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HIGH FLUX ISOTOPE REACTOR

A GENERAL DESCRIPTION

T. E. Cole

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

020-001

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ACKNOWLEDGEMENT

This report was prepared through the cooperative efforts of members of the Oak Ridge National Laboratory Reactor Experimental Engineering Division, Instrumentation and Controls Division and Health Physics Division. The staff of the Oak Ridge Office of the U. S. Weather Bureau prepared information on the meteorological aspects considered.

FOREWORD

This report is issued primarily to make available a general description which reflects the current status of the High Flux Isotope Reactor which is being planned for construction at Oak Ridge National Laboratory. At the date of writing, an Architect-Engineer firm, Singmaster & Breyer of New York, N. Y., has just been selected to perform the facility design. The description set forth herein is based on conceptual design studies only for the facility arrangement and on a preliminary design basis for the reactor proper. It is anticipated that the basic plan for the facility arrangement will not change substantially although many details of arrangement will undoubtedly change. It is expected that few changes will be made in the reactor proper inasmuch as the configuration described is the result of about one year's study on optimization of the arrangement from a nuclear standpoint. Considerably less effort has been placed on preliminary design of the mechanical arrangement of the reactor; however, sufficient effort has been expended to allow reasonable assurance that no major changes will be made.

SUMMARY

The High Flux Isotope Reactor (HFIR) is being planned for construction at Oak Ridge National Laboratory as a supporting facility in the program of investigation of the properties of the transplutonium elements. The reactor will be a flux-trap reactor consisting of a beryllium-reflected, light-water-cooled annular fuel region surrounding a light water island. An irradiation sample of 200-300 g of Pu^{242} will be placed in the island where a thermal neutron flux of approximately 3×10^{15} $n/cm^2 \cdot sec$ can be achieved on the average during an irradiation period of about one year. It is estimated that more than 100 mg of Cf^{252} will be produced by such an irradiation.

In addition to the central irradiation facility for heavy element production, the HFIR will have eight hydraulic rabbit tubes located in the beryllium reflector and four beam holes for basic research.

Preliminary design of the reactor has been based on the results of a parametric study of the dimensions of the island and fuel region, heat removal rates, and fuel loading on the achievable thermal neutron fluxes in the island and reflector. A research and development program including critical experiments, heat transfer, corrosion and fuel element studies has been in progress to verify the important parameters used in the design.

The present design results in an average power density of 2.2 Mw/ℓ in the active core and requires a maximum heat transfer rate from fuel plate surfaces of 1.5×10^6 $Btu/ft^2 \cdot hr$. This heat transfer rate is achieved by flowing H_2O, at an inlet temperature of 120°F, and a pressure of 600-900 psig, through the .05 in. coolant channels at a velocity of 40 fps.

A preliminary analysis of the hazards brought on by a reactor core meltdown shows that a controlled-leakage, filter-scrubber, stack release system of the ORR-type will limit the consequences of such an accident to an acceptable degree.

Construction is scheduled to start in early 1961 with operation at power scheduled for January 1964. The estimated cost of the facility including engineering is $12,000,000.

1. INTRODUCTION

1.1 Historical Background

Review of Requirements. - At a meeting held on December 6, 1957, attended by various members of the staff of the Oak Ridge National Laboratory, the possible need for neutron fluxes an order of magnitude higher than presently available, for the production of transplutonium elements and other isotopes, improved research in neutron diffraction, cross section measurements, and solid state physics, was discussed. As a result of this meeting, a series of informal seminars were held to discuss in more detail the need for an ultra high flux research reactor and to review the technical problems associated with the design and construction of such a reactor. Verbatim transcriptions of these seminars have been issued as as ORNL memorandum.(1) The primary conclusion reached in the discussions was that the most pressing need for high thermal neutron fluxes, i.e., $3\text{-}5 \times 10^{15}$ neutrons$\cdot$cm$^{-2}\cdot$sec^{-1}, exists in connection with the production of transplutonium elements and other isotopes.

The status of the transplutonium production program received its first serious review at a meeting on January 17, 1958, sponsored by the AEC Division of Research and attended by representatives of the various laboratories interested in the program. At this meeting it was proposed to carry out an irradiation program to meet present and anticipated needs for transplutonium isotopes. After a review of this proposed program by an *ad hoc* committee established to assist the Division of Research with the production and distribution of transplutonium isotopes, the irradiation program was initiated. The production goal of this program was to obtain about 20 μg of Cf^{252} by 1962 and milligram quantities by 1967.

Although the U. S. transplutonium production program initiated in 1958 seemed to be adequate to meet projected needs, increased interest in the properties of the heavy elements indicated that a more accelerated production schedule might be advisable. Since any accelerated schedule would require a new isotope production reactor providing thermal neutron fluxes higher than any presently available, a meeting was held in Washington, D. C., on November 24, 1958, to review plans for the construction of such a reactor and other high flux research reactors. The unanimous recommendation of those participating in the meeting was that a high flux isotope reactor be designed and constructed at the Oak Ridge National Laboratory with construction to start in fiscal year 1961.

A preliminary design report was issued in March 1959 for the purpose of review by those persons interested in the heavy element production effort.(2) General agreement that the proposed facility would meet the requirements of the heavy element production program was received and authorization was requested to proceed with plans for constructing the facility.

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(1) J. A. Lane *et al.*, Ultra High Flux Research Reactors, ORNL CF-58-7-117 (July 31, 1958).

(2) J. A. Lane *et al.*, High Flux Isotope Reactor, Preliminary Design Study, ORNL CF-59-2-65 (Mar. 20, 1959).

The Division of Reactor Development has supported research and development on the HFIR during fiscal year 1960 in order to provide for necessary work in establishing a firm basis for the design. This effort to date has been directed primarily at the physics, heat transfer, corrosion, and fuel element development phases of the project and work is continuing along these lines. Continued support for this program is anticipated for the fiscal year 1961.

Directive Authorization, CL-227, dated January 26, 1960, has been received from the USAEC and funds are authorized for preliminary engineering work (Title I) on the entire facility. An Architect-Engineer firm, Singmaster & Breyer of New York, N. Y., has been selected to perform this work. The estimated cost of the facility including the engineering effort is $12,000,000.

Studies of High Flux Research Reactors. - Numerous studies of high flux research and test reactors have been carried out (see Bibliography) which indicate that an annular fuel region with a central moderating region and a good reflector provides the best geometry for maximizing the thermal neutron flux per unit of power. Since the cost of research and test reactors is roughly a function of the power level, internally moderated reactors also provide maximum thermal neutron fluxes for a given cost. It should be pointed out that the maximum thermal neutron fluxes referred to are, in some designs, available over a very limited volume in the reactor.

1.2 The Need for High Thermal Neutron Fluxes

Heavy Element Production. - The production of milligram quantities of americium, curium, berkelium, and californium is important for the development of separations processes for these elements, to study their behavior in existing processes such as Purex, for use as special radiation sources, and to determine their nuclear, chemical, and biological properties. At the thermal neutron fluxes available in existing reactors, however, very long exposure times and large amounts of starting material are required. The irradiation of 10 kg of Pu^{239} at 1.5×10^{14} neutrons$\cdot cm^{-2} \cdot sec^{-1}$ for seven to eight years, for example, results in the production of only 1 mg of Cf^{252}. Fortunately, the production of transplutonium elements in a reactor increases rapidly with increases in neutron flux. This is discussed in the following paragraphs.

The production of transplutonium elements through successive neutron captures in a reactor proceeds according to the following reactions:

$$Pu^{242} + n \longrightarrow Pu^{243} \xrightarrow[5\ hr]{\beta} Am^{243}$$

$$Am^{243} + n \longrightarrow Am^{244} \xrightarrow[26\ min]{\beta} Cm^{244}$$

$$Cm^{244} + 5n \longrightarrow Cm^{249} \xrightarrow[(min)]{\beta} Bk^{249}$$

$$Bk^{249} + n \longrightarrow Bk^{250} \xrightarrow[3\ hr]{\beta} Cf^{250}$$

$$Cf^{250} + 2n \longrightarrow Cf^{252} .$$

Thus, it is seen that ten successive neutron captures are required to produce Cf^{252} from Pu^{242}. Because of this, the initial production rate of Cf^{252} from plutonium goes up as the tenth power of the flux. Although this initial rate drops because of burnup of lighter materials and by the approach to equilibrium of some of the high-cross-section intermediates, after a year's irradiation the flux dependence is still somewhere between the second and fourth order.

Similarly, starting with Cm^{244} eight successive captures are required to produce Cf^{252}, also making the production rate extremely flux-dependent. The incentive for achieving fluxes of 3×10^5 (after insertion of the material to be irradiated) is very great if the desired production, $\sim$ 50 mg, is to be achieved in a time of the order of one year.

The production of Cf^{252} as a function of the amount of source material has been examined, and in addition the production has been calculated as a function of the ratio of slow-to-fast flux. Unfortunately, the depression of the thermal flux in the experiment increases with size of sample. Thus there is an optimum sample size to produce the maximum amount of Cf^{252} in a given time. Calculations indicate that 200 g of Pu^{242} will depress the flux averaged over one year's irradiation about 43%. If it is assumed that the unperturbed flux in the test hole is $5\text{-}6 \times 10^{15}$, the optimum quantity of Pu^{242} that should be exposed for production of Cf^{252} is in the range of 200 to 300 g. The investigation of the effect of the ratio of fast flux to thermal flux showed that the present nuclear design is near to the practical optimum for maximum production.

Fig. 1 shows the calculated californium production as a function of time per 100 g of Pu^{242} in the sample assuming that all the material is exposed to the flux levels stated.(3)

In addition to the production of Cf^{252} and other transplutonium elements, the proposed HFIR can be used to produce special isotopes such as Pu^{244}, which can be used to great advantage for certain nuclear research work because of its long alpha half life and zero spin. On account of the short half life of Pu^{243}, only negligible quantities of Pu^{244} can be produced at fluxes below 10^{15} neutrons$\cdot cm^{-2} \cdot sec^{-1}$, whereas at fluxes of $3\text{-}5 \times 10^{15}$ up to 2% of Pu^{242} can be transformed.

(3) H. C. Claiborne, Effect of Non-Thermal Capture on Californium Production in the HFIR, ORNL CF-59-12-16 (Dec. 3, 1959).

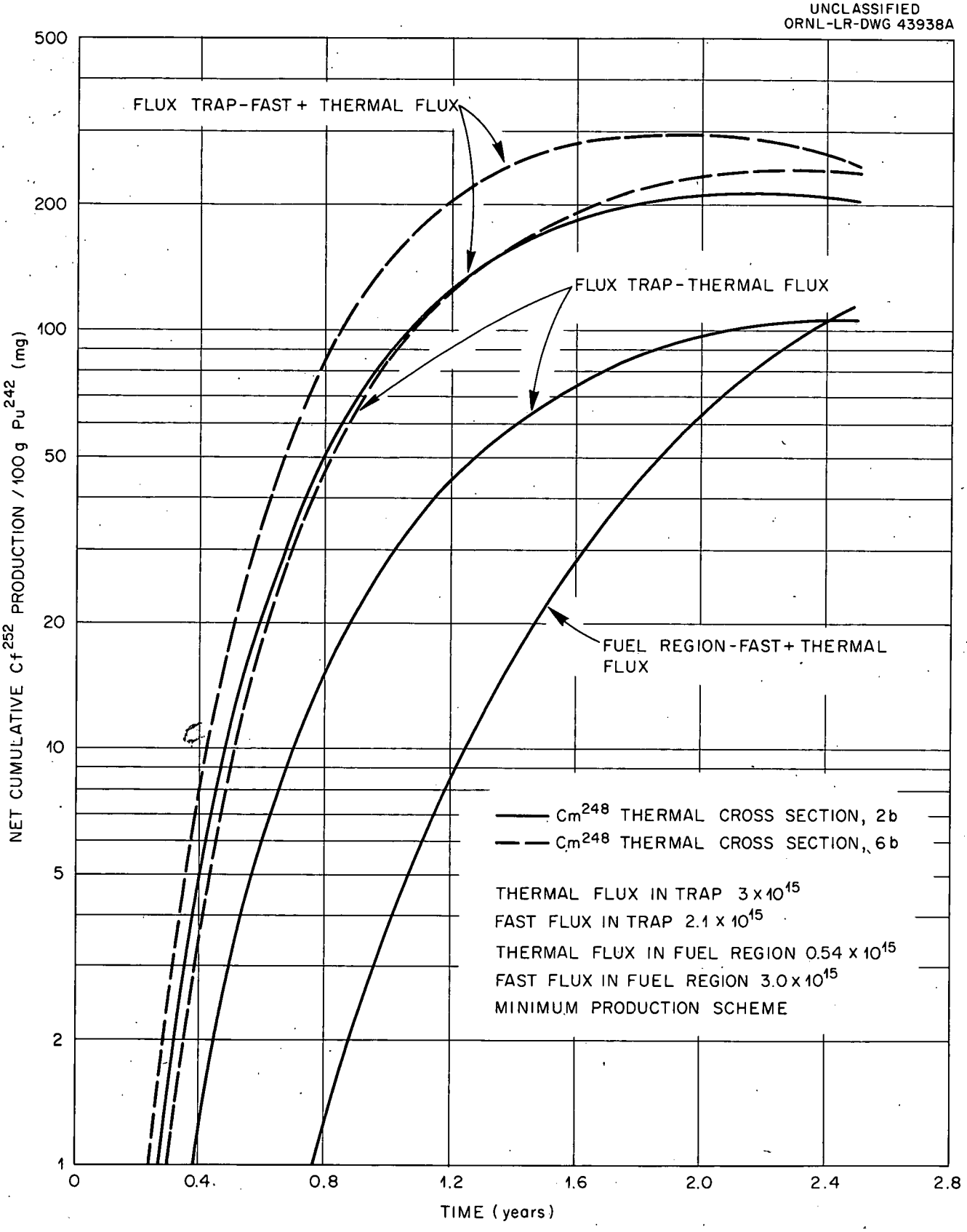

Fig. 1. Californium Production from Thermal and Non-Thermal Captures.

The HFIR can also be used to vary the isotopic ratios of certain transplutonium elements by burnout of high-cross-section isotopes. Curium, which normally contains up to 95% Cm^{244}, 4% Cm^{246}, and 1% Cm^{245}, is an example of this. By irradiating such a mixture at the available high flux, it is possible to obtain curium which is largely Cm^{248} and which contains little or no Cm^{244}.

The Production of Special Isotopes. - 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$\cdot cm^{-2} \cdot sec^{-1}$ 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.

(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-year Ba^{133} and 5.3-year 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 years at 10^{14} flux to 6.6 years at 10^{15} flux. Also, for Co^{60} the time for maximum activity is decreased from 7.8 years to 1.6 years at these two flux levels.

(3) The HFIR can also be used to burn out undesired atoms in an isotopic mixture. Fox 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.

Neutron Beam Research. - Although the HFIR is intended primarily for isotope production, the design will incorporate four nominally 4-in.-dia horizontal beam holes for neutron diffraction experiments and cross section measurements. The estimated thermal neutron flux at the inner face of these beam holes is 1×10^{15}.

Solid State Physics. - The fast flux in the fuel region of the HFIR will be about 5×10^{15} at full-power operation. Materials can be irradiated adjacent to this region to carry out accelerated radiation damage studies; an integrated fast flux of about 10^{23} can be achieved in some samples in one year. The available flux will be of particular interest in determining the effect of radiation on the mechanical properties of metals and ceramics used in power reactors. Studies of solid state reactions and neutron diffraction experiments with metals can also be done expeditiously in the facilities and fluxes available in the HFIR.

Achievable Thermal Neutron Fluxes. - It is seen from the foregoing that thermal neutron fluxes in the range of $2\text{-}4 \times 10^{15}$ are desired for the production of heavy elements and 1×10^{15} for the production of other isotopes. In regard to the former, operating power levels of 100 Mw and maximum power densities up to 4000 kw/liter must be achieved to obtain an effective thermal neutron flux of 3.5×10^{15} in a 200 g sample of Pu^{242}. It is believed that while such conditions may appear to push the technology of water-cooled reactors to their present limit, operation of this reactor at its design power results in reasonable operating conditions.

2. DESCRIPTION OF FACILITY

2.1 General

To accomplish its primary purpose, transplutonium isotope production, the HFIR requires very high thermal neutron fluxes (5×10^{15} nv, unperturbed) although the required space for sample irradiation, by comparison to space needed for engineering tests, is small. Thus, a water-island, flux-trap-type cylindrical reactor fueled with fully enriched U^{235} was selected in the interest of minimizing capital and operating costs. A further reduction in costs was realized from the use of aluminum fuel plates, water coolant and moderator, and a water-cooled-beryllium reflector. A review of previous flux-trap core designs[4] indicated that a core power of 100 Mw should be adequate, and since this was consistent with the amount of money that could probably be made available for the HFIR, the core power was specified as 100 Mw.

The reactor core is cylindrical in geometry and has a water-filled cylinder at the center, the flux trap or island, 5.5 in. in diameter. Surrounding this is an annular fuel region of 4.7 in. thickness. The fuel region is

[4] See Bibliography on Flux-Trap Reactors.

surrounded in turn by a thin control region and a beryllium reflector of 12 in. thickness. This core assembly is housed in a pressure vessel approximately 8 ft in diameter.

Experience gained in the design and operation of the Oak Ridge Research Reactor[5] has shown that the basic idea of locating the reactor tank in and near one end of a pool of water works very well. This general scheme is proposed for the HFIR. As in the ORR, the top of the reactor tank will be below water level, thus greatly simplifying operations in which highly radioactive materials, such as fuel elements or irradiated samples, are transferred from the reactor tank to the storage pools. Beam holes which penetrate the end walls of the canal will be provided for horizontal access to the reflector region surrounding the core.

The heat released in the reactor core will be removed by circulation of ordinary water which is cooled in tube-and-shell heat exchangers with final heat rejection to the atmosphere through cooling towers. The reactor system will be pressurized during operation in order to prevent boiling at the surface of the fuel plates. The reactor coolant will be maintained at relatively low temperature, $\leq 200^\circ$ F, and therefore no great amount of energy storage is present in the water system.

