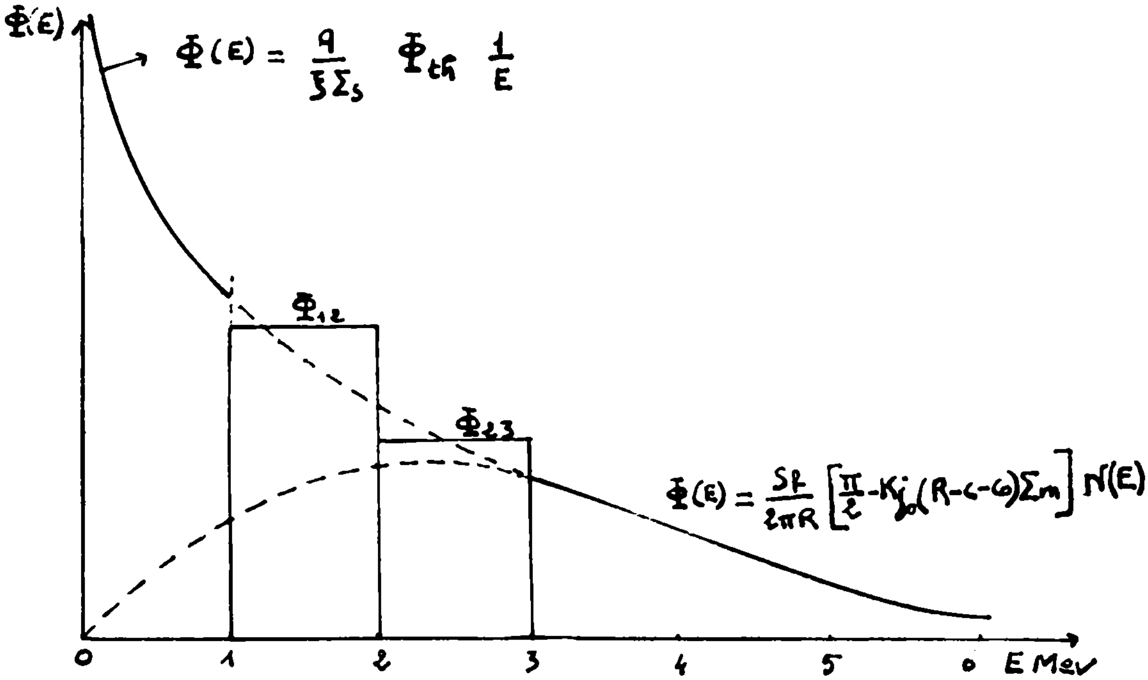
We assume that in each of these energy intervals (1 to 2 Mev and 2 to 3 Mev) the flux $\Phi(E)$ remains constant and has the value Φ_{12} and Φ_{23} respectively. We will then have:

$$\Phi_{23} = \frac{A_2 - K_2}{B\int_2^3 \sigma_2(E)\,dE}$$

and

$$\Phi_{12} = \frac{A_1 - K_1 - C\,\Phi_{23}\int_2^3 \sigma_1(E)\,dE}{C\int_1^2 \sigma_1(E)\,dE}$$

The values of Φ_{12} and Φ_{23}, obtained experimentally are plotted in their respective intervals; through the two segments thus obtained we draw the connecting curve between the spectrum following the 1/E law and the tail of the spectrum with the neutrons that have not undergone collisions.



In the above diagram the connecting curve has been drawn in a completely arbitrary way.

This region will actually show, undoubtedly, a much smaller average slope than the one that fits the connecting curve drawn in the schematic figure, thus justifying the hypothesis of the constancy of the flux in the intervals 1-2 Mev and 2-3 Mev.

We could verify that this curve approaches the true curve by calculating the expressions

$$\int_2^3 \sigma_2(E)\,\Phi(E)\,dE = \frac{A_2 - K_2}{B}$$

$$\int_1^3 \sigma_1(E)\,\Phi(E)\,dE = \frac{A_1 - K_1}{C}$$

where $\Phi(E)$ corresponds to the connecting curve drawn.

The method recommended here will be the more accurate, the greater the number of detectors at our disposal having thresholds conveniently dispersed over the connecting region and that have effective cross-sections well known as functions of energy.

In this perspective we plan to use in the near future:

a) Miniature fission chambers using the following fuels:

U_{238} - threshold at approximately 0.9 Mev.

Th_{232}- threshold at approximately 1.3 Mev.

Np_{237}- threshold at approximately 0.1 Mev.

The effective cross-sections for all these substances as functions of energy are well known.

b) <u>Exact threshold detectors</u>:

The following threshold detectors have been used so far:

13 Al^{27} $(n\ \alpha)$ Na^{24} threshold at approximately 6 Mev

15 P^{31} $(n\ p)$ Si^{31} threshold at approximately 1 Mev

16 S^{32} $(n\ p)$ P^{32} threshold at approximately 1.8 Mev

The effective cross-sections for these detectors as a function of energy are also well known (BNL 325 - second edition, 1958). Other detectors could be used to cover the whole region (0.1 Mev to 10 Mev for instance). The counting technique (4π counter) and the manufacture of the three exact detectors (deposition by vacuum or painting) were described in a report of the Starting Group of Mélusine by P. Leger and B. Sautiez.

IV - CONCLUSION -

The method proposed in this report for fast neutron dosimetry is not undoubtedly a perfect one but we think that it is an improvement over the methods used to date, in which a constant effective cross-section was assumed, regardless of the location of the region where the measurement was carried out. An additional improvement will be made as soon as it will become feasible to determine the neutron spectrum by a more refined method than the one that uses several threshold detectors.

I thank Mr. Delattre and Mr. Hyver for the very useful remarks and suggestions they made during the many discussions that we had together on these problems in fast neutron dosimetry.

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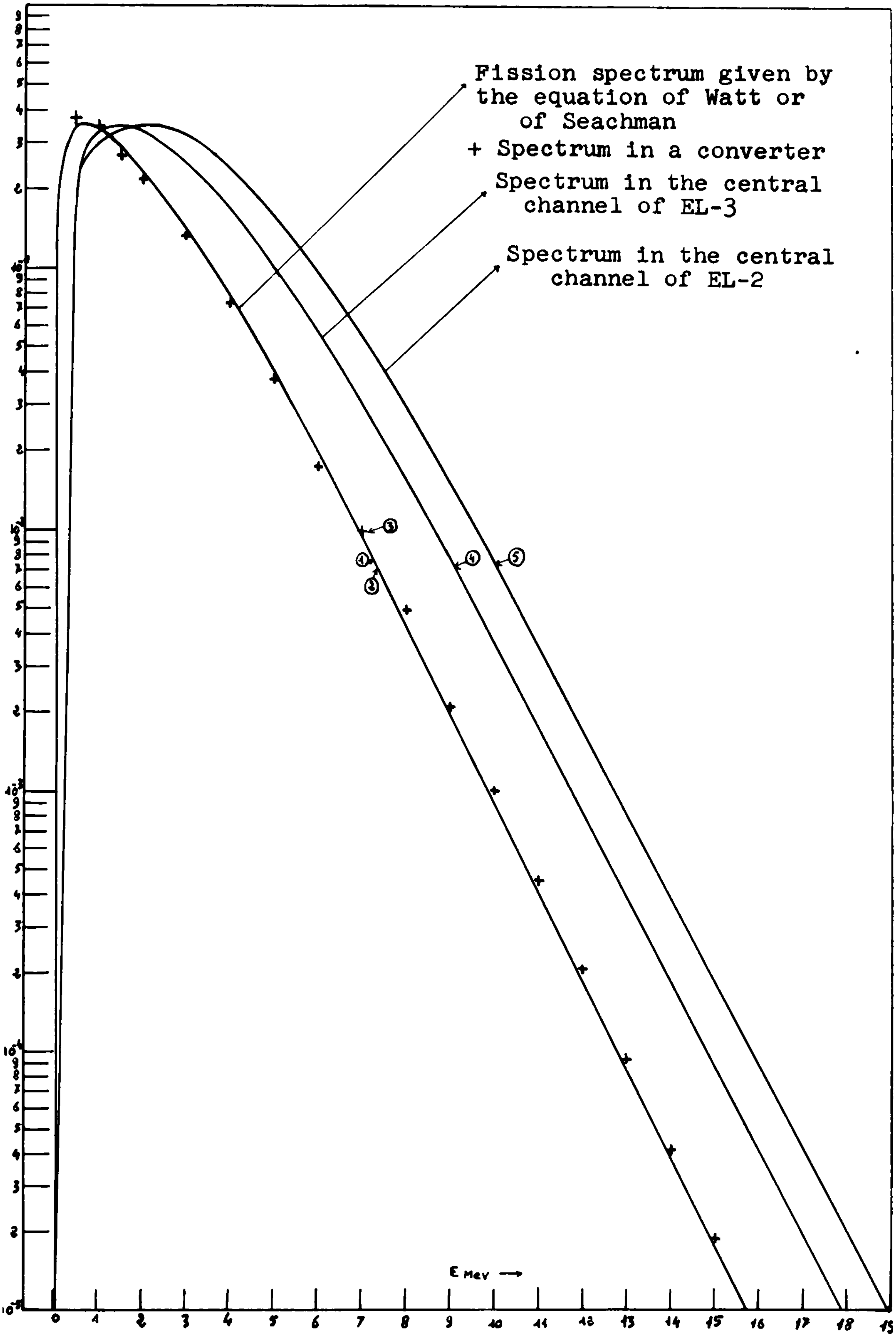

Fission spectrum given by the equation of Watt or of Seachman
+ Spectrum in a converter
Spectrum in the central channel of EL-3
Spectrum in the central channel of EL-2
E Mev →

DISTRIBUTION OF THE FLUX OF THERMAL NEUTRONS IN URANIUM SAMPLES IRRADIATED IN A LOOP COOLED BY DOWTHERM

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EXPERIMENT OBJECTIVE

To know with a maximum accuracy the neutron fluxes to which samples of enriched uranium are subjected in the course of their irradiation in a pile channel.

A certain number of enriched uranium samples are placed inside an irradiation container located in channel H-3 of the EL-3 pile. Under irradiation they produce a certain power of the order of a kilowatt for each sample, and their temperature is adjusted by acting on that of the cooling fluid which consists of dowtherm A (mixture of diphenyl and diphenyl oxide). Knowing the temperatures of the uranium samples on one hand, that of the cooling fluid on the other, the power released by the samples can be calculated and hence the real fluxes, measured experimentally, to which the samples are subjected. Then, knowing the flux existing in the channel at the location of the uranium sample before the introduction of the sample and its container, the decrease of total flux caused by the sample and its container can be deduced.

DESCRIPTION OF THE EXPERIMENTAL CONTAINER

The samples are placed in cylindrical containers made of zircaloy 2, these containers then being filled with liquid sodium so that all the uranium is immersed in the sodium. A glove finger containing a thermocouple which constantly registers the temperature of the sodium bath is immersed in the sodium. The outside wall of the vessel is swept by a stream of dowtherm A circulating at a temperature of 250°C, at a speed of the order of 2 m/s, sufficient to carry off the heat released without any risk of calefaction or carbonization of the fluid in contact with the wall. The container consists of eight smaller containers, the loading details of which are given below, placed in two stacks of four, the axes of these containers and stacks being parallel to the channel axis. The flow of downtherm

A divides in two on entering the container so as to cool each of the two stacks separately. The structural materials of the container are essentially zircaloy 2, making up the case of the container and the fluid channels, and magnesium as filler. The pile channel consists of a glove finger penetrating into the interior of the heavy water reflector of the pile with its axis perpendicular to the axis of the tank.

CALCULATING THE EXPERIMENTAL FLUX

For lack of a more accurate point of reference, the temperature of the uranium samples can be taken as that of the sodium bath as registered by the thermocouple.

Thus the total temperature gradient between the sodium and the circulating fluid is known. The exchange coefficient between the cooling fluid and the vessel wall is calculated. For this, knowing the thermal characteristics of the fluid, the formula recommended by Atomics International for polyphenyls and their derivatives is applied, namely:

$$N\ddot{u} = 0.015\ (Re)^{0.86}\ (Pr)^{0.27}$$

Nü: Nüsselt number -- Re: Reynolds number -- Pr: Prandt number

Then the gradient across the vessel wall is calculated using Fourier's equation; thus the power, and hence the flux is obtained knowing the load in the container.

PRESENTATION OF RESULTS

For each container we give the nature of the load, the total weight of uranium-235 contained, the temperature during irradiation, and, finally, the ratio of the empty channel flux before the introduction of the container to the actual mean flux calculated as stated above, in all the uranium of the container.

Vessel (5) 4 cylindrical samples {D = 6 mm h = 30 mm 5% enrichment	Vessel (1) 1 tubular sample {outside 16 mm inside 10 mm height 30 mm 5% enrichment
Weight of U^{235} contained: 3.18 g Irradiation temperature 425°C	Weight of U^{235} contained: 3.46 g Irradiation temperature 485°C
$\dfrac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}} = 3.72$	$\dfrac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}} = 3.66$

Vessel (6) 5 cylindrical samples: D = 4 mm; h = 30 mm; 10% enrichment	Vessel (2) 1 tubular sample: outside 12 mm; inside 8 mm; height 30 mm; 10% enrichment
Weight of U^{235} contained: 3.55 g	Weight of U^{235} contained: 3.55 g
Irradiation temperature 400°C	Irradiation temperature 430°C
$\frac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}} = 4.42$	$\frac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}} = 4.53$
Vessel (7) 1 tubular sample: outside 16 mm; inside 10 mm; height 30 mm; 5% enrichment	Vessel (7) 1 tubular sample: outside 12 mm; inside 8 mm; height 30 mm; 10% enrichment
Weight of U^{235} contained: 3.46 g	Weight of U^{235} contained: 3.55 g
Irradiation temperature 385°C	Irradiation temperature 395°C
$\frac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}} = 4.95$	$\frac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}} = 4.85$
Vessel (8) 4 cylindrical samples: D = 4 mm; h = 30 mm; 10% enrichment	Vessel (4) 4 cylindrical samples: D = 6 mm; h = 30 mm; 5% enrichment
Weight of U^{235} contained: 2.84 g	Weight of U^{235} contained: 3.18 g
Irradiation temperature 340°C	
$\frac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}} = 4.00$	

DISCUSSION

The preceding results are difficult to interpret in view of the rather great complexity of the container and the variety of materials present. Nevertheless, it is possible to distinguish the following points:

1) The containers, with the exception of No 8, have approximately the same quantity of uranium-235, viz., 3.5 grams and, by and large, the same amount of neutron absorbing material. It was noted that two vessels located at the same distance from the center of the pile present approximately the same total decrease in flux. We shall call decrease of flux the ratio:

$$\frac{\phi \text{ empty channel}}{\phi \text{ actual in the uranium}}$$

This is systematic and is verified for each row of two containers located at the same distance from the center of the pile. In the first approximation, it can be deduced that this decrease of flux depends only upon the amount of absorbing material contained in each container. The geometry of the samples and their location in the container have only little effect.

2) For increasing distances from the center of the pile, the depressions of flux increases appreciably; let us clarify this: the experimentally measured actual fluxes in the uranium of a container decrease more rapidly than the flux in the empty channel as one moves away from the center. This is evidence for a shadow effect caused by each sample on those located behind it relative to the center of the pile. This is probably due to the fact that there exists a neutron drift away from the center of the pile, each sample creating behind it a hole in neutron density.

Can these flux decreases be justified by theoretical calculation? With the existence of the neutron drift conplicating the phenomenon, we performed a calculation on the tubular sample of the No 1 container which, on one hand, is not covered up by any sample, and, on the other hand, appears in a simple geometric form.

The following hypotheses were made:

1) We supposed that the fast neutrons emitted by the uranium in the samples did not play any role at all.

2) We omitted all radiation effects except thermal neutrons.

3) Calculation made as for the calculation of the thermal utilization factor of a lattice cell following the method perfected by Mr. Amouyal and Mr. Benoist, C.E.A. report No 571, with a heavy water moderator.

We are no longer dealing with infinite cylindrical geometry, the tubular sample of uranium being considered as isolated in the center of the channel, its being parallel to the axis of the channel with a fictitious shielding. The characteristics of this fictitious shielding have been set as follows: the effective cross-sections are the averaged cross sections using the so-called method of mixtures on all the material located in the same plane perpendicular to the axis of the channel as the sample considered. This fictitious shielding is limited towards the center by the sample and on the periphery by the heavy water playing the role of moderator and presumed to extend indefinitely around the cell thus defined.

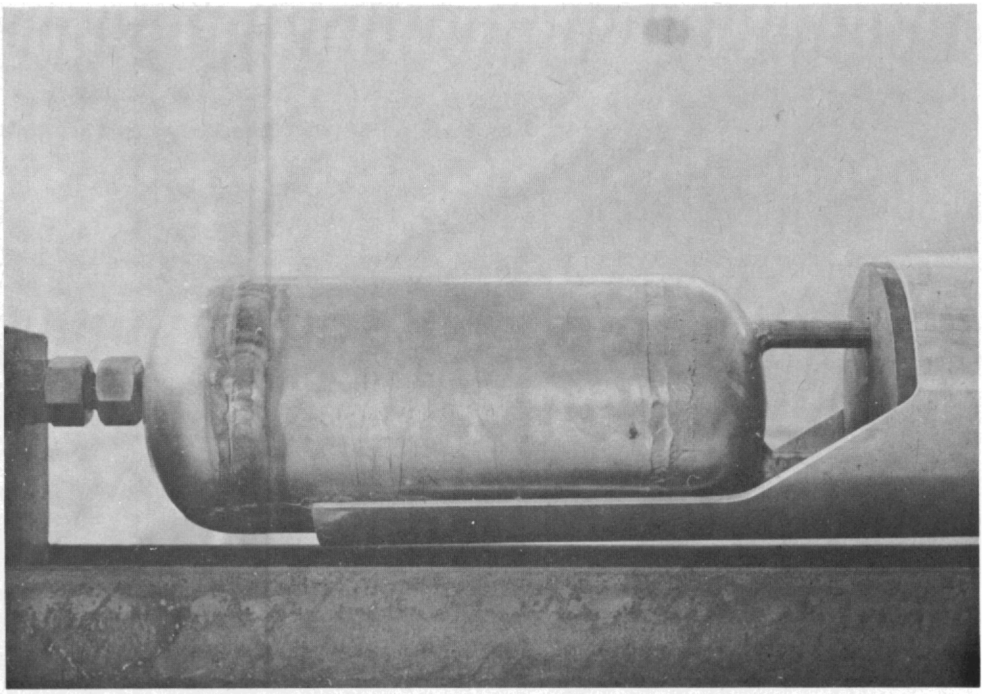
The flux at infinity in the heavy water moderator, i.e., $\phi\infty$ is then taken to be the flux at the location of the uranium sample, with channel empty. The method, then, permits one to calculate the ratio $\frac{\phi\infty}{\phi\text{mean in the uranium}}$ which is none other than the unknown decrease of flux.

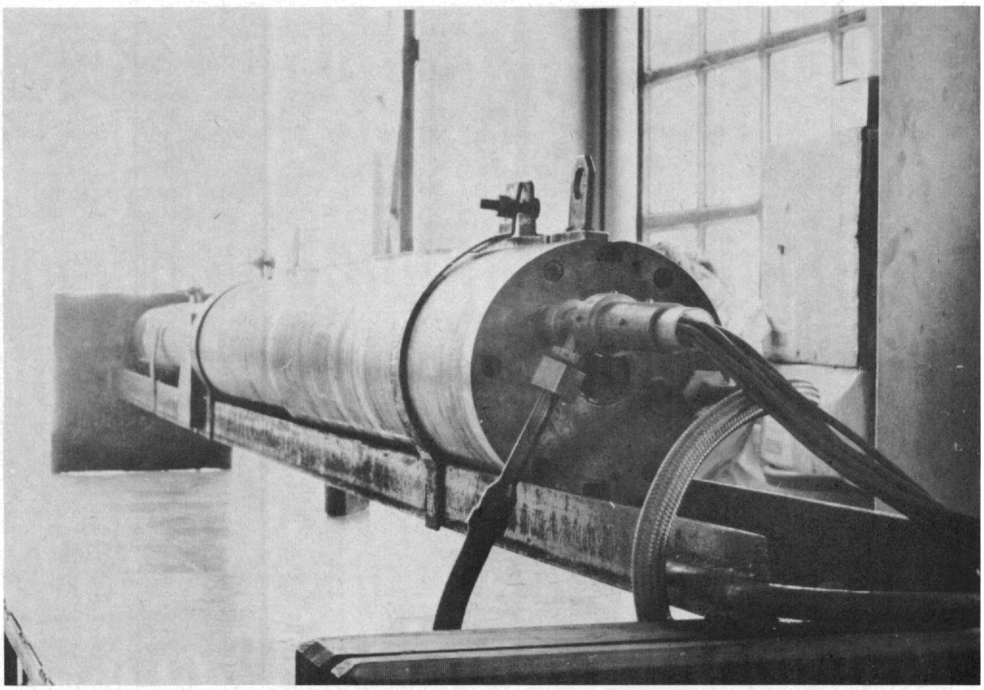
The calculation contains three terms, one consisting of the flux crossing inside the sample itself, calculated by the method called "collision by collision"; let us point out that in the course of the preceding experiments, we verified the agreement of the results given by this method with the experimental results - - a second term: decrease of the flux inside the shielding, a third term: decrease of the flux in heavy water; these last two terms were calculated by the theory of corrected diffusion.

The calculation yields a decrease of flux of 3.40, almost equal to the number 3.66 established experimentally. Nevertheless, too much importance should not be given this result, in view of the hypotheses made and the inaccuracy of experimental measurements. For more complex geometries like that of vessel No 5, we obtained very different results by theoretical calculation: a decrease of 2.5 for 3.72 experimentally. Let us say simply that the order of magnitude can be calculated theoretically, and let us remember that for samples such as ours, we must expect to find fluxes three to four times smaller than those measured when the channel is emptied of any container.

R. Besse and J. Ratier

The authors wish to thank Mr. Delattre, Mr. Eckert and Mr. Levy of the "Service des Grandes Piles" (Large Pile Service) who provided them with all the conveniences for carrying out the experiment.





MEASUREMENT OF THE NUCLEAR CHARACTERISTICS OF EXPERIMENTS

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SUMMARY

The nuclear characteristics of experiments introduced into research reactors can constitute two regions of interest sufficiently different so as to be considered separately:

1. In the use of the research pile data which describe the effect which the presence of the experiment has on the behavior of the reactor are of interest.

2. The experimenter needs to know the nuclear characteristics of the experimental chambers used: the necessary precision will be greater if the experiment is in the stage of a pre-project, of concept or of interpretation.

* * *

The preparation and execution of these tests which will lead to a knowledge of these two categories of information constitute, within the framework of the BR-2, our present program.

The reactor was conceived so as to offer the experimenter the greatest possible diversity of location.

Consequently each functioning cycle of the reactor constitutes today a case special in its power:

- large experimental spaces can be established in the heart of the reactor by the retraction of beryllium plugs pierced by three charging channels,
- each charging channel can receive equally well a control rod, a fuel element, transformed or not into a converter, experimental setups, two different beryllium buttons which permit a total or partial sealing,
- the reflector can be greatly modified by the use of nine irradiation tubes.

In order to use this flexibility to its maximum, without stoping the reactor in the experimental periods required for the knowledge of these various configurations, a nuclear model of low power is coupled to the reactor. This latter combines a rigorous similarity of all the essential constituents of the reactor with an accessibility which is as complete as possible. It also constitutes the ideal experimental field for the study of the reactor.

* * *

We will devote ourselves in what follows only to the physical phenomena without considering the technological consequences which will frequently be their corollaries.

* * *

FIRST PART

EFFECT OF THE EXPERIMENTS ON THE BEHAVIOR OF THE REACTOR

The experimental installations introduced into the experimental reactor have, depending on their importance and nature, different effects on the behavior of the reactor. Their common characteristic is, that they constitute positive or negative sources of reactivity. These effects have to be known to the person in charge of the operation of the reactor, with an accuracy, which is dependent on their importance.

We can attempt to classify them in several categories:

1. The reactive effects are weak, and can be accounted for according to simple and routine procedures.
2. The reactive effects are important, and their occurrence necessitates a special study of the operational cycle with which they are associated.

According to another point of view, the experimental installations can constitute either stable or slowly varying effects, or potential reactivity reserves which can be released suddenly under certain conditions: they are then such that they can influence the safety of the operation of the reactor, and thus have to be submitted to a special analysis.

These different points of view all relate to the neutronic behavior of the experiment; the gamma rays emission can be the subject of another supplementary study: we shall come back to it in the second part of this report.

1. The Reactive Effects Are Weak

An estimate on the variation of the critical factor k_{eff} of a reactor is easy if the perturbation introduced is uniformly distributed throughout the entire lattice: the new value of k_{eff} is calculated from the new microscopic properties of the lattice.

In the case of a thermal absorber the approximations:

$$\frac{\delta\, k_{eff}}{k_{eff}} = \frac{\delta\, f}{f} = \frac{\delta\, \Sigma}{\Sigma} \qquad \text{are valid,}$$

f representing the thermal utilization and
Σ representing the effective absorption cross-section of the lattice.

On the other hand, if the perturbation is not uniform, Such as will be the case upon the introduction of a sample, the computation of the total reactive effect will constitute the basic problem of the theory of perturbations.

This theory can only be applied if the critical factor and the neutron distribution are only slightly affected by the perturbation. Let us assume that we deal with the introduction of an absorber localized in a weak element of volume ΔV of a uniform reactor of volume V; the application of the theory of perturbations to a group of neutrons leads to the expression:

$$\frac{\delta k}{k} = - \frac{\delta \Sigma_a}{\Sigma_a} \frac{\phi^2 \Delta V}{\int_V \phi^2 dV}$$

The expression $\frac{\phi^2 \Delta V}{\int_V \phi^2 dV}$ is the "statistical weight" of volume ΔV expressing the fact that the total effect on the multiplication factor is proportional to the square of the flux at the site of the perturbation.

The concept of statistical weight remains valid in the treatment of two or several groups of neutrons provided that it will be a function of the product of the flux and their adjoints or value functions.

In a heterogenous reactor of small dimensions such as BR-2, the application of this theory is laborious if one does not employ semi-empirical methods. These methods consist of dividing the reactor into a certain number of cells for which a statistical weight relative to a reference cell is determined experimentally.

For this purpose a small quantity of poison is displaced from one cell to the next; in each of these positions the reactive effect is measured as the displacement of the control rod necessary to maintain the reactor critical. The results are normalized to a reference effect in the central cell; one thus knows the statistical weight relative to the central position for each position.

In order that these results can be applied to the calculation of the reactive effects of the experiments that take place in the core of the reactor at different stages of its operation, the same tests are carried out by simulating the exhaustion of the fuel by a uniform distribution of absorbers in the core of the reactor.

2. <u>The Reactive Effects Are Important</u>

The above procedure is no longer applicable, if the experiment is such that it will disturb extensively the flux distribution and will either enhance or decrease markedly the reactivity.

One then has to carry out direct experiments on the actual medium of the experiment or of its facsimile, introduced into the reactor or the nuclear model of the latter.

Thus, during the operation of BR-2, the important experiments can be placed into the nuclear model, which will gradually be brought to criticality by means of a loading arrangement suited to the goals of the experiment.

It is to be noted that at that time the study of the actual configuration or of a very closely related configuration will have been the subject of complete preliminary investigations in connection with the starting of the BR-2, only the effect of the presence of the experiments is new, and their evaluation is the object of these additional tests.

The reactivity associated with the experiment will be determined by a procedure which will depend on the quantity of poison or fuel which causes it, and on the manner in which it influences the rest of the core. A precise knowledge of this factor will only be justified in certain special cases where they could influence the safety of the operation. This will be discussed further on.

The excess of available reactivity will be measured by the progressive withdrawal of the control rods compensated by the distributed poisoning; the efficiency of the rods will be reviewed at this opportunity.

At the same time, the flux charts will be surveyed for several simulated stages of the operation of the load: the relation of the maximum and average flux will be determined, and the temperature at the hot points will be deduced.

The reactive effects of the experiments can be stable or slowly varying, by fuel consumption or by mutation of the neutron absorbing substances of which they are made up, but can also constitute large potential reserves of reactivity.

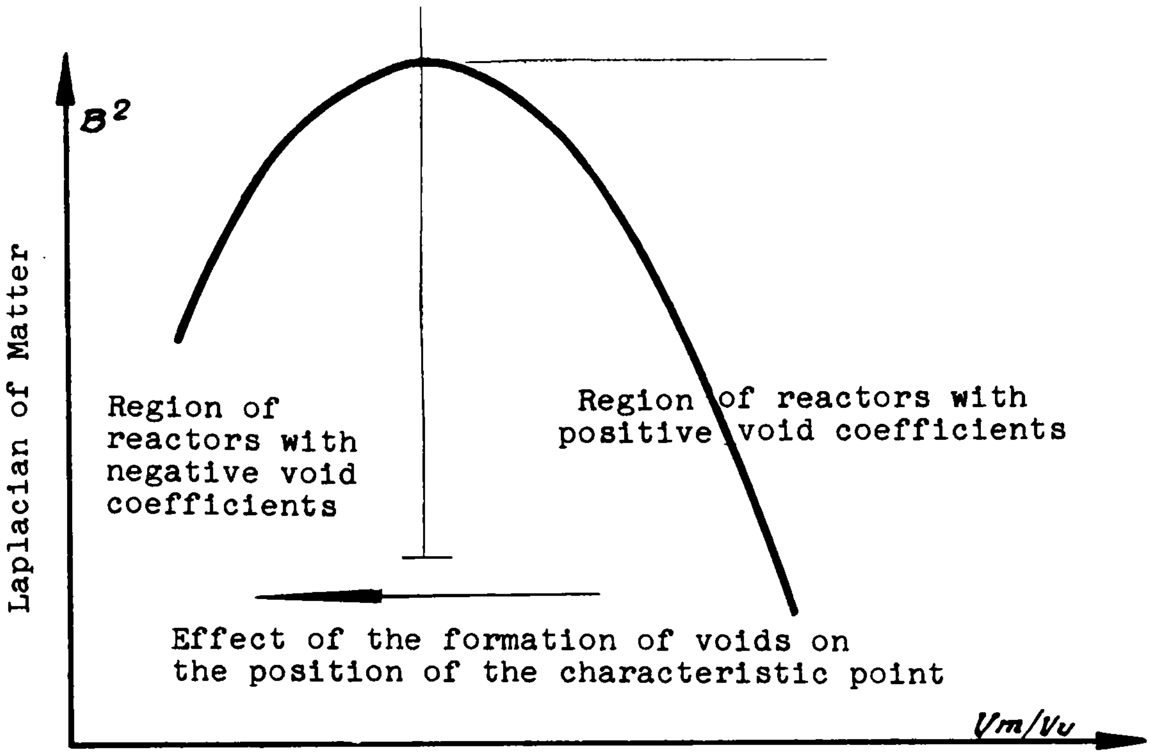
The average void coefficient for the case of voids distributed uniformly in the reactor, and here particularly the local coefficient in the case of large experimental cavities is of determining influence.

By void, in the nuclear sense, one usually refers to inert volumes which are characterized by a zero probability of interaction with slowing down and absorption of the neutrons. In the reactors that we will consider the void will assume at the same time a more general and a more specific significance: we will substitute for the water of the lattice various substances of different moderating power, usually lower, and with a weak absorption efficiency: these will be vapor bubbles, air cavities and certain constitutional materials of the experiments. In order to prove this generalization we want at present only the use of magnesium, aluminum or lead in order to simulate to various degrees of perfection the voids in the studied lattice.

a) Uniformly Distributed Voids

One realizes that the uniform distribution of vapor bubbles in a lattice moderated with water, modifies the relation of the volumes of the moderator and of the fuel. The following curve characteristic of the Laplacian of matter as a function of the ratio Vm/Vu indicates that reactors which have the characteristic point to the right of the maximum will naturally be supermoderated, and will thus have a positive total void coefficient.

The characteristic point can evolve in such a manner that the void coefficient will pass from the positive to the negative region, as the increase of the proportion of the void is influenced by the increase of the power of the reactor and by a decrease of the thermal exchange coefficient. This process could be the cause of instabilities. The reactors in which the characteristic point is to the left of the maximum will naturally be under-moderated and will have a negative void coefficient.



Ratio of the Volumes Moderator-Fuel

The evaluation of the effect of these voids on the multiplication factors of the lattice can be easily carried out by theoretical calculations. The experimental verification is simple and consists of the application of the procedure described further on to the entirety of the core of the reactor.

b) Localized Voids

The measurement of the local void coefficient of the reactor is easily carried out in the case of voids whose volume is small compared to the characteristic dimensions of the lattice.

The same volume of void is successively placed in the different cells of the lattice. One simulates the void by means of a material of weak absorbing section and moderating power, and which does not have other inconvenient probabilities of reaction with neutrons, such as inelastic diffusion. This material will be such that it will permit a good geometric determination of the volumes of the void. Magnesium is rather suitable from the point of view of these conditions: its void volume is 90 percent of its actual volume.

The reactor being maintained at its critical stage, the position of the control rod is recorded before and after the introduction of the simulated void. The results are corrected for the absorption effects by using the relationships of the theory of perturbations.

These tests will be carried out at different temperatures of the moderator.

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This procedure cannot be applied to important localized void volumes such as can be formed in the lattice by the presence of an experimental loop. The large perturbations can only be evaluated with the help of an experimental installation or its facsimile.

Let us consider the case where the local void coefficient is negative: this will occur often; the breakdown of the loop, and its filling up with water are accompanied by a release of reactivity.

One can thus carry out a test on a facsimile of the experiment and measure, by reference to the critical state, the difference of the reactivity between the two states: intact experiment and drowned experiment.

A simple, interpretative model would be to consider that a linear variation of the reactivity between these two extreme cases is established which describes the intermediary states during which the experimental loop is filled with water of densities lying between the void and the density of water at the temperature of the experiment.