The pool will be about 20 ft wide and 80 ft long. At the east end, where the reactor vessel is located, the pool is to be about 40 ft deep with only a minimum floor space in addition to the reactor for the storage of equipment and tools used in reactor operation and servicing. A shallower section, 18 ft deep, common to both the reactor and canal bays of the building, extends an additional 60 ft to provide the space required for fuel element storage, experimental equipment installation, rabbit stations, gamma facility, critical facility, etc. A dividing wall with gates and a transfer port separates the pool into compartments to minimize mixing of the relatively clean water of the storage pool with that of the reactor pool. Operations in which there will be a possibility of the release of large quantities of radioactive contaminants will be limited to the reactor bay and the reactor pool. The relatively clean canal operations will be confined to the canal area and its pool section.

The pool side walls and the pool floor will be provided with a metallic (stainless or stainless-clad) liner to prevent the leakage of water through the concrete walls and to provide a smooth surface for underwater cleaning.

At the center line of the reactor the water in the pool and the thick concrete side walls act as the biological shielding for personnel working in the adjacent experimental area of the beam room. The concrete on this side will be 8-10 ft thick. Above the center line the thickness will be decreased due to the increased water shielding between the core and pool wall, until at the top of the structure only the water in the pool is required for shielding.

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(5) T. E. Cole and J. P. Gill, The Oak Ridge National Laboratory Research Reactor (ORR), A General Description, ORNL-2240 (Jan. 21, 1957).

2.2 Reactor Assembly

General Description. - The reactor core as shown in Figs. 2 and 3 consists of an annular ring or rings of aluminum-clad enriched uranium fuel plates and a beryllium reflector. Together with the supporting structure for the fuel and reflector sections, the reactor can be considered a packaged unit mounted in the lower part of the reactor vessel. Cooling water, about 15,000 gpm, enters the vessel through two nozzles near the top flange, is directed through the fuel, island, control, and reflector regions in parallel and is discharged at the bottom.

The main cover plate of the vessel is to be removed only for major servicing and repair of the reactor. Normal servicing and refueling will make use of a smaller quick-opening hatch in the main cover plate directly over the core.

Several experimental and production facilities are provided in addition to the central island. Four nominally 4-in-dia horizontal thimbles pierce the reflector to within a few inches of the fuel from the beam room at the reactor center line for conducting basic and applied research. Additional isotope production and other irradiations can be performed in or near the reflector through tubes and thimbles which enter the vessel through the main cover plate or through nozzles in the side wall near the upper flange.

Fuel Assembly. - The present design is based on fuel assemblies which consist of an annular ring or rings of involute curved fuel plates. The plates will be 40 to 50 mils in thickness and arranged to provide a uniform 50 mil coolant gap. Other types of elements which are being investigated are discussed in Section 5.2.

Preliminary Design of Target Array. - About 200 g of Pu^{242} as the oxide will be mixed with aluminum powder, made into pellets and packed into nineteen 3/8 in. OD, 18-in. long aluminum tubes. These tubes will be inserted in the island as a 19-rod cluster. The peak heat load will be about 3/4 Mw. This heat is removed by about 1000 gpm of water flow through the island.

Reflector. - The reflector will be fabricated from reactor grade beryllium metal. The outer section which encloses the beam tubes and hydraulic rabbit facilities is considered semi-permanent in that it would be removed only under conditions of major servicing. The inner few inches of beryllium will be subject to radiation damage from the high value of fast neutron flux and will be designed to be replaceable.

Control Plates and Drives. - The control plates consist of two concentric annular rings which move vertically between the fuel and reflector regions of the core. As shown in Fig. 4 the lower section of the inner ring would be fabricated of high cross section (black) material such as cadmium or europium, the middle part of the inner ring of some medium cross section (gray) material such as titanium and the upper section of the inner rod of a low cross section material such as aluminum or beryllium. Each section would be one core length and the total travel of the plate would be three core lengths. The outer ring would be

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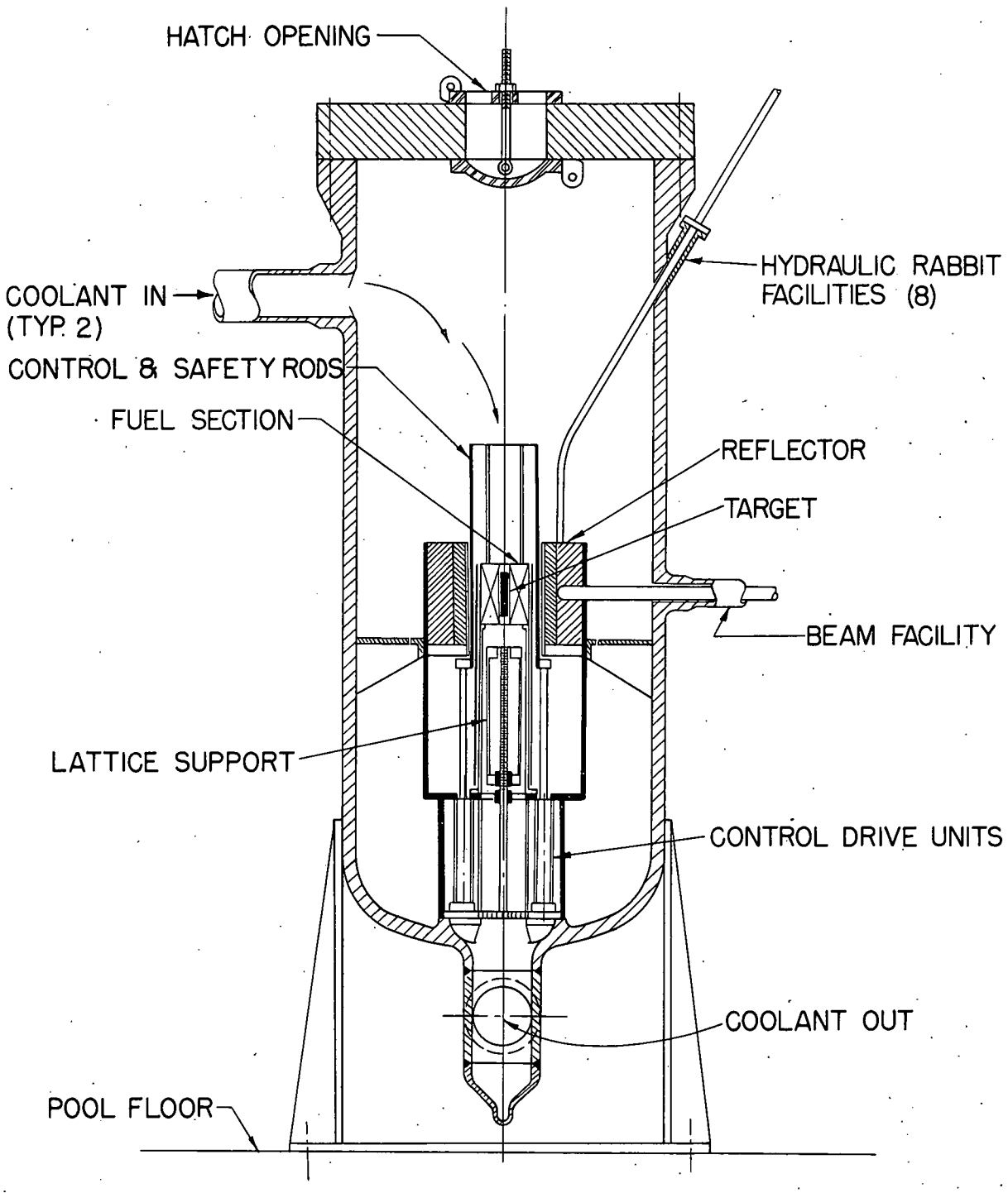

HFIR REACTOR ASSEMBLY
VERTICAL SECTION

0 1 2 3 4
SCALE-FEET

Fig. 2.

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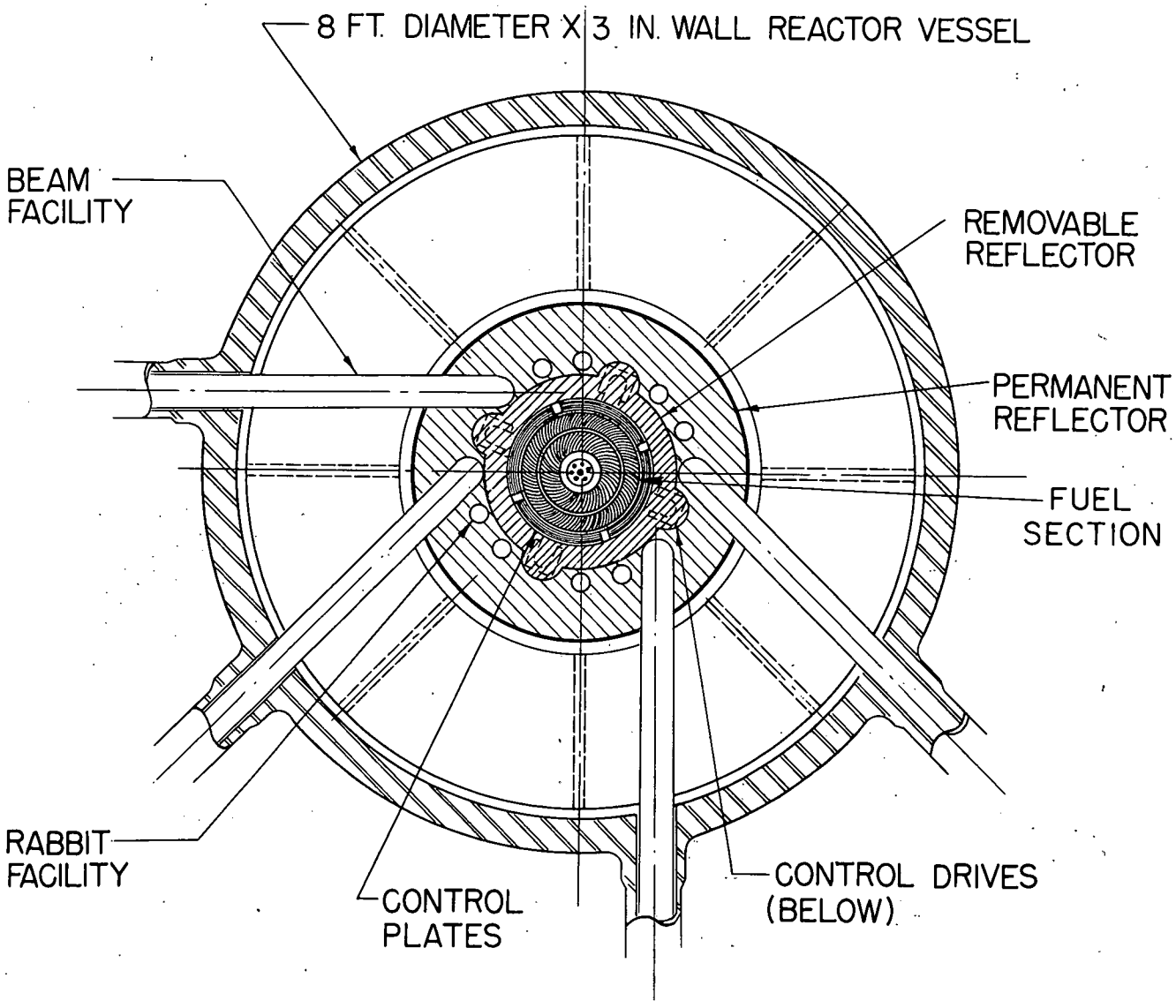

HFIR REACTOR ASSEMBLY
HORIZONTAL SECTION AT ℄

0 1 2
SCALE-FEET

Fig. 3.

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ORNL-LR-DWG 46129

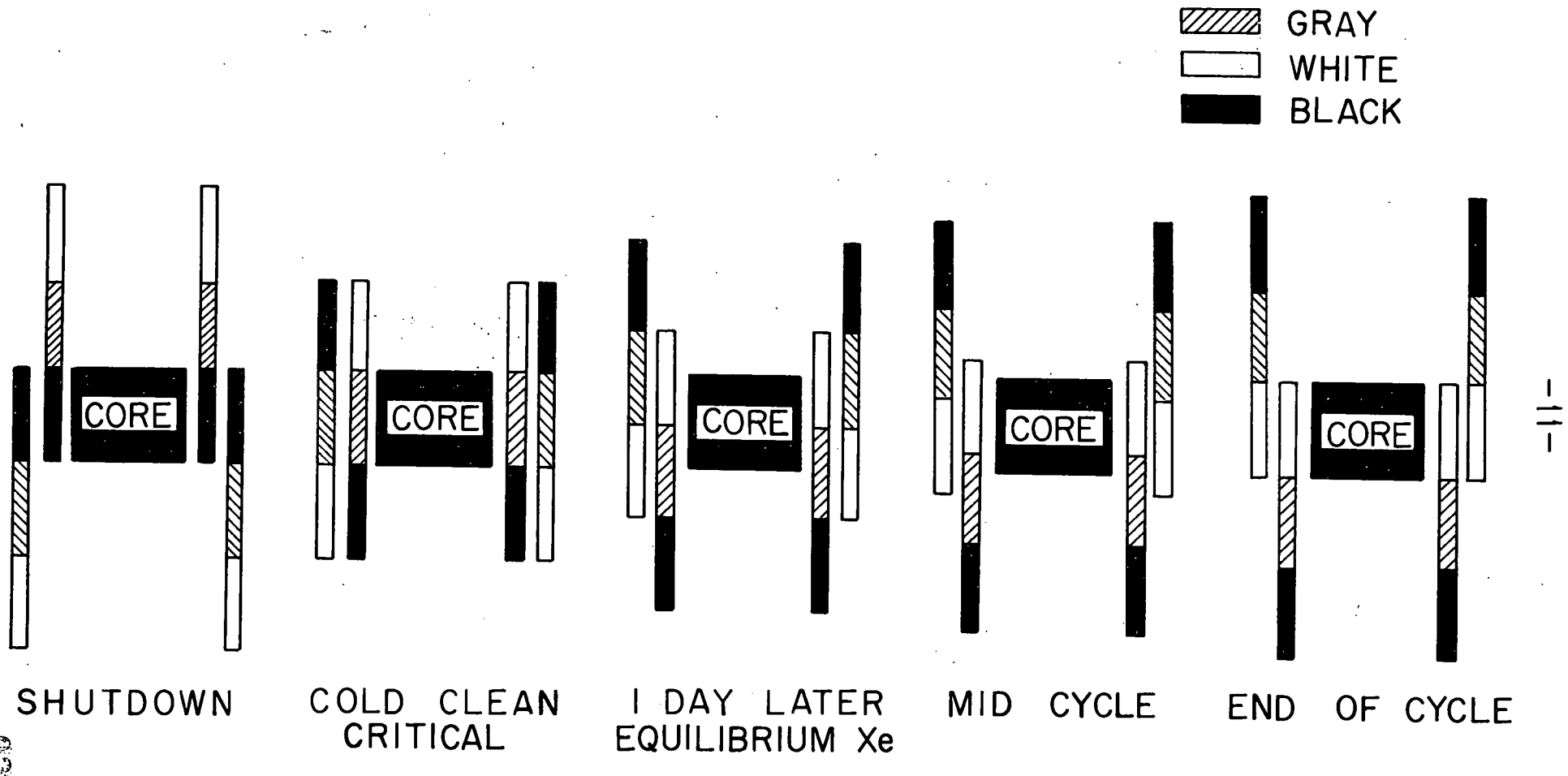

Fig. 4.

HFIR CONTROL PLATES

fabricated of similar materials but reversed in position, the black section being at the top. In operation the rings would move vertically in unison except in opposite directions. Under shutdown conditions the inner ring would be in a raised position and the outer ring in a low position. During the early part of an operating cycle the plates present a medium cross section material to the reactor and near the end of the cycle present a low cross section material.

In an emergency, or when it is desired to shut down the reactor, the outer ring is released and falls due to the action of gravity, spring force, and/or hydraulic load to add poison to the core. For added reliability, the outer ring is divided into segments with individual release mechanisms.

All plates will be accurately located and guided with bearings to provide the required coolant gaps during normal operation and unhindered travel during scram operations.

A number of hydraulic, magnetic and mechanical methods are presently being investigated to provide a reliable system of rod engagement, rod release and position control.

Reactor Vessel. - The reactor vessel is nominally 8 ft in diameter and approximately 20 ft high. A wall thickness of approximately 3 in. will be required for the 1000 psi at 200°F. The material of construction will be stainless steel or stainless-clad steel.

In addition to the two inlet water line connections in the upper part of the vessel, a number of side access holes will be required for hydraulic rabbit tube connections, beam tube connections, and control and instrumentation requirements. A flat head with a replaceable gasket is contemplated. A quick opening hatch in the cover plate will be provided for refueling and servicing.

2.3 Reactor Physics

The preliminary physics studies were conducted to determine a suitable combination of dimensions and materials that would result in satisfactory values of neutron flux and heat flux. In all cases the neutron fluxes of interest were the maximum unperturbed flux in the water island, ϕ_m, and the volume averaged flux in the homogenized experiment, ϕ_E, located in the center of the water island. The subscript m will be used to designate maximum in the following.

In order to include the irradiation sample for producing transplutonium elements in the reactor study, it was necessary to know how production, cross section, and heat generation varied with neutron flux, time, and type of feed material. On the basis of studies made by Claiborne,[6][3] Ergen,[7] and

[6] H. C. Claiborne, Californium Production in the High Flux Isotope Reactor, ORNL CF-59-8-125 (Aug. 31, 1959); also H. C. Claiborne, Effect of Different Sets of Cross-Sections on Cf-252 Production in the HFIR, ORNL CF-59-10-19 (Oct. 8, 1959).

[7] W. K. Ergen, Transplutonium Elements, ORNL CF-59-11-109-Del. (Jan. 11, 1959).

Chetham-Strode[2] it was concluded that heavy isotope irradiation experiments could be adequately represented by considering only thermal irradiation of Pu^{242}, having Cf^{252} as a final product. The experiment average cross section used in the reactor calculations was averaged over a one-year irradiation period at constant flux, in which case the cross section was somewhat dependent on flux level, or over a particular value of the product of flux and time, in which case the cross section was not dependent on the flux level.