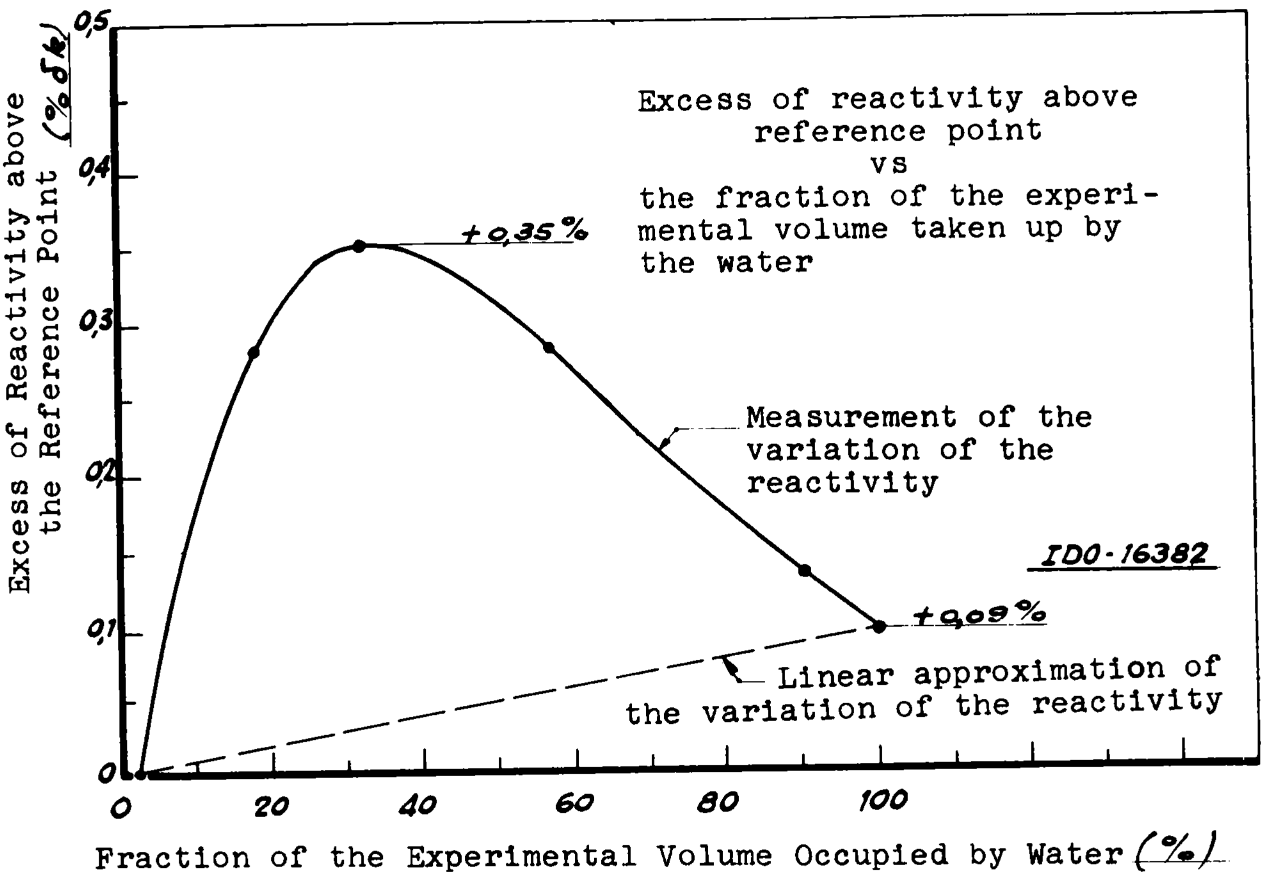
The estimates on the effect of the phenomenon on the reactivity would thus be based on a law of release of reactivity in time, deduced from the linear approximation. The maximum reactivity would correspond to the final state.

It is essential to point out that this procedure is not valid for large cavities since it is absolutely unable to describe the dynamics of an event where the concurrence of several phenomena is such as to create a complex resultant which is otherwise difficult to outline theoretically.

The experimental work of G. O. Bright and C. R. Toole on the Speri-1 have clearly illustrated the uselessness of the approximations. The available reactivity can consequently be more important during the phenomenon than in its final state, and the speed of release will have no relation to the theoretical approximation.

The experimental curve below has been taken from the report IDO 16382 and summarizes the experiments performed by the authors mentioned previously.

The aim of the experiment was the study of the consequences of the putting under water a gas which goes through the core of the reactors SPERT-1 parallel to the fuel elements. It has been assumed that the appearance of a large leak caused the filling up of the loop starting from its periphery to the development of an annular volume of water whose internal zone grew smaller until it was completely filled. This mechanism was simulated by installing in the loop central blocs of Styrofoam (polystyrene of very low density).



These results are mentioned only as illustrations of a general principle, and are evidently not valid except in the particular conditions of the experiment.

One can thus conclude that the determination of the potential reactive effect of a large cavity necessitates after analysis of the mechanism of accidental filling the simulation of a series of intermediate states and the analysis in each one of those of the activity which becomes available.

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Irradiation tubes in neutron beams create, when empty large reactivity losses due to the gaps which they create in the reflector.

One can estimate that in the case of the BR-2 a radial tube inserted dry represents approximately a reactive loss of 1 to 1.5 pcm per cm/square of cross section. The presence of an experimental setup introduced into this tube can obviously modify this estimate, depending on its nature.

In any case, the study of this effect will be carried out experimentally and it will be convenient to analyze the kinetic behavior of the reactor when one or several of its experimental chambers are put under non-moderated [?] water. The test will be carried out in such a way as to make possible an interpretation of its dynamics: the evolution of the reactivity as a function of time.

In order to do this, the filling will be carried out in stages. At each stage the released activity will be associated, by reference to the critical state, with the necessary displacement of calibrated control rods.

One will deduce from this the relation between the reactivity and the volume of water admitted; this will permit to study, making all hypothesis on the speed of filling in case of failure, the variation of the reactivity with time.

* * *

SECOND PART

DOSIMETRY OF EXPERIMENTS

An important stage in the preparation of experiments in a research reactor and especially in a reactor for the testing of materials, is the perfection of the precise techniques of dosimetry.

The interpretation of the experimental results, that is the analysis of the state of the samples after irradiation, can only be based on a quantitative knowledge of the agents which have caused the discovered modifications.

We shall separate the "agents" in four categories:

- thermal neutrons,
- epithermal neutrons,
- fast neutrons,
- gamma rays.

The counting techniques for these particles are quite classical: they make use of known theories, materials and installations. Before entering into a consideration of different aspects of these counting techniques, we want to draw attention to the care which has been given to make sure that the results of the measurements give valuable information, representing faithfully the real situation. We wish to stress the difficulties which are encountered in the realization of this objective and to show the value of approximate solutions whose interest will be a function of the stage of the experimental study which they support.

Let us look at three stages in the carrying out of an experiment.

1. Preliminary Examination

The first contact between the experimenter and the reactor physicist is centered on a choice of the experimental chamber that will be used. This chamber must meet, among others, the specifications of nuclear order required by the experimenter: the flux level, the neutron spectrum, the gamma dosage.

From this moment, flux charts referring to the normal functioning of the reactor should be available: use at the power and nominal temperature of the charging geometry which, as far as the number and disposition of the fuel elements is concerned, that is, as far as the general aspect of the flux is concerned, will at this stage depend on the predictions based on the number and importance of the experiments.

These flux charts will be of only inaccurate value because they give no account of the perturbations caused by the experiments.

Also, during the cycle of operation, the progressive withdrawal of the regulation elements modifies the flux charts very strongly. In this case then, the flux charts only represent the mean time average values.

In the case of BR-2, the preparation of these flux charts will be accomplished in the nuclear model.

Even though the study is carried out taking the remarks made above into consideration, so that the results have a maximum practical interest, one must at this stage recognize that they only have the quality of the right order of magnitude.

2. Study of the Experiment in the Nuclear Model

The second stage can be envisaged only in the case of important experiments, which for reasons envisaged previously will be studied first in the nuclear model before they are introduced into the reactor.

At this occasion, the dosimetry of the experiment will be developed maximally: survey of the flux charts in the reactor, fine structure in the experiments, analysis of the essential points of the spectrum.

The decrease in the flux due to the experiment itself will therefore be included in the measured distributions. The perturbations caused by the modification of the regulation system will, however, be sources of inaccuracies, since the simulation, in order to stay practicable, will be somewhat imperfect.

3. Monitoring of the Irradiation

The direct monitoring of the experiment during the irradiation will constitute the fundamental way of obtaining information. Even though it is the only exact procedure, it could constitute alone only a sufficient support for the work of the experimenter, since it produces its results a posteriori.

The monitors will be small samples whose nature and installation can pose certain technological problems; they must be easily recoverable and manipulable at the time of the removal of the experiment.

The analysis of certain precise and well known changes in their physical-chemical properties should permit the deductions of quantitative data on the actual integrated flux of the experiment.

REVIEW OF VARIOUS DOSIMETRY TECHNIQUES

A. DEFINITION OF TERMS.
B. DETECTION OF THERMAL AND EPITHERMAL NEUTRONS.
C. DETECTORS OF FAST NEUTRONS.
D. INTEGRATING DETECTORS.
E. GAMMA DOSIMETRY.

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A. DEFINITION OF TERMS

The measurements of neutron flux have as their aim the determination of the thermal and epithermal flux corresponding to a nominal nuclear power. As basis of these measures, we accept

1. that the energy distribution of the thermal neutrons is a maxwellian one at a temperature T higher than the physical temperature of the moderator,

2. that the distribution of epithermal neutrons follows a law in $1/E$ limited towards the low energies at a thermal energy cut-off of μkT.

The total spectrum is the sum of these two components.

By definition,

- the thermal flux is the product of the thermal density and the mean velocity of the neutrons in the spectrum at a temperature T; we represent it by $n.\bar{v}.$,
- the epithermal flux is formed by the unit of neutrons whose energy is larger than the thermal cut-off μkT; we will characterize it by the flux per unit energy $\frac{K}{E}$ in the epithermal region near that limit where the hypothesis of variation with $1/E$ is valid. In the region of high energy the epithermal flux will be defined beyond certain thresholds.

We are thus dealing with experimentally accessible quantities. In addition, on a basis of these experimental results one can determine the epithermal flux by means of certain conventions; we propose to adopt

$$\phi_{\text{épith}} = K \int_{\mu kT}^{E_o} \frac{dE}{E} ,$$

where E_o is, for example, the mean energy of the fission neutron; this relation assumes that the variation with $1/E$ is valid up to E_o. The measurement of fast flux with threshold detectors will then supply the information.

B. DETECTION OF THERMAL AND EPITHERMAL NEUTRONS

1. Absolute Methods of Measurements

The absolute measure of thermal flux rests on the definition of a standard thermal flux in a reference pile. In order to compare the unknown thermal flux to the standard flux it has to be noted that:

- Since the temperature of the thermal neutrons in the standard pile is equal to the physical temperature and different from that of the thermal neutrons in the reactor, corrections have to be applied taking into account the influence of the difference of temperature on the reaction rate and possibly the different decreases in the interior of the detector.
- The "thermal" activity of a detector as determined by a cadmium difference is partially due to neutrons whose energy is between the thermal cut-off μkT and the cut-off energy of cadmium E_{cd}; this effect can be made weak for a detector activated at a

point sufficiently far from the source in the standard pile while it has its importance for a detector activated in the reactor.

The rate of reaction per nucleus, for a detector as 1/v, obtained by cadmium difference after irradiation in the standard flux is thus:

$$R_e = [n\ \bar{v}]_e \cdot \sigma_o \sqrt{\frac{\pi\ T_o}{4\ T_e}} \qquad \text{with} \qquad T_e \cong T_o$$

and in the reactor:

$$R_r = [n\ \bar{v}]_r \cdot \left[\sigma_o \sqrt{\frac{\pi\ T_o}{4\ T}} + K' \int_{\mu kT}^{E_{cd}} \sigma(E)\ \frac{dE}{E}\right] ;$$

from which the thermal flux in the reactor:

$$[n\ \bar{v}]_r = [n\ \bar{v}]_e \cdot \frac{R_r}{R_e} \cdot \frac{\sigma_o \sqrt{\frac{\pi\ T_o}{4\ T_e}}}{\sigma_o \sqrt{\frac{\pi\ T_o}{4\ T}} + K' \int_{\mu kT}^{E_{cd}} \sigma(E)\ \frac{dE}{E}}$$

where - $\frac{R_r}{R_e}$ is equal to the ratio of the measured activities, corrected for the decrease of the flux in the neighborhood of the detector;
- $[n\ \bar{v}]_e$ is the standard flux;
- K' defines the relative importance of the epithermal group; its value is deduced from a measure of the cadmium ratio;
- the integral $\int_{\mu kT}^{E_{cd}}$ between the thermal cut-off and the effective cut-off of cadmium E_{cd} can be calculated; E_{cd} can easily be defined for a detector whose cross section varies as 1/v in the neighborhood of this energy;
- $\sigma(E) = \frac{\sigma_o \sqrt{E_o}}{\sqrt{E}}$ such that the last factor is independent of σ_o.

If the deviation between the temperatures T_e and T is important, it can influence the distribution of active nuclei in a detector which has a sizeable total absorption cross section. The ratio of measured activities is different from the reaction rate defined above. This effect has to be considered only when $\Sigma_o\ x > 0.1$ approximately, where x is the thickness of the detector.

The determination of the thermal flux requires the knowledge of the neutron temperature T and of the factor μ defining the thermal cut-off; this estimate, based both on theoretical consideration and previous experimental work will be afflicted with a certain amount of uncertainty which will be related to the value of the thermal flux.

The knowledge of the epithermal flux is derived from a measure of the cadmium ratio:

$$R_{cd} = \frac{\text{Activity of the uncovered detector}}{\text{Activity of the covered detector}}$$

for a detector of zero thickness.
One has, therefore:

$$R_{cd} = \frac{\sigma_o \sqrt{\dfrac{\pi\ T_o}{4\ T}} + K' \displaystyle\int_{\mu kT}^{E_{cd}} \sigma(E)\, \frac{dE}{E} + K' . I . R .}{K' . I . R .}$$

where I.R. is the resonance integral.

The cadmium ratio can be measured by means of gold or indium detectors. We will come back to the use of one or the other of these detectors. It is, thus, possible to calculate K'; associating with the determination of the cadmium ratio a measure of the thermal flux, one can finally determine the value of K which is the flux in a logarithmic interval of energy equal to ln e or the flux per unit energy $\frac{q}{\xi \Sigma_s}$ at 1 eV.

In this manner, the energy distribution of the epithermal group is entirely fixed following the hypothesis of a variation with 1/E.

The validity of this hypothesis is verified, at several points of the core, by measuring the flux per unit energy at energies determined by means of the detectors which have a resonance peak at an energy E_{res}. One has to use an appropriate technique in order to avoid the influence of secondary resonances. In this case also the standard pile will serve as a reference; starting with a spatial integration of the slow-down density at the energy E_{res}, the detectors can be calibrated in an absolute fashion for this measurement.

Finally, since the determination of R_{cd} for a detector in 1/v is calcuable starting from R_{cd} measured by means of a detector like indium, one can now find, at any point of the reactor, the value of the expression $r \sqrt{\frac{T}{T_o}}$ and of r since T is known. One has, indeed:

$$r \sqrt{\frac{T}{T_o}} = A/R_{cd}$$

where r is the characteristic relative to the part of the spectrum in 1/E used by C. H. Westcott. The factor A is tabulated as a function of the thickness of the cadmium covering.

The absolute measurement methods described above are particularly well adapted to the nuclear model which, before the start of the Br-2, will function at a weak power in order not to increase unnecessarily the difficulties in handling the fuel elements.

Later the functioning power of the nuclear model will be increased and the flux to be measured will be such that the reaction rate of a detector will be measured absolutely and with precision by means of β - γ or γ - γ coincidence counters. These techniques can be applied by using gold or cobalt respectively.

Let us finally note that the measurement at low flux cannot be directly extrapolated to high flux except by simulating correctly the effect of the temperature and of the poisoning by fission products. The simulation techniques will be studied on the nuclear model.

11. Analysis of the Detectors in the Two Groups

For the measurement of thermal flux, it is interesting to use detectors whose behavior is as 1/v up to the cadmium energy cut-off and whose activity due to the standard flux is sufficient. From this point of view, gold is useful since its first resonance is sufficiently far from E_{cd}.

In order to measure the cadmium ratio and the epithermal flux gold or indium can be used;

- For the two applications the advantages mentioned above for gold remain of interest; further, the resonance integral has been measured with care and been considered a basis in certain measurements of resonance integrals. One has, however, to use very thin foil (of the order of several hundredth of μg per cm^2) which are obtained by evaporation, for example, in order to avoid the theoretical corrections due to thickness.

- For indium, the thickness correction is well known, especially for the determination of the cadmium ratio; the effective energy E_{cd} and the activation cross sections are less well known. It permits a less exact measurement of the epithermal flux but its half-life makes its use easier at the time of the activation in the standard flux. These two detectors permit the precise calculation of the cadmium ratio for a 1/v detector starting from their respective cadmium ratios since their activity under cadmium in a reactor spectrum is appreciable.

The determination of the deceleration density at definite energies is based on the use of rhodium, indium, gold, and manganese which show a predominant resonance at 1.26 eV, 1.45 eV, 4.9 eV, and 337 eV, respectively. In order to eliminate the influence of other resonances, the detector is activated alone and then covered in one part or the other by foil of the same material and well chosen thickness; the difference in activities is therefore due only to resonant neutrons whose energy is given.

Let us mention that the use of rhodium as well as of dysprosium whose interest lies in that they possess a high activation and do not show any activation resonance, must be accompanied by certain precautions which have been studied specially:

- The activity of rhodium after irradiation up to saturation decreases following a scheme which necessitates a waiting time of 500 seconds before the measurement; the decrease is then purely exponential and the half-life is 4.4 min.

- The activation of dysprosium furnishes two isomers, one of which has a short half-life (1.3 min); after about ten minutes the decrease is practically exponential; the frequency of use of the detector has to be limited as a function of the value of the integrated flux since one of the isomers formed has a high activation cross section and its activity disappears only slowly ($T_{1/2}$ = 82 hrs).

Finally, the flux in conveyors or other accessible places of a high flux reactor can be measured during the functioning at nominal power by means of quartz detectors. Silicon is advantageous from several points of view: it has a weak activation cross section and hence does not pose any protection problems during manipulation; besides, its a β-emission of high energy (1.48 MeV) permit the determination of the total number of disintegrations in a 4π counter without self-absorption being a source of too great inaccuracy.

Another type of detector, the thermopile, can be of interesting usage.

The principle of this detector is very simple: the electromotive force induced in the end of two branched thermocouples in series is a measure of the temperature difference between the two junctions.

One uses the calories released by certain nuclear reactions, of which the currently most utilized one are the capture by boron or lithium or the fission of uranium, by covering one of the two junctions by one of these materials: the electromotive force becomes a function of the neutron flux which, through adequate reaction, causes a heating of the covered junction.

One can raise the sensitivity of the apparatus by using a chain of thermocouples whose junctions are alternately covered and uncovered. One can arrange a setting such that the thermopile will be insensitive to every temperature effect and that its response time will be reduced to a minimum.

These detectors have small dimensions and their relative simplicity of construction permits one to create a variety of models adapted for each use.

They are especially inconvenient for the survey of flux charts in high power reactors. One can imagine that they should also accompany experimental setups which necessitate a continuous flux control.

For these two applications they have been used with success in the BR-I. The technological problem of the passage of the wire outside of the primary of the reactor obviously has to be resolved: this is common to all measurements of temperature in the core.

A thermopile currently used in the BR-I consists of 4 "hot" junctions covered with natural boron.

The sensitivity is 165 $\mu V/10^{11}$ $n.cm^{-2}.sec^{-1}$

The time constant has been measured with precision: it is 8 seconds and can be further improved.

C. FAST NEUTRON DETECTORS

In a material testing reactor, the dosimetry of fission neutrons, that is, fast neutrons which have never yet interacted with matter, is of great importance.

The possibilities of irradiation in the interior of annular fuel elements in the BR-2 permit to envisage a very extended employ of adequate dosimetry methods for the monitoring of experiments which are put there.

The detectors generally used are called "threshold detectors". One understands by this the use of certain nuclear reactions which necessitate the participation of neutrons of an energy larger than the threshold energy of the reaction.

On this side of the threshold the probability of reaction is zero; on the other side it evolves fairly rapidly toward a more or less constant value: this is at least the case for certain reactions like P^{31} (n.p.) and S^{32} (n.p.); several other reactions used present, however, a less favorable situation.

In any case, the reaction probability as a function of the neutron energy is idealized in the form of a step function, a "step" function, on one side of which the probability is constant and on the other of which it is zero.

The "step" is situated at such an energy that according to a hypothesis on the distribution of neutrons the actual number of reactions corresponds to the number of reactions calculated starting from this idealized probability; it is called effective threshold of energy.

In actuality the various detection reactions used follow this idealized model to various degrees of approximation. This idealization is expressed by the relation:

$$\sigma \int_{E_{eff}}^{\infty} \phi(E)\, dE = \int_{0}^{\infty} \sigma(E)\, \phi(E)\, dE$$

where σ (E) is the actual effective cross section, depending on the energy for the considered reaction.

σ is the mean value of the effective cross section above the threshold.

E_{eff} is the value of the energy of the effective threshold calculated by means of the assumption that the neutrons are distributed according to a fission spectrum, that is that ϕ (E) is an adequate analytical expression.

The measured reaction rate will be attributed to neutrons whose energy is higher than the effective threshold energy.

The number of reactions that can be used is large and permits a fairly detailed division of the fast spectrum:

Reaction	Effective threshold energy	Half-life of isotope formed	Particles emitted by the isotope formed
Np^{237} fission	600 keV	-	-
U^{238} fission	1,3 MeV	-	-
$In^{115}(n.n')In^{115*}$	1,5 MeV	4,5 h	β : 0,83 MeV (6 %) γ : 0,334 MeV
Th^{232} fission	2,0 MeV	-	-
P^{31} (n.p) Si^{31}	2,4 MeV	2,62 h	β : 1,48 MeV no γ
S^{32} (n.p) P^{32}	2,9 MeV	14,3 days	β : 1,7 MeV no γ
Al^{27} (n.p) Mg^{27}	4,6 MeV	9,8 m	β : 1,8 MeV(80%) ; 0,9 MeV(20%) γ : 1,0 MeV(20%) ; 0,84MeV(100%)
Ni^{58} (n.p) Co^{58}	5,0 MeV	72 days	β : 0,47 MeV γ : 0,81 MeV
Si^{28} (n.p) Al^{28}	5,5 MeV	2,3 m	β : 2,8 MeV γ : 1,78 MeV
Mg^{24} (n.p) Na^{24}	6,3 MeV	14,9 h	β : 1,4 MeV γ : 2,75 MeV ; 1,38 MeV
Al^{27} (n.α) Na^{24}	8,1 MeV	14,9 h	The same as above

The effective energy thresholds are tabulated only with greatest reserve and it is on purpose that we do not give effective cross sections: there is a considerable confusion among the various references on this point.

We have undertaken to make these data more precise as our techniques are perfected.

It is thus that for the reaction S^{32} (n.p.) P^{32}, we obtain:

$$\frac{\int_0^{\infty} \sigma(E)\,\phi(E)\,dE}{\int_0^{\infty} \phi(E)\,dE} = 63{,}8 \text{ mb},$$

for the fission spectrum of uranium 235.

The precision of the computation is estimated at 5%.

D. INTEGRATING DETECTORS

1. The monitoring of an experiment based on the measurement of the activity of a sample supposes that from this a posteriori measurement one can deduce the expression $\int \phi\,(t)\,dt$ extended over the duration of the irradiation.

Actually, the hypothesis of invariability of flux at the level of the experiment is not very valuable and in any case incompatible with the precision sought.

This condition imposes a selected choice from the materials that are usually used as activation flux detectors.

If N (t) is the number of active nuclei formed by the irradiation, in a flux $\phi\,(t)$, of a detector of volume V and of effective activation cross section Σ, then the differential equation governing the phenomenon can be written:

$$\frac{d\,N\,(t)}{dt} = V\,\Sigma\,\phi\,(t) - \lambda\,N\,(t)$$

$$N\,(t) = V\,\Sigma\,e^{-\lambda t}\int_0^t \phi\,(t)\,e^{\lambda t}\,dt$$

where λ is the radioactive decay constant of the active isotope formed from the initial atom.

The activity of the detector measured by a counter is λ N (t). Let us examine the error introduced by assuming that this activity is expressed by

$$\lambda\,N\,(t) = \lambda\,V\,\Sigma\int_0^t \phi\,(t)\,dt\quad .$$

If it were thus, the activity measured would represent well the integral sought.

In the last relation the exponential terms of the correct expression have been neglected, which is equivalent to implicitly assuming that every active nucleus rests stored as it is up to the time of the end of the irradiation.

We can express the relative error introduced in the form:

$$\Sigma = \frac{\left|\int_0^t \phi\,(t)\,dt - e^{-\lambda t}\int_0^t \phi\,(t)\,e^{\lambda t}\,dt\right|}{e^{-\lambda t}\int_0^t \phi\,(t)\,e^{\lambda t}\,dt}$$

Expanding this expression and neglecting terms in λt of order higher than unity we obtain:

$$\Sigma = \frac{\lambda t/2}{1 - \lambda t/2}$$

The error introduced will be negligible, if the product λt is small. The choice will therefore fall on materials whose half-life is the longer, the longer the time of irradiation and the smaller the admissible error.

Thus, for example, the errors committed will be less than the thresholds below if the irradiation times are not larger than the values mentioned.

Nature of detector	Half-life	Errors smaller than or equal to 1 %	5 %	10 %
Or	2,7 days	1,85 h	8,90 h	17,0 h
Cobalt	5,28 years	55,1 days	265 days	505 days

Cobalt is thus an integrating detector that is very interesting; we use it pure or in the form of its compounds the poorer in cobalt the greater are the integral fluxes to be monitored: thus one avoids the manipulation of too active samples.

Let us mention also the use of beryllium:

Formation from $_4Be^9$ (σ_{act} at 2200 m sec^{-1} : 9$\pm$ 3 mb) from $_4Be^{10}$ ($T_{1/2}$ = 2.7 x 10^6 years) emitter of 0.56 MeV β.

2. The monitoring of the integral flux at the level of an experiment can be based on a series of other methods all having in common the measure in a sample of the number of original atoms used up.

The dosage can be related to the decrease in the number of original nuclei or the appearance of daughter nuclei.

Contrary to the method based on the activity of the integrated detector these dosages do not imply any approximation in principle but make necessary in general delicate manipulation which are sources of inaccuracy.

The precision will increase with the integrated flux and the practical application threshold can easily be calculated: they make the significance of these methods for high flux reactors obvious. Amidst the numerous methods mentioned in the literature we retain here only those whose usage appears to us most practical:

Two detectors of the same nature and dimensions are intercalibrated by activation before and after the utilization of one or the other as integrating detectors. The dissimilarity of the two coefficients of intercalibration confirms the diminution in the integrator of the number of original nuclei: the integrated flux is easily deduced. Gold and dysprosium have adequate characteristics for these manipulations.

The method is applicable starting from the following integrated fluxes.

from 10^{21} n cm^{-2} for gold
from 5 x 10^{19} n cm^{-2} for dysprosium.

E. GAMMA DOSIMETRY

In a reactor of the BR-2 type functioning at full power the energy dissipated by gamma radiation can attain several tens of watts per gram.

This shows sufficiently the technological importance of gamma radiation in the preparation of experiments.

We have thus devoted ourselves to the development of practical methods for gamma dosimetry.

Three different techniques have been developed using as a reference a chemical dosimeter based on the ferrous-ferric oxidation reaction.

1. Measurement of the Opacity of Glass Samples

We have utilized up to now samples of ordinary glass which are normally used for microscopy work (ILFORD Treated Glasses).

The optical density of these detectors is measured, after their irradiation by an intense source of Co^{60}, by means of a spectrophotometer. The wave length of the light source used was about 500 mμ.

The response of the detectors is linear in the range of integrated doses of 10^5 to 10^6 roentgens.

The recovery phenomenon is being studied and gamma irradiations in the presence of neutrons will be carried out.

2. Study of the Response of an Ionization Chamber with Graphite Electrodes and CO_2 Filling

This method is classical: it permits, by measuring the ionization current in an internal cavity of the chamber and through application of the Gragg-Gray relation, to determine the energy absorbed per gram of graphite.

The measurement range can be regulated by varying the gas pressure and extends up to 5.10^5 roentgens/hour approximately.

3. Calorimetric Method

Gamma thermopiles have been developed. They permit the measurement of very strong doses, starting from some 10^5 roentgens/hour in utilizing the calories liberated by the attenuating of gamma rays in the absorbing samples.

The detector consists of a chain of chromel-constantan thermocouples in series. The junctions are alternately covered with a little ball of lead or left uncovered. The assembly of the chain is mounted on a support such that it will be insensitive to every temperature effect and that the electromotive force measured between the ends of the chain will be a unique function of the heating caused by the gamma rays.

A thermopile consisting of 24 junctions and loaded with 12 spheres of lead of 8 mm diameter has been tested in the presence of a Co^{60} source.

The response of the detector is linear in the restricted domain explored up to now from 4.10^5 to $5.2\ 10^5$ roentgens/hour. Its sensitivity is 2.8 μv/ 10^3 roentgens/hour. Its time constant is 2.5 minutes.

SIGNIFICANCE OF AN EXPERIMENTAL MEASUREMENT OF THE ACTIVATION OF DIFFERENT TYPES OF STEEL FOR REACTOR TANKS

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The problem of the activation of the steels in nuclear reactors is extremely complex because of the great variety of radiations to be taken into account and because of the high number of isotopes involved at least in the tracer state. A detailed study of this problem is underway at Harwell in Great Britain the results of which will be published in the near future.

Many specialists have been properly devoting their time to the effect of present day cobalt content of the steels which have been used up to now. Certain metallurgical concerns have already undertaken costly works to perfect steels with a very low cobalt content which they are ready to produce under various commercial names.

The question arises as to what extent it is really necessary to lower the cobalt content of steels used in the construction of nuclear reactors. Without prejudging the detailed and precise elements of the answer that will be made by the Harwell report, it is useful to briefly analyze the conclusions that will permit the steel specialists as well as builders and users of nuclear reactors to form an opinion.

We must first take note of the fact that the steels used in building nuclear reactors are subjected to surface corrosion and erosion by the primary coolant. The corresponding products, which are already more or less radioactive depending on the place in the reactor from which they come, are carried into the circuit and undergo strong irradiation at the time of their passage into the core of the reactor. These products are filtered out as much as possible and are eventually fixed on ion exchangers, but they can also become imbedded at various points in the primary circuit. With respect to the radiations emitted, their action adds to that of the fission products, which can for various reasons be drawn into the primary coolant despite the fact that it is theoretically airtight by the nuclear fuel elements.

Because of the effects of corrosion and erosion, the chemical composition of the steels in contact with the primary coolant affects the level of radioactivity in numerous places in the reactor and its auxilliary systems. However, this effect is a problem in itself and is not taken into consideration in the following analysis.

The activation of steels is produced essentially by the action of neutrons. In many cases the absorption of slow neutrons by a stable isotope transforms the latter into a radioactive and usually gamma emitting isotope. Fast neutrons can be absorbed like slow neutrons, but they produce, in addition, nuclear reactions of the types (n,p), (n,2n) and (n,α).

For each radioactive isotope formed, the specific activity is given after an irradiation time t in a flux of neutrons ϕ, by the expression:

$$S = \frac{c}{100} \times \frac{a}{100} \times \frac{0.6\ \phi\ \sigma}{3.7\cdot 10^{10}\ A} \times$$
$$\times \left(e^{-\phi\sigma t\cdot 10^{-24}} - e^{-\frac{0.69\ t}{T}}\right) \times$$
$$\times \frac{1}{1 - 1.45\phi\sigma T\cdot 10^{-24}}$$

where:

S is the specific activity in curies per gram;
c is the content in percent of the natural element in the steel;
a is the abundance in percent of the parent isotope in the natural element;
ϕ is the neutron flux in n/cm^2, sec;
σ is the effective cross section of neutron capture or reaction of the type (n,p), (n,2n), or (n,α), in barns;
A is the atomic weight of the isotope formed;
t is the irradiation time in seconds;
T is the half life of the isotope formed in seconds.

Table I indicates, for isotopes generally present in steels, the abundance a in percent of the natural body, the effective cross section of absorption of thermal neutrons with a speed of 2200 m/sec σa, the active species produced and its half life T, the maximum energy of γ-rays and X-rays emitted by this active species and the product $a\sigma_a/A$.

Table II indicates, for the same isotopes, the type of nuclear reaction brought about by fast neutrons, the effective cross section σe of fission neutron reaction, the active species produced and its half life T, the maximum energy E of γ and X-rays emitted by this active species, and the product $a\sigma_e/A$.

By means of these data one can calculate the specific activity generated in the steels by irradiation. The following examples refer to typical cases.