Parameter or Optimization Studies. - The first part of the parameter study was concerned with determining the optimum diameter of the water island with and without the central experiment present. The results indicate that the optimum island diameter tends to increase with experiment weight but is essentially independent of other core dimensions. Since the unperturbed flux in the island was also of interest, the effect of the experiment on the optimum island diameter was effectively neglected in selecting the 5.5 in. island diameter. This size water island should result in a maximum value of $\phi_m/(P/V)$. P is the reactor power and V is the volume of the fueled region. The dimension selected agreed very well with the experimental values determined by Feinberg et al.[8]

In the process of arriving at the optimum island diameter consideration was given to the use of a beryllium island in the center of the water island for the purpose of allowing a larger diameter fuel annulus without sacrificing flux. The results indicated that beryllium could not be used without a significant reduction in both ϕ/P and $\phi/(P/V)$. For this reason beryllium was not included in the water island in further studies.

Once the island diameter and composition had been established, the parameter study was continued to determine the best combination of core length and fuel annulus thickness. This study was made with no experiment in the island, a 100 g experiment, and a 200 g experiment in the island. In every case the optimum diameter of the experiment was determined and used in the final determination of the optimum core length and corresponding fuel annulus thickness. The results indicated that increasing the weight of the experiment tends to increase the length of the core. However, over the range of zero to 200 g Pu^{242} feed material the curve is quite flat, the optimum core length covering the range of about 18 to 24 in. A length of 18 in. was selected with a corresponding fuel annulus thickness of 4.7 in.

In the above calculations the fuel distribution was uniform throughout the fuel annulus, resulting in a ratio of maximum-to-average power density in the radial direction equal to about 2. To investigate the effect of varying this ratio the fuel annulus was divided into several discrete, radial regions of fuel. The results indicated that a perfectly flat radial power distribution would decrease ϕ/P by only about 8% but would increase $\phi/(P/V)_m$ by about 90%. Therefore, a contoured fuel distribution in the radial direction was specified for the HFIR.

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(8) S. M. Feinberg et al., "An Intermediate Reactor for Obtaining High Intensity Neutron Fluxes," Proceedings of the International Uses of Atomic Energy, Geneva, 1958, Vol. 10, 296, P/2142.

Most of the parameter study calculations were made with no control surfaces in the core but with sufficient fuel to give a k_{eff} equal to about 1.2. Introducing gray control surfaces between the fuel annulus and the beryllium reflector increases the ratio of maximum-to-average power density in the radial direction and thus increases ϕ/P to some extent. However, the presence of the poison also hardens the neutron energy spectrum sufficiently to result in a small increase in $\phi/(P/V)_m$. In other words, the increased leakage to the island, resulting from spectrum hardening, more than offsets the increase in the ratio of maximum-to-average power density.

As mentioned previously, the experiment that is to be placed in the water island has an optimum diameter, a particular value depending on the weight of feed material involved. Since the production of Cf^{252} depends on the average flux in the feed material and on the total weight of feed material, and since the average flux is also dependent on the weight of the feed material, it stands to reason that there must also be an optimum weight of feed material. Using the earliest available Cf^{252} production curves[2] and assuming only thermal irradiation of Pu^{242}, the optimum weight of Pu^{242} was estimated to be about 200 g. The more recent Cf^{252} production curves[6] result in an optimum weight closer to 300 g of Pu^{242}. The maximum heat generation rates estimated for the two quantities are 540 and 740 kw, respectively.

Most of the parameter study calculations were made for a metal-to-water ratio of 1 in the fuel annulus because high ratios tend to increase neutron leakage. The particular value was also consistent with heat transfer requirements. However, as studies were made on fuel element design, it appeared desirable to increase the coolant channel thickness, making the metal-to-water ratio equal to 0.8. Reactor calculations indicated that the lower ratio would reduce ϕ/P by about 3%. A firm decision on the fuel plate thickness has not yet been made and therefore the metal-to-water ratio is not yet firmly established.

Reactivity Control Studies. - Once the parameter studies had been completed to a point where it was possible to select a typical HFIR core, effort was concentrated on the selection and analysis of a control system.

The requirements for the control of reactivity, in addition to that of being able to adequately control the k_{eff} of the reactor, were that the control surfaces should not, at the beginning of the fuel cycle, result in an excessive value of the maximum power density, and during the fuel cycle the maximum power density should not change too much. These requirements, of course, suggested the use of symmetrical control surfaces as opposed to conventional rods that are withdrawn from only one end of the core, and also suggested that as little excess reactivity as possible be controlled. Since the amount of fuel in the core is determined by the desired fuel cycle time, the use of burnable poison was proposed for partial control of the excess reactivity.

The various symmetrical control systems considered and studied at some length for the HFIR were the soluble poison systems, split core systems, movable side reflector systems and a vertical mechanical reflector control system. All but the latter system were rejected because of several outstand-

ing disadvantages. The use of a vertical mechanical reflector control system was suggested early; however, its real advantage from the standpoint of power distribution control did not become apparent until a detailed two-dimensional study had been made of the power distribution in the core, using a gray control region between the fuel annulus and the beryllium side reflector. The proposed system consists of two thin (about 1 cm each) cylinders that are concentric with the fuel annulus and separate the fuel annulus from the beryllium outer reflector. The cylinders move vertically and in opposite directions so as to maintain symmetry about the longitudinal center line and the horizontal midplane. The cylinders consist of three equal-length, longitudinal regions, one end region being black for safety control, the middle gray for shim and regulation and the other end essentially white for displacing water. The regions tentatively are about one core length long.

This system resulted in essentially constant power density during an entire fuel cycle.

Miscellaneous items studied in connection with reactor control were void coefficients of reactivity, temperature coefficients, prompt neutron lifetime, and xenon instability.

The temperature coefficient of reactivity in the water island without an experiment present is positive from 70°F to some temperature well above the normal operating temperature of about 150°F. Thus, a void coefficient for uniform voids over the island is also positive for small void fractions and is estimated to pass through zero at about 25% voids with a total positive change in reactivity equal to 0.3%. A cylindrical void in the center of the island has been calculated to have a negative effect on reactivity for any diameter void.

The temperature coefficient of reactivity for the fuel annulus has been calculated to be negative (about -1×10^{-4} $\Delta K/K/°F$) and very nearly equal but opposite in sign to the island coefficient. It has not yet been determined whether the over-all coefficient is positive or negative.

The prompt neutron lifetime has been calculated for the beginning of the fuel cycle, when the shim control plates are fully inserted, and for the end of the fuel cycle, when the shim plates are completely withdrawn. The values obtained were 50 and 100 μ sec, respectively.

The problem of xenon instability was investigated by both digital and analogue computer techniques. Step changes in reactivity up to about 2% were introduced in the form of local perturbations without sustaining oscillations in power distribution.

Fuel-Cycle-Time Calculations. - Early in the parameter studies a 10-day cycle was selected on the basis of the desired maximum thermal neutron flux in the island. Increasing the fuel concentration to increase the core life required more burnable poisons and more control surface. The final result was a decrease in island thermal neutron flux.

Fuel burnup calculations were made to determine the variation in maximum power density and neutron fluxes with time, to determine the position of the control plates at different times, to determine a satisfactory initial distribution of fuel and burnable poison, and, of course, to determine the necessary amount of fuel for a 10-day cycle. It was assumed that the core would be operated at a constant power of 100 Mw.

Once the proper fuel and burnable poison distributions had been established, it was found that the maximum power density at any time during the cycle was less than the allowable and remained nearly constant during the cycle. The maximum thermal neutron flux in the water island varied by only 2% during the cycle.

A schematic representation of the HFIR showing typical flux distributions is shown in Figs. 5 and 6. Pertinent characteristics of the core are listed in Section 7.

2.4 Heat Transfer and Cooling

Core. - The heat transfer characteristics were derived on the basis of experimental work done by Gambill[9] in connection with the determination of friction factors, film coefficients, and burnout heat fluxes for the thin HFIR coolant channels; and by Griess[10] in connection with the determination of a significant temperature drop across the aluminum oxide film. Hilvety,[11] using the correlations proposed by Gambill for the film coefficient and burnout heat flux, made a study of the hot spot and hot channel factors for the HFIR and then calculated the maximum temperatures and heat fluxes for the proposed HFIR as presented herein.

The correlation for heat transfer coefficient employed is that proposed by Levy.[12] Recent experimental data[9] indicate that this correlation predicts safe minimum values of heat transfer coefficient under HFIR operating conditions. Calculations of minimum burnout heat flux have been based on the Zenkevich-Subbotin correlation.[13]

Based on the core design with 40-mil plates and 50-mil water gap and considering all design tolerances and engineering and calculational uncertainties, the maximum fuel plate surface (oxide-H_2O surface) temperature is estimated to

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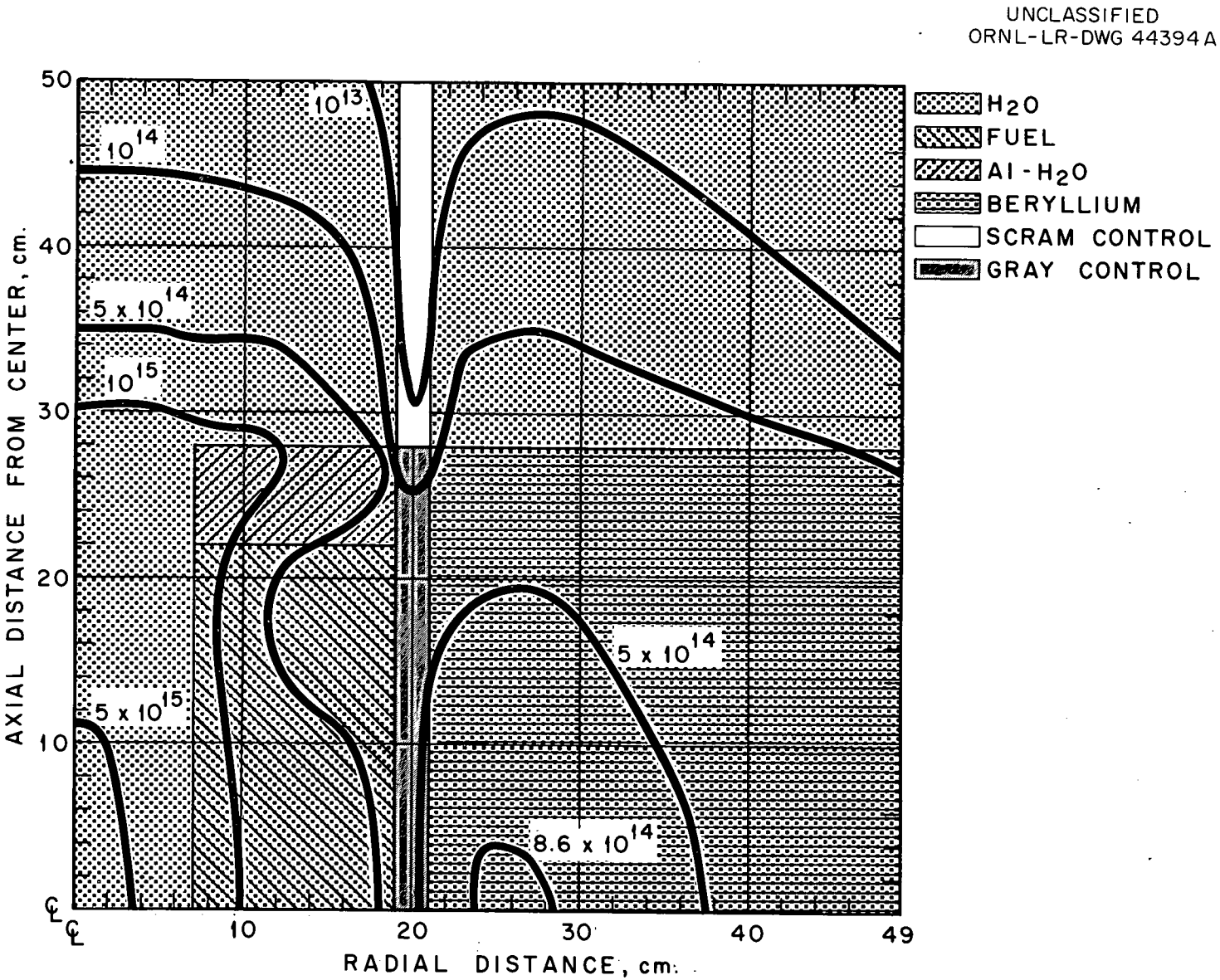
[9] W. R. Gambill, personal communication (Feb. 1960).

[10] J. C. Griess, personal communication (Feb. 1960).

[11] N. Hilvety, "Preliminary Hot Spot Analysis of the HFIR", to be issued.

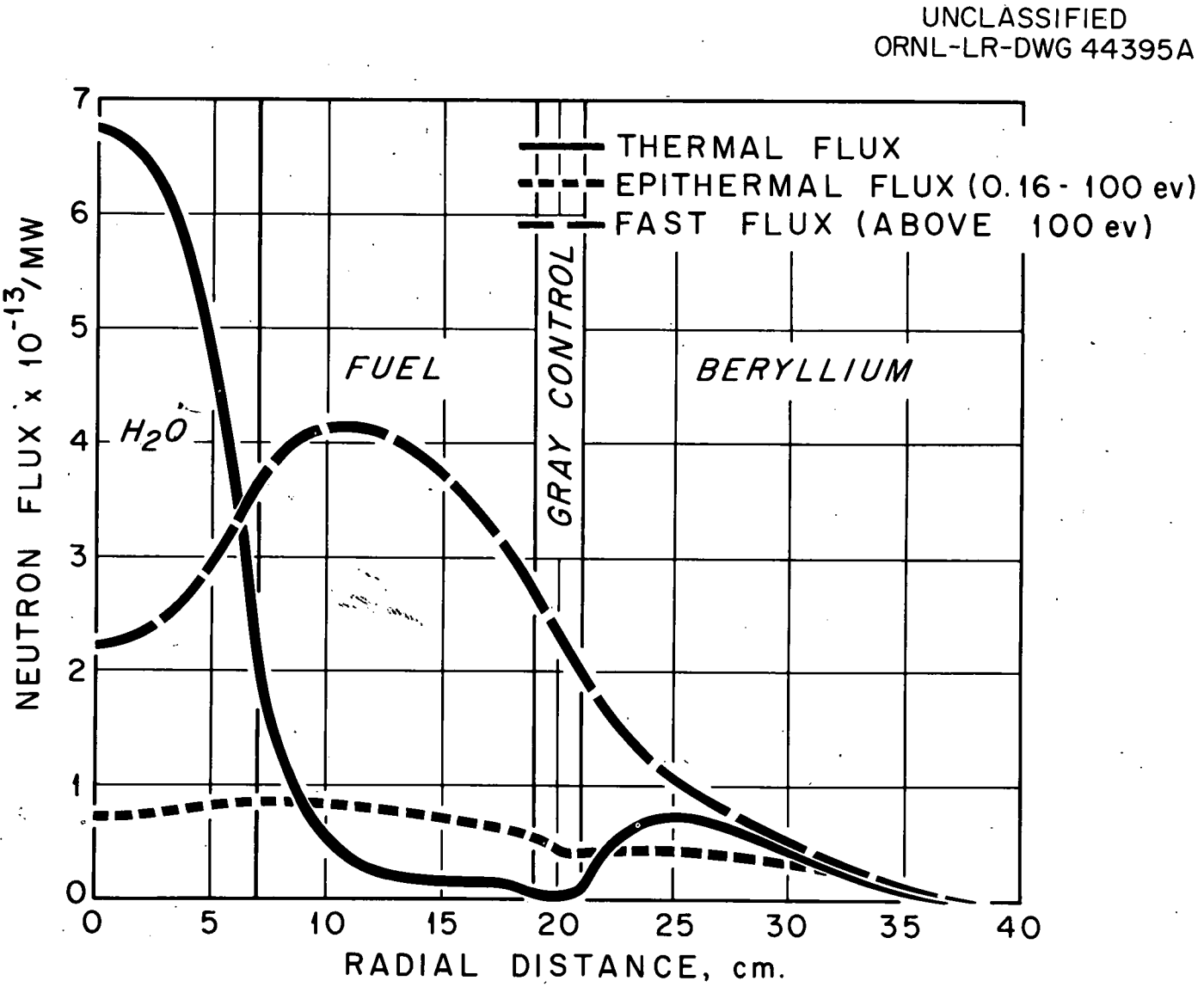
[12] S. Levy, R. A. Fuller, and R. O. Niemi, "Heat Transfer to Water in Thin Rectangular Channels," Am. Soc. Mech. Engrs., Paper No. 59-A-127 (1958).

[13] B. A. Zenkevich and B. I. Subbotin, Atomnaya Energiya 3, 149 (1957); English translation in J. of Nuclear Energy, Part B, 1, No. 2, 130 (1959).



VERTICAL CROSS SECTION OF THE HFIR CYLINDRICAL CORE SHOWING TYPICAL THERMAL FLUX DISTRIBUTION (100 MW)

Fig. 5.



RADIAL FLUX DISTRIBUTION AT HORIZONTAL MIDPLANE OF CLEAN HFIR CORE

Fig. 6.

be 362°F with a maximum hot spot heat flux of 1.4×10^6 Btu/hr·ft^2. The minimum burnout heat flux for this reactor geometry based on an increase in power level is calculated to be a factor of 2.5 above the maximum operating heat flux. The minimum burnout heat flux for localized spots such as might be brought on by fuel segregation over and above the design tolerance is estimated to be a factor of 3.6 above the maximum operating heat flux. The calculated average heat flux is 7.7×10^5 Btu/hr·ft^2.

Reflector. - The reflector is cooled by a portion of the primary coolant water flowing downward in parallel with the fuel region and flux trap coolant. The total available pressure drop across the reflector is thus the same as that of the fuel region. Flow in the reflector region will be controlled by orificing the reflector support grid if necessary. It is expected that about 7% of the primary coolant water will flow through the reflector region. Adequate cooling of the beryllium reflector will require that this region consist of about 5% by volume coolant water.