Table I

Parent Isotope	a %	σa Barns	Active Species	T	E Mev	$a.\sigma_a/A$
Ti 50	5.3	0.04	Ti 51	72 j*	1.0	0.00416
Cr 50	4.4	16	Cr 51	27.8 j	0.323	1.38
Mn 55	100	13.3	Mn 56	2.6 h	2.1	23.8
Fe 54	5.84	2.3	Fe 55	2.94 a*	0.07	0.244
Fe 58	0.33	0.8	Fe 59	46 j	1.3	0.00447
Co 59	100	37	Co 60	5.3 a	1.33	61.7
Ni 58	67.8	4.2	Ni 59	8.10^4a	X of Co	4.8
Cu 63	69	4.3	Cu 64	12.8 h	1.35	4.73
Zn 64	48.9	0.5	Zn 65	250 j	1.12	0.377
Zn 68	18.5	0.1	Zn 69	15.8 h	0.4	0.0269
Zr 94	17.4	0.1	Zr 95	65 j	0.92	0.0183
Mo 98	23.8	0.45	Mo 99	67 h	0.84	0.105
Ta 181	100	21.3	Ta 182	113 j	1.2	11.8
W 184	30.6	2.0	W 185	74 j	0.132	0.33
W 186	28.4	34	W 187	24 h	0.76	5.12

*j - days; h - hours; a - years

Table II

Parent Isotope	a %	σe mbarns	Active Species	T	E Mev	$a.\sigma e/A \times 10^3$	Type of Re-Action
Al 27	100	3.43(?)	Mg 27	10.2 m	1.02	12.7	(n,p)
Al 27	100	0.6 (?)	Na 24	15 h	5.3	2.5	(n,α)
P 31	100	31.2 (?)	Si 31	170 m	1.26	100	(n,p)
P 31	100	1.43(?)	Al 28	2.4 m	1.78	13	(n,α)
Si 28	92.18	4	Al 28	2.4 m	1.78	0.44	(n,p)
Si 29	4.71	2.7	Al 29	6.7 m	2.43	5.1	(n,p)
S 34	4.215	3 (?)	Si 31	170 m	1.26	0.400	(n,α)
Ti 46	7.99	4.1	Sc 46	85 j*	1.12	0.712	(n,p)
Ti 47	7.32	0.21	Sc 47	3.4 j	0.17	0.0328	(n,p)
Ti 48	73.99	0.077	Sc 48	44 h	1.33	0.119	(n,p)
Ti 50	5.3	0.0002	Ca 47	4.7 j	1.31	0.000023	(n,α)
V 51	99.76	0.08(?)	Sc 48	44 h	1.33	0.167	(n,α)
Mn 55	100	0.05(?)	Mn 54	291 j	0.835	0.093	(n,2n)
Fe 54	5.84	56 (?)	Mn 54	291 j	0.835	6.06	(n,p)
Fe 54	5.84	0.37	Cr 51	27.8 j	0.323	0.0425	(n,α)
Fe 56	91.68	0.44	Mn 56	2.6 h	3.0	0.72	(n,p)
Co 59	100	5.7 (?)	Fe 59	46 j	1.29	9.67	(n,p)
Ni 58	67.8	225 (?)	Co 58	72 j	1.62	263	(n,p)
Ni 58	67.8	13	Co 58m	9 h	0.025	15.1	(n,p)
Ni 58	67.8	0.17	Fe 55	2.94 a	0.07	0.209	(n,α)
Ni 58	67.8	0.0012	Ni 57	36 h	1.38	0.00142	(n,2n)
Ni 60	26.2	5 (?)	Co 60	5.3 a	1.33	2.19	(n,p)
Ni 62	3.66	0.14(?)	Fe 59	46 j	1.3	0.00869	(n,α)
Cu 63	69	0.72	Co 60	5.3 a	1.33	0.829	(n,α)
Zn 64	48.9	35 (?)	Cu 64	12.8 h	1.35	26.8	(n,p)

Table II [continued]

Parent Iso-tope	a %	σe mbarns	Active Species	T	E Mev	a.σe/A x 10^3	Type of Re-Action
Zn 67	4.11	0.27	Cu 67	59 h	0.39	0.0166	(n,p)
Zn 68	18.5	0.02	Ni 65	2.56 h	1.5	0.0057	(n,α)
Mo 92	15.86	1.3	Nb 92	10 j	1.83	0.224	(n,p)
Mo 92	15.86	0.017	Zr 89	79 h	0.51	0.00304	(n,α)
Mo 95	15.7	0.1 (?)	Nb 95	35 j	0.76	0.0165	(n,p)

First Case

Activation of a stainless steel of the type 18/8 used as a shielding material for fuel elements.

Hypotheses:

time of irradiation $t = 10^8$ sec (or 3.17 years)

thermal neutron flux $\phi t = 10^{14}$ n/cm², sec

fast neutron flux $\phi r = 10^{15}$ n/cm², sec

steel content Fe = 74 %

Co = 0.1 %

Ni = 10 %

Table III indicates the specific activity S in curies per gram of steel of isotopes whose half-life is greater than 70 days, at the end of irradiation, then, after periods respectively of:

$S_{0.3}$: 10^7 sec (0.3 years)

S_3 : 10^8 sec (about 3 years)

S_6 : 2.10^8 sec (about 6 years)

S_{16} : 5.10^8 sec (about 16 years)

S_{32} : 10^9 sec (about 32 years)

S_{63} : 2.10^9 sec (about 63 years)

S_{158} : 5.10^9 sec (about 158 years)

Table III

Parent Isotope	Type of Reaction	Active Isotope	Activities in curies per gram of steel			
			S_0	$S_{0.3}$	S_3	S_6
Ni 58	(n,p)	Co 58	4.4	1.4	7.10^{-6}	-
Fe 54	(n,γ)	Fe 55	1.5	1.4	0.71	0.337
Fe 54	(n,p)	Mn 54	0.67	0.51	0.043	0.003
Co 59	(n,γ)	Co 60	0.28	0.27	0.186	0.123
Ni 58	(n,γ)	Ni 59	2.10^{-5}	2.10^{-5}	2.10^{-5}	2.10^{-5}
Ni 60	(n,p)	Co 60	0.012	0.011	0.008	0.005

Table III [continued]

Parent Isotope	Type of Reaction	Active Isotope	Activities in curies per gram of steel			
			S_{16}	S_{32}	S_{63}	S_{158}
Ni 58	(n,p)	Co 58	-	-	-	-
Fe 54	(n,γ)	Fe 55	0.036	0.0009	-	-
Fe 54	(n,p)	Mn 54	-	-	-	-
Co 59	(n,γ)	Co 60	0.036	0.005	8.10^{-5}	-
Ni 58	(n,γ)	Ni 59	2.10^{-5}	2.10^{-5}	2.10^{-5}	2.10^{-5}
Ni 60	(n,p)	Co 60	0.002	2.10^{-4}	4.10^{-6}	-

Second Case

Activation of stainless steel of the type 18/8 used as reactor tank plating material.

Hypotheses:

irradiation time $t = 10^9$ sec (or 31.7 years)
thermal neutron flux $\phi t = 7.10^{10}$ n/cm^2, sec.
fast neutron flux $\phi r = 1.5 \cdot 10^{11}$ n/cm^2, sec
steel content Fe = 74 %
Co = 0.1%
Ni = 10 %

Tavle IV indicates the specific activities S in microcuries per gram of steel, of the isotopes whose half-life is greater than 70 days, at the end of irradiation, then, after the same periods as defined in the first case.

Table IV

Parent Isotope	Type of Reaction	Active Isotope	Activities in microcuries per gram of steel						
			S_0	$S_{0.3}$	S_3	S_6	S_{16}	S_{32}	S_{63}
Ni 58	(n,p)	Co 58	672	214	0.0013	-	-	-	-
Fe 54	(n,γ)	Fe 55	2060	1920	980	460	50	1.2	-
Fe 54	(n,p)	Mn 54	110	84	6.4	0.45	-	-	-
Co 59	(n,γ)	Co 60	690	660	460	300	89	11	-
Ni 58	(n,γ)	Ni 59	1.5	1.5	1.5	1.5	1.5	1.5	1.5
Ni 60	(n,p)	Co 60	5.2	5.0	3.5	2.3	0.67	0.084	-

Third Case

Activation of low alloy steel of the type used as material for a reactor tank.

Hypotheses:

irradiation time $t = 10^9$ sec (or 31.7 years)
thermal neutron flux $\phi t = 7.10^9$ n/cm^2, sec
fast neutron flux $\phi r = 1.5 \cdot 10^{11}$ n/cm^2, sec
steel content Fe = 97 %

Co = 0.01%
Ni = 1 %

Table V indicates the specific activity in microcuries per gram of steel of the isotopes whose half-life is greater than 70 days, at the end of irradiation, then, after periods identical to those defined in the first case.

Table V

Parent Isotope	Type of Reaction	Active Isotope	Activities in microcuries per gram of steel						
			S_0	$S_{0.3}$	S_3	S_6	S_{16}	S_{32}	S_{63}
Ni 58	(n,p)	Co 58	67.2	21.4	0.0001	-	-	-	-
Fe 54	(n,γ)	Fe 55	270	252	128	61	6.5	0.16	-
Fe 54	(n,p)	Mn 54	144	110	9.2	0.59	-	-	-
Co 59	(n,γ)	Co 60	6.9	6.6	4.6	3.0	0.89	0.11	-
Ni 58	(n,γ)	Ni 59	0.015	0.015	0.015	0.015	0.015	0.015	0.015
Ni 60	(n,p)	Co 60	0.52	0.50	0.35	0.23	0.067	0.008	-

COMMENTS

We must, first of all, underline the approximate character of the figures given in Table I to V. In fact, the effective cross sections of activation are far from being known with accuracy. In addition, real activation is brought about by a neutron flux whose energy spectrum covers a very wide range varying from one reactor to another and according to operating conditions. We did not take into consideration secondary reactions which isotopes undergo due to neutron bombardment for lack of appropriate data.

Having made these reservations, one can comment on the figures in tables III to V as follows.

About five years are necessary before the high activity of Co 58 and Mn 54 diminishes to a negligible level with respect to the level of activity of other isotopes.

Fifteen to twenty-five years are needed before the activity of Fe 55 comes down to a level comparable to that of Co 60 derived from Co 59.

After about 50 years the activity of Co 60 derived from Co 59 becomes inferior to that of Ni 59 in the materials out of which are built the reactor tanks. The corresponding time for shielding materials is about 75 years.

The activity due to Ni 59 is practically constant, the half life of this isotope being 80,000 years.

Co 60 derived from Ni 60 has, in the material out of which the reactor tank is built, an activity in the order of 10% of that of Co 60 from Co 59.

It should be noted that the isotopes Fe 55 and Ni 59 are not γ emitters but emitters of X-rays more easily stopped and, therefore, initially less dangerous than γ-rays. In fact, in the basic standards set up by Euratom, the isotopes Fe 55, Co 60, and Ni 59 are classed in the same category of radiotoxicity.

CONCLUSION

The advocates of the development of low cobalt steel seem not to have taken into account the reaction of Ni^{60} (n,p), Co^{60}, Fe^{54} (n,γ) Fe^{55} and Ni^{58} (n,γ) Ni^{59} in calculating the activation of steels.

Except, perhaps, in very particular cases and with the reservation of a more detailed study, it does not appear useful to lower the cobalt content of steels for nuclear reactors below the values specified and obtained at the present time, namely, 0.1% in stainless steels, and 0.01% in low alloy steels. A cobalt content 2 to 3 times higher is still permissible because, in any event, the activity due to cobalt in such concentrations remains comparable to that due to the essential components (iron and nickel) of the steels considered.

In any event, taking into consideration the lack of certainty which characterizes the values of certain effective cross sections of activation, it appears that an experimental measurement of the activation of various types of steels for reactor tanks is very desirable.

SAFETY PRINCIPLES IN THE INSTRUMENTATION OF REACTORS

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I take it that in on this Symposium I need hardly explain that a nuclear reactor - especially a materials testing reactor - is a dangerous apparatus. The total radioactivity in a reactor is soon in the order of 10^8 curies, and moreover the power level in the reactor is essentially an exponential function of time, which can be dangerous when power is increasing too rapidly.

We know that a reactor accident does not necessarily imply an atomic bomb explosion. A small explosion, however, is still possible, accompanied by - and this is by far the most serious aspect - the possible escape of radioactive material.

The safety measures which should therefore be adopted for a reactor are the following.

In the first place the reactor should be located in a thinly populated area. The hazard of an accident is then confined to the operating personnel, and as few "innocent" people as possible are affected in the surrounding area.

In the second place the construction and operating principle of the reactor should be as safe as possible. With a materials testing reactor, however, it is precisely on this point that we are not free in our choice, and we shall return to this subject presently.

In the third place we must ensure that operating personnel receives proper instructions. Regulations must be issued that <u>can</u> and <u>must</u> be complied with. Good administrative control is necessary, and a proper division of responsibility.

And then in the fourth and <u>last</u> place we must apply automatic safety devices which rule out the possibility of human errors, or at least minimize the seriousness of their consequences. And moreover these devices must also intervene promptly, where human reactions would be too slow.

Now in a materials testing reactor, automatic safeguards are particularly important because of its principle less attention can be paid to safety in its construction. A materials testing reactor operates with a high neutron flux, which entails a high degree of xenon poisoning. After a scram it is therefore desirable to start-up again as quickly as possible before the xenon begins to build up. A rapid start-up means a considerable increase in reactivity per unit time, and this can be particularly dangerous, involvius as it does, a short reactor period, and therefore makes automatic safeguards most essential. In connection with xenon poisoning, too, the excess reactivity is fairly high. Moreover, experiments constantly call for modifications to the design, and the presence of irradiated specimens and experimental loops involves the risk of a sudden increase of reactivity, even when the reactor is shut down. All these are factors that adversely affect safety.

The phenomenon of xenon poisoning leads us at once to formulate a further requirement to which the automatic safety system should conform : as far as possible the system should not operate needlessly, because if rapid restarting is not practicable for one reason or another, xenon poisoning may necessitate a fairly long shutdown, lasting several days. Needless shutdowns can also spoil experiments being carried out. Therefore the automatic safety system must accordingly be safe, that is to say it must operate when it is really necessary; but it must also be reliable, that is to say it must not intervene unnecessarily.

In this paper I should like to say a few words about how a safety system can be made both safe and reliable with more or less unsafe and unreliable instruments. Many of the points I shall make are not new. Fortunately perhaps, a certain tradition is gradually being established in the design of protective systems for nuclear reactors.

I have said that an automatic protective system must, in the first place, be safe (it must operate when it is necessary) and in the second place it must be reliable (it must not operate unnecessarily). The first requirement of safety implies that any internal fault in the safety system ought in a certain sense to lead to the shutdown of the reactor. The system must be fail-safe. Evidently, this is in conflict with the requirement of reliability, for the system now operates, although it should not really do so. The reactor itself is not unsafe at all.

There are in fact two ways of reconciling both requirements. The first is to make the automatic protective system so simple as to be both safe and reliable. Usually this can only be done at the expense of accuracy. The second method is to use not one but several protective devices, thereby increasing the safety and generally reducing the reliability. The latter drawback can be partly eliminated by means of coincidence circuits.

In 1957, Siddall in Nucleonics worked out numerically the degree of safety and reliability in a simple system, a 1 out of 2, a 2 out of 2, a 2 out of 3 system and some other systems that are not of interest for us now. In a 1 out of 2 system there are two equivalent channels. When only one of these channels gives an indication of a unsafe condition, the reactor is scrammed. In a 2 out of 2 system the indication of both channels is needed for scramming. In a 2 out of 3 system, at least 2 channels here to give an indication. I should like to show you the table in which the results of the calculations are listed but I should point out that I have made some changes in the assumed numerical values of certain quantities.

System	Frequency of accidents due to unsafe failure of one of the chanoels	Numerical value Measure of safety	Frequency of scrams due to safe failure, tests and maintenance	Numerical value Measure of reliability
1 out of 1	$n_e n_u t_u$	1 per 175 years	n_s	2 per year
1 out of 2	$n_e(n_u t_u)2$	1 per 31000 years	$2n_s$	4 per year
2 out of 2	$2n_e n_u t_v$	1 per 4400 years	$2n_s(n_s t_s+n_t t_t+n_o t_o)$	1 per 190 years
2 out of 3	$3n_e(n_u t_v)2$	1 per 25.10^6 years	$6n_s(n_s t_s+n_t t_t+n_o t_o)$	1 per 64 years

After E. Siddall - Nucleonics June 1957.

n_e = Frequency of situations requiring intervention of automatic safety system (1 per year).

n_u = Frequency of unsafe failure of one channel (one per 2 years)

t_n = Average duration of unsafe failure before it is noticed (no test facility) (100 hours).

t_v = Same as t_u, but with test facility available (2 hours).

n_s = Frequency of safe failure of one channel (2 per year).

t_s = Average time needed for exchanging defective apparatus (6 minutes).

n_t = Frequency of testing (1 per 4 hours = 2200 times per year).

t_t = Time needed for testing a channel (10 seconds).

n_o = Frequency of maintenance (50 times per year).

t_o = Average duration of maintenance (6 minutes).

A 2 out of 2 and a 2 out of 3 system both offer facilities for testing during operation. If this is done every 4 hours, for example, a fault remains in existence for an average of 2 hours.

The table makes it clear that, compared with a 1 out of 1 system, a 1 out of 2 system increases safety but not reliability - rather the contrary - whilst a 2 out of 2 system noticeably increases reliability but only slightly improves safety. In principle, the whole system is really made less safe, but owing to the possibility of testing during operation there is in fact a net gain in safety.

The only system which provides both increased safety and improved reliability is a 2 out of 3 system, that is one consisting of three channels with a coincidence circuit.

Of course, the assumptions in the table are evidently somewhat simplified. Moreover, the data are valid only when no correlation exists between the faults. Correlation can adversely affect both the safety and the reliability of the system, and must therefore be guarded against. I am thinking here, for example, of supply voltages, which are more or less common to all apparatus.

The table, then, points clearly to the superiority of the 2 out of 3 system. Of course, such a system is admittedly expensive, but since the entire instrumentation of a big reactor amounts to little more than 1% of the total costs, this economic consideration is not very important.

The foregoing remarks apply to only one subsystem, as for example a temperature or a flux monitoring system.

Now a complete safety system very soon adds up to a large number of subsystems. With a materials testing reactor, for example, the aim will be to safeguard the following quantities : neutron flux, low-power period (of C.R. channel), high-power period (of logarithmic currentmetering channel); primary flow; secondary flow; reactor outlet temperature; fuel element temperature; helium pressure; heavy-water leakage.

In general it can be said that some ten subsystems are needed, and the question we must now consider is how we are going to connect these systems one with the other. Diagrams show the possible ways in which this can be done.

The two contacts 1a in the first two posibilities are contained in the same apparatus, and there is thus always a certain chance of a short-circuit occurring between these contacts and their supply leads. In case (1) a short-circuit of this kind can put one subsystem entirely out of operation. In case (2) the consequences of such a short-circuit may be much more serious. For this reason, and also because it is simple and straightforward, system (3) is the one generally chosen. It does, however, make use of follow-up relays (A,B and C), which causes en additional time lag of 5 to 6 milliseconds. It might now be remarked that a short between two contacts of could put the whole system out of operation. This is quite true, but the chance of this happening, however, can be minimized by devoting great care to the construction of the coincidence circuit.

Having accepted circuit (3) we can again try to evaluate numerically the reliability of the whole system. There would not be much point in evaluating the degree of safety. This is certainly better than the safety of one subsystem, but the subsystems complement each other or overlap.

As regards the reliability of system (3), it is of course true that if one of the contacts a is opened together with one of the contacts b or one of the contacts c, the result will be a scram. With ten subsystems we thus arrive at a reliability 100 times lower than that of one subsystem. A false scram can therefore be expected once every 7½ months. Evidently, then, the reliability of the system is now relatively low.

For a power reactor the figure is in fact unacceptable. But even for a materials testing reactor this degree of reliability is on the low side.

The reliability of a 2 in 3 system was thus $6\ n_s(n_s t_s + n_t t_t + n_o t_o)$.

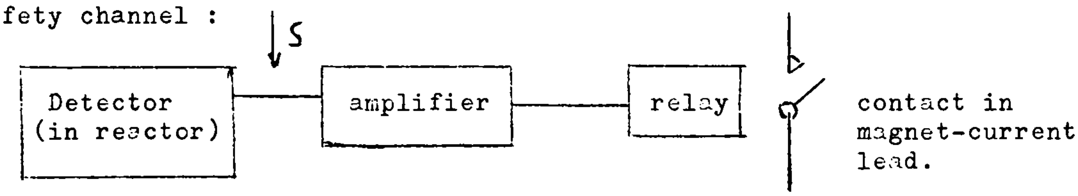
Whith the assumed values, this is numerically :

$$6 .\ 2 .\ (2.3 + 70 + 57)\ .\ 10^{-5}.$$

It can thus be seen that the reliability is virtually directly proportional to n_s .

In the case of power reactors, therefore, one channel will certainly have to give appreciably less than 2 false alarms a year.

I should like to return for a moment to the subject of safety. We saw that the possibility of testing during operation provided a high degree of safety, but really we were a little <u>too</u> optimistic, for the testing referred to did not extend to the <u>whole</u> channel. In this connection, let me draw for you a block diagram of a safety channel :



When testing it is usually only possible to apply a signal to the position S. This means that the detector is <u>not</u> included in the test. We do not actually increase the flux in the reactor during a test. On the other hand we do not let the rods drop either. We really only test the "amplifier" and the relay, in which process we can ascertain whether the armature is sticking and perhaps even see whether the contact is open.

To ascertain what happens in the magnet-current chain, Siddall proposed the following solution. In series with each contact in the coincidence circuit he put a resistor R. In the normal state there is thus a resistance of 1.5 R included in the chain. When one of the relays opens, this resistance is raised to 2.5 R. The values of the resistances can now be selected such that, as a result of this resistance increase, the magnet current drops by, say, 10%. The presence of faults can then readily be determined.

But this still leaves the danger that a fault at the detector side will remain unnoticed for a very long time. The way to get around this difficulty is to compare the three signals from the detectors, preferably at the output of the amplifiers, and when one of the signals deviates impermissibly from the other two to give an immediate alarm. In this case, even a defective detector will remain unnoticed for only a short time. The test now takes care of the amplifier and everything that comes after it, and the signal comparison takes care of the detector and the amplifier, so that there is a certain amount of overlapping.

Let me now briefly recapitulate : the 2 out of 3 system provides a high degree of safety, mainly because of the facility for testing "in situ". Safety also demands careful attention to various subsidiary factors, such as the spacial arrangement of the relays, the position of the leads, and so on. The 2 out of 3 system also provides reasonable reliability, which, however, is endangered when there are several subsystems employed. For a whole system to be satisfactorily reliable it is therefore necessary that the individual channels should give a false alarm appreciably less than twice a year.

Testing should be extended as far as possible over the whole channel, and it is desirable to be able to observe changes in the magnet current. A faulty detector can be quickly discovered by comparing the test signals.

I should now like to say a few words about a protective system that has been devised by our team. In the design of this system, efforts were made to reduce the number of components to the minimum in the interest of both safety and reliability.

The circuit in question is a flux safety channel, the underlying requirement being that an ion chamber current of 75 /uA should cause a relay contact to open within 5 milliseconds.

The anode current of a thyratron Th contains a relay coil A and is supplied with a square-wave voltage of 2000 c/s frequency. The current I from the ionization chamber flows through the resistor R, which connects the grid to the cathode. The grid voltage is thus a direct voltage proportional to the current I, and hence proportional to the neutron flux at the position of the ionization chamber. As long as I is smaller than a critical value the thyratron passes a constant current during half of each cycle - owing to the square-wave form of the supply voltage - and this current actuates the relay. If I exceeds the critical value, the grid voltage goes so negative that the thyratron can no longer ignite, and consequently the relay cuts out, thereby opening the contact A in the safety chain.

The thyratron, type PL2D21, has a screen grid, which is connected to the cathode. The relay coil is shunted with a capacitor C for by-passing the alternating component of the anode current. The point at which the circuit enters into operation (the scram point) can be adjusted by the variable resistor R_1. This resistor also serves to correct the spread in the characteristics of the thyratrons.

The electronic circuit is duplicated in order to increase the degree of safety.

I should now like to mention some points to which particular attention was paid in the design.

The screening of the grid circuit is not connected to earth but to a point at a potential of -300 V. This ensures that a scram signal will certainly appear in the event of a short-circuit, and from the screening point of view the 300 V voltage source has the advantage of a low impedance to earth.

Another feature is that no point in the circuit has a positive potential with respect to the thyratron cathode. There is therefore no possibility of a positive direct voltage appearing on the anode as a result of a short-circuit; this would be dangerous inasmuch as it would keep the thyratron ignited and hence the relay would remain energized.

Another important point worth mentioning is the deliberate use in the circuit of a rectifying element, namely the thyratron. This enables the circuit, in which a DC relay is to be energized, to be fed from a source of alternating voltage. In the event of a short-circuit between anode and cathode, or between anode and screen grid, the relay will therefore be de-energized.

As I have said, the safety amplifier is fed by a square-wave voltage, the reason being that the anode-current pulses then always last for half a cycle and thus the average anode current is independent of the grid voltage. The frequency is chosen as 2000 c/s because one complete cycle can elapse before the relay current is interrupted, 2000 c/s being equivalent to 0.5 milliseconds. Added to the 4 milliseconds opening time of the relay (type T51C), this gives a total time of 4.5 milliseconds. The requirement was less than 5 milliseconds.

The square-wave voltage generator employed, consists of a multivibrator which delivers a square-wave voltage of 280 V peak-to-peak. This is almost the maximum obtainable with 300 V supply voltage, and it is virtually independent of the valve characteristics. The power gain is obtained with cathode followers; these have a poor efficiency, but they have the advantage that they can never give a voltage gain greater than unity, again independent of the valve characteristics.

Finally, a tight coupling in the transformer ensures that the secondary voltage cannot give rise to overshoot transients.

This, then, is an example of how special circuits are being designed to improve safety. Reliability, too, is improved by the use of fewer components, by chosing these components with particular care and, above all, by using them correctly.

On the question of correct usage, I should like to make the following remark. Kilbey, in his article in Nuclear Engineering of May 1957, quotes a nice example of a scaler, originally developed with "Inter Service approved components", but later completely re-designed to a somewhat lower specification, it having been found that the high requirements were not needed. The maximum temperature above ambient was reduced from 50° to 20°C, better components were used, an improved layout was made and the tolerances made more critical. These modifications reduced the annual failure rate in percentage population for the valves from about 10% to roughly 1%. This means to say that, of every ten valves used in the apparatus, an average of one per year previonsly became defective, whereas after the modification the average failure rate was only 0.1 per year. This numerical example shows clearly how very important it is to use the right components in the right way.

The circuit I have just discussed is merely one example, chosen because I was closely connected with its development. It is certainly not the last word in this field, and in any case it was developed quite some time ago. But even in the new transistorized safety system developed by Mullards you will still find one of the principles I have mentioned, namely the deliberate use of a rectifying element.

To sum up, I have tried to show you that, although it is an art to make a safety system that will in fact function at the rare moments when it is required to function, it is perhaps an even greater art to ensure that, during all the time that it is not needed, it will in fact not function.

TECHNOLOGICAL ASPECTS OF CONSTRUCTION OF THE BR-2 REACTOR TANK

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I. DESCRIPTION OF THE TANK (Fig. I)

The body of the tank presents the general aspect of a devil made up essentially of an upper tie for fixing the cover, of a truncated cone part attached to a cylindrical collar which, in turn, is prolonged by a truncated cone and a cylinder provided with a terminal tie for fixing the bottom.

It also comprises piping with reinforced sheets for the feeding and draining of water, a support for the skirt holding up the tank, interior rings welded to the wall for the fixing of the interior parts, and a deflector facing the water feed-in piping.

Construction Particulars

In order to meet the requirements of the ASME standards, the circular assembly joints have been separated from the intersections between the cones and the cylinders. This arrangement has made it necessary to effect an operation of restricting the truncated cone collars at their end largest in diameter and the widening out of the central collar at its two ends.

II. PROBLEMS OF MANUFACTURE

A. Supply of Raw Materials

The primary consideration was to reduce to a minimum the number of welds to be made on the tank.

It was therefore advisable to obtain:
- the two ties in the form of rings forged in one piece;
- sheets permitting to build the structure of the tank with a minimum number of elementary collars, that is to say, four, each one of them having a single longitudinal joint.

Ties: The suppliers were confronted with two difficulties. The first one related to the weight and the dimensions of the parts; the second concerned the elastic limit of the material which had to reach 11 kg/mm^2 in the annealed state.

The solution was given by a French firm, the Société Forgeal.

They delivered to us two forged rings, without welding, with the dimensions indicated on Figure II, of an alloy having the elastic limit desired and whose composition was as follows:

Mg : 3.5 to 4.5 percent
Mn : 1 percent maximum
Fe : 0.5 percent maximum
Si : 0.5 percent maximum
Cu : 0.05 percent maximum

These rings, examined fully with ultra-sonic equipment, were found to be free from any internal physical defects.

Sheets: Alloy chosen from the range standardized in the USA, under the designation, 5052 - 0.

Acceptance according to ASTM standard, with ultra-sonic checking by complete sweeping of the two surfaces of each sheet, to be sure of the absence of all fissures, flaws or inclusions.

Desired Dimensions in Order to Have Only a Single Longitudinal Joint (Figs. III and IV)

for frustum of upper cone: one sheet of 7.400 m x 3.400 m x 48 mm; weight: 3300 kgs.

for frustum of lower cone: one sheet of 6.800 m x 3.200 m x 63.5 mm; weight: 3700 kgs.

frustum of lower cylinder: one sheet of 7.100 m x 1.550 m x 63.5 mm; weight: 1900 kgs.

There was no problem for the central collar.

Among those consulted, the best outfitted dealer had only scarified ingots of a maximum weight of 2 T. and his capacity of rolling was limited to 2.800 m in width.

It followed that the design with one single longitudinal joint could not be maintained except for the lower cylindrical collar and that for the cone frustums we could be forced to use two longitudinal joints.

Finally we ordered four main sheets:

two of dimensions 3.620 m x 2.600 m x 48 mm, each weighing 1,225 kgs.

two of dimensions 3.800 m x 1.920 m x 63.5 mm, each weighing 1,300 kgs. Fig. V.

each cone frustum being obtained from two shells, as shown in Fig. VI.

The realization of the central collar presented a somewhat different aspect. Indeed, its relatively small dimensions would have permitted us to design it with a single longitudinal joint. This solution which was envisaged for a while, was abandoned because we feared cracking in the welds during the widening operation.

We therefore preferred making it up likewise in two shells shaped, before welding, by hot stamping (Fig. VII).

All of the rolled sheets were delivered by Usine d'Issoire de la Société Francaise CEGEDUR, the materials exactly meeting our specifications.

B. Fabrication in the Shops

1) Roughing Work on the Ties on the Vertical Lathe (No Problem)

2) Cutting up the Sheets

- Cutting with band saw and handling of the sheets on ball-bearing table: abandoned for the following reason:

The sheets not being perfectly flat, their weights carried only on some ball bearings which had a tendency to be blocked in their housings; this made the guiding practically impossible.

- Cutting with argon arc: abandoned because at that time the process was not developed to the point of cutting thicknesses of 48 and 63 mm.

- Cutting with portable circular saw: adopted for the rectilinear ridges.

- Drilling on vertical lathe: adopted for the curved sections. This operation was carried out on a vertical lathe having a plate diameter of 12 m.

Shaping of Collar Components

The bending of the half-cones and of the large cylindrical collar was done on an hydraulic press of 350 T by means of successive crushings, according to the sources; the tools used were covered with rubber of one cm thickness in order not to injure the sheets.

The shells of the central collar were obtained by stamping with the aid of a shaping tool, the blanks being previously heated to about 350°. No special precaution was necessary to guarantee the surface condition as this component, whose initial thickness was about 54 mm, had to be brought back to 21.4 m when machining the tank.

Execution of Longitudinal Welds

The method of welding adopted was that of argon-arc welding with fusible electrode, the torch being guided manually and fed by direct current from a set with flat characteristics.

The welding line was slightly richer in magnesium than the base metal (3.5 percent instead of 2.8) in order to reduce the tendency to cracking.

The diameter of the welding wire was 1.6 mm for the angle welding and 2.4 mm for the end-to-end joints.

Before being used, each roll of wire was dried in the oven for one hour at a temperature of 80° in order to eliminate all traces of moisture.

The shielding of the arc was attained with the help of pure argon and the necessary precautions taken to assure the continuous protection of the melt bath in order to avoid its oxidation (avoid air currents, even such arising from the natural convection due to preheating).

To maintain the pressure of the arc and of the welding stream within very narrow limits was one of the determining factors of the quality of the joints (of the magnitude of $\pm$ 5%) with reference to the correct values gave rise immediately to the appearance of pits in the metal deposited.

And finally, the shape of the chamfer, the scraping of its blanks and the examination of the latter by means of penetrating dye, showed themselves to be of extreme importance for obtaining fully sound welds.

The welding itself was carried out in the following manner (Fig. IX):

- Preheating of the pieces to about 175°C maintained at that temperature by means of a blow-torch during the entire duration of the welding.
- Filling in of the chamfer, interior side, up to mid-height (Fig. IX-1).
- Chiselling of the bottom up to a depth required in order to eliminate the defects (Fig. IX-2).

- Radiographic examination and penetrating dye test.
- Possible retouching.
- Filling in of the chamfer, exterior side, up to mid-height (Fig. IX-3).

- Radiographic examination.
- Possible retouching.
- Finishing of the weld on the inside (Fig. IX-4).

- Radiographical examination - retouching.
- Finishing of the weld on the outside.
- Radiographic examination of the weld completed finished.
- Examination of the latter with ultra-sonic equipment.

It should be noted that between each layer of weld, a careful cleaning with pneumatic chisel was carried out in order to remove all traces of oxide which might have been deposited on the surface.

Restriction of Truncated Cone Collars

Of all the methods considered, we gave preferrence to the method of chasing on the lathe used at the present time in the fabrication of bulged bottoms.

To eliminate the risks of cracking, the pieces were preheated with a blow-torch to a temperature of about 300°C which was maintained for the complete duration of the operation.

The temperature was carefully checked during the whole time so that the dimension measurements taken in the course of the operation might be corrected for the expansions due to preheating.

The Making of Circular Assembly Welds

The methods followed with respect to the longitudinal joints were repeated exactly as in the work on the longitudinal joints.

It should be noted, however, that in this special case the pieces, mounted on a roller handling device, could be moved by means of a very slow rotational movement making it possible for the welder to always work in the most comfortable position.

Two types of assembly were conceived:

The first (Fig. X), used for the welding of a tie on a cone frustum, had to fulfill the following conditions:

1. To turn the part about its axis without lateral displacement.

2. Leave the parts free to make their transverse contraction.

3. Roll on the tie with the rollers subjected to heating of 175°.

4. Roll on the cone without injuring the surface of the parts.

This was carried out as follows:

Tie: Cylindrical rollers of steel, floating in a housing filled with water and serving as driving rollers.

Cone: Conical aluminum rollers, free, arranged in such a way that the vertex of the cone sheathing of the part coincides with that of the cone sheathings of the rollers.

The second type of assembly (Fig. XI) was adopted for the welding of the central collar, on the lower cone.