Irradiation Sample. - The present design for the HFIR target array consists of a 19-rod cluster of 3/8-in. OD aluminum tubes. Each tube is filled with tightly fitting pellets made from a mixture of PuO_2 and aluminum powder. These rods, which are suspended vertically in the central island, are cooled by 1000 gpm of water flowing through the island; the water velocity produced at the sample is approximately 30 fps. At the maximum heat generation rate of 37 kw in each rod, the maximum heat flux at the rod surface will be approximately 1.1×10^6 Btu/hr·ft^2. At this heat flux, coolant velocity of 30 fps, and a coolant inlet temperature of 120°F, the rod surface temperature will be about 290°F. The pellet temperature is not expected to exceed 600°F.

Heat After Shutdown. - Calculation of the heat release following shutdown of the reactor combined with the results of experiments on natural convection cooling of the reactor core indicates that provision must be made to maintain a small ($\sim$ 10%) coolant flow for a maximum time now estimated to be about five hours following reactor shutdown.

The heat capacity of the coolant system is sufficiently large that even if no heat were lost from the system the average water temperature rise would be only about 50°F three hours after shutdown. As a practical matter, the heat loss from the reactor tank and coolant lines to the pool water is estimated to equal the heat release from the core under these conditions after about one-half hour, and as the heat capacity of the reactor pool is approximately 5 times that of the reactor primary coolant loop the total rise in average temperature should only be about 10°F.

2.5 Instrumentation and Control System

The main instrumentation problems appear at this time to be those concerned with providing assurance that adequate cooling is present for the reactor core and experimental facilities. This concern is a natural consequence

of the fact that the heat transfer rate has been pushed very high in order to minimize the reactor power level needed to achieve the desired flux. It is not anticipated that a great deal of development work will be required for instrumentation of this type; the project will require, however, a considerable amount of careful engineering of the instrumentation for process water control.

Nuclear instrumentation does not appear to require a design effort much greater than that normally expended on a reactor of the MTR-ORR type. It does appear that somewhat more effort must be expended during the design to assure that the gross average steady state power and the integrated power input resulting from any excursion will be limited to a value which will not result in core damage. The limitation arises from the fact that the high power density does not allow as much margin for errors in power as is usual in most of the MTR-ORR-ETR class of reactors. It is also considered highly desirable to investigate the application of modern techniques in electronics, such as solid state devices (transistors) in order to reduce maintenance problems and to give added assurance against unnecessary shutdowns.

An additional complication brought on by the extremely high power densities and the small size of the reactor core is the problem of control rod location. This problem resolves itself into one of obtaining sufficient control of k in the machine and at the same time not unduly perturbing the flux distribution in the fuel region. The approach to this problem is to use segments of concentric cylinders located between the core and the reflector as described in Section 2.2.

Nuclear calculations have shown the possibility of a positive temperature coefficient of reactivity associated with the temperature of the water in the flux trap. At the present time this does not appear to be serious inasmuch as the calculated coefficient is small and the response to changes in power is slow. The temperature coefficient of the fuel region is negative and therefore for rapid changes in power the behavior will be determined by the negative coefficient in this region. Work is continuing on this problem and it is planned to measure these effects in a critical experiment which will be a mock-up of the reactor from a nuclear standpoint. The results of the calculations and experimental measurements will be analyzed by means of the ORNL Reactor Control Department analogue computer in order to observe the effects of the reactor constants during normal and abnormal operation.

2.6 Cooling Systems

Four separate water systems will be utilized to remove the heat generated in the HFIR and associated experimental facilities. These four systems are:

(1) The reactor primary coolant system.

(2) The reactor pool coolant system.

(3) The auxiliary pool coolant system.

(4) The secondary coolant system.

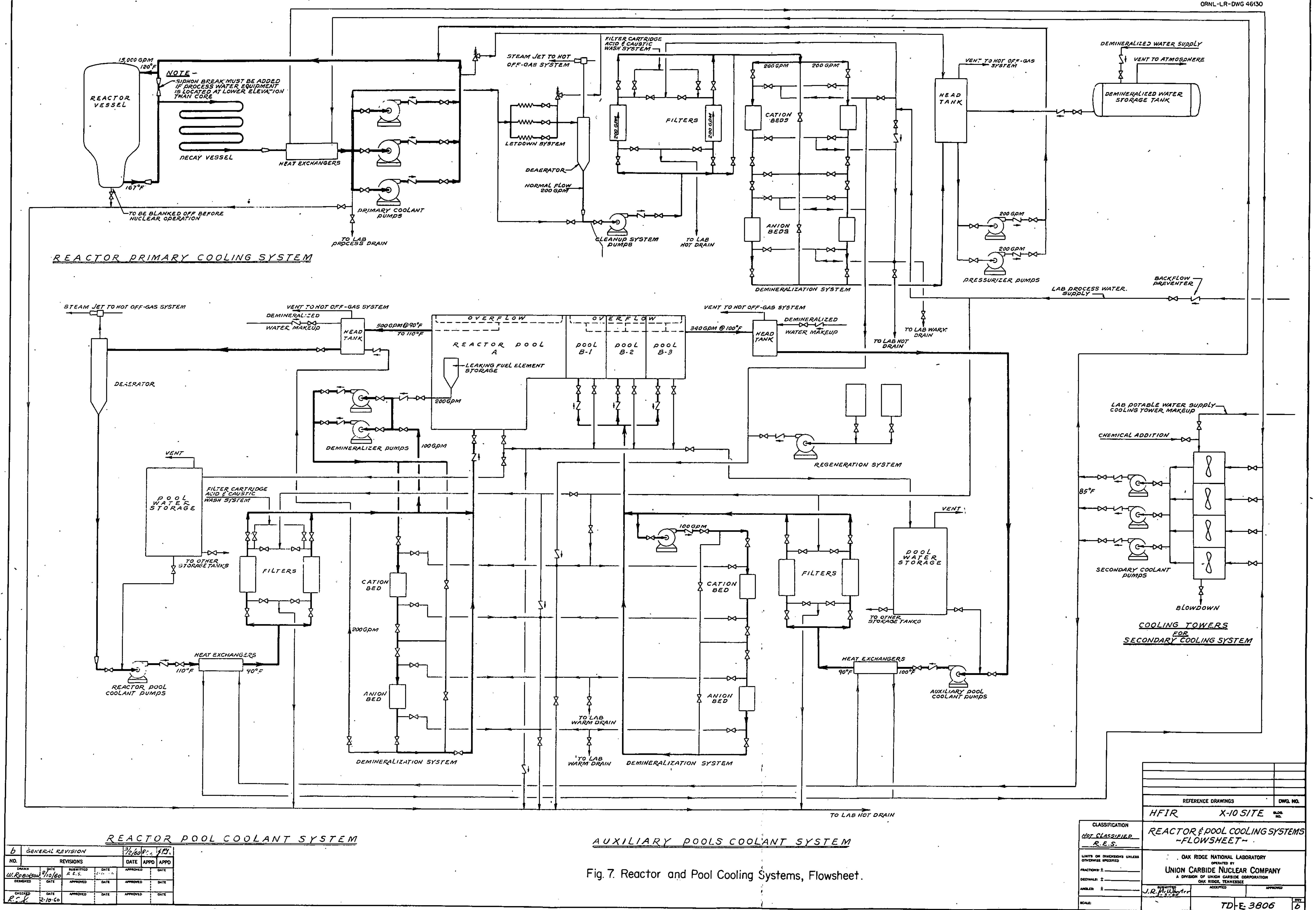


Fig. 7. Reactor and Pool Cooling Systems, Flowsheet.

Heat absorbed by the first three systems will be transferred to the secondary system by shell and tube heat exchangers and then dissipated in the atmosphere through the secondary system cooling towers. Fig. 7 is the flowsheet for the cooling systems and the pertinent design data are presented in Section 7.

Reactor Primary Coolant System

General Description. - In the primary coolant system approximately 15,000 gpm of H_2O will be circulated to permit operation at the design power level of 100 Mw. The primary coolant loop is to be designed for operation at 1000 psig although it is probable that normal operation can be at lower pressure. The estimated reactor core pressure drop is about 57 psi. The system flow path is from the reactor vessel outlet through a decay vessel, to the heat exchangers, thence to the circulating pumps and back to the reactor vessel. A by-pass cleanup stream is to be taken from the pump suction, let down to a low-pressure filter and demineralizer cleanup system, and then fed back to the main stream on the return line to the reactor vessel by the pressurizer pump. The pressurizer pump and the letdown system combine to act as the prime pressure controller for the primary system. All components in contact with the primary water are to be constructed of stainless steel with the possible exception of the low-pressure piping which may be aluminum if more economical. A storage tank of sufficient capacity to fill the primary system is included in the design.

Decay Vessel. - At present the inclusion of a decay vessel between the reactor and the heat exchangers is planned. This vessel would provide the necessary holdup time between the reactor core and the heat exchangers for decay of N^{16}.

Primary Heat Exchangers. - The heat exchangers are designed to transfer the full power heat load of 350×10^6 Btu/hr from the primary water to the secondary system with one exchanger out of service. The primary system flow of 15,000 gpm will enter the exchangers at 167°F.

Primary Coolant Pumps. - Two 7500 gpm centrifugal pumps will provide the coolant flow for the reactor at design power. In addition a third identical pump will be installed as standby. Emergency shutdown flow will be provided by auxiliary D.C. motors directly connected to the main pumps and floating on the line during normal operation. Upon loss of normal power supply the D.C. motor will assume the load corresponding to approximately 10% of rated pump flow automatically upon coast-down of the pump. Power for the emergency drives will be supplied by one or more battery banks.

Primary Cleanup System. - The cleanup system will operate at essentially atmospheric pressure, thus permitting conventional demineralizer operation. The letdown stream will be de-gassed in a barometric type deaerator which will provide the suction head for the cleanup system circulating pumps. The system will include two full capacity filters in parallel. The demineralizer unit will contain two 200 gpm separate bed arrangements in parallel. The entire system will be designed for a normal by-pass stream of at least 200 gpm; under special conditions such as cleanup of a contaminated system a much larger flow

rate can be obtained for quick cleanup. The cleanup stream will flow to a head tank where makeup will be added as necessary. The pressurizer pumps will transfer the water to the high pressure system from this head tank.

Pool Coolant Systems

Reactor Pool Coolant System. - The reactor pool coolant system is designed to maintain the temperature of the reactor pool water between 90 and 110°F. The water will overflow the weirs at the upper perimeter of the pool, flow to a head tank where makeup demineralized water is added, and then to a barometric deaerator. The suction head for the circulating pumps will be provided by the deaerator barometric leg. The main stream of 500 gpm will then flow through the heat exchangers where heat will be transferred from the pool water to the secondary system. The cooled water will then pass through filters and back to the pool. A by-pass clean-up stream of 100 gpm will be taken from the main stream following the filters and passed through a separate bed demineralizer. A direct pass through the demineralizer may be made in the event of a leaking fuel element, by placing the fuel element in an especially designed rack in the pool where a flow of water can be drawn over the assembly for cooling, passed through the demineralizer and then introduced to the main stream at the head tank. With the exception of the demineralizer which will be constructed of stainless steel, all components in contact with the pool water will be constructed of either stainless steel or aluminum, whichever proves to be more economical.

Auxiliary Pool Coolant System. - The auxiliary pool coolant system is designed to maintain the temperature of the auxiliary pool between 90 and 100°F. The design flow of 340 gpm will pass over the weir arrangement at the top of the pools; flow to a head tank where make-up demineralized water can be added and then to the circulating pumps. The heat exchangers are designed to transfer the heat load from the pool water to the secondary system. The main stream will pass through the heat exchangers, then filters, and back to the pools. A by-pass stream of 100 gpm will be taken from the main stream and passed through a separate bed demineralizer unit. Material selection will be based on the same economic evaluation as that for the reactor pool equipment.

Storage Tanks. - Provision will be made for storage of reactor pool and auxiliary pool water. The capacity of the tanks for storage of reactor pool water will be sufficient for draining the pool to the top head of the reactor.

Secondary Coolant System

Water at essentially atmospheric pressure is to be used as the coolant in the secondary system. This water will be circulated through the heat exchangers and over the cooling tower. The cooling tower will be designed to cool the circulating water to 85°F off tower at an atmospheric wet bulb temperature of 77°F and at 125% of the summation of all component secondary side design flows and heat loads.

Makeup water will be laboratory potable water and will enter the system along with chemical treatment solutions at the cooling tower basin.

The design load of the cooling tower is such that full reactor load can be carried while cooling tower basin compartments are being cleaned. In addition, at full reactor load there will be one spare circulating pump for the secondary system.

The basic material of construction will be carbon steel and cast iron.

2.7 Experimental Facilities

This reactor is primarily a facility for heavy element production; therefore the central flux-trap region is considered to be the major experimental facility. In addition to the central facility, however, space in the reflector will be useful for the irradiation of small capsules and samples at relatively high thermal neutron fluxes. The four small-diameter beam tubes terminating in the outer reflector region will provide relatively high beam intensities for basic research investigations requiring collimated beams. It is believed that these secondary facilities will increase the over-all utility of the reactor without significantly compromising the primary objective.

In the present stage of design, the planned sample configuration for the flux trap is that of an assembly of small cylinders with proper coolant channels and metal-to-water ratios to achieve the maximum flux averaged over its volume. Insertion and removal of the sample will be accomplished during shutdown by utilizing the shielding provided by the water in the pool. Present indications are that the samples will generally be in the reactor for long-term irradiation, that is, a year or more. It will be possible, however, to remove the sample easily and transfer it to a hot cell area for inspection for possible corrosion, erosion, or other damage at much shorter intervals.

Eight irradiation facilities are presently proposed for the outer reflector region. These facilities are vertical cylindrical holes penetrating the beryllium reflector, each having a nominal diameter of $1\frac{1}{2}$ in. The present plan is to equip each of the holes with a re-entrant-tube hydraulic rabbit installation which will allow for the insertion and removal of samples during operation. The hydraulic rabbit tubes will pass through the cover plate or side walls of the reactor vessel near the top and extend along the walls of the canal sufficiently below water level to provide the necessary shielding. Terminal stations are located in the canal where loading and unloading operations can be done easily. As the detailed design of the facility proceeds, it is anticipated that additional capsule and sample irradiation facilities may be located in regions farther removed from the core. Provision will be made for inserting and removing holders in these locations with special tools in a manner similar to that employed for handling the fuel and the central irradiation samples.

The four beam holes are arranged as shown in Figs. 3 and 8. The present plan is to have beam tube liners which extend from flanged faces at the inside surface of the cubicles to a region separated from the fuel by about 3 in. of beryllium reflector. Outer sleeves of stainless steel which extend from nozzles on the pressure vessel to flanges in the cubicles allow for replace-

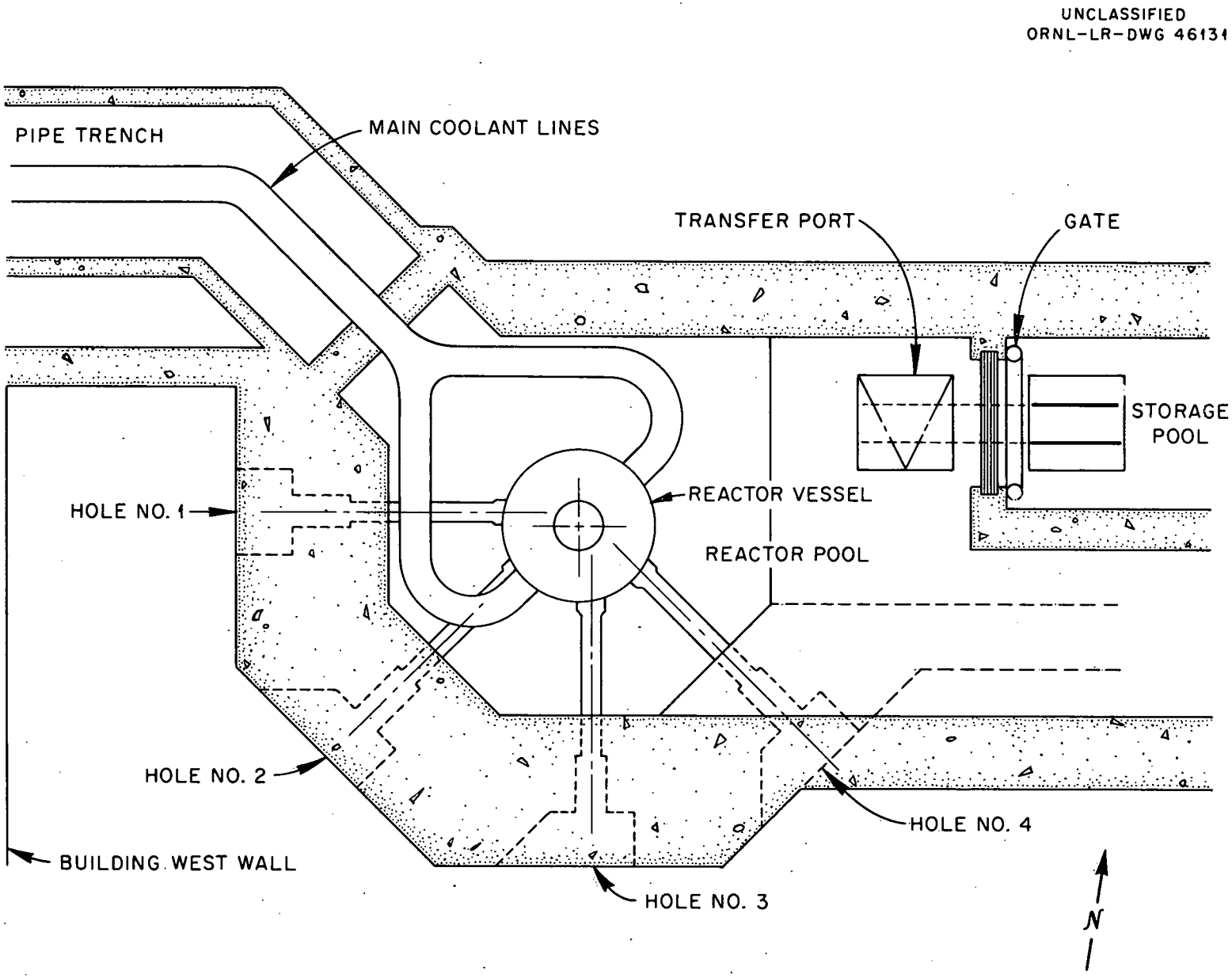


Fig. 8. HFIR—Reactor Structure—Horizontal Section

ment of the beam tube liners. Material selection for these liners has not been made although Al, Be and Zr are being considered. The beam tubes are to be located on the horizontal center-line plane of the reactor core and will be approximately 42 in. above floor level.

The beam hole liners are positioned in the beryllium reflector so that if the liners were extended they would be approximately tangent to the outer surface of the fuel region. This arrangement will minimize the number of uncollided neutrons and high-energy gamma rays in the beam emerging from the beam tube without significantly reducing the lower energy neutron beam. This approach was arrived at through consideration of the types of experiments which might be performed and the desired neutron spectrum in the beams for these experiments and will minimize the shielding problem which is usually encountered in stopping the high-energy neutrons and gamma rays in the emergent beam of experiments which may be typified by neutron diffraction set-ups.

The reactor shield will be arranged so that each beam hole emerges perpendicular to a shield face about 13 ft wide, thus providing adequate space for experimental equipment and shielding. A minimum of 25 ft will be provided between the face of the biological shield and the building wall. The beam hole shown to be pointed west may be extended beyond the building wall for several hundred meters without interference from existing laboratory installations.

The floor area will be designed for exceptionally heavy floor loadings, as found necessary in previous beam hole experimental equipment design. Since the facility is located on a hillside, the beam hole floor level is also at grade level, although it is below the operating floor level.

2.8 Buildings

The space within the reactor and canal buildings will be subdivided into four separate areas: (1) the reactor bay; (2) the beam room; (3) the canal bay; and (4) the control room and office area. Figs. 9, 10 and 11 show a preliminary version of the proposed plan.

The building walls, roof, partitions, doors, truck entrances, etc., will be designed for a better than average degree of air tightness in order to make provision for containment as will be described in a later section. The intent is to make each of the four areas essentially independent in regard to the spread or release of contamination.

Each of the four main areas of the building will have individual air handling equipment without interchange of air. Appropriate provision will be made to handle any release of activity which might occur in any of the areas except the control room and office area which will be designed as a region outside of the contained area.

The building structures are to be mill type, without windows, of the

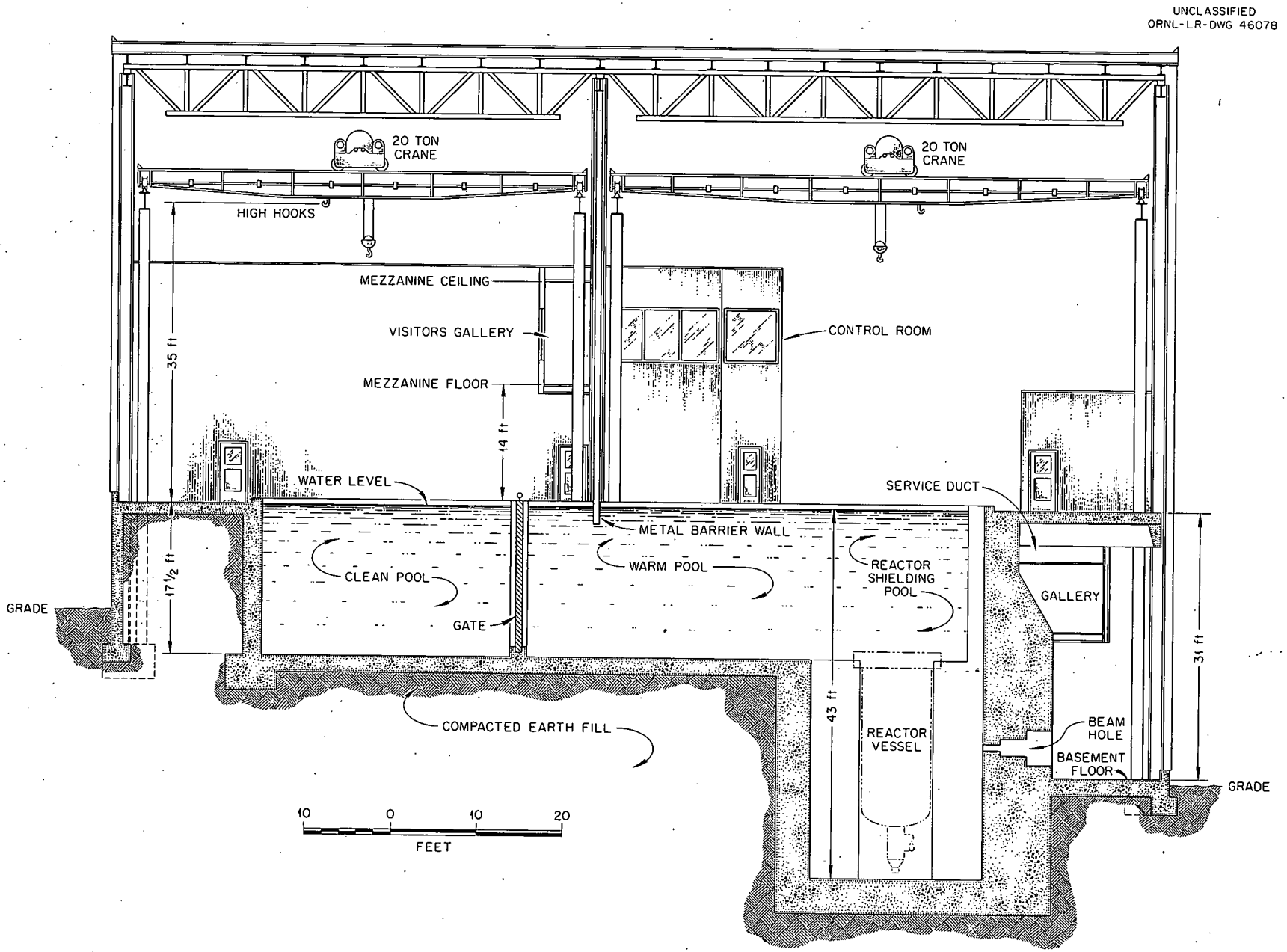


Fig. 9. Vertical Section Through Building and Structure.

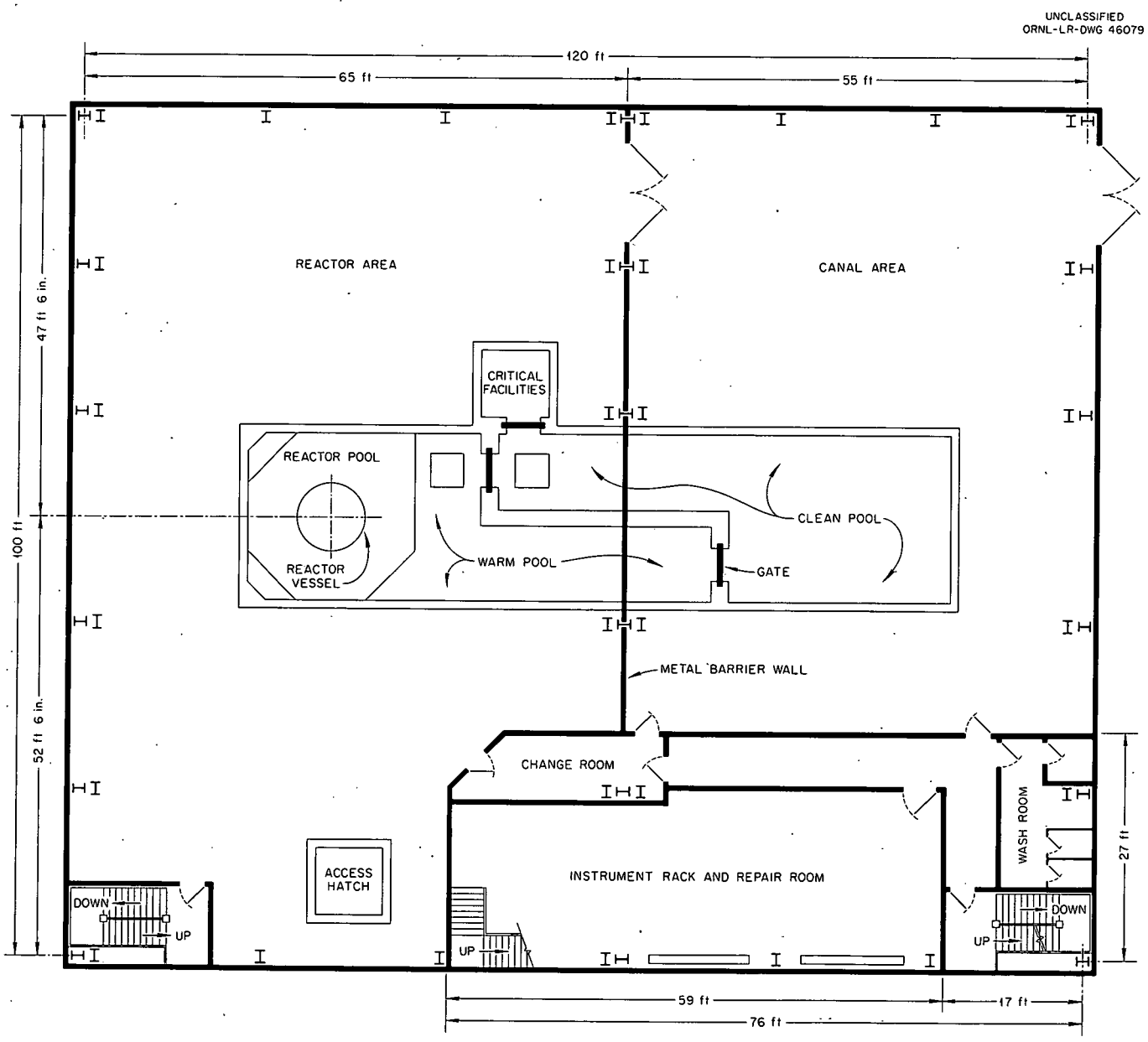


Fig. 10. Main Floor Plan.

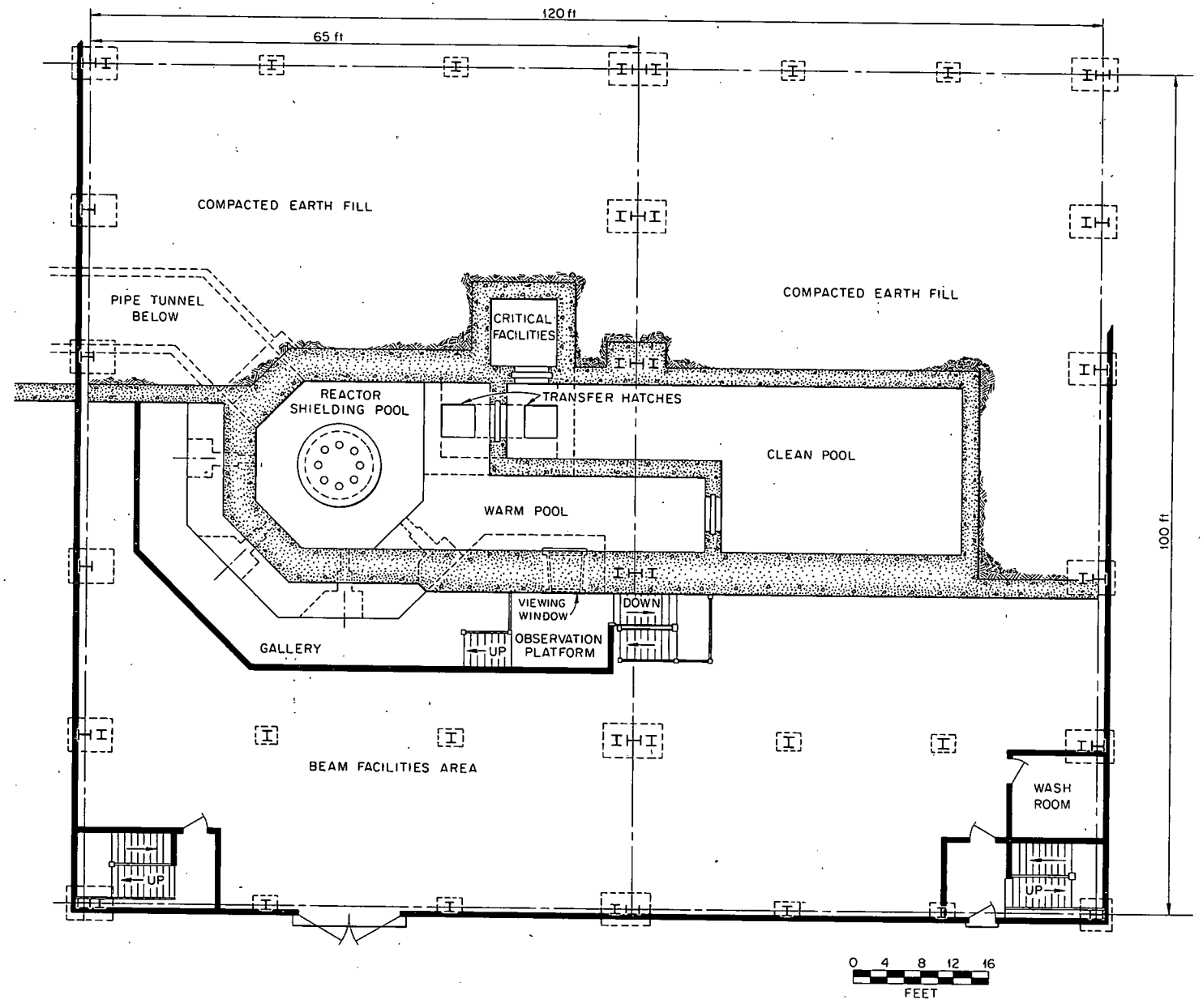


Fig. 11. Basement Floor Plan.

most economical type within the requirements of the Southern Building Code, construction below grade and to the start of the siding will be reinforced concrete. Floors are to be concrete slab throughout, with high floor load requirements as are necessary for the handling of heavy equipment, shielding, etc.

A hillside site is preferred as the most economical location in order to provide for truck access to both the reactor operations and canal area at the upper level and to the beam facility area on the lower level. These levels have an elevation difference of approximately 31 feet.

Reactor Operations Area. - The reactor operations area as shown in Figs. 9 and 10 is that portion of the construction at the upper level which encompasses the reactor pool structure. The floor area including the pool is approximately 66 ft by 100 ft. The height of the building will be determined by the crane and roof truss design to provide a minimum crane hook clearance from the operating floor of 35 ft.

A 30-ton crane with a much smaller auxiliary hook will be provided for full length and width travel of the room.

Beam Room. - See Figs. 9 and 11. The floor level of the beam room is located $3\frac{1}{2}$ ft below the center line of the reactor. This is considered a convenient distance between the facility tubes and the floor for installation of equipment necessary to conduct beam-type research.

Enclosed stairways are to be provided to the reactor and canal operating areas on the east and west ends of the building. A truck entrance and the personnel doors will be placed at convenient locations as indicated in the Figures.

Canal Area. - The floor area is approximately 55 ft by 100 ft neglecting the control room area which is presently shown inside this area although it is separate in regard to air handling equipment, etc. A crane with auxiliary hook similar to the one specified for the operations area will be provided.

Control Room and Office Area. - The arrangement of control room, instrument repair room and offices within the building has not been definitely determined. Figs. 9 and 10 give an indication of the planning at present. The arrangement shown for the control room allows the console operator to view the work on the operating floor from an elevated level. An auxiliary instrument and instrument repair room will be located adjacent to the control room in order to centralize operations of this type.

Office space is to be the minimum required for the staff actually engaged in operating the facility. Additional office, laboratory and maintenance space for the staff not directly engaged in operation will be provided in other buildings so as to minimize the extraneous activities in the HFIR building. The building design will incorporate provision for visitors to view all phases of the facility and control room without interfering with the work being performed.

Process Water Building. - No layout of the process water building is available at the present time; however, outline criteria have been developed for use by the architect-engineer, Singmaster & Breyer, in making studies of building arrangement.

Preliminary studies are to be based on the following criteria:

(A) Shielding will be placed around the entire HFIR primary cooling system to reduce the radiation levels to the least of the following:

(1) 1 r/hr at shield surface 24 hours after a complete core meltdown.

(2) 1 r/hr at shield surface immediately after meltdown of 1% of the fuel.

(3) 1/3 of ORNL tolerance at shield surface with one defective (leaking) plate in the core during otherwise normal operation.

(B) A decay vessel will be provided to allow the high level N^{16} activity to decay before the primary coolant reaches the main pumps and heat exchangers. This provision is made even if the shielding listed in item (A) should be adequate to protect anyone outside of the shielding in order that inspection and maintenance operations may be more easily accomplished.

(C) Individual shielded compartments will be provided for major items of equipment which may require maintenance during operation. An example would be the main pumps in which the installed spare could be overhauled during operation. The degree of compartmentation necessary has not been specified in detail at the present time; however, the intent is that adequate shielding will be provided to allow maintenance on those items which may require longer periods of time for maintenance than is available in a normal refueling shutdown and which do not otherwise require that the reactor be shut down.

(D) All compartments will be vented to a radioactive gas handling system and drained to a radioactive drainage system in order that uncontrolled release of activity will not result from a leak in the system.

(E) All equipment components except piping will be readily removable from the compartments.

3. OPERATION

3.1 Operating Cycle

The present design is based on 24-hour per day operation with a 10-day core life and shutdown only as required for reloading of fuel and necessary maintenance. It is anticipated that with future improvements in fuel loading and the use of burnable poison the core life may be extended to about 15 days; however, this appears to be the maximum that can be attained without making a sacrifice in the integrated flux received by the heavy isotope pro-

duction capsule for irradiation times of the order of one year.

The entire core will be changed each shutdown due to the high rate of burnup and due to the requirement of maintaining a minimum ratio of max/avg power density. This procedure will yield a minimum amount of fuel handling and will eliminate the possibility of accidentally adding more than the desired amount of fuel. It is planned to provide a critical experiment facility capable of checking out new cores prior to placing them in the reactor and in addition it is believed that flux and burnup measurements on depleted cores will provide information of considerable value in improving the fuel element design.

3.2 Operating Personnel

Operation of the reactor will be the responsibility of the Reactor Operations Department of the ORNL Operations Division. This department is currently responsible for operation of the ORR, LITR and X-10 Graphite Reactor. Operating personnel will be chosen from this organization which now has many years of experience gained in operation of water-cooled and -moderated reactors. It is anticipated that the operating crew of this reactor will be very small inasmuch as the types and numbers of experiments are limited and do not include large test loops of the type installed in the ORR and LITR. This operating crew will be supplemented as necessary by other members of the department for the performance of operations such as maintenance, fuel reloading, etc.