Such assembly has to fulfill similar conditions as those of the preceding one; moreover, it had to avoid subjecting the bond to bending before the welding is completely finished.

It was for this reason that supplementary rollers were provided to the right of the central collar, on a base having an elastic suspension, allowing the exertion of a support force upward, by following the variations in rise due the small excentricities of the raw pieces.

At the present time the tank, being completely welded, is ready to undergo machining which, in several months, will bring it to the final stage of testing.

Nevertheless, we do not wish to wait any longer to thank the various organizations who in various ways took part in the carrying out of the work we have just described.

We wish to make special mention of the Bureau d'Etudes Nucléaire which approved most of the decisions taken in the matter of the solutions which we proposed; to the Association VINCOTTE which cooperated in a very active manner in the various checkings done both on the raw materials as well as in the workshop operations; and, finally, the Centre National d'Information de l'Aluminium which offered us precious assistance particularly in the supply phase and in the study of methods and procedures to be followed.

fig. I

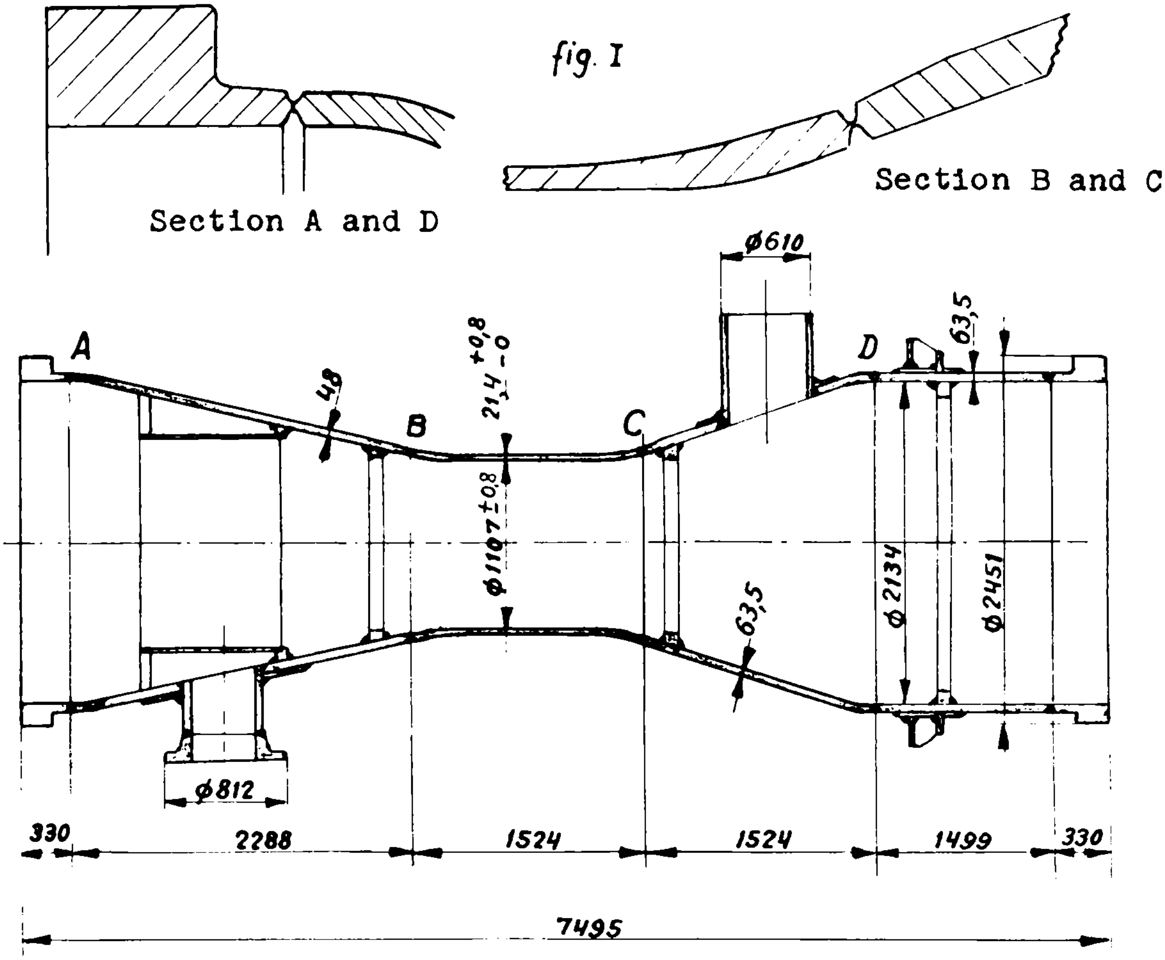
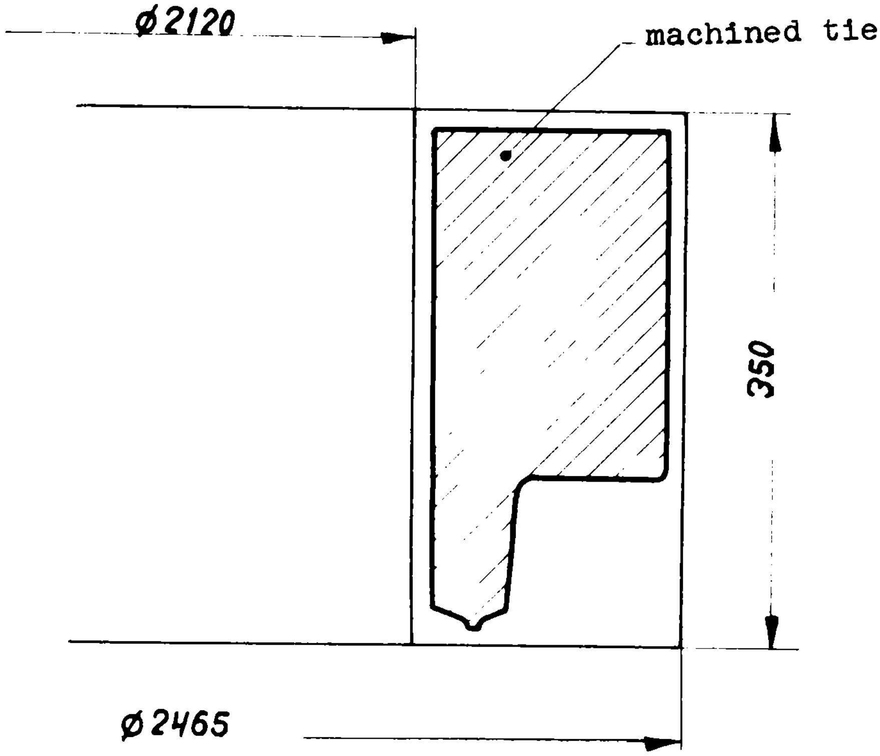
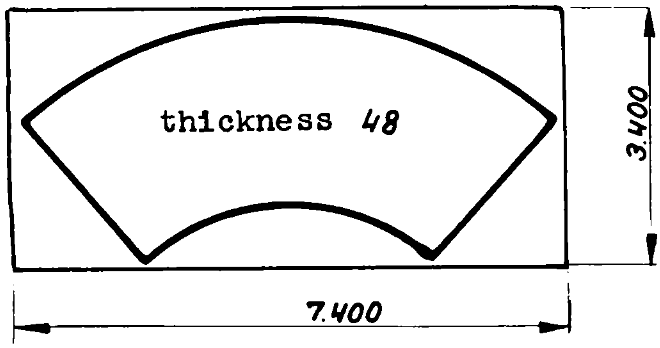


fig. II



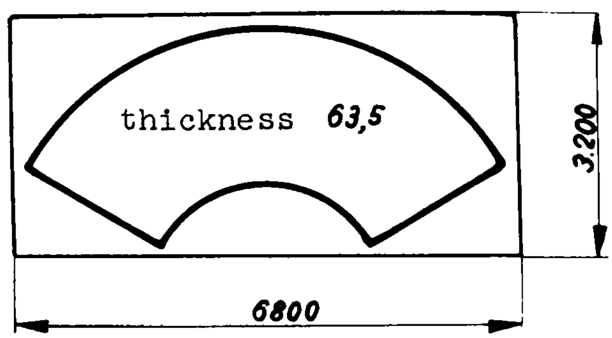
Dimensions of forged, raw ring
Weight of raw ring 1180 kgs

Upper truncated cone collar (1 seam)



Gross weight 3.300 kgs

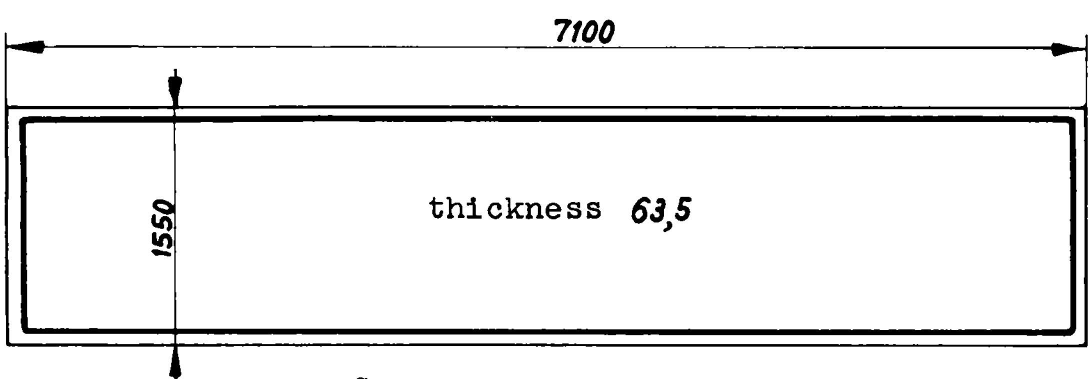
Lower truncated cone collar (1 seam)



Gross weight: 3.700 kgs

fig III

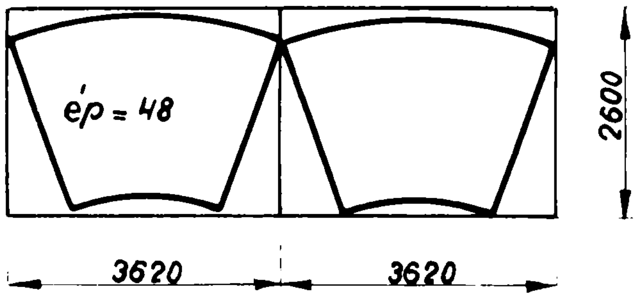
Lower cylindrical collar



Gross weight: 1900 kgs

fig IV

Upper truncated cone collar of 2 sheet metal plates



Gross weight of 2 sheet metal plates = 2450 kgs

Lower truncated cone collar of 2 sheet metal plates

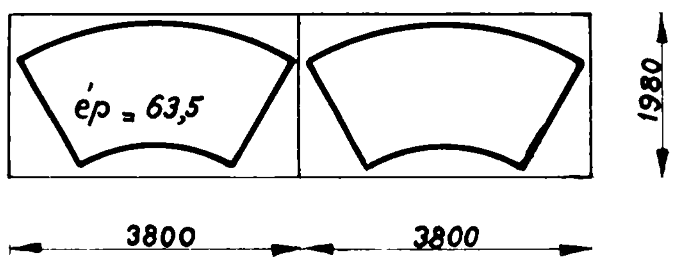


fig V

Gross weight of 2 sheet metal plates = 2600 kgs

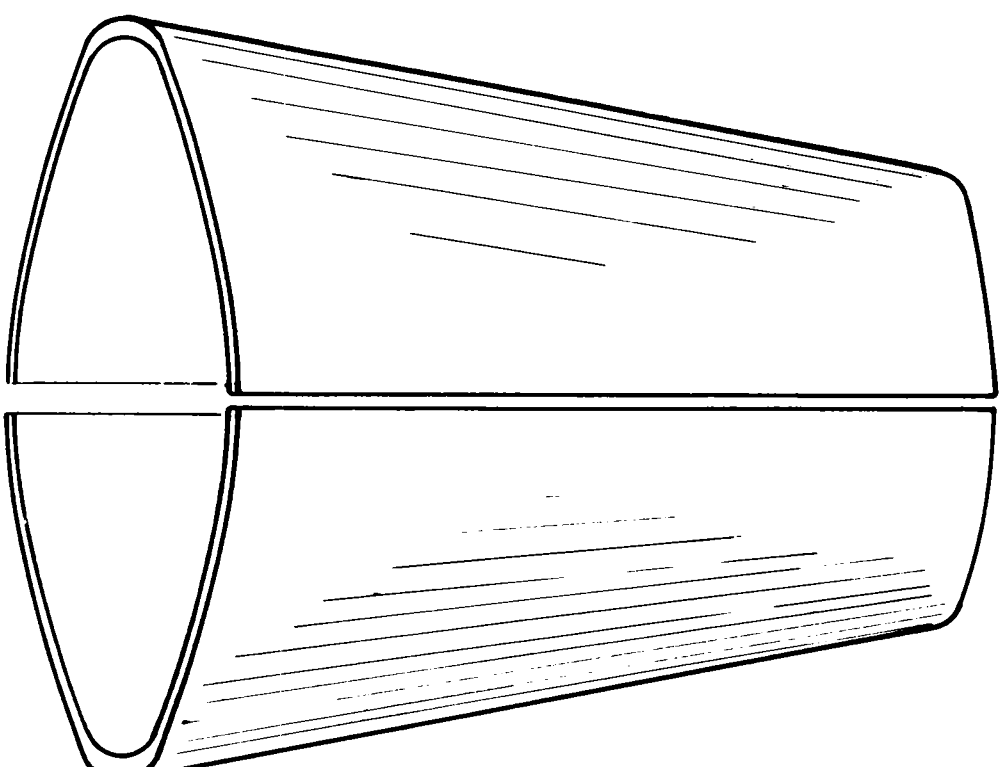


fig VI

Shaping of truncated cone half collar

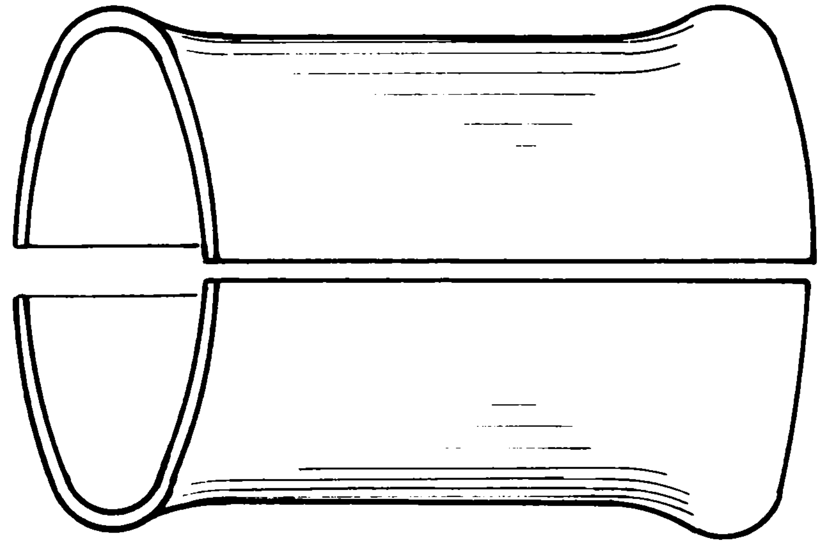
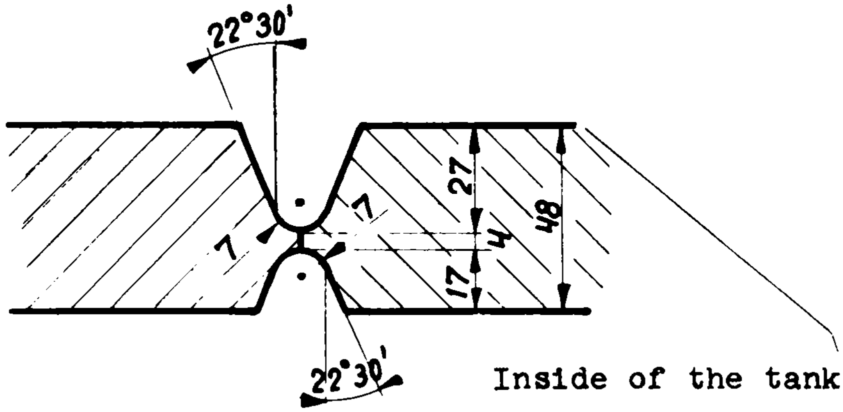


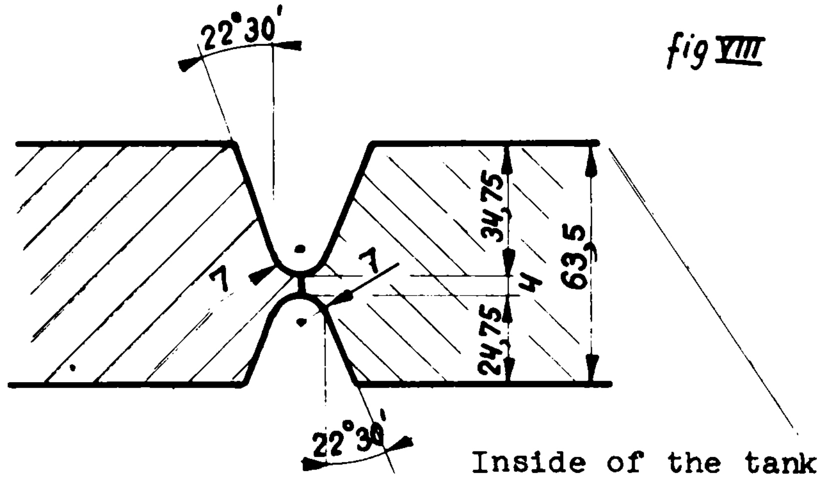
fig VII

Shaping of central half collars



Preparations for welding

fig VIII



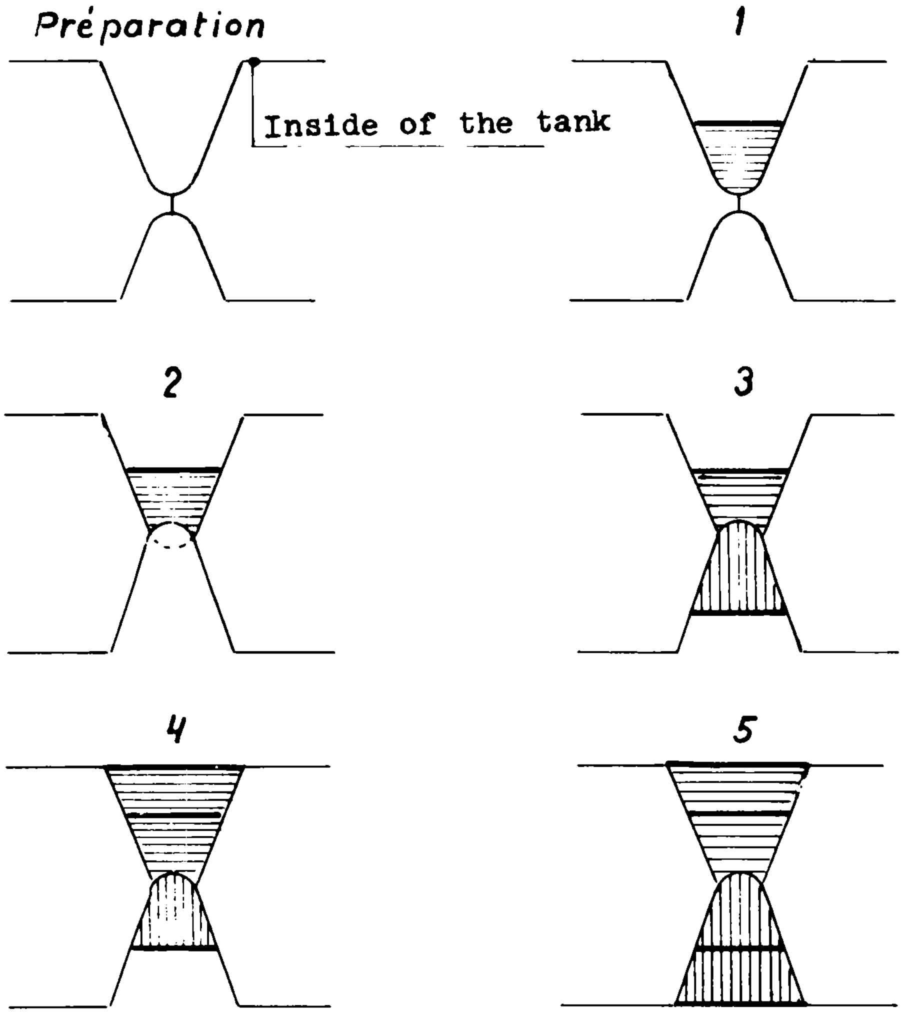


fig IX

Bonding of upper cone-collar

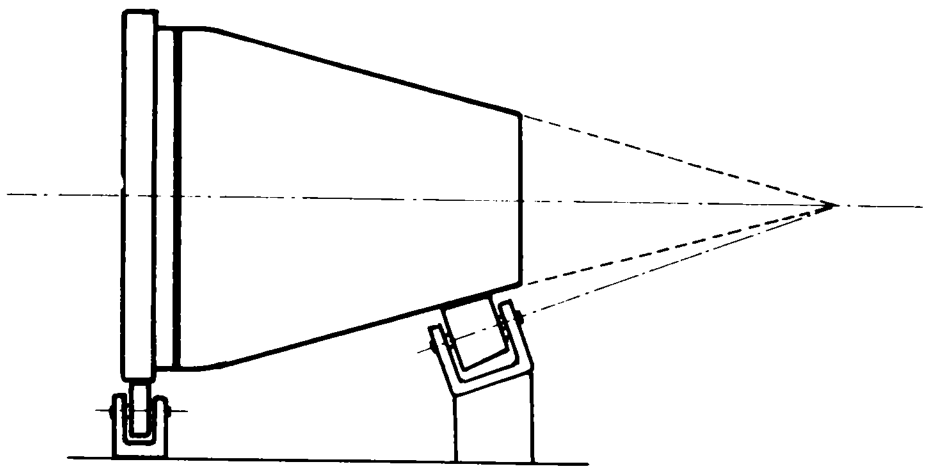


fig. X

Bonding of central collar-lower cone

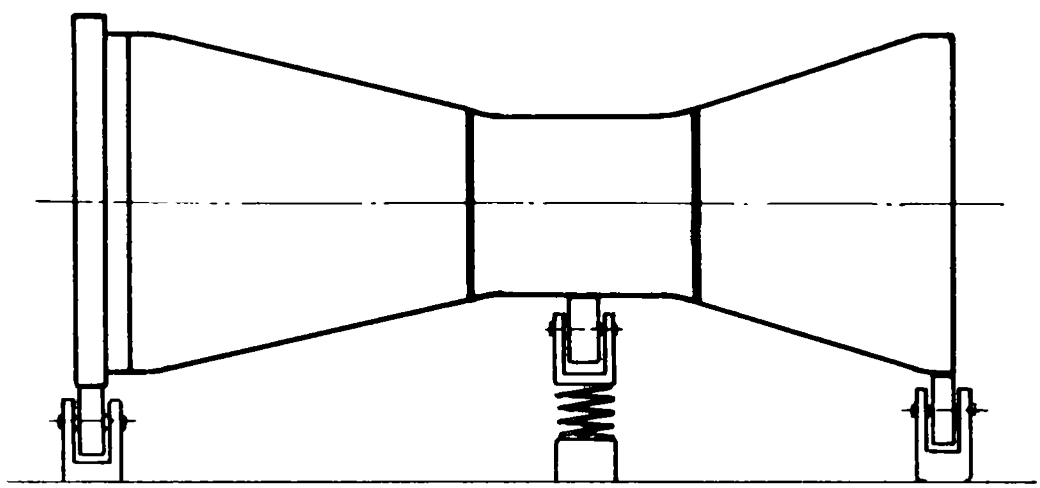


fig XI

OPERATING EXPERIENCE WITH IN-PILE LOOPS AT ORNL

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ABSTRACT

A series of in-pile loops have been operated in the ORNL research reactors over the past fifteen years. Among these are loops cooled with water, gas, single-pass air, sodium, and helium. Fuel loops of aqueous uranyl sulfate and molten salt have also been operated. There has been no major difficulty, but a number of minor incidents have occurred, and these have been the basis for development of a safety system designed to prevent interference with reactor operation or hazards to personnel.

A small organization has been developed to evaluate hazards and to provide a continuing safety surveillance of loops and other experiments. It has been necessary to provide standards for containment, temperature and pressure monitoring, shielding, release of radioactivity, and other safety factors. These, together with reactor design features found desirable, are described. The instrument safety system used for loops is a double-channel type requiring response from only one channel to cause corrective action and having fail-safe features. With this system a very complex loop may cause spurious shutdowns of the reactor unless the instrumentation is reliable. Methods of improving the instrumentation and the results achieved are presented.

Operating and safety experience with various loops is described.

INTRODUCTION

Loops of various types have been operating in ORNL research reactors for the past 15 years. These have ranged from the simplest type of systems such as air and water-cooled loops for the irradiation of nonfissionable target materials to the very complex liquid-fuel loops containing molten salt or aqueous fuel solutions. These loops have all had properties which might have developed into hazards to personnel, the reactor, and valuable equipment had safeguards not been employed. Even with the use of safeguards considered adequate by the experimenters and by technical reviewers of the loop designs, some minor incidents have occurred — due primarily to malfunction of equipment but sometimes to an oversight by the reviewers. As experience has been gained, the liklihood of an oversight by the reviewers has been nearly eliminated and equipment selection has been greatly improved.

As the loops have become more and more complicated the need for standards of general design, materials, welding, instrumentation, operational practices, and safety devices has become apparent. Since safety devices fail occasionally, the increasing number of such devices increased the number of malfunctions which shut down or "scrammed" the reactor. Loops which operate at considerable fission power have become commonplace. Often these experiments contain the fission products at high temperature and pressure and in a liquid state so that they could be released in hazardous quantities if a leak occurred.

The purpose of this report is to describe the loops and their use and to list some of the causes of the incidents which have occurred — with the standard methods which have been devised to prevent their recurrence. There is also in nearly every experiment a long list of recognized possible failures which might have occurred had they not been prevented by design and fabrication changes made as a result of reviews at the time the experiment was designed.

Some Types of Failure

In order to emphasize the importance of the application of the standards which have been developed with respect to design detail, material and equipment selection, fabrication, operational procedures, and personnel competence, a representative list of failures and hazards follows.

1. A relay in a safety circuit was a type which used easily bent leaf springs to hold the contact points together. Jarring of the relay cabinet caused momentary opening of the relay and several unnecessary reactor scrams before the trouble was located and the relay replaced by one using coil springs.

2. A glass insulating seal cracked, allowing leakage of radioactive gas to the coolant region of an air-cooled molten salt loop. A later surge of air pressure displaced a gasket on a flange causing the radioactive air to be blown out into the surrounding area. Evacuation of personnel from the area was necessary, and a day of reactor operating time was lost.

3. A capsule containing an aqueous fuel solution ruptured due to ignition of accumulated hydrogen and oxygen gases. The resulting pressure surge in the cooling-air line blew a hose connection loose and released radioactive gases into a room containing personnel. Decontamination of personnel and the room was necessary. A half day of reactor operating time was lost.

4. A shield used to remove a loop section had large voids in its walls, and personnel were exposed to radiation during subsequent operations. The exposure was not excessive for any one person, but a large number of people had to be used to divide the exposure time.

5. An air pocket collected in a horizontal water-cooled loop. This air oscillated, causing the reactor power to oscillate so that the data from some other experiments became useless.

6. A capsule in a vertical air-cooled loop oscillated up and down, causing reactor power oscillation.

7. A flow monitor connected to a safety circuit was found stuck in the full flow position during a routine check. Had the flow failed, no safeguard action would have been initiated.

8. A dynamic type of flow monitor which generated a pulsing signal from a rotating vane operated by the flow of the liquid being monitored was frequently stopped by particles in the liquid. This caused several unnecessary reactor power reductions.

9. On many occasions failure of thermocouples due to their breaking off the surface being monitored or developing bypass shorts in the lead wire have robbed a test specimen or entire loop of its safeguards and/or data indicators so that the run had to be terminated. These occurrences have been expensive in loss of data and loss of reactor operating time.

EXTENT OF DESIGN REQUIRED

The design of a loop must be as thorough as possible, with nothing left to the discretion of the fabricators except final sizing or fitting of those portions of the work which can only be done in the field. This means that all dimensions must be set; all routing of piping and wiring must be exactly prescribed; the location of each piece of equipment must be described; the material of each piece of pipe, screw, plate, gasket, weld, etc., specified exactly; and the specifications given for each pump, motor, sensing element, electrical switch or relay, valve, etc.

It is necessary that the design layout be done on standard blueprints so that the information given to the fabricators will be identical to that given the reviewers of the experiments. The use of crude sketches must be avoided except to provide information to draftsmen. This is necessary to ensure that competent engineers and experiment reviewers will be provided copies of the drawings and to ensure that copies will be available in reference files.

It is important that designers be familiar with the special problems encountered in nuclear reactors. For example, equipment is sometimes designed with very small clearances which may be perfectly acceptable in ordinary mechanical applications. In a reactor, however, if experiment apparatus fits very tightly it may jam and gall, and because all operation must be done remotely the removal of such apparatus becomes a very serious problem. An example is the shielding plugs used to shield radiation from beam holes. Such plugs have jammed inside beam holes and great force has been necessary to loosen them. Another example is that of a pneumatic tube approximately 1 1/4-in. in diameter which was put inside a liner with 1/32-in. clearance. This tube was approximately 25 ft long, and when attempts were made to remove it after 5 yr the tube jammed, galled, and finally pulled apart. It was necessary to remove the jammed section by drilling from approximately 12 ft outside the shield. The consequences of jammed components such as this may be serious, and very careful attention to these features must be given by the persons inspecting experiments.

In tying thermocouples to a high-temperature loop, it is sometimes found that, if sufficient slack is not provided in the thermocouple, thermal expansion in the loop will pull the thermocouples off.

Gamma heating of loops and experimental components is sometimes very serious. In the ORR, for example, an uncooled aluminum component in an experiment hole inside the lattice would quickly melt. At the LITR, where the average thermal flux is about 2×10^{13} n/cm^2/sec in a similar lattice, the temperature of uncooled aluminum components will rise to about 400°C.

Welding specifications on experiments should be standardized, depending on the material and the application. Failure due to this can result in leakage of capsules and loop components. Such a capsule in the ORR containing fissionable material developed a slight leak, causing the radioactivity in the building air to rise quickly to a factor of 10 above normal. Although this was not sufficient to warrant evacuation of the building, it is still an undesirable situation. Such leaking of welds is often very difficult to find. In this particular case it was not located until the experiment was being removed from the reactor. This operation apparently released enough gas that the building air activity rose above tolerance and the building had to be evacuated for several hours.

SAFETY

Hazards Evaluation

The evaluation of hazards must begin with the original conceptual planning of a loop, since a great deal of the design work is for safeguard equipment to eliminate or minimize the possibility that an incident will develop from such hazards. As a first approach, all conceivable hazards (however remote) should be listed by the planners of the loop. Those which appear to be realistic, after they have been reviewed carefully, must be considered in the design.

Reactivity Effects of Experiments. It is obvious that one of the hazards of experiments is in the reactivity changes which may occur when an experiment tube inside the lattice of a reactor fills with water, moves, or otherwise changes in such a manner as to affect the reactivity of the reactor. Some experiments are designed so that portions may be moved into or away from the high-flux zone. In making a hazard analysis, the question always arises as to how much effect such changes may have. It is difficult to calculate reactivity effects due to other experiments, burnup of fuel, nonsymmetrical lattices, build-up of fission products, etc. The most successful method is to test a mockup of the experiment or the experiment itself in each of the conditions which may result in changes in reactivity. A critical test is made with the experiment in each of the conditions to be compared. The difference in the critical position of the control rods in each of these conditions gives a measure of the change in reactivity which might occur if the conditions of the experiment suddenly changed in operation. In general, reactivity-effect measurements made on a clean cold lattice at the ORR have given higher values than similar measurements under conditions of high burnup in a lattice containing a number of other experiments in the reflector.

To eliminate the possibility of any experiment causing dangerous power fluctuations, the reactivity change which could conceivably occur from any experiment is limited to a value which can be safely controlled by the reactor safety system.

Safeguards

The safeguards employed to protect personnel and equipment from hazardous conditions must be as reliable as possible and must interfere as little as possible with other work. This means that controls which act as safeguards must have components which are proved reliable by special tests or long experience and that they should act locally if possible.

The proper application of the standard safeguard depends, to a great extent, upon circumstances. There is frequently a choice of safeguards which might be used in any one instance and, when this is true, the one employed should be selected on the basis of dependability, noninterference with other work, and economy. Where corrective action must be performed by the safeguard equipment, the time of response is sometimes the deciding factor in making the selection. The standard methods for accomplishing safety action and for providing safe static parts of systems are listed and discussed.

Temperature Reduction. The operating temperature of a test specimen may be reduced by:

1. shutting down the reactor,
2. increasing the coolant flow,
3. reducing the temperature of the coolant,
4. turning off auxiliary heaters (if any),
5. retracting the specimen from the reactor,
6. shadowing the specimen with a neutron absorber.

Any one of these methods could be employed, depending upon the circumstances. Methods 2, 3, and 4 are generally used by temperature controllers for the normal operation of the loop and can cope with most minor temperature excursions. If these fail, provisions to use item 5 should be made, if possible, to avoid reactor shutdowns. If this cannot be done then item 1, reactor shutdown, is necessary. Item 6 can only be used infrequently and is seldom considered. If the temperature elevation were due to some cause such as total loss of coolant flow, then a fast reactor scram might be required.

Pressure Reduction. Pressure reduction can be accomplished by:

1. pressure relief valves,
2. reduction of temperature if vapor pressure is responsible,
3. rupture disks,
4. pressure regulating valves on the pressure source if external,
5. increasing the container volume,
6. decreasing flow rate if due to pressure differential in a dynamic system.

Items 2, 3, and 6 are managed by the normal controls on most systems in which pressure is a variable. Where pressure increases can occur suddenly, items 1 and 3 are normally used in addition to the controller and exhaust to the off-gas or drain system (through whatever gas or liquid cleanup or holdup systems are required).