4. HAZARDS CONSIDERATIONS

4.1 Facility Location and Arrangement

The proposed location for this reactor facility is shown in Fig. 12, which is a plant layout drawing of the Oak Ridge National Laboratory. Fig. 13 is a preliminary site plan. The indicated site and the area to the west of it have been reserved as a reactor area in the plant planning program for several years. As can be seen from Fig. 12 most of the buildings in this part of ORNL are temporary structures which are to be removed within the next two or three years.

The Laboratory center is shifting to the east as far as general office and laboratory space are concerned, and the 4500 area, as indicated on this plan, is the administrative, office and laboratory area. Contracts have been let for approximately $10 million to provide additional office and laboratory space in the 4500 area. The site selection, therefore, is consistent with the over-all plan to separate operations involving the handling of large quantities of radioactive materials from the normal office and laboratory functions.

In this proposed location the land slopes down from Hilltop Circle toward Central Avenue, and therefore the location is advantageous for a building to which grade access is desirable from several levels. The area select-

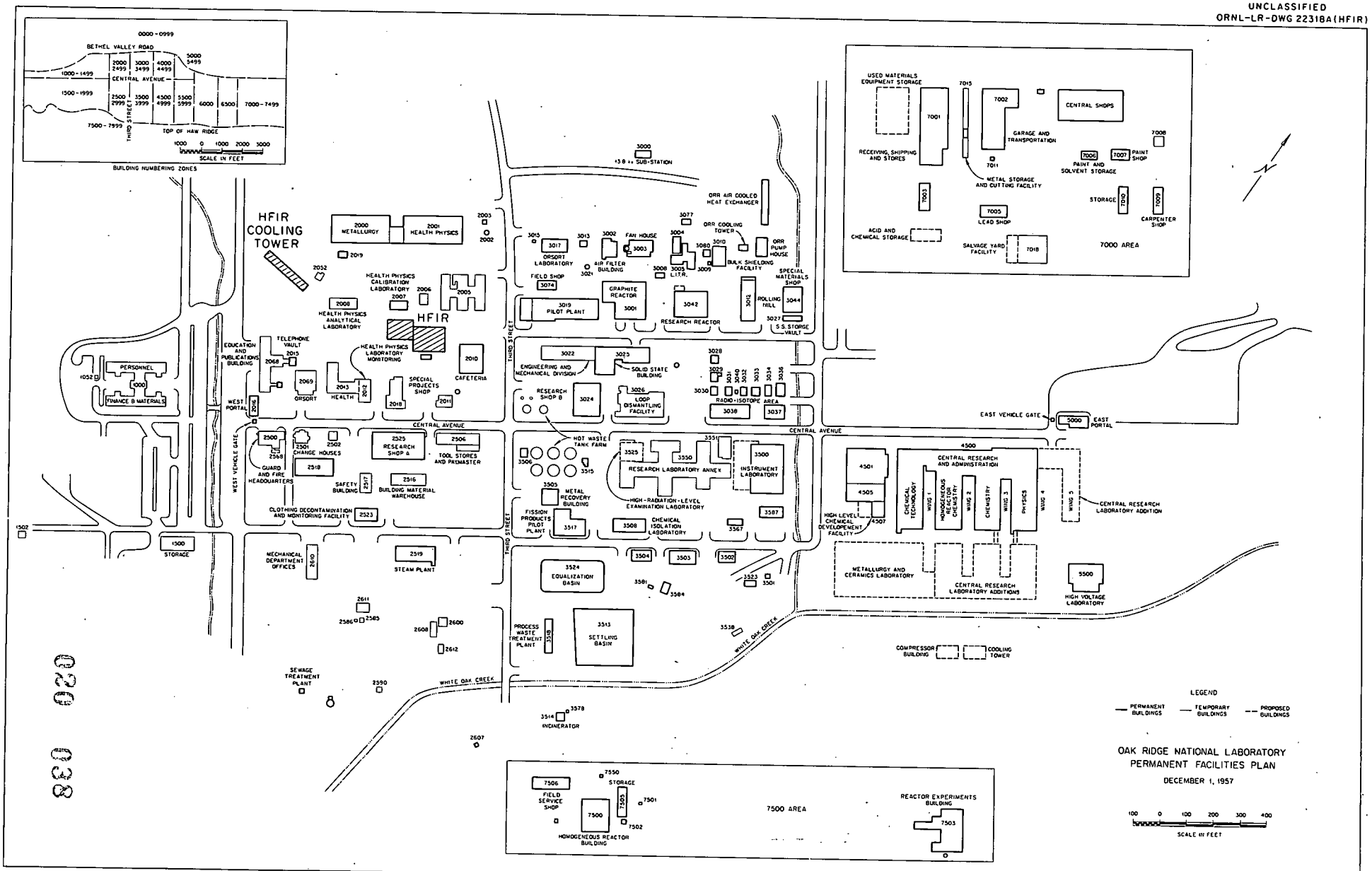

UNCLASSIFIED
ORNL-LR-DWG 22318A(HFIR)
BUILDING NUMBERING ZONES
BETHEL VALLEY ROAD
CENTRAL AVENUE
THIRD STREET
TOP OF HAW RIDGE
SCALE IN FEET
HFIR COOLING TOWER
HFIR
CENTRAL AVENUE
WEST VEHICLE GATE
WEST PORTAL
EAST VEHICLE GATE
EAST PORTAL
THIRD STREET
WHITE OAK CREEK
7000 AREA
7500 AREA
LEGEND
PERMANENT BUILDINGS
TEMPORARY BUILDINGS
PROPOSED BUILDINGS
OAK RIDGE NATIONAL LABORATORY
PERMANENT FACILITIES PLAN
DECEMBER 1, 1957
SCALE IN FEET

Fig. 12.

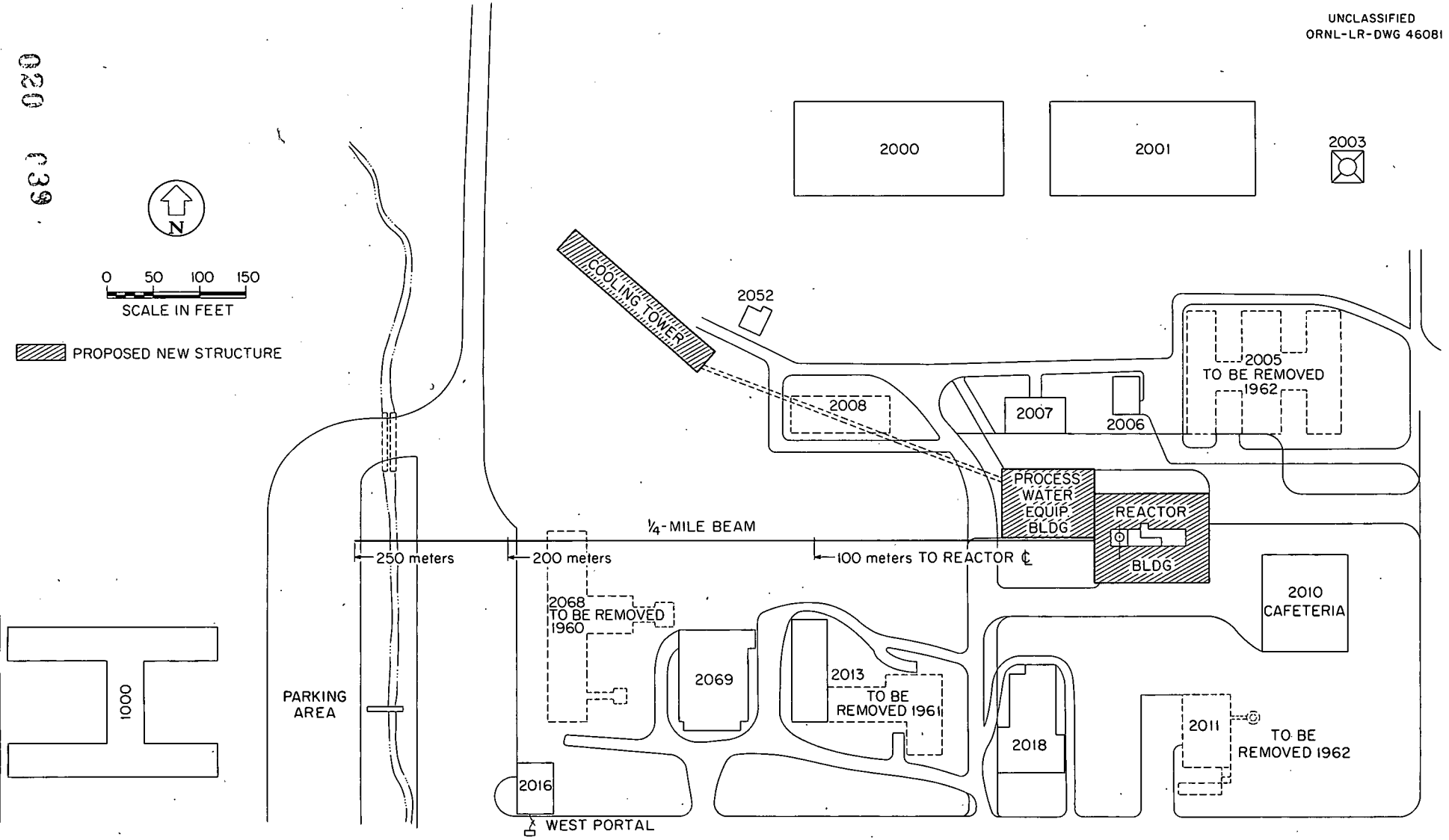


Fig. 13. HFIR Site Plan

ed is conveniently located with respect to services such as waste disposal, power, and water. In addition, a 175-ft stack is located near the site. This stack is not presently being used; however, it is anticipated that it will be used for discharge of ventilation air from areas in which small radioactive gas releases may occur. The short distance (approximately 250 yards) from other reactor installations (X-10 Graphite, ORR, LITR) will allow the Operations Division to utilize their operating personnel and equipment more efficiently, thereby reducing the operating costs to a minimum. If arrangements can be made to move the cafeteria (Building 2010) to the 4500 area it is planned to modify the present cafeteria building in order to provide space for offices and laboratories needed by the Operations Division. Thus a minimum of such space will be needed in the new facility; this is considered highly desirable from the standpoint of control of possible contamination.

In considering the many possibilities for arrangement of the facility, the conclusion has been reached, on the basis of the operating experience of the Reactor Operations Division and the Isotopes Division, that spread of contamination can and will occur as a result of handling materials in and around reactors. This has led to the arrangement previously described in Section 2.8, in which the reactor area, the beam hole area, the canal area, the office and control room space, and the process water building will be separate areas insofar as this is practical. Each of these areas will be a separate enclosure with respect to ventilation and each will be designed to provide the degree of containment judged to be adequate for the particular operations that will be carried on in that area. This principle of isolation has been successfully used in the Isotopes Division for years and has been proved to be a sound approach.

A possible alternate site in the ORNL area is situated about 200 yards north of the proposed location. This site is considered less desirable than the proposed site due to the greater difficulty in connecting to Laboratory facilities such as the various drainage systems due to an intervening hill of nearly solid rock. At the present time no advantage is seen for the alternate site from a construction, operation or hazards standpoint.

Although no detailed cost studies have been made it has been estimated that additional costs of about $1,000,000 would be incurred if the reactor were to be built anywhere in the Oak Ridge area outside of the ORNL fence. This estimated cost does not include the added expense of operation and is the result of the necessity of developing a new site, providing water, drainage, electric and other utilities and of providing an adequate waste disposal system. The incentive to locate the reactor in the ORNL area can be seen to be economic as well as to have greater convenience and utility in operation and use of the facility.

Information on the Oak Ridge area is generally available and is not repeated here. A recent report is available which gives information on the geology, hydrology, seismology and meteorology of the general area in addition to information on the population of the surrounding cities and counties.[14]

(14) Kaiser Engineers, Allis-Chalmers Mfg. Co., Experimental Gas-Cooled Reactor, Preliminary Hazards Summary Report, ORO-196 (Rev. Aug. 11, 1959).

More complete information on the meteorology of the Oak Ridge area is given in ORO-99, "A Meteorological Survey of the Oak Ridge Area."

Perhaps the most significant point in relation to the proposed site is the fact that it is in the Oak Ridge area and therefore a controlled area of considerable extent separates the proposed location from the general public. The nearest distance from the proposed site to land outside the Oak Ridge controlled area is greater than two miles and the general region in this direction of closest approach is sparsely settled farm or undeveloped land.

In regard to the Oak Ridge National Laboratory it is believed that the HFIR will not substantially increase the hazard to personnel or property. In terms of fission product inventory, the ORR, LITR and X-10 Graphite Reactor each contain equal or greater quantities of fission products. The HFIR will have a greater proportion of material of short half life - i.e., those materials which reach a saturation concentration during the nominal 10-day HFIR operating cycle - and therefore the potential hazard due to core content is greater. As described in various sections of this report the HFIR complex is being designed to handle, as nuisance-type operations, accidents which would be serious in other facilities in this area. This philosophy of designing for fuel element ruptures of considerable severity is the result of our belief that the extremely high-power densities may cause this type of accident to be more frequent than in the LITR or ORR; however, the consequences will be much less serious than if such an accident were to occur in the ORR or LITR.

Another important point which must be considered is the fact that HFIR is being designed as a facility specifically for high flux irradiation of small samples with some provision for basic research. The experimental program planned does not now include the insertion of engineering test loops of the type ordinarily associated with engineering test reactors such as MTR, ORR and ETR. Inasmuch as it is our conviction that failure of this type of experimental set-up is one of the most probable ways of getting the associated reactor into serious trouble, the HFIR should be many times safer than any of the engineering test reactors.

4.2 Waste Disposal

One of the major advantages of locating the facility in the ORNL area is the fact that an extensive radioactive waste disposal system for liquid and gaseous wastes is available for handling the normally small quantities of effluent. In the event of fuel element melt-down even a partial release of fission products to the coolant system would result in large quantities of highly contaminated liquid waste which would be disastrous to operation in an area which did not have the capability of handling these wastes. The ORNL system is capable of handling these very large quantities of activity without seriously disrupting operations. Information on the waste disposal facilities at ORNL is available in the following reports: (1) Cowser, Morton, and Witkowski, "The Treatment of Large-Volume Low-Level Waste by the Lime-Soda Softening Process", 1958 Geneva Conference, P/2354, Vol. 18; (2) deLaguna, Cowser and Parker, "Disposal of High Level Radioactive Liquid

Wastes in Terrestrial Pits - a Sequel," 1958 Geneva Conference, P/2351, Vol. 18; (3) Morgan, "Waste Management Program at Oak Ridge National Laboratory," Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, 86th Congress, Vol. 1, pp 427-514 (1959).

4.3 Accidents

While detailed studies have not yet been made on all types of accidents which could result in the release of substantial quantities of fission product activity from HFIR, sufficient work has been done to enable criteria to be established for the control of cases which appear to be maximum in terms of a release. Studies will be continued in the investigation of all types of accidents; however, the following discussion will indicate the present status.

In view of the fact that HFIR is basically a reactor of the MTR-ORR-BSF class - i.e., in regard to fuel plates, coolant, moderator, neutron prompt generation time - and as a consequence of the fact that it is not planned to operate large engineering test experiments in or near the reactor core, it is believed that a catastrophic excursion as a result of introducing large amounts of excess k is not sufficiently credible to constitute a basis for criteria on fission product release. Studies will be made of this class of accidents for HFIR; however, it is believed that sufficient information can be set forth to justify this position.

During normal operation of the HFIR it is possible that coolant flow may be restricted in one or more coolant channels in the fuel region. Any substantial reduction in flow to a channel will result in at least a partial melting of the fuel plates forming the channel and adjacent plates may be affected also. Due to the fact that this may occur, the shielding of the coolant system will be designed to cope with this situation without dangerous radiation dosage to operating personnel as described in Section 2.8. The reactor would be shut down immediately and clean-up operations started on the coolant. The clean-up and off-gas systems will be designed to handle failures of this type without exceeding normal radiation exposure limits to employees. The activity will be confined by the reactor pressure vessel and pressure piping system and provision for disposal of the contaminated wastes will be a part of the design of the facility. The primary coolant system can be depressurized immediately, thereby eliminating any major concern over a coincident leak in the high-pressure system. In this regard it should be pointed out that depressurization due to a loss of the pressurizer pumps during normal full-power operation will result in a reactor scram and preliminary estimates indicate that adequate time is available to shut down the reactor before any damage to the fuel results; the same is true for loss of main pump power; adequate emergency coolant will be available to provide for removal of afterheat.

The most serious accident for which a mechanism clearly exists would be the result of losing all coolant flow to the core during or following operation. The heat transfer requirements following shutdown of the reactor

are sufficiently high that some part or all of the core may melt during a period of about 2 hours following shutdown if coolant flow is greatly restricted. After this time natural convection cooling will be adequate to remove the afterheat even if the recirculation path is blocked. If the reactor pressure vessel is closed at the time of melting, the activity will be confined to this system as previously discussed. If the reactor pressure vessel is open or if the core has been removed to the storage pool and free convection cooling is somehow lost, radioactive gases will be released to the water and from the water to the air at the surface of the pool. For this type of accident to release activity to the atmosphere it appears that the reactor will have been shut down for at least an hour and possibly longer due to the fact that at least this much time will be required to depressurize, clean up the coolant and open the pressure vessel. It appears therefore that at least an hour's delay can be counted on from reactor shutdown to release of activity. For this case the activity release would be under water.

A much less credible case would be the result of complete loss of coolant to the core - i.e., loss of water from the reactor tank and the pool. It is not quite clear to us just what sequence of events might bring this about; however, it might be presumed that an earthquake could rupture the pool with a coincident brittle fracture of the reactor tank. Consideration of a catastrophe of this type leads one to suspect that the containment provision would no longer be intact either but, as will be shown later, if the containment does remain intact the airborne hazard to the nearby areas of interest is still acceptable considering the very small probability of occurrence. This is a result of the fact that the isotopes of interest released to the atmosphere through the scrubber and filter system are primarily xenon and krypton. An accident of this type would, however, release large quantities of contaminated water. If such an accident is considered to have sufficient probability of happening provision would have to be made to insure that the release of contaminated water would be contained in the building. No study has been made on this latter point at present; however, it is believed that, if found necessary as a result of study, provision can be made in the design to insure that any gross release of contaminated water from the pool can be contained in the building.

4.4 Fission Product Release from Fuel

The release of fission products from fuel plates of the MTR-ORR-HFIR type under various melting conditions has been studied by a group at ORNL during the last several years.[15]

The following tabulation represents, in the opinion of the group working on this problem, a conservative estimate based on experimental results, of the per cent release of fission products from the fuel, either to the water if the

[15] G. E. Creek, W. J. Martin and G. W. Parker, Experiments on the Release of Fission Products from Molten Reactor Fuels, ORNL-2616 (July 7, 1959).

element is under water or to the air if the element is in the air, in the event of a meltdown.

Table 1

Element	Release, %
Krypton	100
Xenon	100
Bromine	10
Iodine	10
Selenium, Molybdenum, Technicium, Ruthenium, Tellurium	< 1 for oxide fuel; ~ 0 for alloy fuels
Rubidium	< 0.1
Strontium	< 0.1
Cesium	< 0.1
Barium	< 0.1
All others	0

If the core is melted under water it is estimated that the only material of consequence which would escape from the water to the air would be less than 1% of the iodine and bromine content of the core and less than 50% of the core content of xenon and krypton. All other materials would be retained in the water.

It is highly probable that much less than 1% of the iodine and bromine would get through the water; however, this value is taken for underwater meltdown in order to assure a highly conservative basis for the hazard estimate. The same is true of the xenon and krypton; the figure of 50% is to be used in the case of an underwater meltdown in order to assure a highly conservative estimate.

A tabulation of the fission product inventory in the HFIR core at the end of 10 days' operation at 100 Mw power level has been made, based on the data given in ORNL-2127.(16) The table presents the concentration at the time of reactor shutdown and for several times of interest following shutdown.

(16) J. O. Blomeke and M. F. Todd, Uranium-235 Fission-Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time, ORNL-2127 (Aug. 19, 1957).

Table 2

Isotope	Half Life	Core Content, Curies Time after shutdown, Minutes 0	1	10	60
Xe 131m	12.0 d	4.3×10^3	4.3×10^3	4.3×10^3	4.3×10^3
Xe 133m	2.3 d	1.18×10^5	1.18×10^5	1.18×10^5	1.24×10^5
Xe 133	5.27d	3.73×10^6	3.73×10^6	3.73×10^6	4.1×10^6
Xe 135m	15.6 m	1.54×10^6	1.54×10^6	1.54×10^6	1.39×10^6
Xe 135	9.13h	1.12×10^5	1.12×10^5	1.23×10^5	2.91×10^5
Xe 137	3.9 m	4.93×10^6	4.24×10^6	8.38×10^5	1.13×10^2
Xe 138	17 m	4.59×10^6	4.31×10^6	3.4×10^6	3.95×10^5
Xe 139	41 s	4.02×10^6	1.89×10^6	1.53×10^2	0
Xe 140	16 s	3.16×10^6	2.37×10^5	0	0
Kr 83m	114 m	3.98×10^5	3.98×10^5	3.9×10^5	3.85×10^5
Kr 85m	4.36h	1.22×10^6	1.22×10^6	1.12×10^6	9.27×10^5
Kr 85	10.27y	4.6×10^2	4.6×10^2	4.6×10^2	4.88×10^2
Kr 87	78 m	2.28×10^6	2.28×10^6	2.14×10^6	1.41×10^6
Kr 88	2.77h	3.1×10^6	3.1×10^6	2.95×10^6	2.29×10^6
Kr 89	3.18m	4.02×10^6	3.38×10^6	4.62×10^5	10.5
Kr 90	~33 s	3.52×10^6	1.3×10^6	12	0
Kr 92	3.0 s	2.19×10^6	1.7	0	0
Rb 86	19.5 d	50	50	50	50
Rb 88	17.8 m	3.03×10^6	3.03×10^6	3×10^6	2.48×10^6
Rb 89	15.4 m	3.97×10^6	3.89×10^6	3.14×10^6	3.26×10^5
Rb 90	2.74m	4.9×10^6	3.82×10^6	4.31×10^5	1.85
Rb 91	14 m	3.26×10^6	3.13×10^6	2.08×10^6	1.76×10^5
Rb 92	80 s	4.61×10^6	2.95×10^6	2.49×10^4	2.7×10^{-2}
I 130	12.6h	2.48×10^3	2.48×10^3	2.46×10^3	2.35×10^3
I 131	8.05d	1.29×10^6	1.29×10^6	1.29×10^6	1.29×10^6
I 132	2.4 h	3.3×10^6	3.3×10^6	3.3×10^6	3.3×10^6
I 133	20.8 h	5.5×10^6	5.5×10^6	5.5×10^6	5.5×10^6
I 134	52.5 m	6.42×10^6	6.42×10^6	6.42×10^6	4.82×10^6
I 135	6.68h	5.08×10^6	5.08×10^6	5×10^6	4.4×10^6
I 136	86 s	2.66×10^6	1.7×10^6	2.02×10^4	0
I 138	5.9 s	2.88×10^6	2.53×10^3	0	0
I 139	2.7 s	3.82×10^5	1.8×10^{-1}	0	0
Se 77m	17.5 s	1.5×10^2	1.5×10^2	1.5×10^2	1.4×10^2
Se 79m	3.9 m	1.28×10^4	1.23×10^4	8.45×10^3	1.51×10^2
Se 79	6.5×10^4y	9.7×10^{-3}	9.7×10^{-3}	9.7×10^{-3}	9.7×10^{-3}
Se 81m	56.5 m	6.61×10^3	6.61×10^3	6.21×10^3	3.17×10^3
Se 81	17 m	1.73×10^4	1.73×10^4	1.52×10^4	6.57×10^2
Se 83	25 m	1.5×10^5	1.47×10^5	1.17×10^5	3.45×10^4
Se 84	~2 m	9.37×10^5	6.93×10^5	3×10^4	0

Table 2 (continued)

Isotope	Half Life	Core Content, Curies Time after shutdown, minutes 0	1	10	60
Mo 99	67 h	4.67×10^6	4.67×10^6	4.67×10^6	4.58×10^6
Mo 101	14.6 m	4.13×10^6	4.05×10^6	3.06×10^6	2.27×10^5
Mo 102	12 m	3.49×10^6	3.21×10^6	1.68×10^6	9.77×10^4
Tc 99m	6.04h	4.74×10^5	4.74×10^5	4.74×10^5	4.74×10^5
Tc 101	14.0 m	4.13×10^6	4.13×10^6	3.71×10^6	6.4×10^5
Tc 102	<25 s	3.44×10^6	3.3×10^6	1.89×10^6	1.1×10^5
Tc 107	< 1.5 m	1.31×10^5	8.4×10^4	1.28×10^3	0
Ru 103	41 d	3.66×10^5	3.66×10^5	3.66×10^5	3.66×10^5
Ru 105	4.5 h	7.29×10^5	7.29×10^5	7.29×10^5	6.7×10^5
Ru 106	1 y	6.24×10^3	6.24×10^3	6.24×10^3	6.24×10^3
Ru 107	4 m	1.72×10^5	1.65×10^5	4.47×10^4	7.6
Br 82	35.9 h	1.03×10^2	1.03×10^2	1.03×10^2	98
Br 83	2.4 h	3.94×10^5	3.94×10^5	3.94×10^5	3.07×10^5
Br 84	30 m	9.47×10^5	9.28×10^5	7.86×10^5	2.46×10^5
Br 85	3.0 m	1.27×10^6	9.65×10^5	1.27×10^5	2.5
Br 87	55.6 s	2.87×10^6	1.34×10^6	1.43×10^3	0
Br 88	15.5 s	2.4×10^6	1.7×10^5	0	0
Cs 136	13 d	2.23×10^3	2.23×10^3	2.23×10^3	2.23×10^3
Cs 137	26.6 y	3.49×10^3	3.49×10^3	3.49×10^3	3.49×10^3
Cs 138	32 m	4.7×10^6	4.7×10^6	4.21×10^6	1.9×10^6
Cs 139	9.5 m	4.88×10^6	4.32×10^6	2.44×10^6	5.22×10^4
Cs 140	66 s	5.13×10^6	3.5×10^6	1×10^4	0
Ba 137m	2.6 m	3.48×10^3	3.48×10^3	3.48×10^3	3.48×10^3
Ba 139	85 m	5.11×10^6	4.19×10^6	3.47×10^6	1.94×10^6
Ba 140	12.8 d	2.24×10^6	2.24×10^6	2.24×10^6	2.24×10^6
Ba 141	18 m	5.03×10^6	4.78×10^6	3.4×10^6	4.83×10^5
Ba 142	6 m	4.64×10^6	3.7×10^6	1.5×10^6	4.3×10^3
Ba 143	30 s	4.12×10^6	9.5×10^5	4.15	0
Sr 89	54 d	5×10^5	5×10^5	5×10^5	5×10^5
Sr 90	28 y	3.27×10^3	3.27×10^3	3.27×10^3	3.27×10^3
Sr 91	9.7 h	4.95×10^6	4.95×10^6	4.95×10^6	4.4×10^6
Sr 92	2.7 h	5.2×10^6	5.2×10^6	4.8×10^6	3.7×10^6
Sr 93	7 m	5.4×10^6	5.08×10^6	2×10^6	1.46×10^4
Sr 94	2.0 m	5×10^6	3.8×10^6	1.6×10^5	0

Table 2 (continued)

Isotope	Half Life	Core Content, Curies, Time after shutdown, minutes: 0	1	10	60
Te 125m	58 d	2.24×10^{-1}	2.91	5.82	26.8
Te 127m	90 d	1.91×10^{3}	1.91×10^{3}	1.93×10^{3}	1.93×10^{3}
Te 127	9.3 h	1.34×10^{5}	1.34×10^{5}	1.47×10^{5}	1.74×10^{5}
Te 129m	33 d	5.45×10^{4}	5.45×10^{4}	5.45×10^{4}	5.9×10^{4}
Te 129	72 m	6.05×10^{4}	6.1×10^{5}	3.6×10^{5}	9.7×10^{4}
Te 131m	30 h	3.81×10^{5}	3.8×10^{5}	3.8×10^{5}	3.5×10^{5}
Te 131	24.8 m	2.44×10^{6}	2.4×10^{6}	2.4×10^{6}	1.37×10^{6}
Te 132	77 h	3.31×10^{6}	3.31×10^{6}	3.31×10^{6}	3.2×10^{6}
Te 133m	63 m	5.1×10^{6}	5.1×10^{6}	4.8×10^{6}	2.8×10^{6}
Te 134	44 m	5.6×10^{5}	5.6×10^{6}	5.2×10^{6}	2.1×10^{6}
Te 135	<2 m	3.6×10^{6}	2.8×10^{6}	1.1×10^{5}	0

4.5 Containment

Type. - Containment of the controlled leakage type[17] (ORR type) is proposed for the HFIR.

This type of containment is based on the premise that if air is removed from the containment region at a rate sufficient to establish a significant negative pressure with respect to the external surroundings, the leakage will be from the outside of the building to the inside. Contaminated air in the building will therefore not leak out. In practice it has been found practical to achieve a negative pressure of the order of 0.5 inches of water in large buildings of normal mill type construction if reasonable attention is given to minimizing the leakage through seams, around door jambs, ventilating system closures, etc. The ORR building, which has a gross volume of about 8×10^5 ft^3, can be maintained at a negative pressure of 0.3–0.5 in. of water with an exhaust rate for the emergency system of about 5,000 cfm.

It is anticipated that for the HFIR building a much lower leakage for a given pressure differential can be achieved due to the fact that the low leakage requirement will be part of the criteria for design of the building from the beginning of design; provisions for low leakage were not incorporated in the early stages of ORR design.

The exhausted air will be passed through an elaborate filter and gas removal system to decontaminate the air before discharging it to the atmos-

(17) F. T. Binford and T. H. J. Burnett, A Method for the Disposal of Volatile Fission Products from an Accident in the Oak Ridge Research Reactor, ORNL-2086 (Aug. 2, 1956).

phere via a stack.

In the event of an accident in any of the contained regions the normal ventilating system for the building will be shut down automatically. It is planned to provide a radiation detection system specifically for this purpose. The details of such a system have not been worked out at this time; however, a highly reliable system has been designed for the ORR which will initiate emergency action on radiation levels sensibly in excess of any which will occur in normal operation.

The design of the HFIR emergency air handling system will be based on continuous operation and therefore it will be capable of handling an accident at any time. The requirement that it be in continuous operation will mean that a considerable quantity of the equipment will have to be in duplicate in order to provide for filter changing, maintenance, etc., but it is believed that the reliability of the system will be increased manyfold. Adequate emergency power will be provided so that this system will be independent of the normal plant power system.

In the ORR the emergency air handling system is being operated continuously at the present time in order to provide for cell ventilation requirements for various of the engineering test loops. This system removes air from the reactor building through air inlets to the experiment enclosures and then discharges it through the filter and scrubber system to the stack. In this way the emergency containment feature for the building is enhanced by being in continuous operation and in addition provision is made for containment of an accident involving the release of activity from an experiment. It should be noted that the probability of a release of activity from the experiments is considered to be much greater than from the reactor. ORR containment is based on release into the building and containment in the building. Due to the fact that the exhaust system takes suction at many places in the building a major release would contaminate the entire building, thus constituting a considerable hazard to nearby persons from direct gamma radiation from the building and an almost impossible job of cleaning up the building following a major release should one ever occur.

Preliminary studies of the HFIR emergency system are based on an additional degree of containment over that employed in the ORR. This additional step in containment consists of a cover over the pool with the emergency ventilation system taking suction between this cover and the pool surface. Details of how this cover will be designed in order to allow access for normal operations in the pool have not been worked out at present. The problem does not appear to be too difficult, however, as the requirements for building containment will be for an air flow of several thousand cubic feet of air per minute, and based on studies made at ORNL on the required air velocities for containing fumes in hoods it appears that with an air flow of 5,000 cfm an opening in the cover as large as 100 ft^2 would be allowable. After carefully establishing the allowable opening it is believed that a design for this cover could be made which would in effect allow the opening to be moved around as desired for the necessary handling operations in the pool while maintaining the containment feature. This problem will have to be

studied at considerable length before a decision is made as to whether or not a practical arrangement can be achieved.

The addition of a cover to the pool would prevent radioactive gases from spreading through the building in the event of an accident and thus the hazard from external radiation would be reduced considerably over the situation which exists for the ORR. In the event of a serious accident in the ORR very prompt evacuation of the ORNL area would be required due to the gamma radiation from the building. It is further believed that one of the major advantages of this added cover comes about due to the anticipated occurrence of small leaks or failures in fuel plates. The anticipated leaks would not release sufficient activity to constitute a widespread hazard; however, they would present a considerable operating problem in handling of the leaking fuel element. Finally the fact that large quantities of radioactive gas would not be released into the building proper as a result of a meltdown would reduce the consequence of any small leakage from the building through doors or walls.

Containment Areas. - The present plan is to have the HFIR building compartmented to as great a degree as can be reasonably done without seriously interfering with the normal work of operating and using the facility. As previously described there will be five main areas; with the exception of the control room and office area, each of these will be provided with an appropriate degree of containment.

The reactor operating area (reactor pool, etc.) is considered to be the area in which the greatest probability of a large activity release will exist and for this reason our preliminary investigation of activity release has concentrated on a release in this region.

The beam room is also considered a possible release area due to the presence of the beam tubes which penetrate the reactor pressure vessel. It is believed that the probability of a major release is much smaller here than in the operating area. The beam room will constitute a separate containment area under emergency conditions and even in normal operation it will be isolated from the operating area in terms of ventilation, air interchange, etc., in order to minimize the probability of contaminating more than one area in the event of a small release of radioactive gas or particulate activity.

The canal area is the third containment region and as is the case with the beam room a separate ventilation system will be provided. This area will be separated from the reactor operating area by an essentially air-tight wall which will extend down into the pool water for a sufficient distance to prevent any leakage of air at any differential pressure which can be achieved by the normal and emergency ventilation system. The canal or pool will be divided into several regions by gates in order to limit the interchange of water between the reactor operating area and the canal room area. The degree of containment which will be required for the canal area relative to the reactor operating area or the beam room area has not yet been established. It appears that the requirements will be comparable

even though the probability of a large release in this area will be much smaller than in the reactor operating area.

The control room and office area will not be a containment region in the same sense as the three previously described areas. This area will have its own ventilation system; however, the design will be such that this area is outside the containment region from the standpoint of an activity release.

The process water building will also constitute a containment area. The criteria for the degree of containment have not yet been established for this region. As described in Section 2.8 the equipment containing primary coolant will be compartmented and shielded on the basis that the primary coolant may at some time contain a large fraction of the fission product activity of the core. In addition it is planned to exhaust these compartments to a radioactive gas handling system in order to cope with the small releases of radioactive gas which would occur as a result of a leak from the system. It appears at present that the probability of a major release of activity in this area is sufficiently small to constitute an acceptable risk but further studies will be made before final design is initiated. Of course, it will be necessary to provide for handling the results of a meltdown which is confined to the primary coolant system and the necessary features will be incorporated in the design of the cleanup and off-gas system.

Assumed Accident. - For the purpose of this preliminary investigation to determine the results of a major release of activity in the HFIR facility it is assumed that the reactor core melts down in air. This should provide an extremely conservative estimate of the hazard even though it is not clear how such an accident is brought about. The fission products are assumed to be released from the fuel in the percentages and quantities described in Section 4.4. It is not intended that this be interpreted as our evaluation of the maximum credible accident inasmuch as it is our belief that the maximum credible accident will fall considerably below this in terms of activity release.

Based on our discussion of accidents it appears that a minimum length of time after shutdown on which to base the release to the atmosphere would be one hour; however, in order to present as complete a picture of the time dependence as is available at this time we have investigated a release from the stack at 1 minute, 10 minutes, and 1 hour after reactor shutdown. The one-minute interval taken as minimum is the time required for the emergency exhaust air to pass through the ducts, filters and stack based on reasonable values of air velocity and the distance from the proposed site to an existing stack.