Reduction of Radioactivity. The reduction of radioactivity is more subject to the dictates of circumstances than is control of temperature and pressure since it is not normally variable by continuous controllers. Some causes of high radioactivity and the appropriate actions are:

1. induced radioactivity in a coolant – addition of more shielding to exposed piping if the coolant flow rate cannot be changed;
    a. short-lived induced radioactivity – provide a shielded retention tank in the exit line to allow decay, decrease the flow rate to allow decay, or increase the flow rate to provide dilution.
    b. long-lived induced radioactivity – increase the flow to provide dilution (in a closed or recirculating loop, this is not a permanent solution);
2. failure of a test specimen releasing radioactive materials to the coolant – withdraw the loop or specimen from the neutron flux if provisions have been made for doing this (if the specimen or loop cannot be withdrawn, shut down the reactor and remove the loop and/or specimen; the designed shielding should be such that radiation through it should not be excessive in the event of such a failure);
3. leak of radioactive gas or liquid through containment walls – shut down the reactor and remove the loop from the reactor unless repairs can be made within a very short time (such repairs must not involve the risk of worsening the situation).

Supplying Emergency Coolant. Because of after-heat, large heat capacity, or inadvisability of reactor shutdown it is frequently necessary to guarantee an emergency coolant supply if the primary supply should fail. The type of supply depends, of course, upon the type of loop being operated:

1. If the coolant is a fuel-bearing liquid as in the case of the primaries of the aqueous or molten salt fuel loops, there is little that can be done to supply emergency coolant except to provide a spare pump or emergency power if the failure is due to loss of electrical power. For such cases, the loop and/or its container must be designed to withstand the consequences of coolant loss or the reactor must be shut down.

2. If the loop is closed and the coolant must be maintained under high pressure or single-pass flow cannot be substituted for recirculation for any other reason, spare pumps and emergency electrical power are the only

emergency flow provisions. Such loops should, if possible, be double-contained. This is actually the same situation as item 1.

3. If the loop is closed but single-pass flow can be substituted for recirculation in an emergency, quick-acting valves can admit coolant from an external source and provide an exhaust or drain opening for the system.

4. If the loop has single-pass coolant flow, the provisions for emergency cooling are fairly simple. A water-cooled loop has a choice of at least two supplies of water – the process system and the demineralized system. If one of these is used normally, the other can be considered an emergency supply since it is unlikely that both systems will fail simultaneously.

5. If the loop is air-cooled the normal supply would be from the regular compressed air system or a special compressor. If the regular plant air supply is normally used, emergency cooling must come from a special standby. If a compressor is the normal supply, then the plant air supply or bottled gas become the emergency supplies.

Supplying Emergency Electrical Power. In order to maintain instrument readings, pump power for coolant, etc., some loops require a source of emergency electrical power if the commercial supply should be interrupted. Although loss of the commercial supply is infrequent, the consequences of such loss can be bad enough to justify the presence of the emergency supply. Emergency power can be supplied by the following ways:

1. If a delay of several seconds can be tolerated between the loss and rejoining of power, a diesel or gasoline motor-driven power generator can be used as the emergency source. A general-purpose, diesel-operated generator is in use for the reactors and can supply power to loops if desired.

2. If delays must be kept to a minimum, storage batteries must be used. When this is done, it is usually best to use the batteries as an interim supply until the diesel-operated generator has attained full capacity and the load can be switched to it.

Supplying Emergency Compressed Air. If pneumatic instruments are used for important functions in controlling the operation of the loop, an emergency supply of compressed gas may be advisable. Such a supply can best be obtained by the use of bottled nitrogen or air.

Containment. Practically all loops are a source of airborne radioactivity in the form of gases, vapors, or dust. Some can be made to operate at below atmospheric pressure and these generally cause only minor containment problems unless they are subject to pressure increases due to internal explosions or sudden temperature excursions. A great number of loops, however, operate at pressures ranging from slightly above atmospheric to more than 2000 psig, and containment becomes a major requirement.

Two methods of guaranteeing containment are being used at present:

1. Double containment provides a second wall if the inner wall should fail. The space between the two walls is monitored to detect a break in either wall and usually requires a connection to the off-gas or drain system. If the outer wall should be broken, the loop operation must be halted until repairs are made. Double containment is prescribed for systems having high temperatures, very radioactive gases, or containing corrosive fluids or gases.

2. Single containment walls make use of all-welded construction and/or monitored flanges. When this type of system is used, all welds must be thoroughly checked for reliability and the entire system proved leak-tight by the use of pressure checks and very sensitive leak indicators such as a helium leak detector. A monitored flange is one which has two concentric gaskets with a

space between them. This space is monitored by blowing a sweep of air through it by way of holes drilled through the flange metal. If the inner gasket should leak, radioactivity will be detected in the exhaust sweep gas. A single-wall system must be carefully protected from wall damage by external forces.

In addition to providing a guarantee of containment for the loop itself, sampling, filling, and drain lines and valves must be enclosed within chambers with limited access openings and vented to the off-gas system.

Fire and Explosion Prevention. Loops that contain or generate combustible materials must be carefully designed to prevent fires or explosions. Any water-cooled loop will have free hydrogen and oxygen in the exit water. If the water is recirculated, the accumulating gas must be disposed of or recombined. Any gas void, such as that over the water in reservoirs or surge tanks, can accumulate explosive gas mixtures and so must be purged sufficiently frequently to prevent such accumulations. Loops which contain organic materials either in solid, liquid, or gaseous form must make use of inert gas atmospheres in any chamber where an accumulation of vapors might occur.

Effluent Cleanup. The coolant wastes from any loop contain radioactive materials in the form of liquids, gases, or solids. The standard waste disposal systems are equipped to handle some portion of the radioactivities but additional specialized cleanup provisions must frequently be a part of the loop equipment:

1. If long lengths of unshielded pipes are necessary between the loop and the off-gas or drain system, the radioactivity of the wastes must be reduced sufficiently to prevent the pipes from being a radioactivity hazard.

2. If the liquid wastes contain dissolved or entrained radioactive gases, these must be removed to prevent their being released through openings from the drain system.

3. If the gaseous wastes contain large amounts of radioactive gases which would not be removed by the normal cleanup system, they must be removed from the gas before it enters the off-gas system.

4. If the liquid effluent is very radioactive and is of such large volume that it cannot be handled by the waste disposal system, the radioactivities must be mostly removed from the large-volume fraction and concentrated to be sent to the hot drain. The slightly radioactive large-volume fraction may then go to the warm drain (at ORNL the liquid radioactive waste disposal system is composed of a small-volume system called the hot drain for very radioactive wastes and a large-volume system called the warm drain for slightly contaminated wastes).

The cleanup of gaseous effluents is done by one of or a combination of three methods, depending upon the circumstances:

1. Most gas streams from loop coolant systems contain entrained particles. These may be removed either by filters or caught in the gas removal traps if such are used. If the particle content of the gas stream is great or the gas removal traps are to be used for a long period of time, the use of filters ahead of the traps is advisable.

2. If the gaseous radioactivity of a gas stream consists only of shortlived nuclides which decay to nonradioactive daughters, then a simple decay tank in series with the exhaust system may suffice for radioactivity reduction.

3. If the gaseous radioactivity is long-lived, the treatment it receives must be suited to the type of gas, but some sort of retention may be required.

   a. The radioactive argon-40 is of no great concern unless its release to the atmosphere raises the radioactivity level sufficiently to make the direct exposure rate objectionable. Generally several hundred curies per day can be exhausted directly to the off-gas system.

b. Mixed fission product gases must be retained in charcoal traps or retention tanks until their concentration in exhausted gases is such that the concentration at ground level is not above tolerance for any one of the gases or its daughter products. For radioiodines the discharge rate for the off-gas system must be kept below 5 curies per day.

Liquid effluents are cleaned of radioactivities in ways that depend upon the liquid and the volume:

1. Aqueous effluents and other liquids, if of small volume, may be stored in "dump" tanks until the radioactivity is sufficiently low for disposal to a drain if still aqueous or by burial if not.

2. Aqueous effluents of large volume must employ ion exchange columns to concentrate the dissolved radioactivities or use large-volume storage tanks which can be very expensive. Particulate radioactivities can be removed on filters.

3. If dangerous radioactive gases are dissolved or entrained in aqueous wastes, these must be removed by a degasifier and disposed of as gaseous effluent.

Approval Procedure For Experiments

When an experimenter wishes to perform a complicated experiment, he confers with the Assistant Reactor Operations Superintendent who acts as Experiment Coordinator. One of the technical personnel of the Operations group is appointed to assist the experimenter with data on the reactors as required and assist in the outlining the general plan of the experiment. As soon as this is done the Operations technical group reviews the plan to make sure it is feasible, does not conflict with operations or other experiments, and that insertion and removal can be done within a reasonable time. Early review for operability is important to ensure that any conflicts are found before design has progressed to a point that it cannot be changed without large loss of time or money. Following this the design of the experiment is completed, and a questionnaire (Appendix I) is executed for safety review.

This questionnaire is an aid to the reviewer and includes questions concerning the safety and operation of the experiment. After review, the request, if complicated, is generally sent on to the Reactor Experiment Review Committee. This committee is composed of senior personnel from seven divisions at the Laboratory.

Experience has shown that close attention must be continued after initial approval to make certain that any changes are properly reviewed and that the safety devices are checked regularly. This is the responsibility of the reactor supervisor. The order of responsibility is as follows:

Experimenter → Operations Assistance in Experiment Design → Operability Approval (early in design) →

→ Operations Safety Approval → Safety Approval by Reactor Experiment Review Committee (not required on more simple experiments) →

OPERABILITY

The operability of a loop must be one of the prime considerations during its design and installation. Certain requirements of design and fabrication which might be taken for granted actually must receive close attention to ensure that they are followed. This attention must begin with the design itself and not be relaxed until fabrication and installation are complete. Even after operation of a loop has begun, careful supervision for safety is necessary to prevent some change which may render safety devices inoperable.

Equipment Layout and Component Location

The placement of the loop components and the routing of piping and wiring must be done in such a way that:

1. it does not interfere with other equipment which is already installed or planned;
2. the components are accessible for servicing;
3. radioactive portions are isolated or located where they can be shielded with adequate supports provided for the shielding;
4. installation and removal can be done without lowering the water if in the LITR tank or ORR pool (this provision is necessary because of the possibility that lowering the water may expose portions of other, unshielded, loops).

Reliability of Instruments and other Components

In order to minimize loss of operating time for the reactor, the instrumentation used for data collection, and all other components required for operating the loop must be of types proved reliable by experience or test. This will be emphasized with respect to safeguard instrumentation and components, but is mentioned here since there is frequently a tendency for reviewers to disregard the importance of those parts not obviously related to safety. Not only may the failure of any part of the loop cause a loss of data but may initiate a chain of events resulting in a threat to safety.

The materials of fabrication for the piping and component parts which contact coolant or the reactor system must be compatible with each other and with the other components of the reactor. Moving parts of stainless steel must not rub against stainless steel of the same type where close tolerances are necessary because of the possibility of galling. The same is true of aluminum.

Since all three of the ORNL research reactors contain aluminum parts, care must be exercised to ensure that no material which is damaging to aluminum is allowed in the reactor system. For example:

1. Mercury greatly accelerates the corrosion of aluminum in air or water and so must be kept out of all three reactors. Due to the extensive use of aluminum in the ORR reactor and pool, any mercury brought into the reactor building must be in containers that guarantee that it cannot be spilled.

2. Carbon in the form of graphite or amorphous carbon will enhance the corrosion of aluminum in water or a moist atmosphere. All pencil marks must be thoroughly cleaned from aluminum surfaces which are to be wetted and noncapsulated pieces of graphite, such as graphite bearings, must be kept away from aluminum. Organic material placed in radiation will be partially changed to free carbon in time and therefore must be kept from contact with aluminum in a water system.

3. Copper and copper ions may produce a pitting type of corrosion of aluminum in a water system due to the formation of galvanic cells.

4. Steels which rust should be avoided in water systems for several reasons. The rust causes the water to become very dirty and requires treatment, and if the parts ever have to be removed, the rusty surfaces make the mechanical operations difficult. The rust layer may also become very radioactive by absorbing radioactive ions from the system.

Fabrication and Installation of Parts

Responsibile personnel must be assigned the task of ensuring that the fabrication and installation is done in strict accordance with the approved design. Any deviations from the design must be approved by the experiment reviewers. The following must be guaranteed by actual observation by the inspectors:

1. Conformity to design is imperative.

    a. Only the materials specifically approved for each item can be used. There must be no unapproved substitutes.

    b. The equipment location must be exactly as shown on the design prints. No additional space may be used without specific approval by the operating group.

    c. The routing paths of all pipes and wiring must not deviate from the design.

2. Critical dimensions must be checked by measurement to prove that they are within the prescribed tolerances.

3. Where routing of piping within a reactor is complex, trial installation of components or mocked-up components should be made before final installation is attempted. This is to ensure that time will not be wasted during the reactor shutdown time devoted to the installation.

4. All welds must be checked by approved methods such as by dye-penetrant or x ray where containment or support are important for safety.

Time Scheduling of the Installation and Removal of In-Reactor Parts

The procedure for inserting and removing experiments must take into account possible interferences with neighboring experiments, the time required and other interferences with operations. Experiments which require too long for installation delay other work, and once installation has begun it is often impossible or undesirable to stop, so that days or even weeks of operation may be lost. All experiments should, therefore, be designed so that they can be installed and removed within the normal shutdown period (or periods if more than one is required). Since other experiments generally have to be installed and removed during the same period, the experiment designer should not expect to use the whole shutdown period for his experiment. In cases where the experiment cannot be easily installed within the normal shutdown period, the cost of redesigning the experiment should be balanced against the cost of downtime. In some cases it may be more economical to redesign experiments so that they can be installed more readily than it would be to hold the reactor down.

Preoperational Checks

All parts of a loop must be checked for performance and operability as well as can be done without use of reactor power before operation is attempted. Much of this testing for the external components can be done during installation. These checks include circuit continuity, instrument calibrations, flows through pipes, operability of pumps and other such equipment, and containment. The in-reactor section must be pressure-checked before installation, and the test specimens must be tested to insure that they are not likely to fail due to operational conditions not related to irradiation.

Since much of the knowledge of any loop's operation depends upon calculations and extrapolations from existing data, it is almost always necessary to make performance checks while starting up the reactor. The power level of the reactor is held at some fraction of full power until an evaluation of the conditions are made, then raised to a higher level. It is advisable to do this at two or more low-power levels to observe trends. In addition to checking temperatures and pressures and their controllers, very careful attention must be given to the radioactivity of the external parts of the loops and to ensuring that there are no leaks of radioactive gases or liquids.

Operating Standards

To ensure safe and well-planned operation of the loop and its components in the reactor, the operating procedures must be devised during the planning stages of the loop. These procedures must be written. For complex loops where a particular phase of the operation, such as obtaining a sample of radioactive solution, involves careful preparation and the sequential use of several valves, check lists of the procedure steps are required to ensure that each step is performed and in the proper sequence.

During a loop's operation, the safety standards must be maintained. The best guarantee of this is to know that the operating personnel are adequately trained and are trustworthy. Experience has shown that the following items must be emphasized:

1. The approved maximums or minimums for temperatures, flows, pressures, etc., must not be exceeded without due approval. Operating conditions frequently show that adjustments are necessary, but such changes must not be made arbitrarily.

2. Safeguard instrumentation must not be deactivated without authorization or without employing auxiliary safeguards.

3. Proper servicing of components must be scheduled and performed as required to minimize failures. This includes lubrication of mechanical parts, renewing batteries in instruments, cleaning recorders, changing filters, replacing burned-out signal lights, recalibration of indicators, etc.

Neutralization of Experiments

One of the most important features in the operation of loops and other experiments in a nuclear reactor is the provision for neutralizing the loop in the event of trouble, so that the reactor can continue to operate. This is generally done by retracting the sensitive portion of the experiment from the high flux zone, but may sometimes be accomplished by flooding with water, use of neutron shutters, or other means. The retraction or other action may be done automatically or it may be done only when the reactor is shut down. With a high-flux reactor such as the ORR, it must be accomplished within about 10 minutes to be effective. If a longer time is required, sufficient xenon may be produced (as a daughter of fission-product iodine) to poison the reactor.

Reactor Building Features Important to Loop Operation

A number of building design features have been found to be important in the accommodation of loops and large experiments. Wireways have been necessary to provide easy connection between the reactor, the instrument panel, and the equipment cell. Since the instrument panels for a loop may require 400 ft^2 of space, these often have to be located at some distance from the reactor. Loops may also require shielded equipment cell and space for auxiliary equipment. In some cases the cell may be adjacent to the reactor shield if the experiment is in a horizontal beam hole, but in a vertical loop it may have to be located some distance away from the reactor.

Provision should also be made for access by piping from the reactor to the equipment cells. Since this may become radioactive with certain experiments, the piping should be shielded or easy to shield. Enough separate routes should be provided so that piping from one loop can be readily installed or removed without disturbing that of other loops.

The building space required for experiments varies considerably. The following table shows the space requirements for a series of representative loops at the ORR.

TABLE A-I

BUILDING SPACE REQUIRED BY VARIOUS LOOPS AT THE ORR

Loop	Panel	Equipment Cell	Auxiliaries Space	Total
		(ft^2)		
Aqueous Uranyl Sulfate	360	100		460
MSR Pressurized Water	360	360	360	1080
Gas-Cooled Reactor	180	20	360	560
Nitrogen	150	360		510

Space is needed for loop fabrication and check out. Preferably this should be near the reactor since some loops have to be assembled in large pieces which are difficult to move.

The reactor building may be very large if all the necessary instrument rooms and equipment cells are inside the building. This offers a distinct disadvantage when radioactive contamination is released. At such times, it is almost impossible to prevent all parts of the building from becoming contaminated. Whenever feasible the reactor building should be compartmented into reasonably gas-tight sections having separate ventilation systems. It is almost impossible to prevent occasional releases of radioactivity, and a compartmented building is a great help in preventing the spread of contamination. If possible, the different sections of the building should be connected only through outside entrances or hallways. Separate, adjacent buildings are much better from the standpoint of contamination control.

Safety Instrumentation for Experiments

Safeguards which require a corrective action because of some change in a property can only be guaranteed by having reliable instrumentation.

With high-flux reactors such as the ORR, the reliability of experiment instrumentation becomes especially important not only from the standpoint of safety but from the standpoint of continued operation. Failure of a safety instrument may shut down the reactor, and if the instrument is not immediately remedied the xenon in the fuel will increase to a point such that the reactor cannot be restarted for about two days. In order to start the reactor sooner the fuel must be changed; a procedure which may take five hours at the ORR or longer at other reactors where the fuel handling operation is more difficult. Furthermore, additional troubles often occur during a shutdown or a start-up so that the probability of additional downtime increases markedly.

This then, places a high premium on reliability of safety instrumentation which may scram the reactor. The early experiments at the OGR and the LITR were provided with safety instrumentation designed as the occasion demanded without special standards except that it be fail safe and that it would adequately protect the reactor and personnel. While such instrumentation gave adequate safety protection, it often caused spurious scrams of the reactor and gave other troubles.

The need was soon apparent for instrumentation which had a number of improved features.

The instrumentation should be reliable. It should operate for months at a time (if possible) without failure.

If an instrument failed, a definite procedure should be provided to: (a) substitute another instrument, (b) block out the defective instrument, or (c) leave the reactor shut down until repairs could be made. While the last of these is a reasonably acceptable procedure at a low-flux reactor, it is distinctly unacceptable in a high flux reactor except when no alternative system can be provided.

Experiments should be designed so that they require as little special attention as possible. It becomes very costly to man an experiment continuously since this requires two men per shift or eight for four shifts. When shift relief is counted, as many as ten may be required.

Because of the higher personnel requirements and other reasons, most experiments are completely controlled by instruments. This means that during a reactor startup instruments must adjust temperature, cooling, or make any other changes necessary for the operation or safety of the experiment. With a high-flux reactor, the reactor must be able to go from zero to full power within a few minutes. Unless the reactor can be started up rapidly it may be caught by xenon, requiring a lengthy fuel change.

All equipment should restart automatically in the event that the electrical power should be lost. Occasionally experiments may be designed with certain equipment requiring manual restart. If the reactor is started up after a scram, unattended experiments having manual restart features may shut down the reactor again if the experiment is attached to the safety system. The reactor operators do not have time to attend to such details since there may be only a few minutes to get the reactor to power.

If one of a pair of duplicate safety devices fails, the safety decision as to whether to shut down the reactor while the instrument is being repaired or to block out the instrument, repair, and reattach it to the safety system with the reactor operating should be made before the experiment is inserted. The exact condition for doing this must be fully specified. It is not considered safe to have such decisions made hurriedly.

If an instrument is removed from the safety system during operation, repaired, and reinserted, some provision must be made to check its operation on the reactor safeties.

Too much instrumentation should be avoided. The more safety devices attached to the reactor the more likelihood there is that the reactor will be shut down due to instrument failure. Only the minimum number of devices required for safety should be used.

Where possible, enough spare instruments should be available so that if one fails another can be substituted immediately. This is especially important in the case of thermocouples. On high-temperature loops, thermocouple failures are sometimes common and the minimum number of thermocouples consistent with safety should be determined before the experiment is inserted into the reactor, not at the time the thermocouples fail.

Instruments should be able to control an experiment over the rapid power changes of scrams, startups, the gradual flux changes with control rod movement due to fuel burn up, and the growth and burn-out of xenon following power changes.

Reliability of instrument components has proved to be one of the most important factors in operation of experiments. Cases have occurred where components have been used which have had a long history of failure because new designers were not familiar with the failure records. Since experience has shown that even the best instrumentation is subject to failure, a list (based upon experience and tests) of the most reliable sensing elements, indicators, relays, and other circuit components has been compiled. This list is not the result of testing all the available types of these components, and so does not actually eliminate any particular type. It does, however, eliminate their usage unless special tests prove them to be reliable for their intended purpose. A requirement other than reliability of the component itself is the guarantee that spare parts are on hand and that maintenance personnel have been trained in the servicing of the component. When these requirements have been met, any new type of component is added to the list of acceptable components.

Where signal response must be guaranteed for safety, two independent systems are required. This applies to such actions as heater cutoff, pressure reduction, automatic withdrawal from the reactor, setbacks, and scrams. If, however, two different such actions can be employed on a loop, and both accomplish the same purpose, one can be called the duplicate of the other. A further requirement is that fail-safe features be incorporated into the instrumentation. For example, up-scale response on the temperature indicators is generally specified if a thermocouple develops an open circuit.

Standards have been set up for designing experiment safety systems and for attaching these to the reactor. Construction practices are specified which will minimize shutdown due to shorts, loose connections, etc. Standardized and complete maintenance drawings are provided for experiments and will shorten the time required for troubleshooting. Signals to the reactor controls requiring a scram are double-tracked in such a manner that either closing a normally open relay or opening a normally closed relay will cause a scram. This guarantees that a scram will occur if the tie-in to the reactor controls is inadvertently disconnected or broken. In order to maintain this arrangement and to standardize the various connections from the loop instrumentation to the reactor control system, circuits of the desired type have been designed and built into standard chassis which are supplied as a stock item. These contain all the necessary relays and switches to accommodate the transmission of signals from the loop monitoring equipment to the reactor control room. In the reactor control room another standard chassis is used to make up the connections to the reactor controls. One complete channel accommodates one loop or experiment.

In order to deactivate the circuit from the loop to the experiment, a locked switch built into the chassis mounted in the control room must be turned from the "on" position to the "off" position. A third position called "test" allows signals to be sent from the loop to the control room without actually causing a setback or scram; it is used for checkout purposes prior to the actual operation of the loop. The key needed to operate the switch is kept by operations personnel.

The choice of whether to use a scram or setback to lower the reactor power for safety purposes is dependent upon the time-response required. A scram is much faster than a setback, but recovery of reactor power takes longer. Since false signals sometimes occur, a scram should be required only if a setback will not lower the power level rapidly enough. At the ORR where the production of xenon during shutdown may poison the reactor, this is very important. In slowly varying systems it is customary to require a sequential order of actions in order to minimize scrams. This is done by requiring that an audible alarm, setback, and scram signals be initiated at successively higher or lower values for the property being monitored. The alarm alerts personnel who might, by manual action, correct the condition. The setback may avert the scram by lowering the power somewhat. If a fast change should occur, however, the scram would likely be initiated before the other actions become effective. If a condition which is being monitored can only exist or not exist, such as electrical power, the scram signal would not be preceded by the setback signal. In all cases an audible alarm is sounded and monitor lights turned on to identify the source of the trouble.

The safeguard philosophy used at ORNL is to guarantee reactor shutdowns when required since the consequences of not having a shutdown when required can far outweigh the consequences of having an unnecessary scram. The occurrence of unnecessary shutdowns from instrument failure has been reduced to the status of a minor problem by the use of reliable components, regular servicing, and standardized safety circuit designs.

Services Required for Loop Experiments

Loop experiments are the largest users of services in the ORNL research reactors. Among the services required are for hot off-gas, cell ventilation, hot drain, warm drain, ordinary electric power, special electric power, emergency electric power, compressed air, potable water, process water, and demineralized water. Each of these is available to all the experimenters as required. Their usage depends upon the experiment in the reactor at any given time.

Hot Off-Gas

This is a small-volume system used for highly contaminated gas or as a static suction on systems which might give off large amounts of contaminated gas. Approximately 1000 cfm at -30 in. w.g. are required for the ORR. The lines are shielded, and the gas is filtered and cleaned in a caustic scrubber or Cottrell precipitator before going to the 250-ft stack.

Cell Ventilation

At the ORR this is a 5000-cfm system operating at -5 in. w.g. It is used primarily for ventilating the shielded equipment cells. These contain equipment which is usually part of loops and which may release large quantities of gaseous activity.

Hot Drain

This is a small-volume system of stainless steel for highly radioactive liquids. A 2000-gal holdup tank is provided to serve the ORR, OGR, and LITR. Waste from this tank is pumped to the central system for the Laboratory.

Warm Drain

This is a large-volume waste system for slightly contaminated liquid waste. The floor drains in the ORR drain to this system.

Ordinary Electric Power

Approximately 800 kw of electric power is used at the ORR (exclusive of the main pumps and coolers).

Special Electric Power

A separate power system at the ORR is fed by a 75-kva transformer from the 2400-v Laboratory system. This is reserved for all critical instrumentation, such as high-gain pulse equipment, regulated power supplies, etc., where noise-free line voltage is required. Special outlets are provided on this system to prevent its use for other equipment.

Emergency Electric Power

A 350-kw diesel-electric generator provides emergency power to the ORR, LITR, and OGR.

Compressed Air

Varying amounts of compressed air from the 100-psi Laboratory system are used by experiments at the ORR.

Process Water

This is fed by the main water system through an air-break and is used in cooling equipment and other applications where there is a possibility of contaminating the system during a loss of water pressure.

Potable Water

This is a plant-wide system used for drinking fountains, fire, and general non-process use.

Demineralized Water

A central plant supplies the ORR with amounts up to 20 gpm. This is used in certain applications where the water passes through a neutron field and it is not desired to use process water which would become activated.

ORGANIZATION

Reactor loops and experiments at ORNL are designed and operated by groups which are separate from the reactor operating group. The reactor operations group, however, has responsibility for safety and operability of experiments. For this a highly trained staff experienced in the hazards of experiments in nuclear reactors is absolutely essential to safe operation. All personnel must be trained to a certain degree, but the higher supervisors should have considerable experience as well.

With such personnel it is possible to operate a high-flux reactor and supervise the safety of experiments with a rather small staff. The operating staff for three research reactors at ORNL consists of approximately 43 persons of whom 26 are nontechnical and the remaining 17 provide technical supervision and support. In addition, approximately 30 persons from other groups give the instrument, maintenance, and other support needed to operate the Graphite Reactor, Low-Intensity Testing Reactor, and Oak Ridge Research Reactor. This does not include the personnel required to build and operate loops and other experiments.

The Assistant Superintendent of Operations supervises safety and operability reviews of experiments. Each experiment is assigned to one of the technical personnel who assists in design, furnishes information on the reactor as needed by the designers, and checks on all points of interference. An early review of the general plan is made for operability. It is extremely important that the general plan be reviewed before detailed design begins, to ensure that the design is compatible with operation, scheduling, and safety. When the experiment design is complete, a questionnaire (Appendix) is obtained on the experiment. This is reviewed by the Technical group for safety. Following this review, the questionnaire may be referred to the ORNL Reactor Experiment Review Committee for a final review.

After the experiment is installed, the operation and supervision becomes the responsibility of the reactor supervisor. The organization is shown below.

Personnel Requirements for Loop Experiments

Loop experiments vary in personnel requirements according to their complexity, whether they are manned, the frequency with which new loops must be built and the number of personnel necessary for supervision and analysis of data.

For example, a circulating solution loop in the LITR required 32 men for operation while a smaller molten-salt loop required only 8. The first loop was manned full time and required 9 engineers and chemists, 6 craftsmen, and 17 technicians. The second required 6 engineers and physicists and 2 craftsmen. The requirements for most loops appear to fall between these figures.

Superintendent

Operations Superintendent

Assistant Superintendent (Experiment Coordination)

Development Supervisor

2 Physicists

Technical Assistance Supervisor

3 Engineers

ORR Supervisor

LITR Supervisor

Graphite Reactor Supervisor

Shift	Shift	Shift	Shift	Day Shift
Engineer	Engineer	Engineer	Engineer	Foreman
Foreman	Foreman	Foreman	Foreman	Ass't Foreman
4 Operators	4 Operators	4 Operators	4 Operators	4 Operators

DOWNTIME EXPERIENCE

The number of shutdowns at the ORNL reactors has decreased as the design and instrument standards went into effect. Over a period of eight years the LITR has had a rather large number of reactor scrams due to loops and other experiments. Shutdowns were not so serious at this reactor because it is always possible to override xenon. However, it was apparent that better reliability would be required at the ORR where any shutdown of greater than 10-min duration might require that the fuel be changed before the reactor could be started up again. The standards developed from the experience in the LITR have resulted in much greater reliability for the ORR experiments.

Table A-II compares the number of shutdowns from loops in the LITR with those at the ORR. It is seen that a marked improvement has occurred in the ORR due to the standards being applied.

TABLE A-II

CAUSES OF SHUTDOWNS DUE TO OPERATION OF LOOPS IN LITR AND ORR

	Instruments	Thermocouples	Mechanical	Experiments	Human error	Total
			LITR			
1951				1		1
1952	1		1	2	2	6
1953	1	1	1	1	2	6
1954						9
1955	6	1	7	8	3	25
1956	6	1	9	5	4	25
1957	1	9	6	3	5	24
1958	3			12		15
			ORR			
July '58– June '59	1					1

URANYL SULFATE IN-PILE SOLUTION LOOP[1,2]

The purpose of the uranyl sulfate loop is to investigate the radiation-corrosion behavior of aqueous homogeneous fuel solutions and potential materials of construction under thermal neutron radiation in the range of 10^{13} to 10^{14} $n/cm^2/sec$ at ~300°C and ~2000 psia. The work is a part of the Homogeneous Reactor Project program.

Basically the experiment is an enclosed loop (Figs. 1 and 2) using a 5-gpm canned-rotor pump to circulate a fuel solution of enriched uranyl sulfate across corrosion specimens located in various sections of the loop. Fuel is circulated through the loop core and heat exchangers by a small, canned-rotor pump. The core is adjacent to the reactor lattice. Heat exchangers between the pump and core are used to add or remove heat as required. A small tank located above the main loop in a bypass stream is maintained at a temperature greater than that of the loop to permit steam pressurization of the system. Metal specimens are located in holders in the core and in an enlarged section of the line near the pump outlet. The loop is designed as a compact assembly to permit enclosure in a small leak-tight container for insertion in an 8-in. beam hole.

The corrosion in the loop is measured by the consumption of O_2 which is periodically charged to the loop. Samples of fuel are periodically removed from the loop to obtain information on fuel stability and the type of material which is corroding. The fuel removed is replaced to maintain a constant fuel inventory in the loop.

Because the loop becomes radioactive, elaborate equipment, instruments, and procedures are required to ensure complete safety to the experimenter, reactor, and experiment. Most of the hazards of this loop are associated with the possibility of high pressures which might rupture some of the loop components and release radioactivity. The loop, itself of stainless steel, is contained in a stainless steel liner for preventing escape of radioactivity should the loop develop a leak. The auxiliary components such as valves and pressure lines to and from the loop are also enclosed in sealed and shielded chambers. Both the equipment chambers and the loop container are vented to the plant off-gas system. To prevent all the fission products of the loop from being vented to the off-gas system at one time, some of the vents pass through charcoal traps. These traps are capable of holding up a major portion of the fission gases for several weeks.

Because of the number of varied operations to which the loop is subjected, approximately 22 procedures have been written (in check-off form to avoid any mistakes). One procedure requires over 280 separate operations. All the procedures involve the manipulating of over 100 valves. The loop is equipped with about 30 instruments to monitor radiation, control and record temperatures and pressure, and give the necessary alarms for putting the reactor either in setback or scram. The safety devices guard against loss of circulation, loss of pressure, excess radiolytic gas, overheating of the loop from coolant failure or partial loss of flow.

There are about 20 fast set-back (FSB) points and about seven scram points. Due to the nature of the experiment, set points are set within several degrees and 25 psia of control values.

This experiment is unusual in that low temperatures can be as dangerous as high temperatures since the recombination of radiolytic gas decreases with a reduction in temperature. If the rate of generation of radiolytic gases is greater than the recombination rate, an explosive mixture of hydrogen and oxygen will develop along with excessive pressures which could rupture the loop.

In the event of an emergency (i.e., pump or heater failure) with the reactor at power the loop can be drained to a dump tank which is vented to the off-gas

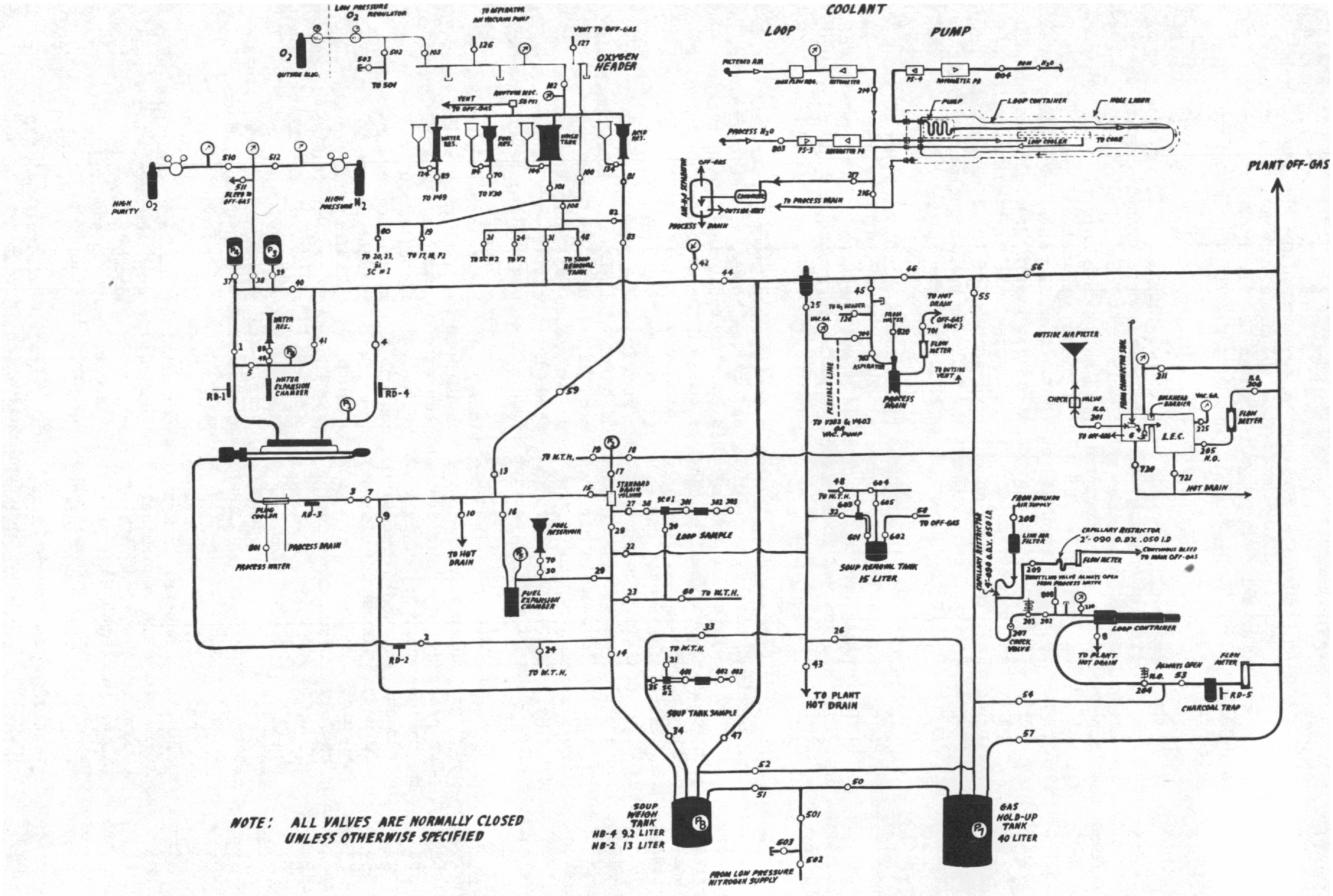

Fig. 1—Flow diagram of uranyl sulfate loop.

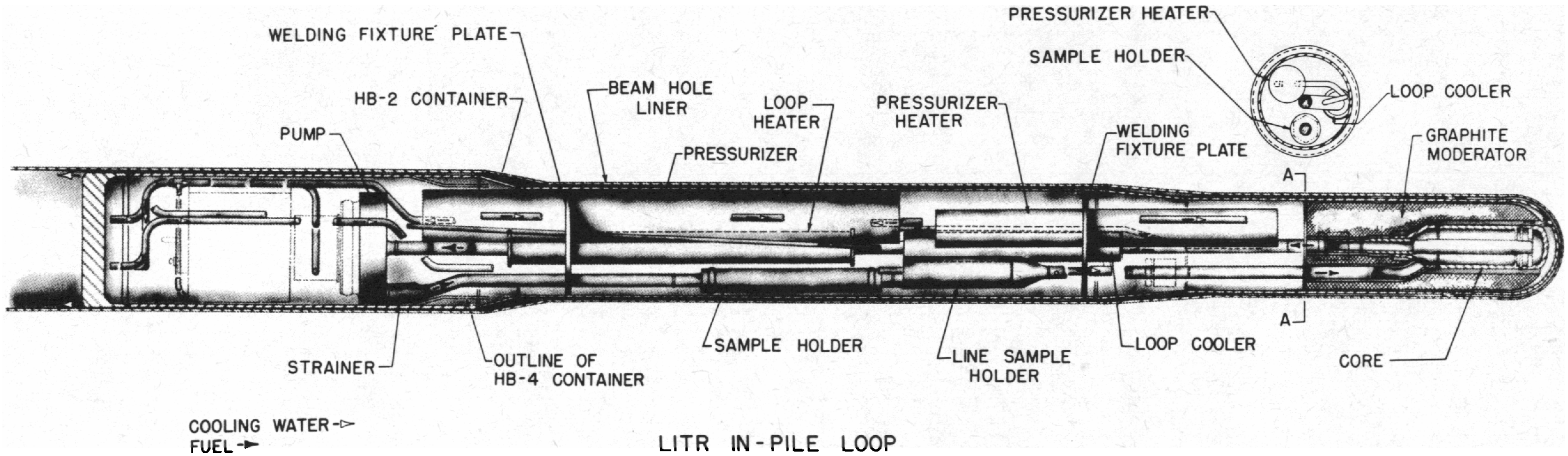

Fig. 2—Drawing of uranyl sulfate loop.

system. For minor difficulties the loop can be retracted, reducing the neutron flux by a factor of 50.

Many such loops have been operated at the LITR and ORR over a period of six years and no serious release of radioactivity has occurred. Major causes of failure of this loop have been associated with pumps and loop circulation. Experience with these and similar loops has shown, however, the need for improved instrument reliability. The following table shows the number of scrams of more than 15 min duration for 10 loops operated prior to 1958.

TABLE I

SCRAMS IN LITR CAUSED BY TEN UO_2SO_4 LOOPS

Radiation-detection instrument trouble	12
Simplytrol interaction	3
D-C power supply failure	2
Human error	6
Instrument slidewire	1
Pump failure	5
Pump power high	2
Coolant flow plugged	1
Total	32

Since a large number of reactor scrams were due to instrumentation failure, instrument procedures and practices were improved, using proved standard circuits and reliable components. The development of records on scrams caused by experiments has proved to be of great assistance in improving instrument design.

The loop requires two technicians per shift and must be continuously manned.

MOLTEN SALT LOOP

The molten fluoride loop[3] was designed and operated primarily to demonstrate the feasibility of such a system, and secondarily to measure the effect of radiation on a corrosion of the system by a high-melting liquid salt.

The loop (Fig. 3) was basically a U-bend tube carrying the molten fuel through a region of high neutron flux. The flow was supplied by a pump which was outside the reactor shield. From the pump discharge, fuel passed through a venturi flowmeter, through the U-bend in the neutron flux, and back to the pump through a heat exchanger.

Among the special hazards of this experiment, which was the first of its type, was the problem of criticality. Approximately 2.5 kg of enriched uranium was present in the loop and extreme precautions were taken to ensure that the molten salt could not be mixed with water to form a critical mass. However, it was determined that a mixture of the fuel charge with water would be very unlikely even if the fuel circuit and the water jacket formed leaks at the same time. Safety devices were, of course, provided to ensure that any leak in the fuel circuit could be detected before it caused a failure of the water jacket.

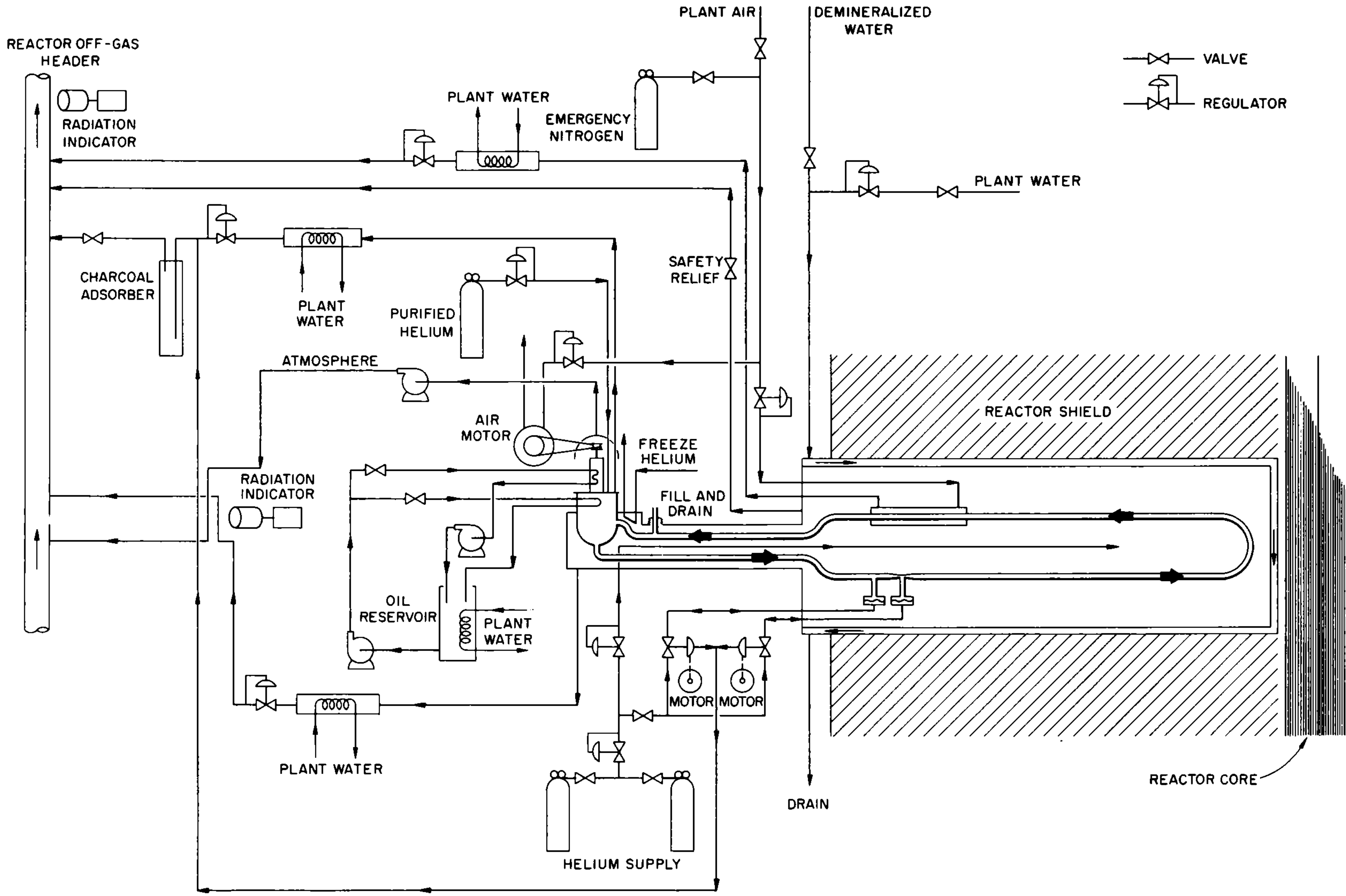

Fig. 3—Schematic diagram of fluoride fuel loop.

Failure of the pump or stoppage of the fuel line would result in extreme overheating of the U-bend in the neutron flux, and it was necessary to provide safety devices which measured temperature, as well as flow, in this region. If the flow stopped or the temperature rose above acceptable limits, the safety circuits would shut down the reactor.

Because part of the fuel circuit was outside the reactor shield, radiation from the pump and associated piping presented a serious problem. The flow circuit time was about 19 sec, with the result that delayed neutrons were emitted from the pump section, and it was necessary to provide large amounts of neutron shielding in addition to the lead gamma shielding. A number of regions in the experiment were swept with air or inert gas to provide immediate warning if leaks should occur. In addition, the surge tank was swept with helium. Since the helium passed over molten fuel in the surge tank, large amounts of fission gases were carried out and it was necessary to pass these through a charcoal trap before releasing to the off-gas system.

The safety devices included approximately 70 thermocouples located in various sections of the loop, with the greater number concentrated in the U-bend where the highest temperatures were expected. Other thermocouples were placed on the loop shell where molten salt might leak on the stainless steel jacket. Some of these thermocouples were spares. The jacket in the nosepiece (Fig. 4) of the loop was designed so that if any melted fuel should leak in this region it would flow away from the region of high flux. Radiation monitoring instruments were arranged on the various purge-gas lines to indicate leakage through the charcoal absorber or out of the loop into the jacket from either the pump or the heat exchanger. The charcoal trap was fabricated from 6-in.-OD pipe approximately 10 ft long and holding 2 ft^3 of activated charcoal. This was placed in a vertical hole in the floor and the inlet gas line entered at the bottom so that the section containing most of the radioactivity was deep in the ground. Some shielding was, of course, also necessary over the top. A rough measurement of the release of gas from the loop indicated that approximately 0.001 curie/sec was discharged.

In order to increase the heat capacity of the U-bend section, the tubing walls were made quite thick. In case of a stoppage of flow, this would give additional time for the reactor scram to take effect before the tubing reached a hazardous temperature. Inconel fins were brazed to this section of the loop and the insulation was omitted so that the U-bend section could lose heat more readily. The entire in-pile section was jacketed with a hermetically sealed stainless steel jacket with water cooling outside the jacket. The space between the jacket and the loop was provided with a helium purge to provide an inert atmosphere and to detect any fuel leaks by radiation monitoring of the exhaust helium. The only suitable pump available at the time was too large to place inside the reactor beam hole so that it was necessary to place the pump outside the shield and build an auxiliary shield around this. A 12-ft beam hole together with the piping to the pump made the over-all loop about 15 ft long and considerably increased the amount of fuel required. The entire loop had to be provided with electric heaters to prevent freezing of the molten salt which had a considerable expansion on melting, making it impractical to remelt the fuel after it had once frozen in the loop. The salt had a low thermal conductivity, requiring turbulent flow in the irradiated section of the loop to give proper heat transfer.

Temperatures of coolant water, coolant air, pump oil, lead shield, and charcoal absorber were recorded. Two separate instruments continuously monitored maximum temperature in the hot portion of the loop. Each of these instruments was provided with spare thermocouples which could be switched in if necessary. Pump-motor air and heat-exchanger air were both controlled manually. The pump shaft speed was indicated by a Tachometer. An auxiliary supply of gas cylinders was provided to give a 15-min supply to the pump if the plant-air supply dropped below 15 psi. Fuel flow was monitored by a Speedomax attached to the venturi flowmeter. The instrumentation is shown in Fig. 5.

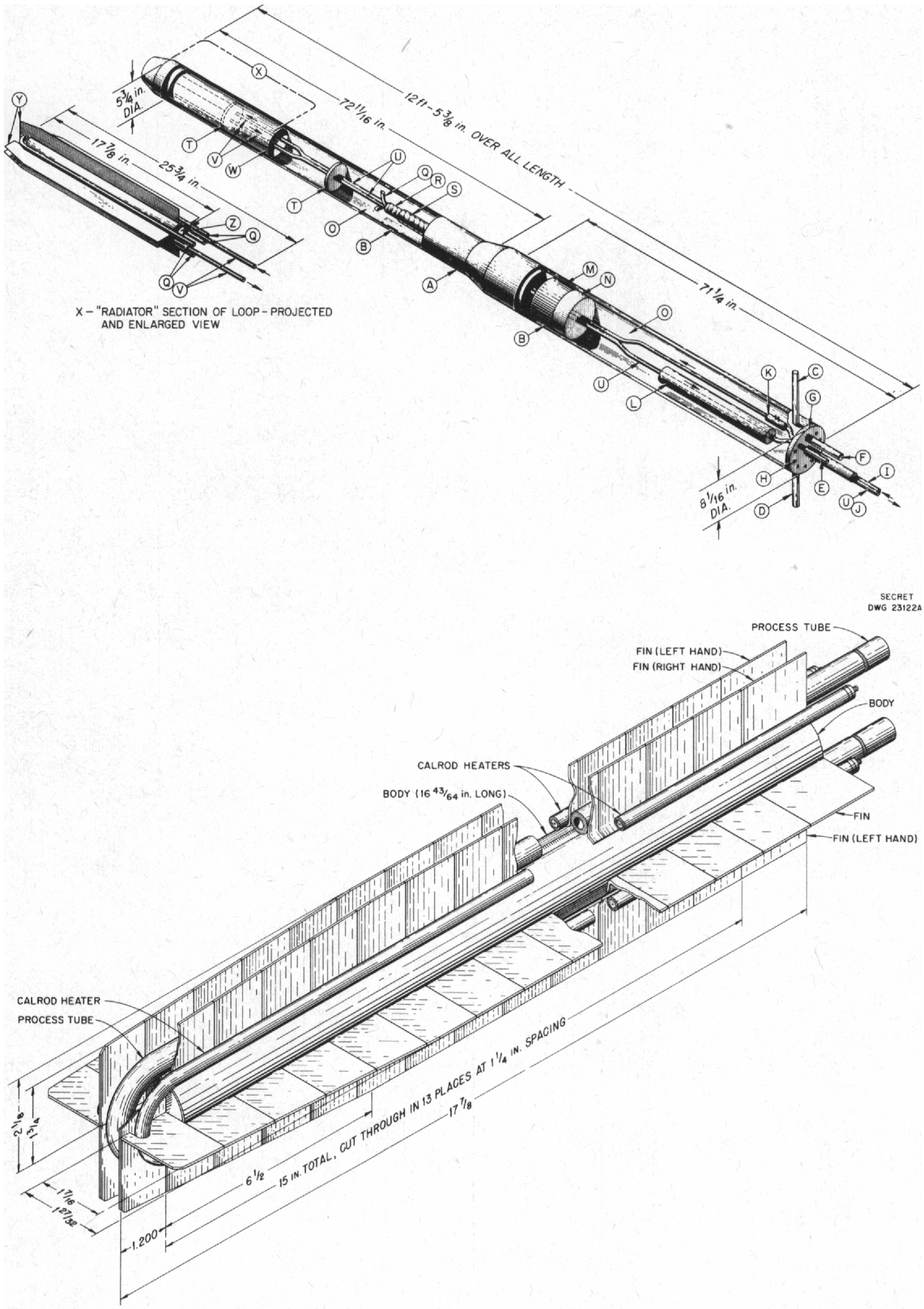

Fig. 4—Irradiated section of nosepiece of fluoride fuel loop.

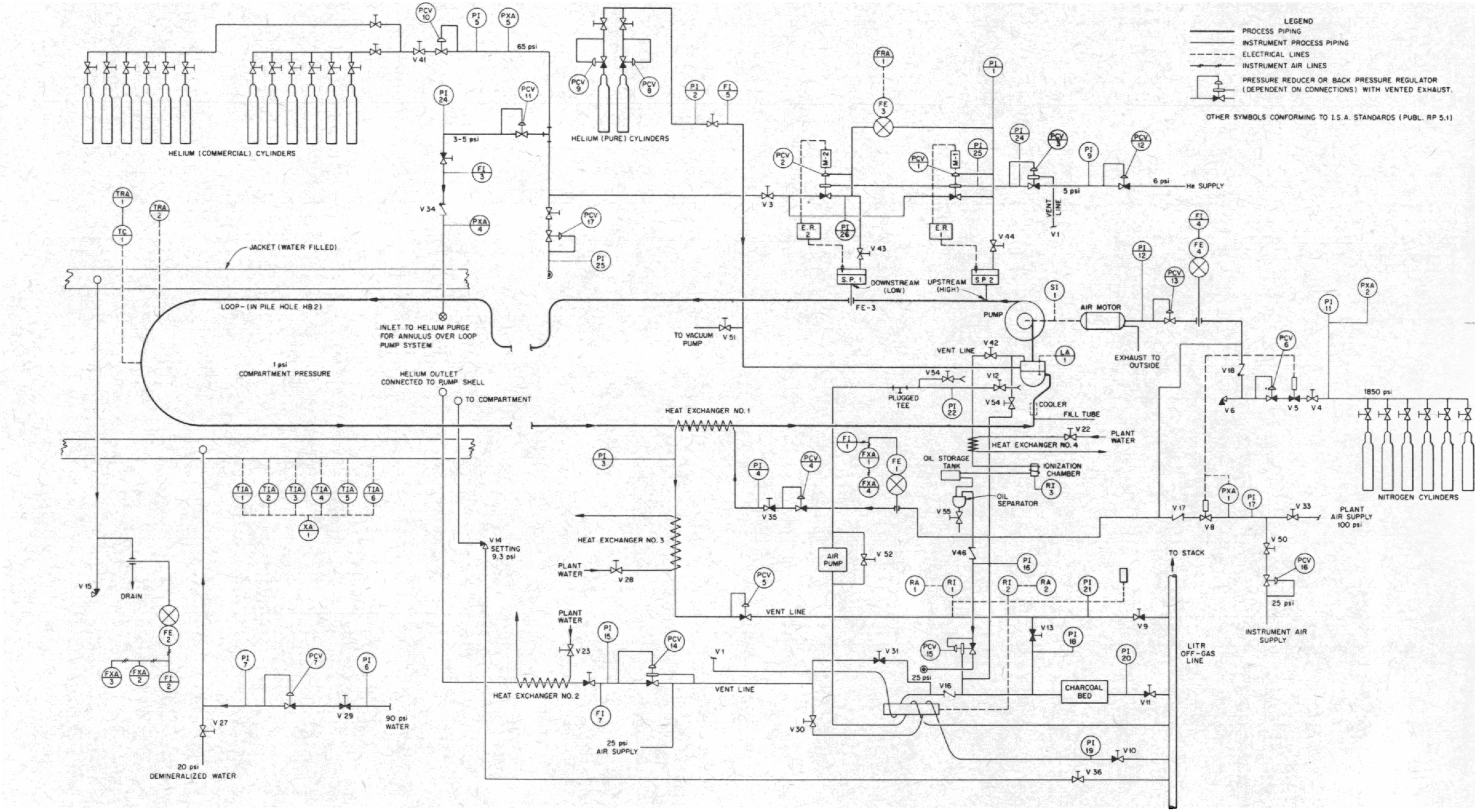

Fig. 5—Flow diagram of instruments and controls of fluoride fuel loop.

TABLE II

CHARACTERISTICS OF MOLTEN SALT LOOP

Operating time, hr	645
Time at power, hr	475
Location	LITR, HB-2
Reactor power, kw	3
Fission power in loop, kw	2.7
Maximum power density, kw/cc	0.4
Temperatures, °F	
Maximum flux region	1485
Outside reactor	1450
ΔT in loop	35 ± 5
Flow, gpm	1.1
Velocity in high-flux region, fps	8.6
Pressure	atm
Cycle time, sec	19.6
Composition, mole %	UF_4-ZrF_4-NaF (25-$12\frac{1}{2}$-$62\frac{1}{2}$)
Melting point, °F	1170
Density, g/cc	3.93 at 1500°F
Viscosity, centipoises	10.0
Uranium enrichment, %	93
Container material	Inconel
Fuel inventory, kg	5.0
Volume of system, cc	1290
ID of tubing in flux, in.	0.225

Some trouble was encountered in welding the tubing joints. All welds were inspected for cracks by the dye-penetrant method and by x rays. Each section was checked with a helium leak detector as it was fabricated. The first such loop fabricated failed in a welded butt joint during the bench test. The leak occurred in a flange-weld joint which had been vacuum tested, and the failure was attributed to a combination of stress concentration and incomplete weld penetration. After this fillet-welds were employed for the final loop. The section behind the U-bend portion was packed with shaped graphite blocks to provide shielding. It was found necessary to coat the graphite with sodium silicate to prevent shorting of the electrical connections. This section of the loop was also shielded with ferro-boron and cadmium.

After almost one month in the reactor, failure of the belt driving the pump terminated the operation of the loop.

Aside from the failure of the belt, the major difficulty was in the shield around the portion of the loop outside the reactor shield. The capture gamma produced in the paraffin required that 2 ft of high-density concrete blocks be added around 15 in. of lead, 12 in. of paraffin, and a sheet of Boral.

Table II shows the characteristics of the molten salt loop.

LIQUID METAL LOOPS

Liquid metal loops[4] were operated at ORNL to study the compatibility of molten alkali metals with stainless steel and Inconel at high temperatures and under reactor radiation. Five loops were irradiated, one of stainless steel (containing lithium) and four Inconel (containing sodium), as shown in the accompanying table. All loops were provided with heaters so that the metal could be melted, and an electromagnet pump was used for circulation. A flowmeter and a surge tank for thermal expansion were provided (see Fig. 6). Special attention to fire protection was required in designing the loops. Inert gas or a vacuum jacket was used around the in-pile parts, and an off-gas jacket with activity monitors surrounded the external portions (see Fig. 7.). To protect the reactor structure, water cooling was provided in the reactor hole liner for the loop jacket. The loop tubing was 0.225-in. OD x 0.025-in. wall except in the lithium-

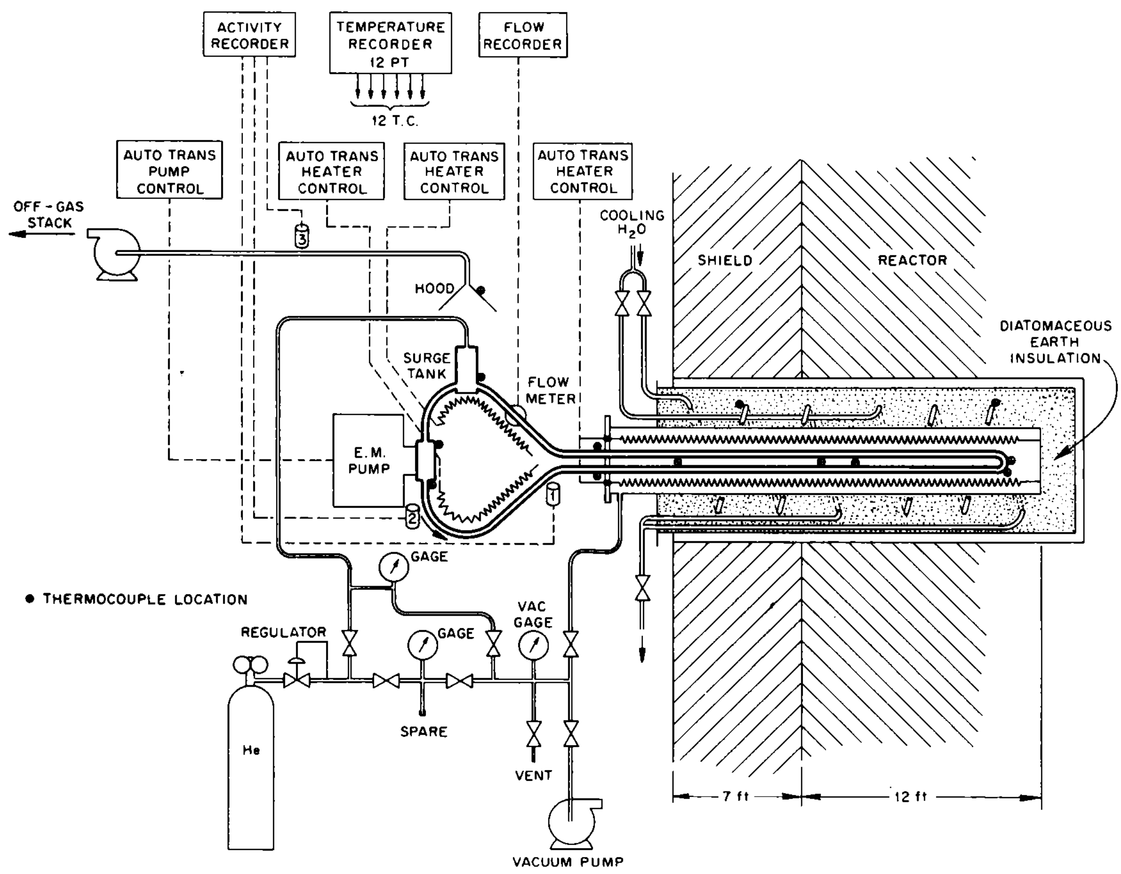

Fig. 6—Liquid metal loop system for the graphite reactor.

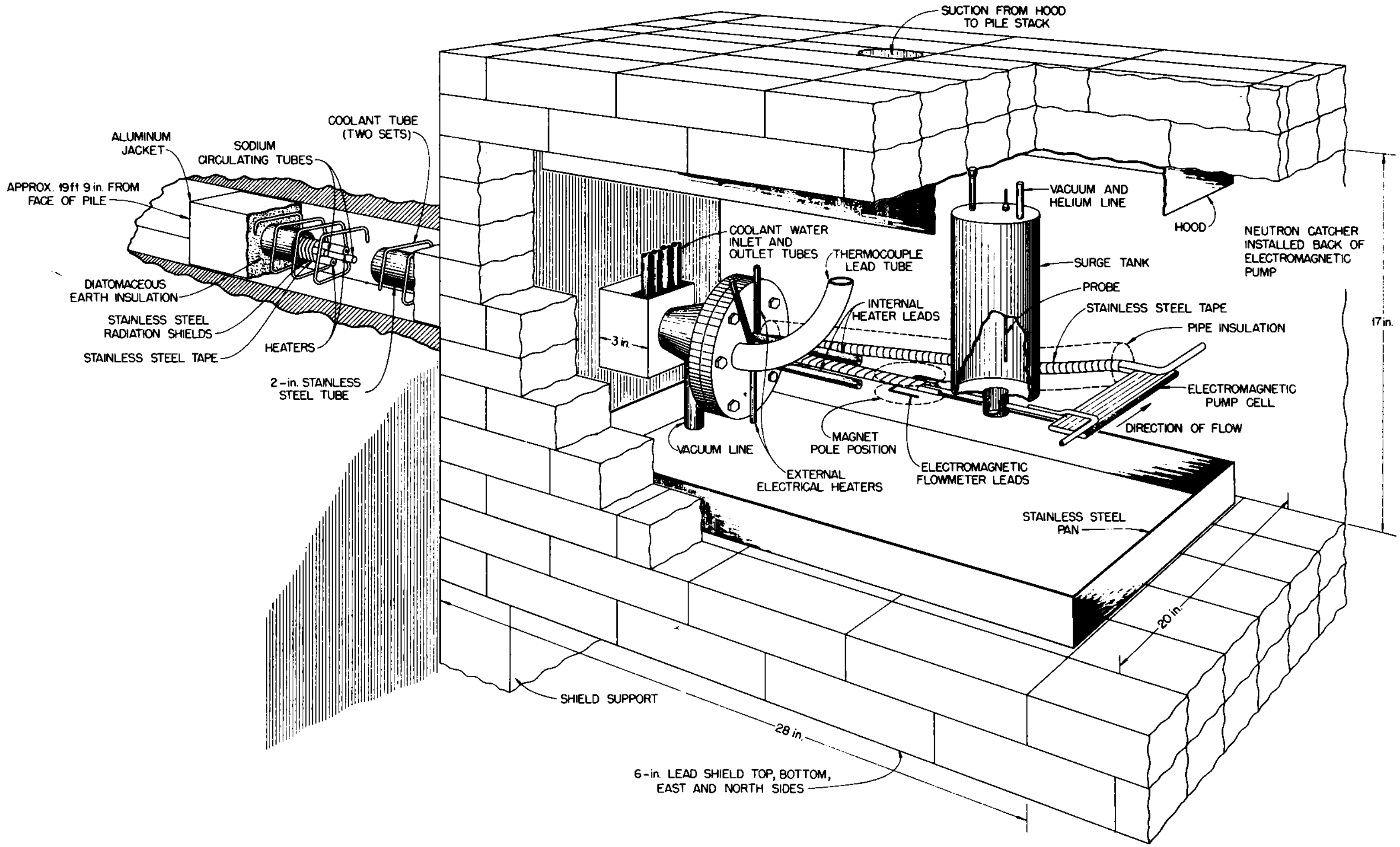

Fig. 7—Liquid metal loop in the graphite reactor.

stainless steel loop which was 0.385-in. OD x 0.0625-in. wall. The loops extended 18 to 19 ft into Hole 58 of the Graphite Reactor and approximately 12 ft in Hole HB-2 at the LITR. Helium or vacuum jackets were of stainless steel and were attached to the shield plug to furnish mechanical support to the specimen surge tank portion. The LITR loop (Fig. 8), which included a stress-corrosion specimen, was provided with additional instrumentation beyond that of the Graphite Reactor loops. This included automatic temperature control for the irradiated section and safety circuits for emergency shutdown of the reactor. Excessive temperatures indicated by thermocouples in the loop water jacket would first sound alarms and shut off the heaters; slightly higher temperatures would turn off reactor power.

Jacket water flow and jacket vacuum were other quantities monitored to produce a reactor setback and alarm.

Before the loops were filled, they were first flushed with alcohol, leak tested, and evacuated; then were filled with molten metal from a special metering apparatus. The liquid metal was forced by gas pressure from the melt tank through a filter to a monitoring tank and finally to the surge tank of the loop. After being filled, the loops were bench tested in the laboratory for 40 to 100 hr. The loop was cooled, until the liquid metal solidified and inserted into the reactor. The metal was then slowly remelted, beginning at the surge tank to avoid rupture from expansion.

Operation of the first loop of lithium-stainless steel was fairly uneventful. Failure of the loop occurred when the connections between the heaters burned out and circulation stopped. This occurred shortly after the temperature was raised from 1000 to 1500°F.

The first sodium-Inconel loop failed after approximately 160 hr due to stoppage of the flow. This was due to an obstruction apparently caused by mass transfer or a metallic reduction product of weld scale.

The second sodium-Inconel loop was found to have a small leak in the loop tubing before being inserted into the reactor. The leak was sealed by welding and the loop was operated for approximately 100 hr until another small leak developed at the junction of the tubing and the shield. This was detected by the monitor on the shield off-gas line.

The third sodium-Inconel loop was provided with two specimen chambers and a bypass purification system to filter out oxide in the spiraled tubing section to relieve stress. After 218 hr of reactor operation, the heaters failed, the loop was removed, and the heaters replaced. The loop was again inserted in the reactor and operated for 17 hr when a leak in the inpile section into the vacuum jacket was detected.

The stress-corrosion loop in the LITR was charged with sodium with the same procedure used in other loops. Temperature of the specimen in the in-pile section was held down to 1070°F because concurrent tests of the nickel-palladium braze with which the loop tubing was joined to one end of the specimen showed the braze to be susceptible to sodium corrosion. Part of a heater failed after 16 hr of operation, but the temperature was maintained by increasing power to one of the heaters. The electrical contacts (tungsten) of the extensometer oxidized to the extent that closure was uncertain, which rendered creep measurement impossible. After a total of 110 hr at 1070°F, the temperature was increased to 1280°F. After 49 hr at this temperature, circulation stopped. Examination after disassembly of the loop disclosed a leak in the brazed joint of the specimen with destruction of the heater around the leak.

Table III gives the operating conditions for the loop.

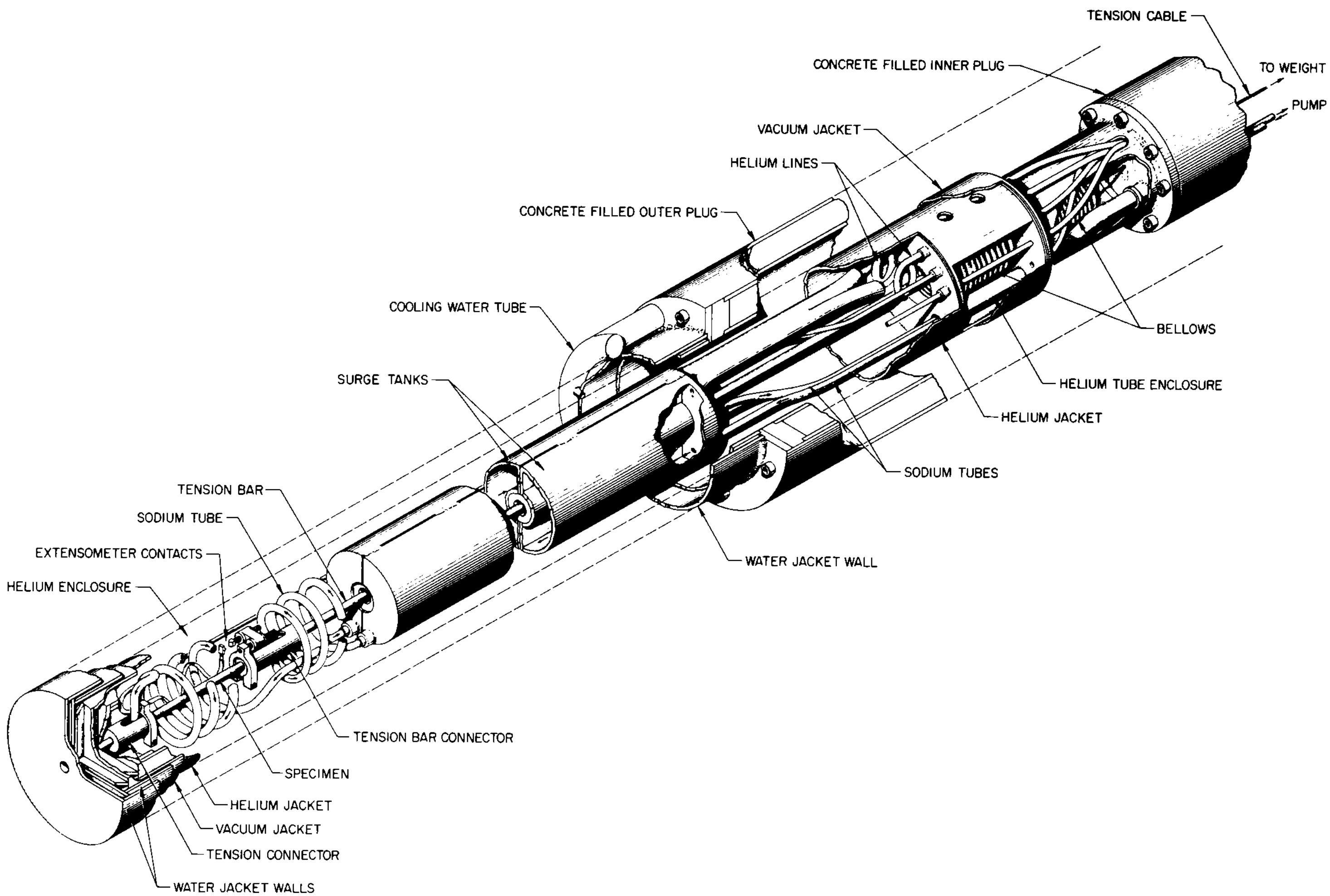

Fig. 8—Liquid metal stress corrosion loop in the LITR.

TABLE III.

OPERATING CONDITIONS OF IRRADIATED LIQUID METALS LOOPS

(Loop 1–3, OGR; loop 4, LITR.)

Loop No.	Liquid Metal	Loop Material	Bench Test		Irradiation Conditions							
			Temperature (°F)	Duration (hr)	Temperature (°F)		Duration (hr)	Velocity (fps)	Flux (neutrons·cm^{-2}·sec^{-1})		Exposure (neutrons·cm^{-2})	
					External	In-Pile			Thermal	Fast	Thermal	Fast
1	Lithium	316 SS	1000		1000	1000	160	~2	1.3×10^{11}	5×10^{10}	7×10^{16}	3×10^{16}
1	Sodium	Inconel	1000–1200	40	1100	1500	115	1	5×10^{11}	5×10^{10}	3×10^{17}	3×10^{16}
					1000	1000	50	1				
2	Sodium	Inconel	1000–1200	40	1000	1000	6	1–1.5	5×10^{11}	5×10^{10}	2×10^{17}	2×10^{16}
					1100	1500	95	1–1.5				
3	Sodium	Inconel	1000	84	1100	1500	235	1	5×10^{11}	5×10^{10}	4×10^{17}	4×10^{16}
4	Sodium	Inconel	900–1100	150	1000	1070	110	1	8×10^{12}	3×10^{12}	5×10^{18}	2×10^{18}
					1000	1280	49	1				

MSR PRESSURIZED WATER LOOP[5]

The MSR Pressurized Water Loop (see Figs. 9 and 10) will consist of an in-pile test section, a main heat exchanger to remove heat from the water, canned-rotor pumps for circulation, 60 kw of electric heat for temperature control, a surge tank and makeup system, purification system, and an out-of-pile control test specimen section. All process equipment except the in-pile section makeup system and sampling station is to be located in the ORR basement.

The test loop will provide a facility in which fuel and other materials can be subjected to radiation and other conditions simulating a pressurized water reactor. The objectives of the test program are:

(1) Prototype testing of fuel pins for use in the first nuclear powered merchant ship.
(2) Development and testing of other fuel element designs.
(3) Development and testing of other structural and control materials of interest in water-moderated reactors.
(4) Studies of water chemistry and activity buildup in pressurized water systems.
(5) Other studies of a basic or applied nature which require the combination of radiation and high-temperature water.

The in-pile section will consist of a U-tube of $1\frac{1}{2}$-in.-ID, 316 stainless steel pipe. This U-tube will contain up to six fuel pin test specimens, three in each leg. The cooling water will flow through the U-tube and over the specimens.

A stainless steel vacuum jacket around the in-pile and in-pool piping will serve as a thermal insulator.

There will be at least six feet of reactor pool water over the test specimen during removal; thus, no additional shielding is required.

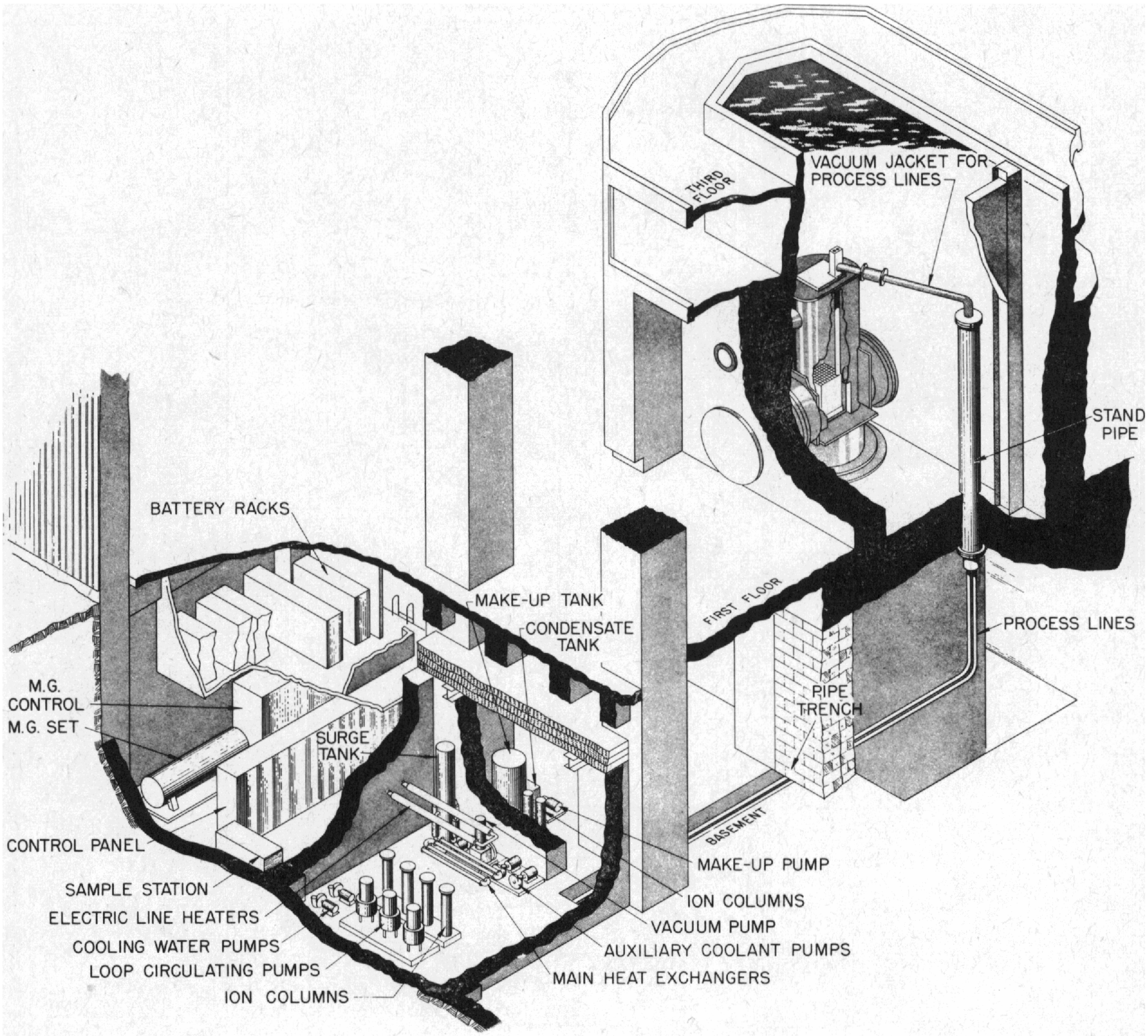

Fig. 9—MRS pressurized water loop in the ORR.

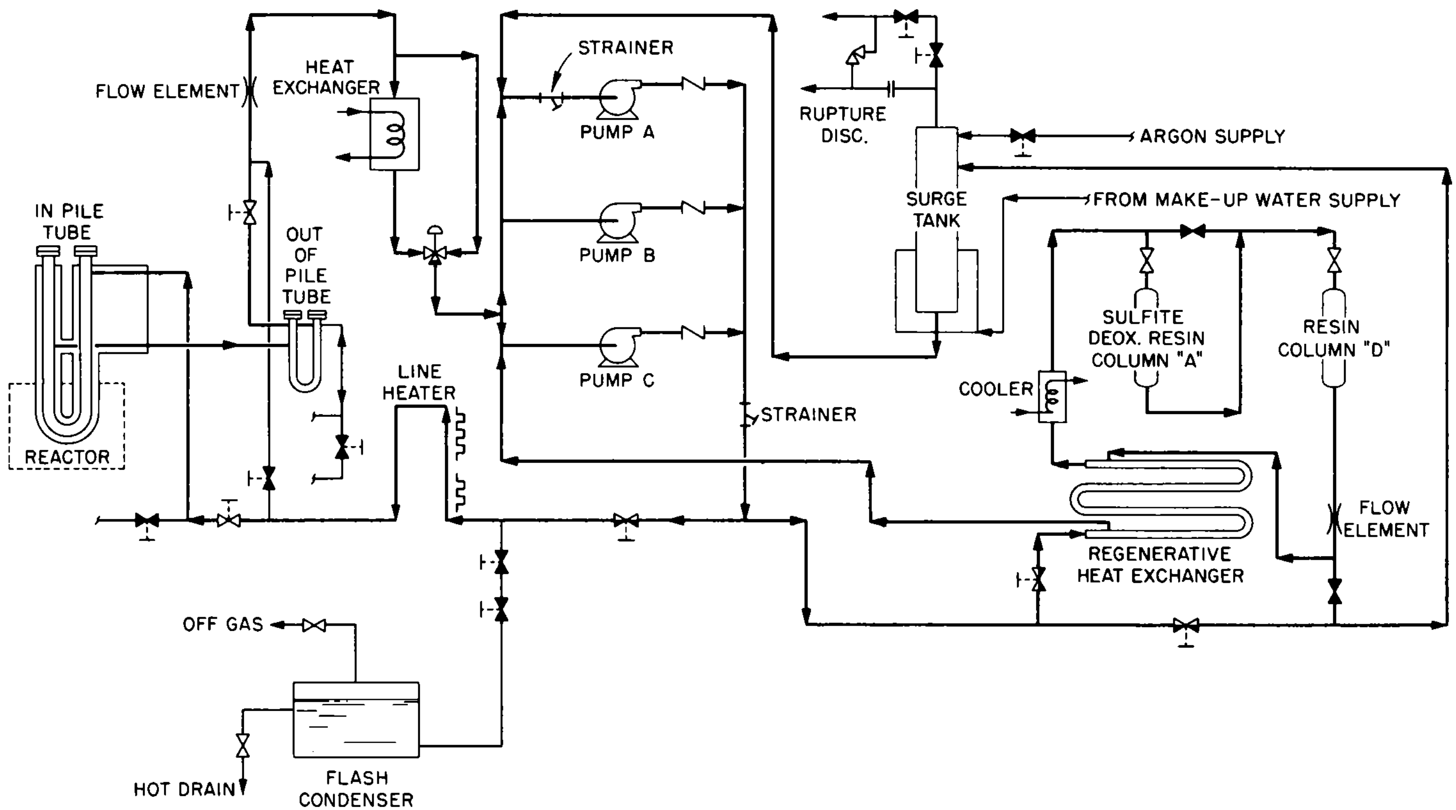

Fig. 10—MSR pressurized water loop at the ORR, simplified flow diagram.

Design pressure and temperature of the in-pile section is 2250 psig and 625°F. Combined thermal and pressure stresses will be limited to values specified by ASA code for piping.

A water-cooled heat exchanger will be used to remove the heat generated within the system by fission and gamma heating and for emergency cooling.

The three primary system water pumps will be water-cooled canned-rotor Westinghouse Model A-150D. These pumps are to be in parallel, with one normally operating and the other two on standby.

Clamp-on heaters will furnish up to 60 kw of heat to the system. The heat can be added at a controlled rate so as to maintain the desired recirculating temperature. Also, the heaters will serve to maintain loop temperatures when the reactor is down, to minimize thermal cycling.

The surge tank is provided to absorb volume changes during heating or cooling of the system and to provide and maintain the desired system pressure.

The makeup system will provide purified water for loop filling and replace loop losses during operation.

A bypass ion exchange system will be provided for the control of water purity in the loop. The entering water will have been cooled down to 95°F to prevent damage to the resin. Water leaving the column will be passed through the regenerative heater before being returned to the system.

Provisions are made to switch over to reactor cooling-water circulation if circulation of water through the loop is not possible.

TABLE IV.

SAFETY CONTROL OF THE MSR-PRESSURIZED WATER LOOP

Property Being Monitored	Normal Condition	Safety Responses		
		Alarm	Setback	Scram
Main loop flow, gpm	40	32		22
Reactor water flow, gpm	25	17.5		15
Main pump cooling temperature, gpm (3 pumps)	>6 each	4	4 (2 of 3)	
In-pile tube temperature, °F	600	625	700	
Saturation temperature, °F	521	550	580	
Main loop pressure, psig	1750	2150 (high) 1600 (low)	1450 (low)	

Loop instrumentation is designed to provide:

(1) protection of the reactor, loop, and personnel at all times from damage or injury;
(2) as complete automatic control of the loop as is practical;
(3) indication and recording of all necessary data.

Loop pressure is controlled by the steam pressure formed in the surge tank which is heated by a 24-kw-capacity electrical heater.

The loop temperature is controlled by a water-cooled heat exchanger and electric heater on the loop piping.

The greatest hazard associated with the experiment is considered to be loss of water circulation through the in-pile section. Unless corrective action is taken immediately, boiling of water in the in-pile section would cause violent "chugging" and a rapid increase in temperature. The resulting reduced strength of the stainless steel and increased stresses would endanger the in-pile pressure tube. Burnout of the fuel specimens would also be probable under this condition.

Relief valves are provided to prevent excessive pressures in the loop.

A brittle fracture in any part of the high-pressure system would be serious; however, a brittle fracture in such a system is considered highly improbable.

NITROGEN COOLED LOOP[6]

This loop was designed for the study of irradiation effects on ceramic fuels at elevated temperatures. (Table V).

The experiment is located in a stainless steel tube, extending through the top of the ORR tank into the core (see Fig. 11). Gas is circulated by a 30-hp compressor through the reactor section via connecting pipes. Coolant storage tanks, vacuum pumps, coolers, and other equipment are located along with the compressor in a shielded cell in the ORR basement. The system contains about 160 scf of gas. The top flange of the reactor section extends about 8 ft above the reactor tank and 6 ft below the surface of the pool. In removing samples a lead cask is supported by the crane above this flange and the radioactive samples pulled into it.

The hazards of the experiment include compressor failure which would reduce but not stop flow since a pressure tank would provide flow for about three minutes. The reactor power would, of course, have to be reduced in this event.

Fission products may escape from the sample and settle out on the piping walls. All piping in the reactor pool is shielded by about 8 ft of water. Piping in the basement is shielded by concrete. A removable filter in the compressor room is provided to remove a large fraction of any fission fragment release. This is shielded with 5 in. of lead in addition to the one foot of concrete cell wall.

If the reactor power is not reduced in event coolant flow stops, as provided by the safety system, the sample would melt and drop to the bottom of the experiment tube which is below the lattice in a zone of low flux. Reactivity effects of inserting and removing the sample and flooding the tube with water were measured. No large effects were found.

The basement cell is provided with an air purge designed to sweep any release of radioactivity to the off-gas stack.

Double seals are provided on all gasket flanges in the reactor pool. A purge gas flows between the seals and is monitored to detect any leakage of gas past the first seal.

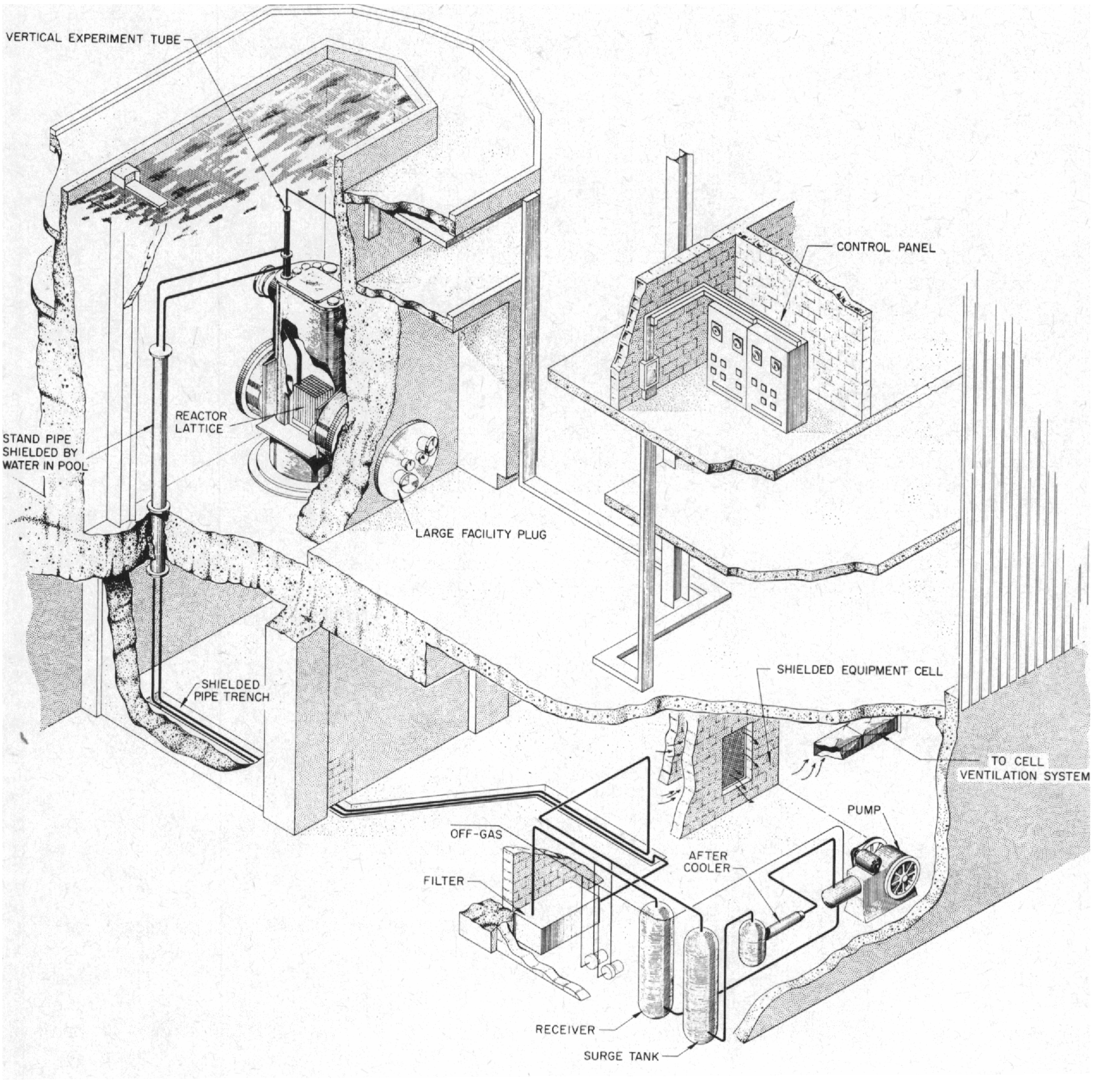

Fig. 11—Nitrogen-cooled loop in the ORR.

One of the most important features from the standpoint of reactor operation is a provision for withdrawing the sample above the high-flux zone within a few minutes. Thus, if some failure of the experiment requires that the reactor be shut down, the experiment can be quickly neutralized and the reactor started up again with little loss of time. In the ORR, this shortens such a shutdown to 10 - 15 min compared with the 5 - 10 hr which would be required if the reactor were caught by xenon and the fuel had to be changed.

An occasion to use this feature occurred when the loop was first put into service. A Kovar seal leaked, allowing gaseous activity to escape into the reactor building. Within a short time the sample was pulled up above the flux zone and the reactor started.

LOW TEMPERATURE LOOP

Although most of the loops in ORNL research reactors have operated at high temperatures, a series of low temperature loops[7,8] have been used since 1955 for irradiating materials at very low temperatures. With the latest loop of this type, temperatures as low as 3.8°K have been obtained. The loop (Fig. 12) is located in the ORNL Graphite Reactor in a vertical hole, extending approximately 20 ft from the top of the shield to the center of the reactor. This facility permits investigation of radiation-induced defects at temperatures sufficiently low that the defects are retained in the original configuration in which they were introduced.

The low-temperature section of the loop was designed as a cryostat around a helium refrigerator originally designed by the Arthur D. Little Company and Cambridge Company to deliver refrigerated helium gas at temperatures as low as 10°K and a capacity of 100 watts at 20°K. The original capacity was increased by the use of additional compressors to deliver a maximum of 100 pounds of helium per hour with a resulting refrigerator capacity of about 40 watts per degree and with a terminal temperature of 8°K. Expansion engines with improved valve design were also incorporated and have been quite reliable.

TABLE V.

CHARACTERISTICS OF THE NITROGEN-COOLED LOOP

Operating Conditions

Volume	160 scfm
Maximum temperature of specimen	~2000°F
Gas flow	~17 scfm
Gas pressure	95 - 120 psig

Sample Devices

	Alarm	Setback
Sample temperature	x	x
Sample temperature (duplicate)	x	x
Capsule temperature	x	x
Coolant radioactivity (2)	x	
Supply tank pressure	x	
Sump tank pressure	x	
Coolant flow	x	x
Compressor water flow	x	
Flange purge gas radioactivity	x	

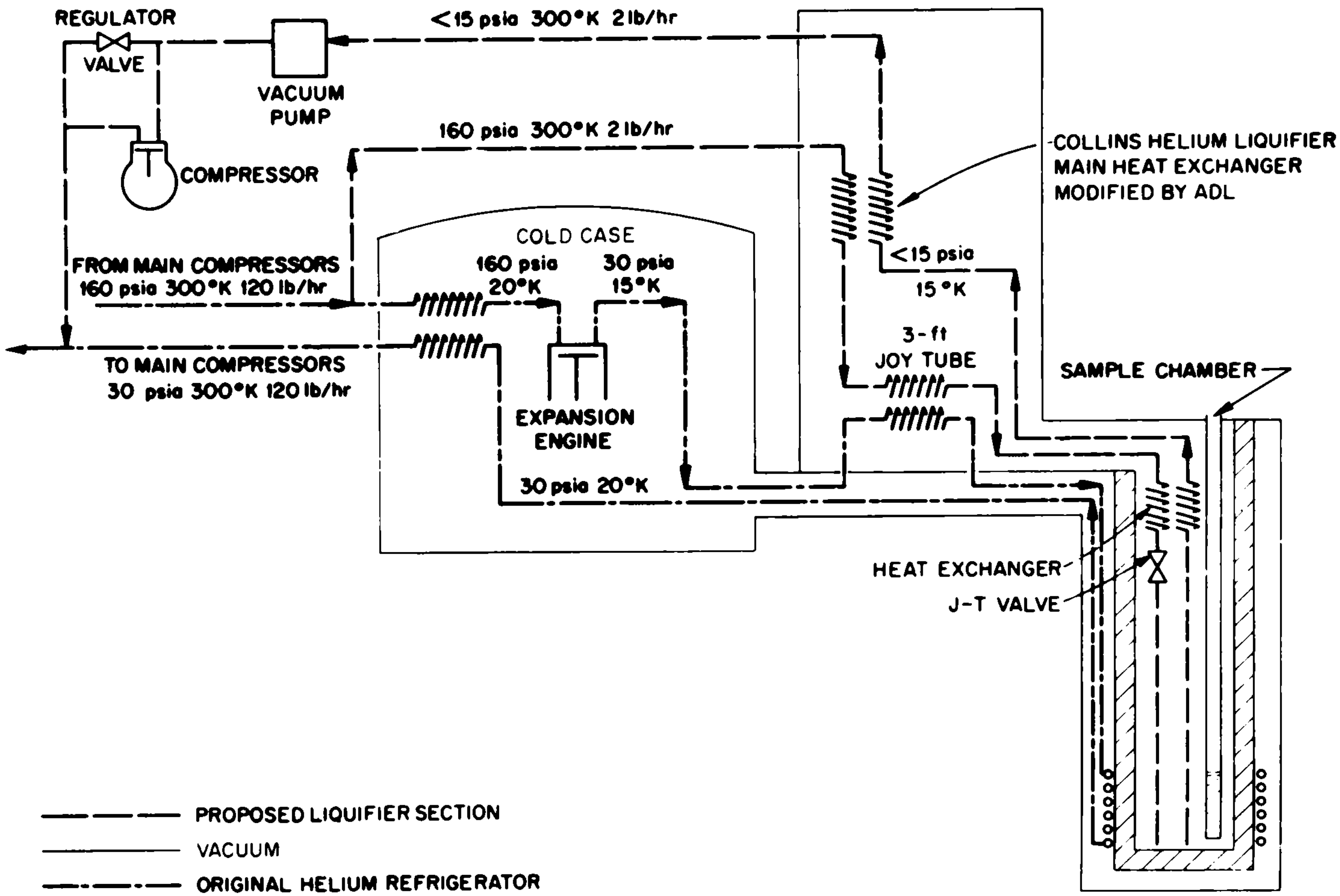

Fig. 12—Low-temperature loop flow diagram.

Some of the considerations imposed on the design by the reactor were the following: (a) the loop extended into a very small hole of a relatively long length, (b) the ambient temperature in the reactor in the vicinity of the loop was often above 150°C, and (c) the loop had to be constructed of materials in which there would be no long-lived neutron-induced activity.

In removing the very long irradiation tube, as was occasionally necessary, it was a considerable advantage to be able to handle the tube without shielding. This could be done only if 2S aluminum and other such materials having low induced activities were used. The in-pile section (Fig. 13) of the loop as finally designed consists of four parts: the sample tube leading to approximately 20 feet below the top of the reactor, a sample chamber heat exchanger, inlet and outlet heating lines, and a vacuum jacket which houses the entire assembly. Exces-

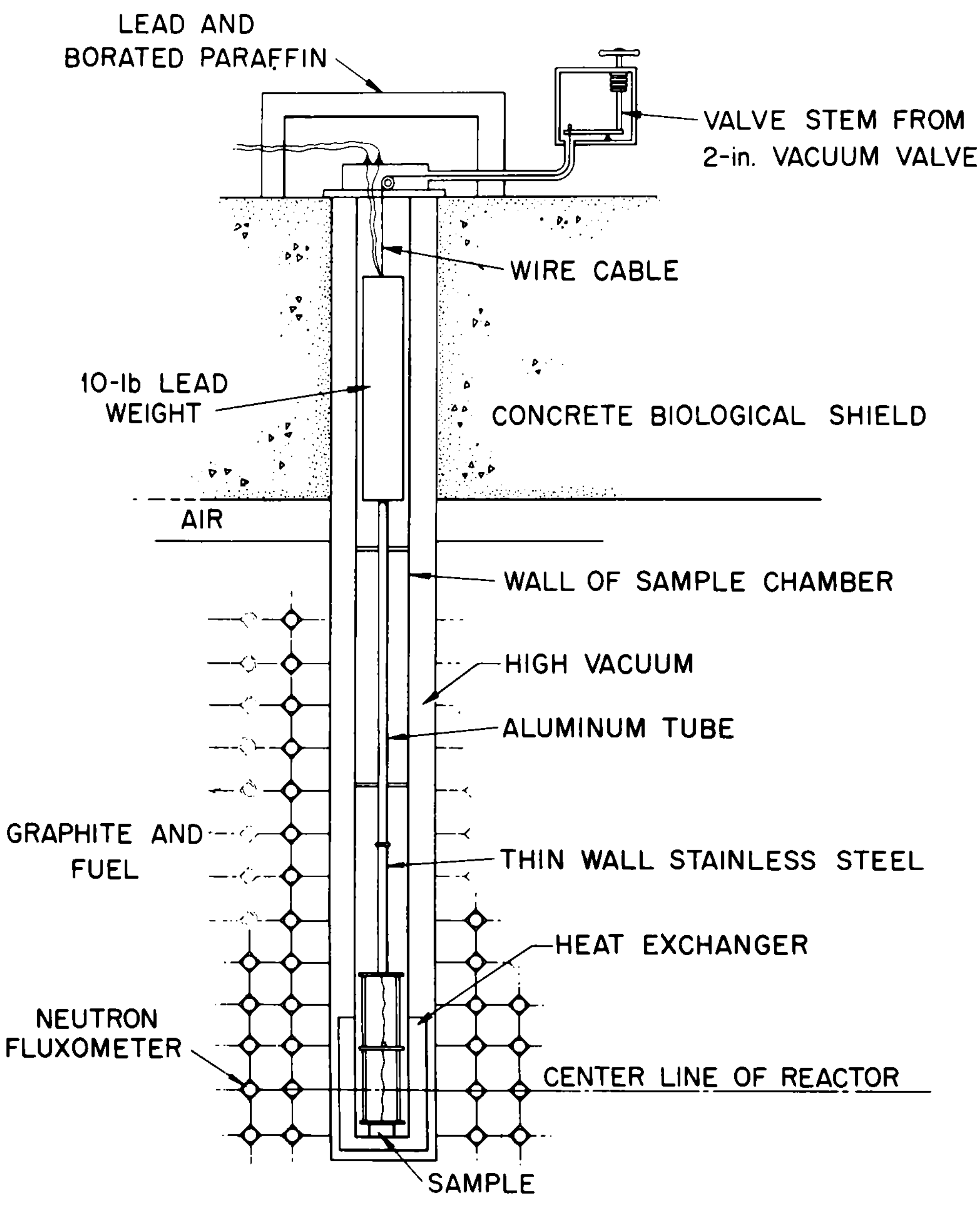

Fig. 13—In-pile section of low-temperature loop.

sive heat loads due to high conductivity of the aluminum were reduced by incorporating a thin-walled stainless steel tube into the sample tube about two feet from the top of the cryostat. This section of the tube was inside the reactor shield where the neutron flux was very low. The helium lines also were aluminum in the lower 17 feet. These are connected to a copper heat exchanger. Aluminum surfaces were chemically polished by a bright dip to reduce emissivity. To obtain still lower temperatures, an attachment was built which can be slid into the cryostat to produce temperatures as low as 3.8°K for continuous periods of 300 hours. Approximately 2% of the high pressure gas, cooled to 10°K by the refrigerator, is liquidified by expansion through a Joule-Thompson valve.

In order to keep the heat load at a minimum, the Joule-Thompson valve was located out of the flux in the concrete shield of the reactor approximately 12 feet above the sample. The parts of the liquidifier in the flux are composed of thin-walled stainless steel with a mass in the order of 200 grams, representing a heat load of one watt.

The sample chamber is a closed tube extending to the liquefying gas chamber. The sample is cooled by filling the tube with helium at a pressure sufficiently high to produce liquefication.

Besides the advantages of operating at lower temperatures, helium loops have a great safety advantage over earlier static cryostats filled with liquid nitrogen. On one occasion an open-mouth liquid nitrogen dewar exploded after three days' operation in the reactor. An intense odor of ozone, probably associated with nitrous oxide, could be detected. The second explosion occurred during the initial run of the helium cryostat described in reference 7. The vacuum line leading to the sample ruptured, and several hours elapsed before repairs could be made. Air froze in the sample chamber, and during warmup after termination of the run the cryostat exploded. The characteristic odor described above was again observed. It is believed that in each incident oxygen which condensed in the cryostat was converted to ozone by the ionizing radiation of the reactor. The ozone may decompose violently, react with nitrogen, or with any organic materials present.

The major safety device found necessary with this type of loop is a vacuum safety to ensure that air does not reach and condense on cooled surfaces inside the high-flux region of the reactor. Also, since the sample tubes are straight, radiation from the reactor may be dangerous if the reactor is started while the hole is open. It is, therefore, necessary to monitor the shielding plug with such experiments to be sure it is in place before the reactor is started.

SINGLE-PASS AIR-COOLED LOOP[9]

These loops are designed for use in testing a variety of fuel-element configurations and types under various conditions of air flow, fuel temperature, and general gas-cooled reactor environments. Information desired includes fission product release rates, corrosion data, and other factors related to general fuel element integrity.

Figure 14 shows, isometrically, the location and function of various components of the loop which has been in use at the LITR for several years. Most of the instruments and controls are not shown here but consist primarily of standard commercially available components.

The experiment access tube consists of a 4-in.-OD flexible metal hose which enters the reactor tank through a side flange as shown in Fig. 14. The lower end of this hose is flanged to a 6-ft length of 2-in., schedule-40, 2S aluminum pipe having the lower end closed. The lower section of this pipe is

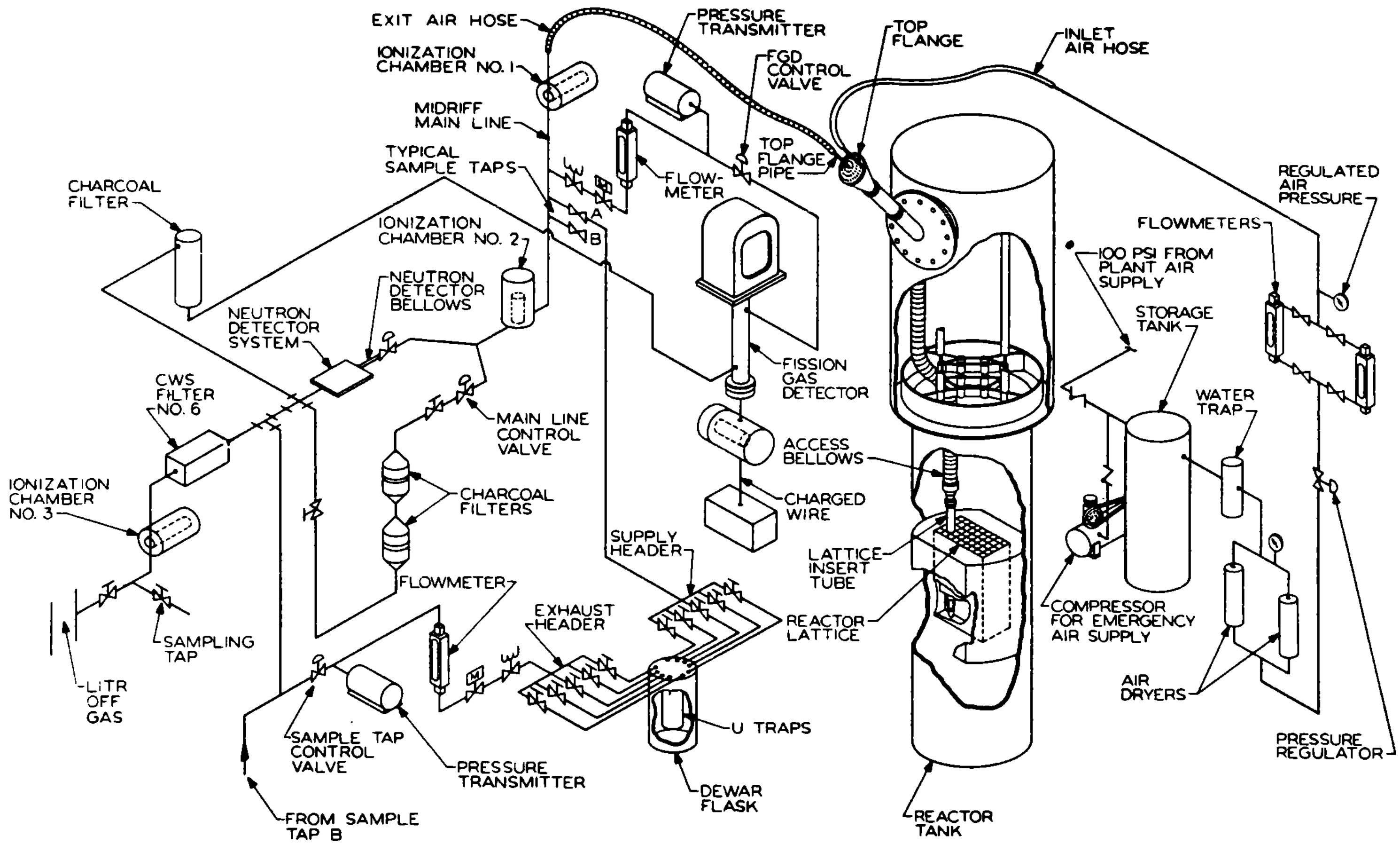

Fig. 14—Drawing of single-pass air-cooled loop.

inserted into a hollow beryllium piece having the same outside dimensions as a reactor fuel element and is located in one of the core lattice positions.

The test specimen is generally mounted in a ceramic housing which forms the cooling-air passages and serves as a mounting for thermocouples. The ceramic assembly is then provided with additional thermal insulation and is enclosed in a metal container which has air entrance holes at the bottom and a pipe connection at the top. The top section is then equipped with a length of flexible metal hose which serves as an exit air line and permits insertion and removal of the test capsule through the various bends in the access tube. The upper end of the exit air hose connects to a straight section of rigid tubing at a point a few feet below the top of the access tube. The rigid tube passes through a packing gland in a flange which seals the access tube entrance. Thermocouple leads connect to gas-tight connectors which penetrate the flange. The inlet air is supplied to the access tube which operates under internal pressure and serves as the in-pile inlet air line. Thus air flows down the access tube around the outside of the test capsule, enters the capsule through the bottom entrance holes, flows around the test specimen to provide cooling, and exits through the flexible hose arrangement which penetrates the access tube flange.

Inlet Air System

The major source of cooling air is the 100-psig plant-air supply which is backed up by an electrically operated compressor that is automatically started if the supply pressure decreases to a predetermined value. Lines connecting both sources to the experiment system contain check valves to prevent back flow from either source to the other.

Air enters the loop through a 60-ft^3 storage tank which, in the event of air supply failure, will provide sufficient coolant flow to overcome shut-down heat in the experiment.

From the storage tank, air flows through a water trap to remove water droplets and thence through one of two air dryers which are alternately switched so that one dryer is in use while the other is being regenerated.

From the dryers the air passes through a hand-operated control valve which is used to establish the system operating pressure through one or both of two parallel connected Fischer-Porter, variable-area-type flow meters, then to the experiment access tube through a final length of neoprene hose which permits quick installation or removal of the access tube flange.

Exit Air System

Exit air from the experiment leaves the access tube through a length of flexible metal hose which allows for vertical adjustment of the experiment (metal hose is necessary due to the high temperature of the exit air). The section of exit-air line immediately following the flexible hose is monitored by a remotely instrumented, air-filled ionization chamber (No. 1) which is used for determining the general radioactivity level (gamma detection only) of the exit air. Following this point sampling stations permit the continuous removal of air from the main exit line for purposes of radioactivity analysis. The main line flow is again monitored by an ionization chamber (No. 3) and then flows through a pneumatically operated control valve which maintains the desired flow rate, thence through a bank of charcoal traps to the entrance point of a main-line, paper, particulate filter. The purpose of the charcoal traps is to reduce the fission gas concentration of the exit air in the event of a large fission product release from the experiment.

One of the sampling stations supplies air to a charged-wire device designed to indicate the fission product gas concentration in the air stream. In this device, air flows axially through a cylindrical chamber along whose axis a recorder wire is moved at a constant rate. The wire is insulated from, and is charged negatively (>1000 v) with respect to, the cylinder. Thus, fission gases which decay while in transit through the chambers are attracted to the wire. The traveling wire then transports these products to a shielded gamma scintillation detector where the radioactivity is determined. Any activity detected must come from fission product daughters which are also radioactive; thus argon activity, for instance, does not contribute to the background. This device has proved to be very useful in detecting low concentrations of fission product gases in the exit air from these experiments.

Another sampling station provides air samples to a bank of small refrigerated charcoal traps which are later removed from the system and transported to a laboratory where more refined radioactivity analysis can be performed.

A third sample is supplied to a neutron-sensitive detector for determining the delayed-neutron counting rate and thus certain information relating to the fission product concentration and age in the exit air stream.

Sample air lines return to the main exit-air line at the entrance to the main particulate filter. Following this filter, the general gamma activity level is again monitored by an ionization chamber to indicate the filtering efficiency.

Hazards

The major hazards associated with this type of experiment are:

1. release of dangerous quantities of fission products to the building and surrounding atmosphere (this would most likely result from a leaky flange connection, a defective weld, or severe mechanical damage to the exit air piping),
2. direct radiation fields from the exit-air system due to the accumulation of a large quantity of fission products in some inadequately shielded region,
3. excessive heat production in the in-pile assembly resulting in local boiling of the reactor cooling water with consequent oscillation of reactor power,
4. extreme overheating of the in-pile assembly resulting in thermal damage to the aluminum pipe which separates the in-pile assembly from the reactor system (rupture of this pipe could result in any or all of the following:
    a. severe contamination of the reactor cooling system by fission products,
    b. injection of a large quantity of air into the reactor tank, producing collapsible voids that could result in a violent fluctuation of reactor power and a possible power excursion,
    c. damage to the lattice structure and/or surrounding core pieces,
    d. damage to other experiments located nearby in the core).

Safeguards

Safety instrumentation and precautions are dictated primarily by (1) excessive temperature and (2) high radioactivity levels.

1. Excessive temperatures produce the following corrective or preventive action:

Condition	Action
Loss of main air supply	Reactor scrams; alarm sounds; emergency compressor starts

Condition	Action
Low air flow	Alarm sounds; emergency compressor starts; reactor is shut down manually if normal air supply is not restored, or the experiment is partially withdrawn and operation of the reactor is resumed
Excessive fuel temperature	Alarm, setback, and scram sequentially as temperature increases
Excessive experiment housing temperature	Alarm, setback, and scram sequentially as temperature increases

2. High radioactivity levels result in the following:

Condition	Action
High direct radiation reading in the exit-air piping area	Experiment instruments sound alarms; reactor monitron sounds alarms; investigation made and proper action taken
High concentration of radioactivity in building air	Reactor building constant air monitor sound alarms; investigation made and proper action taken

In addition to these safeguards, the exit-air piping from these experiments is enclosed in lead shielding in all areas where personnel hazards might result from high direct radiation. Further, the design and construction of such a facility is reviewed critically by a committee of experts in various scientific fields, and approval of this committee is necessary before the experiment can be operated. The instrument and built-in safeguards of the experiment system result from close co-operation between the experimenters and the reactor operations personnel and from the final experiment reviews.

During the operating history of this facility fission products have been released to the building atmosphere on several occasions. The usual causes of these releases were defective welds, failure of flange gaskets, and poor connections in the air sampling systems. No dangerous personnel radiation dose or contamination has been received during these incidents, however, and enclosure of the charcoal trap samplers in a box which is vented to the reactor off-gas line has reduced the frequency of such occurrences.

ORGANIC LIQUID LOOP

The organic liquid loop was designed and operated for the study of the effects of energy deposition[10,11] in organic liquids during in-reactor irradiation upon the liquid viscosity, gas evolution rates, boiling point, and other engineering characteristics of several organic liquids such as alkylbenzene and Dowtherm-A.

Figure 15 schematically describes the recirculating hydrocarbon loop which was operated in horizontal beam hole HB-2 of the LITR. Basically the loop flow was as follows. Liquid entered the pump intake through a pipe to the bottom of the sump. The pump discharged into a cavity containing a continuously recording viscosity probe. From there the liquid flowed through an electrically powered heater where the liquid temperature was raised to the desired value, thence through several feet of tubing to the in-pile container. The liquid was returned to the external system via a second length of tubing. The flow was approximately 3 gpm.

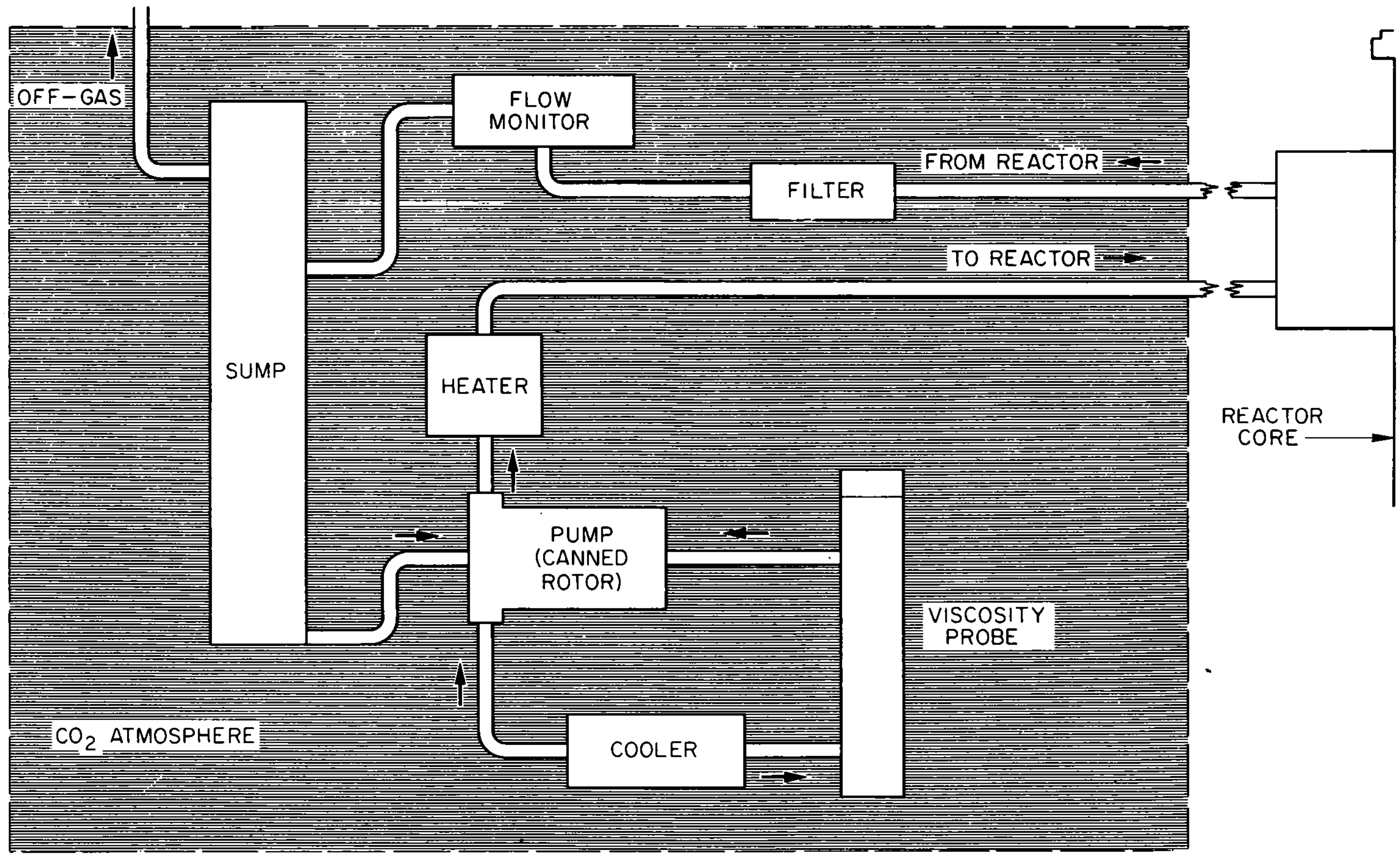

Fig. 15—Drawing of organic-liquid loop.

From here the liquid was passed through a water-cooled, counter-flow heat exchanger where sufficient temperature and flow data were taken to permit a heat balance calculation and thus determination of the nuclear heat deposition in the liquid. The liquid was then passed through a fine-mesh stainless steel screen, thence through a flow sensing element and returned to the sump.

The pump was a canned-rotor, electric-motor-operated type which permitted complete isolation of the organic liquid from the motor windings and electrical connections which were made through explosion proof conduit and terminal boxes, thus reducing the possibility of ignition of the organic liquid. It was necessary to install cooling-water coils on the external housing of the pump motor and in the pump intake line to prevent thermal damage to the motor windings during operation of the loop at maximum liquid temperatures.

The vibrating-reed viscosity probe was installed immediately after the pump discharge since the pump cooling coils permitted rather close control of the discharge liquid temperature and thus standard viscosity readings in accordance with the probe manufacturer's specifications.

The heater unit was of shop construction, consisting of a helical coil of stainless steel tube (approximately 3-in.-ID) surrounding a cylindrical external-helix-wound heater form. The entire assembly was wrapped with thermal insulation (about 1/2 in. of Thermoflex) and enclosed in a mild steel cylinder. Heater leads were brought through the external loop closure to the heater housing through a sealed insulating sleeve. This construction reduced the possibility of a fire due to shorting of the heater leads.

Return liquid from the in-pile section was passed through a fine mesh (~300 mesh) stainless steel screen for removal of any foreign material or coke which may have accumulated and which could result in partial or total plugging of lines and/or interfere with the operation of the flow-sensing element.

The flow-sensing device used was a Potter Aeronautical propeller type utilizing a tiny rotating magnet and a fixed coil for signal production. The frequency of the voltage induced in the coil was determined by a count-rate meter to give a direct indication of flow rate.

The sump liquid level was determined manually by use of a calibrated dip stick. Makeup liquid was added as needed to make up for sample removal and volume loss due to gas evolution. The gas pressure in the sump was maintained within the desired range by a differential pressure switch which operated a solenoid valve in a line leading from the top of the sump to the reactor off-gas system. The gas volume released was measured by an integrating gas meter (wet-test type) located in the same line.

All the external portion of the loop except the filler access tube and sample removal top was enclosed in a mild steel housing having a Plexiglas window for visual observation of the components. This housing was flanged to the in-pile tube sheath, the in-pile end of which was welded to the in-pile container water jacket. A tube supplied CO_2 gas to the extreme end of the insulated space surrounding the in-pile hydrocarbon container. Gas was released here at a controlled rate and allowed to flow along the annular region surrounding the in-pile container and then through the tube sheath and discharge into the external equipment enclosure. A connection from this enclosure to the reactor off-gas system allowed continuous flow of the gas.

Samples of the CO_2 were removed periodically and analyzed to determine if a leak existed in the organic liquid system and if a combustible mixture of gases was present.

Hazards

The hazards associated with the organic-liquid loop are:

1. fire -- due to ignition of liquid or gas which may have leaked from the system,
2. explosion -- due to the accumulation of evolved hydrogen gas which may have leaked from the system,
3. contamination of personnel and building -- due to the spilling of removed samples or to leaks from the loop.

Safeguards

Fire or explosions were not considered very probable if the system developed no leaks and/or if the CO_2 blanket was maintained sufficiently. Therefore, those factors which could ultimately lead to fire or explosions dictated the major safety actions required:

Condition	Action
Loss of flow	Alarm, setback, and scram sequentially as flow decreased; turn off heaters
Excessive liquid temperature	Alarm, turn off heaters; setback and/or scram if condition is not corrected
Excessive heater temperature	Alarm, turn off heaters
Excessive in-pile shell temperature	Alarm, turn off heaters; setback and scram if condition is not corrected
Loss of in-pile shell coolant flow	Alarm and setback sequentially as flow decreases to preset minimum required values
Loss of CO_2 pressure	Alarm, condition to be corrected by attendant personnel; reactor shut down manually and heaters turned off if condition cannot be corrected
High concentration of hydrogen in blanket case	Situation evaluated and appropriate action taken by attendant personnel
Loss of electrical power to instruments or the pump	Alarm (battery operated), reactor scram, and turn off heaters.

These loops were manned 24 hr per day during operation and therefore many corrective or preventive actions were to be performed by the attendant personnel.

Incidents

Some contamination of the floor in the LITR building resulted from spilling irradiated liquid; however, this was easily cleaned.

Leaks developed at connections in the external equipment enclosure, but the CO_2 blanket maintained a safe condition with respect to fire or explosion.

GAS-COOLED REACTOR LOOP

The gas-cooled reactor loop (Fig. 16) is being designed and is scheduled to begin operation in 1960. The following description is, therefore, necessarily of a general nature, and, since all the possible hazards associated with the experiment have not been completely evaluated, more safety equipment and procedures will be specified as the design proceeds and evaluation of each new phase is completed.

The purpose of the proposed facility is to provide data for the heat transfer and physical characteristics of clad gas-cooled fuel elements at power ratings

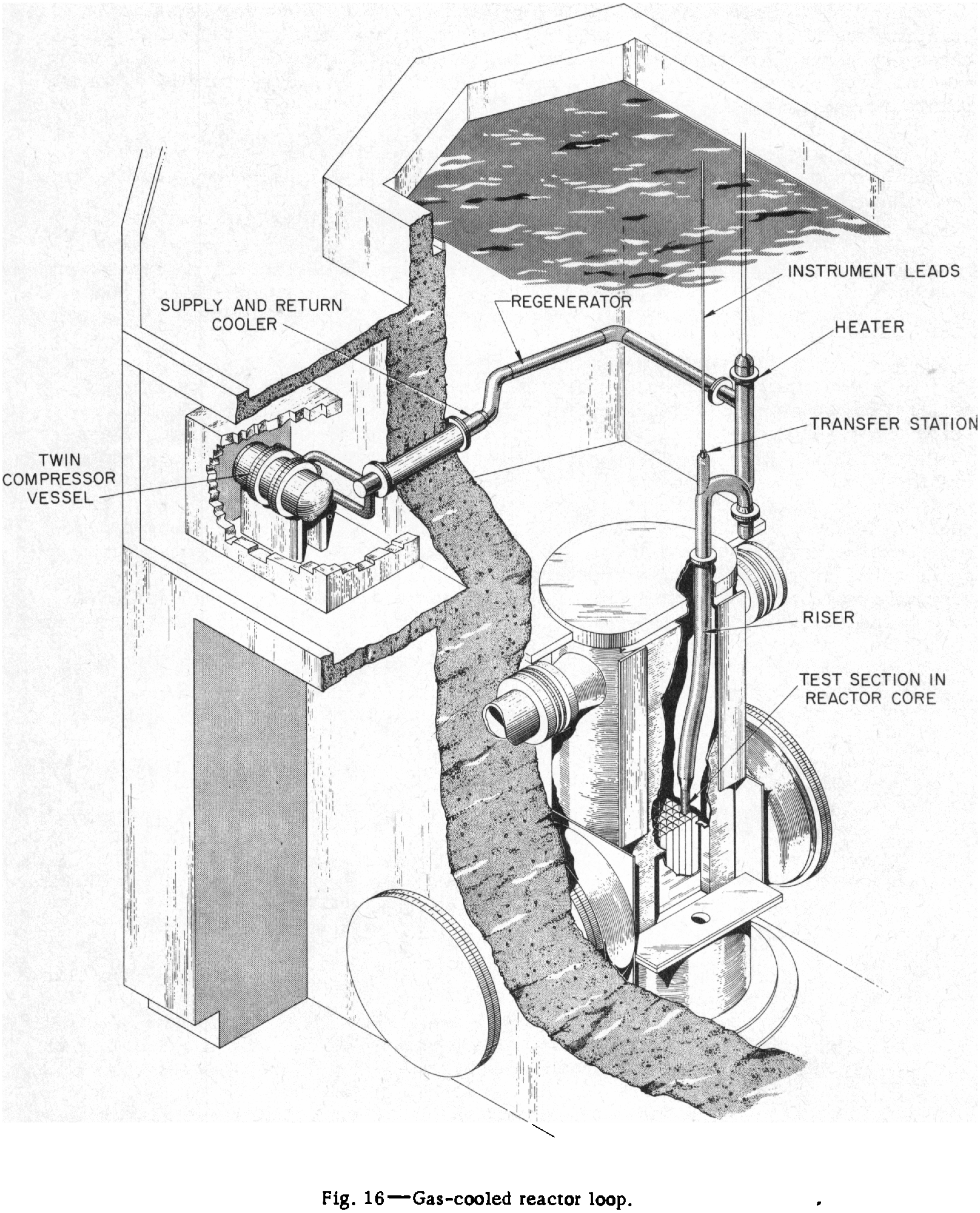

Fig. 16—Gas-cooled reactor loop.

up to approximately twice that of a maximum GCR-II fuel element, with surface temperatures up to that required for destruction of the cladding material.

The loop will be a helium-filled system containing a compressor set for circulating the helium gas, an in-pile section for containing the test elements, necessary piping for transporting the gas to and from the in-pile section, and several heat exchangers for maintaining the desired gas temperature at various points in the system.

The compressor set consists of a pair of regenerative turbine compressors in series, arranged so that either compressor will supply emergency cooling in the event that one compressor fails. Both compressors can be removed and replaced without breaking any external piping or electrical leads.

The compressors discharge into the center pipe of a concentric arrangement leading to the in-pile test section. These lines penetrate the reactor pool wall via an access thimble at a level approximately five feet below the normal reactor pool water level.

Here the gas passes through a water-cooled heat exchanger where the gas temperature is reduced to the desired value. Following this, the gas flows through a regenerator where it absorbs heat from the return gas stream, then flows through an electrically powered heater which produces a controlled temperature rise, then through a vertical regenerating (riser) section where it again absorbs heat from the return gas stream. The gas then enters the test section, flows over the test fuel element surfaces to provide the desired cooling, reverses direction and returns to the compressor set through the outer connections of the concentric tube arrangement. During the return the gas temperature continually decreases due to heat transfer to the supply gas stream and to external losses. The supply and return cooler further reduces the temperature to the desired turbine inlet gas temperature.

Some design criteria for the loop are as follows:

Fission heat load	20 kw
Gamma heat load	20 kw
Maximum fuel clad surface temperature	1300-1700°F
Maximum gas temperature	1000-1400°F
Clad-to-gas ΔT (in all cases)	300°F
Coolant (helium) pressure at test element	400 psig
Mass coolant flow rate	348 lb/hr

Hazards

Possible hazards are:

1. direct radiation from external equipment due to a high concentration of fresh fission products in the gas stream,
2. building-air contamination due to a leak of gas from the coolant system following the failure or destruction of the test fuel element,
3. damage to the reactor due to:
    a. rupture or leaking of fission products into the coolant system or pool water,
    b. rupture or leaks in the in-pile section which could release helium gas in the core section, producing voids (this could lead to violent reactor power oscillations and/or a possible power excursion),
    c. violent rupture of the in-pile section, resulting in damage to the reactor lattice structure and elements and/or damage to other experiments.

Safeguards

The safeguards employed will include the following:

1. All the loop gas handling equipment will be of heavy duty construction with welded connections where possible. Radiographs will be made of all welds to assure that they are sound. Prior to use, the system will be pressure- and leak-tested and thoroughly checked for operability.

2. The external loop will be shielded by a lead enclosure. Piping in the reactor pool will be shielded by approximately five feet of water during operation. Additional shielding may be necessary.

3. All flanged connections will be equipped with a leak detection system.

4. Loop temperatures and pressures will be monitored by recording instruments equipped to initiate corrective or preventive action if certain limits on these properties are exceeded. Some of these will undoubtedly require a reactor setback or scram.

5. Radioactivity monitors will be located at desirable points to warn of high direct radiation or air contamination.

A final safety review of the experiment will be made by the Reactor Experiment Review Committee after all design has been completed.[12]

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APPENDIX

Questionnaire for ORNL In-Pile Experiments (Revised 11-6-58)

In order to simplify the obtaining of necessary approvals and facilities for putting experimental equipment into ORNL research reactors, the following questions should be answered in as much detail as practicable. All questions do not necessarily apply to all experiments and necessity of answering any particular question, where not obvious, should be decided with the aid of Operations Division supervision. Additional questions may be added to the list by the Operations Division if needed. The answers requested should be appended to the questionnaire.

1. What is the purpose of the experiment?

2. What division and department is sponsoring the experiment?

3. What charge codes and work order numbers apply to the fabrication and operation of the experiment?

4. List the names, business phone, and home phone numbers of supervisory personnel in charge of the experiment in the order they are to be called in the event of emergency.

5. List the names and experience of all personnel qualified to operate the experiment.

6. Will the experiment be continuously manned? Why?

7. What is the estimated monetary value of the experiment equipment? What is the value of each in-pile assembly?

8. Which reactor and what facility will be used?

9. What neutron flux and/or gamma flux are desired?

10. What preliminary neutron flux and gamma heating measurements must be made? By whom are they to be made?

11. What will be the duration of the experiment program?

12. What will be the desired operating time for each irradiation?

13. What are the amounts and kinds of materials which will be within the reactor? List both the physical materials and their elemental breakdown. Also list the neutron absorption cross sections for each material and the total in cm^2, i.e., number of atoms times the microscopic cross section. Include dimensioned drawings.

14. List the major components of external equipment and provide a layout sketch of the floor space required.

15. Utilities:

    a. Electricity
        (1) Amperes at 440 v________
        (2) Amperes at 220 v________
        (3) Amperes at 110 v________
        (4) D-C Supply______________
        (5) Emergency supply________

b. Compressed Air
   (1) Pressure __________psi
   (2) Flow __________scfm
   (3) Emergency__________

c. Water
   (1) Process Water
      Pressure ________psi
      Flow ________gpm
      Emergency________

   (2) Demineralized Water
      Pressure ________psi
      Flow ________gpm

   (3) Reactor Water
      Pressure ________psi
      Flow ________gpm

   (4) Pool Water
      Pressure ________psi
      Flow ________gpm

d. Drains
   (1) Normal drain________gpm
   (2) Warm drain ________gpm
   (3) Hot drain ________gpm

   (2)–(3): Estimate quantities of radioactivity

e. Off-gas______scfm. List gases and contaminants expected, rate of release, and amount of radioactivity.

f. Steam________scfm, pressure________psi

16. List all monitoring equipment under the proper headings:

Property being monitored and expected value	Reason for monitoring (safety, control, or data)	Sensing element and its location	Indicator or recorder (name & type)

Safety response (if any) and at what value (alarm, setback, scram or combination of these)

17. Does all instrumentation conform to the recommendations of the Operations Division and the Instruments and Controls Division?

18. What instruments constitute that group which could be classed as the minimum safety requirement?

19. If there is a manual scram provided with the instrumentation, under what circumstances will it be used?

20. Can the experiment be easily and rapidly removed from the neutron flux so that the down time for the reactor will be short in the event of experiment failure? What length of time will be required? If such retraction is automatic, what safety provisions are made to limit the withdrawal speed?

21. Provide complete calculations of the fission heating and gamma heating within the parts of the experiment that will be within the reactor. Ascertain the amount of heat removed by the coolant and the amount which goes into the reactor coolant system.

22. Provide a flow pattern of each coolant path and include complete calculations

of pressure drops and temperatures within the coolant stream, or provide bench test data for the experiment which include these values.

23. If the experiment is a fuel loop, provide a flow pattern and include all volumes, flow rates, and pressure drops. Include detailed calculations or actual measurements.

24. Provide complete calculations of the pressures within the various internal and external components of the experiment. Under what circumstances do these vary?

25. If pressures are above atmospheric describe the containment features incorporated in the experiment.

26. Provide complete calculations of the radioactivity of all effluents and specify the radioactive elements involved. Specify the number of curies per day to be discharged to the off-gas system and to the warm or hot drains. This must be done both for the routine condition and for experiment failure.

27. Provide complete calculations and the necessary supporting data for any effluent cleanup systems that may be required.

28. Provide complete calculations for the thicknesses and materials of all external biological shielding.

29. Are any parts of the experiment subject to radiation damage? If so what is the expected life of the parts?

30. What effect will the experiment have upon the reactivity of the reactor? Will there be any internal motion of the experiment during its operation which can cause reactivity variations?

31. What influence will utility failure have upon the equipment? What must be done to restore operation? What is the heat-up rate in the event of coolant failure?

32. What are the recognized hazards associated with the experiment both with respect to personnel and to the reactor and its components, and what measures will be made to eliminate or minimize these hazards?

33. What procedures will be followed to assure maintenance of instruments and safety devices?

34. Prior to installing the experiment in the reactor will a bench check of the apparatus be made using the actual instrumentation to be installed?

35. What special conditions, if any, must be met with respect to reactor operation? Method of startup? Method of shutdown?

36. Will any special attention by Operations personnel be required?

37. Will any special Health Physics monitoring or services be required?

38. Include copies of the installation, operating and removal procedures for the experiment. Also insertion, sampling, testing, and removal check lists.

39. Are "dry-runs" to be made of the insertion and removal procedures?

40. What transport shields will be provided for the removal and disposal of the experiment? Include dimensioned drawings.

41. Provide complete calculations for the required thickness of the experiment-handling shields and the expected radiation through the shields.

42. What special handling tools are to be provided? Include dimensioned drawings.

43. Will the hot cells be required for dismantling the experiment? What operations will be done in them?

44. What will be the final disposal of the radioactive portions of the experiment?

(Name of person completing questionnaire)

(Date)