Treatment of Released Activity. - As previously described the building containment features would prevent the activity from leaving the building except by way of the emergency air handling system. This system will provide for filtering the air to remove particulate matter and scrubbing and/or adsorption methods such as copper or silver-plated copper mesh and activated

charcoal filters for the removal of gases such as iodine and bromine. After passing through the decontamination system the air will go through blowers and be discharged from a tall stack.

Discussion with members of an ORNL group which has been working on the problem of fission product gas handling has resulted in the opinion backed by experimental evidence[18,19] that almost any desired degree of decontamination for particulates and iodine or bromine can be achieved. Estimates based on the assumed quantities released from the core and the preliminary meteorological studies indicate that all exhausted air can be decontaminated to such a degree that no significant problem in discharging material to the atmosphere is encountered with the exception of the "inert" gases, xenon and krypton. These gases cannot readily be removed from the air stream and therefore release of these gases to the atmosphere constitutes the present basis for concern regarding contamination of the atmosphere.

Many details of the criteria for the air handling system remain to be worked out and many problems are anticipated during the detailed design; however, no insurmountable problems have shown up to date. One of the major problems which is anticipated is the requirement that the decontamination system be designed to handle the heat load brought about by the fission products which may be accumulated in the system. It is believed that this problem can be handled by an appropriate sequence of filters so arranged that the early stages are made of materials which will withstand moderately high temperatures followed or preceded as necessary by water or chemical scrubbers which will control the air temperature in addition to removing fission products themselves.

Stack Release of Activity. - The meteorological parameters for the Oak Ridge area are quite well known as a continuing study has been in progress for many years. ORO-99, "A Meteorological Survey of the Oak Ridge Area," presents the most complete compilation of data for this area although more recent information is available on certain aspects from the Oak Ridge Office of the U. S. Weather Bureau.

A preliminary study of the release of activity from several existing stacks in the ORNL area and for two stacks, one of which may be constructed at some time in the future, has recently been made by F. A. Gifford of the U. S. Weather Bureau. This work was done as a result of interest by several groups at ORNL in obtaining estimates based on the best available information regarding a release of activity from the existing and possible future stacks.

The existing stacks which were studied are the "old" Steam Plant stack

(18) R. E. Adams and W. E. Browning, Removal of Radio-Iodine from Air Streams by Activated Charcoal, ORNL-2872 (to be issued).

(19) W. E. Browning, R. E. Adams and R. D. Ackley, Removal of Fission Gases From Reactor Off-Gas Streams by Adsorption, ORNL CF-59-6-47 (June 11, 1959).

(2061) which has not been used for several years, the Isotope Area (3039) stack which is presently in use as the major facility for handling contaminated air and hot off-gas in the Laboratory, and the X-10 Graphite Reactor stack (3018) which is used for discharge of coolant air from that facility and also as the off-gas stack for the LITR.

The possible future stack facilities which were investigated were: (1) a short stack located on top of the highest hill (Melton Hill) in the nearby area; this hill is located about 7000 ft from the ORNL area, and (2) a tall stack located on a ridge (Haw Ridge) adjoining the ORNL area; this ridge is located about 2000 ft from the center of the Laboratory area. This stack would consist of a tower ~500 ft high, similar to a radio or television antenna tower, with an exhaust duct passing up through the middle of the tower. Table 3 gives the elevation of the top of the various stacks, the height above the base of the 3039 stack (Laboratory ground reference) and the meteorological parameters used in the study. The values of $\bar{u}$, wind velocity, are the most probable values in this area for the particular elevations. In each case values are given for daytime (lapse) conditions and night (stable - i.e., moderation inversion) conditions. The lines for which n = 0.25 are for daytime conditions and those for which n = 0.33 are for nighttime conditions. The values given for n and C^2 are also representative of the most probable values.

Table 3

Stack Designation	Elev. of Top, ft above msl	Height Above Base of 3039, meters	n	$\bar{u}$ m/sec	C^2
3039	1070	76.2	.25	3.6	1.3×10^{-2}
			.33	2.8	2.6×10^{-3}
3018	1075	77.7	.25	3.6	1.3×10^{-2}
			.33	2.8	2.6×10^{-3}
Old Steam Plant (2061)	970	45.7	.25	3.1	1.5×10^{-2}
			.33	1.8	3.6×10^{-3}
Haw Ridge	1470	192	.25	4.2	9.4×10^{-3}
			.33	4.9	1.6×10^{-3}
Melton Hill	1450	192	.25	4.2	9.4×10^{-3}
			.33	4.9	1.6×10^{-3}

For evaluation of the dosage which would be received it was assumed that the radioactive material would be released in a period of time which

was short compared to the cloud passage above the receptor - i.e., a quick release.

The nomogram given as Fig. 8.3 in AECU-3066, "Meteorology and Atomic Energy," was used in combination with Equation 8.7 of the same report to obtain values of gamma dosage for each of the xenon and krypton isotopes of interest.

The nomogram given as Fig. 9.1 of AECU-3066, "Meteorology and Atomic Energy," was used to estimate the concentration at ground level on the basis of unit release rate. The assumed total release was used with these data to obtain a time-concentration product which was evaluated at several points of interest in terms of beta dosage using the following equation.[20]

$$Z = 2.6 \times 10^{-7} E_s \bar{Q} t$$

where

Z = External exposure, rem

E_s = Effective beta energy (Mev/dis)

$\bar{Q}$ = Concentration, ($\mu c/m^3$)

t = time, seconds.

It should again be noted that the radiation dosage of interest is that brought about by external radiation from beta and gamma emitters due to the fact that the assumed release to the atmosphere will be restricted to xenon and krypton. Of course, Kr^{89} and Kr^{90} decay through intermediates to Sr^{89} and Sr^{90}; this problem was investigated on the basis that all Kr^{89} and Kr^{90} were released from the stack as Sr^{89} and Sr^{90}. The results of this investigation showed that a receptor at the point of maximum concentration, even under these conditions, would inhale less than the allowable quantities established for a normal 40-hour working week.

A detailed analysis of the contributions to external beta and gamma dosages of the daughter products of the xenon and krypton has not been made at this time. Preliminary estimates based on the major contributors, such as Rb^{88}, indicate that the maximum dosage figures arrived at on the basis of exposure at short times following reactor shutdown will not be increased by more than about 30%. The maximum dosage figures for longer times after shutdown will be increased in greater proportion; however, they are smaller dosages to start out with so the over-all effect is small.

The maximum radiation doses to a receptor were investigated for the several stacks previously described; the following information is based on

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[20] T. J. Burnett, Reactors, Hazard vs. Power Level, Nuc. Sci. & Eng., Vol. 2 No. 3, 382 (May 1957).

a release from the existing 3039 stack. A discussion of the release from the other stacks will be presented in relation to these data.

The maximum gamma dose for our assumed release of Xe and Kr would be received at locations close to the stack due to the fact that the shorter half-life materials would not have had an opportunity to decay. The maximum gamma dose received at distances close to the stack assuming only 1-minute delay was found to be 27 r; for 10-minute delay, 16.7 r; for 1-hour delay, 10 r. For distances which fall outside of the Oak Ridge controlled area (greater than 3200 meters from the 3039 stack) the maximum gamma dose will be less than 25 r for any of the three decay times investigated.

The maximum beta dose resulting from the assumed release of xenon and krypton from the 3039 stack was investigated for the same delay times after reactor shutdown as for the gamma doses.

The point of maximum dose is reached, for the daytime case, at a distance of about 1500 meters and for the nighttime case at a distance of about 5000 meters. The maximum integrated dose from beta radiation was found to be about 32 rem for the daytime case and about 20 rem for the nighttime condition with a 1-minute delay after shutdown before release. The values for a 10-minute delay would be somewhat smaller but due to the time required for the cloud to reach the distance of maximum concentration the reduction due to this effect is not great. For the case of 1-hour delay before release, the values for dosage received at the point of maximum concentration drop to less than 13 rem for either the daytime or nighttime cases. The distances between which values of 50% or greater of the maximum may be observed for the daytime case are 1000 to 4000 meters and for the nighttime case from 3500 to 15,000 meters.

Use of the old Steam Plant stack would lead to maximum external gamma doses about 2.6 times higher than that from 3039 and use of either of the two possible future stacks, Melton Hill or Haw Ridge, would lead to external gamma doses about 5 times smaller. For the dosage from beta radiation, use of the old Steam Plant stack would lead to maximum values about 4 times higher than those found using 3039. Use of either of the possible future Melton Hill or Haw Ridge stacks would lead to a maximum beta dose about a factor of 8 smaller than for 3039.

As stated earlier the meteorological parameters used are the most probable values. It is believed that presenting the information in this fashion leads to a less ambiguous situation in regard to the dosage values reported. Although it is true that poorer dispersion conditions, C smaller and n larger, may exist - particularly at night - this will not affect the value of the maxima but only the location. In fact observational experience on very stable conditions[21] indicates that the vertical dispersion values

[21] G. R. Hilst and C. L. Simpson, Journal of Meteorology, Vol. 15, No. 1, 125 (Feb. 1958).

are actually so low under very stable conditions that the dispersed material does not reach the ground. It is therefore quite probable that actual ground concentrations will be much less than those predicted by the methods which we have used. In early morning, shortly after sunrise, as conditions are changing from stable to lapse conditions, it is probable that strong mixing and tearing of the poorly dispersed cloud may occur with the result that higher concentrations at ground level may be obtained on an intermittent basis than those calculated. The integrated effect of this "fumigation" or "looping" is estimated not to exceed the values obtained using the most probable meteorological conditions.

In summary, the results of our preliminary investigation indicate that containment of the HFIR by means of a controlled leakage-stack release system with filters and adsorption methods of cleaning the contaminated exhaust air is adequate to handle the assumed accident which involves a meltdown of the core.

It is realized that a much more complete study must be made in order to establish detailed criteria for the decontamination system and to determine the consequences of many different accidents. It is believed that a number of factors will be found, some of which will tend to increase the consequence of the accident assumed for our preliminary study; however, it is also believed that there will be much larger factors which will reduce the consequences and result in confirming that our preliminary study is conservative.

5. RESEARCH AND DEVELOPMENT PROGRAM

5.1 Critical Experiments

Critical experiments on flux-trap configurations have been made at ORNL and in the USSR. Although the configurations tested prior to 1959 were not identical with the proposed design, they did indicate the feasibility of obtaining very high thermal neutron fluxes in flux-trap reactors. With no regard to power density limitations, the early ORNL results[22] extrapolated to an unperturbed thermal neutron flux of 3.8×10^{15} at 100 Mw and the Russian results[8] to 4.4×10^{15} at 100 Mw.

Critical experiments were started at ORNL, in late 1959, on a flux-trap configuration having the same geometry as the proposed HFIR. The experiments are intended as a test to determine the validity of the calculational procedure being used to predict HFIR conditions. No attempt was made to "mock up" the HFIR except for geometry and dimensions. The experiment is set up in the form of concentric cylinders having 1/8 in. aluminum wall thickness. The central cylinder is 6 in. OD and is filled with water. The fuel region is an

[22] F. T. Binford, "Flux-Trap Tests," Operations Division Quarterly Report, July-September, 1958, ORNL CF-58-9-88 (Jan. 5, 1959).

annulus approximately 4.9 in. thick formed by the OD of the inner region tube and the ID of the next cylinder. The fuel used is a solution of uranyl nitrate in a mixture of H_2O and D_2O. Surrounding the fuel is a D_2O reflector of 7.9 in. thickness and surrounding the D_2O reflector is an H_2O reflector of approximately 5.9 in. thickness. The fuel region is 17.9 in. high and provision is made for an H_2O reflector on top and bottom.

Neutron flux measurements are made using very small U^{235} fission chambers, small BF_3 proportional counters and foils, all bare and cadmium-covered. These detectors are so arranged that they may be positioned in the various regions in order to observe the neutron flux and power distribution.

The calculated values for the critical mass and neutron flux distribution agree with the observed data quite well. The observed concentration of U^{235} for criticality was 92.6 g of U^{235} per liter of solution with an atom ratio of D/D+H in the solution of 0.958. The calculated value of the fuel concentration to reach criticality was 85 g per liter. The purity of the D_2O in the reflector region was 99.8%.

Critical experiments on a mock-up of HFIR are being planned as part of the program for fiscal year 1961. Present plans are to fabricate a complete fuel loading based on the present design of HFIR and to assemble this with as close a mock-up of the proposed materials for the reactor as is practical. From this mock-up it is expected that complete information can be obtained on the flux distribution, power distribution and perturbations on these caused by the irradiation facilities and the control rods. In addition information will be obtained on the effectiveness of the control rods and it is hoped to obtain the effects of temperature variation in the several regions of interest. This series of experiments will provide an advance test of the design and allow construction of the reactor with confidence that the nuclear parameters are well known.

5.2 Fuel Element Development

The development of fuel elements capable of operating at maximum power densities up to 4000 kw/ℓ and average power densities up to 2200 kw/ℓ is necessary in order to achieve the desired thermal neutron flux at the specified total power of 100 Mw.

All of the present effort is being devoted to development of fuel elements which are basically aluminum-clad, aluminum-uranium core plates. Both dispersion and alloy fuels are under consideration for the fuel-bearing portion of the plate. In order to minimize the peak-to-average power generation it is planned to vary the U^{235} concentration in the radial direction and to add burnable poison primarily to help minimize the variation of power density during burnup of the fuel.

Three types of fuel elements are being investigated: the first is Mark I which can be built at the present time using proved techniques for

fabrication and assembly, but which would probably limit operation in the proposed geometry to less than 100 Mw if conservative operating conditions are to be observed. This element is wedge-shaped and the annular core would be made up of a number of such elements. The fuel plates would form arcs of concentric cylinders and thus every plate in a given element would have a different radius of curvature and different width. This type of element is advantageous in that no problem exists in varying the fuel concentration in the radial direction as each plate is already different. The disadvantages lie in the estimated high cost of fabrication and the fact that the side plates reduce substantially the amount of heat transfer area which can be obtained.

Mark II is an assembly of fuel plates, each formed to the shape of an involute of a cylinder. These plates are held between two concentric cylinders and all plates are the same shape and have the same fuel loading and distribution. In order to limit the plate width the reactor core will be made up of two such concentric fuel annuli.

It is probable that the fuel loadings for initial operation will be of this type. An assembly of this configuration has been made with aluminum plates and, although the water passage tolerances were not held as closely as desired, it is believed that refinements in fabrication techniques will overcome this difficulty. Individual fuel plates have been fabricated with the desired radial variation in fuel loading and no unexpected problems were encountered. This type of element has the advantage that all plates are the same in a given annulus and therefore only two types of plates are required for the entire core assembly.

Mark III core consists of concentric fuel plates in the form of cylinders. The fuel plate cylinders would be formed from large flat plates containing fuel which are rolled up into a cylindrical form. Several problems remain to be solved before such an element could be used. The major difficulty lies in achieving uniform spacing of the cylindrical fuel plates in order to maintain coolant passages and in solving the heat transfer problem in the vicinity of the spacers. This type of core shows considerable long-range promise as it should be simpler and very much less expensive to fabricate than either of the other two types. Work on this version is just beginning, although an aluminum model was made to determine whether the idea was at all workable.

As described under engineering tests a part of the fuel element development program consists of mechanical and flow tests on individual plates and on plate assemblies in order to experimentally verify the strength and stability of the design. In addition a considerable program on corrosion and heat transfer is complementary to the fuel element development effort.

5.3 Engineering Experiments

Corrosion Studies. - Corrosion studies are being conducted to select materials for construction of the HFIR core and to determine as far as

possible (out-of-pile) the expected life of these materials. Of primary consideration is the selection of an aluminum alloy for cladding the fuel elements. As a result of the high-power density of this reactor, the cladding materials will be subjected to very high heat fluxes and high coolant flow rates. Furthermore, the corrosion of aluminum leads to a more or less adherent oxide coating which has a very low thermal conductivity. Thus as corrosion proceeds at constant heat flux, the temperature of the unaffected cladding material increases. It is the primary objective of this program to determine the effects of such variables as heat flux, coolant flow rate, additives to the coolant, alloy composition, system pressure (boiling), and time on the corrosion of aluminum alloys and on the nature of the corrosion-product oxide film.

Several isothermal loop runs were made, using modifications of existing test loops, to determine the corrosion resistance of four commercially available aluminum alloys at temperatures between 338 and 554°F and at flow rates from 20 to 100 fps. (Although the maximum coolant temperature in the HFIR will be less than 220°F, the surface temperature of the aluminum will be considerably higher because of the high heat flux and the poor thermal conductivity of the oxide corrosion product.) During 10-day tests aluminum alloys 1100, 6061, and 8001 were shown to be equally resistant to corrosion up to 500°F, whereas alloy 5154 corroded excessively even at the lowest temperature (338°F). At 550°F only alloy 8001 was satisfactory during a 10-day test. The effect of water flow rate in the range of 20 to 60 fps on alloys 1100, 6061, and 8001 at temperatures as high as 500°F was negligible; maximum penetration (local and general attack) was between 0.002 and 0.004 in. At flow rates greater than about 65 fps the extent of corrosion was much greater on all alloys tested and at all temperatures.

Isothermal loop runs will be continued to evaluate newly developed alloys and to determine the effect of water purity, including intentional additions of acid, on the corrosion of aluminum alloys generally. It is known that the inclusion of low concentrations of phosphoric acid in the water will inhibit corrosion of aluminum at flow rates less than 20 fps, and it is probable that the same effect will be noted at higher flow rates. Furthermore, by inhibiting corrosion, the acid addition may reduce the thickness of the corrosion-product layer and thereby reduce the aluminum temperature substantially as compared with that in an uninhibited system.

To determine the effect of heat flux on the corrosion of aluminum, a test loop in which an electrically heated aluminum section was cooled by flowing water was constructed and several tests were completed. These tests have clearly demonstrated the poor thermal conductivity of aluminum oxide. For example, in one test with a constant heat flux of 1.6×10^6 Btu/hr·ft^2 and constant operating conditions, the outside surface of an electrically heated test specimen increased by more than 200°F during a 10-day test as a corrosion-product layer developed on the inside or water-cooled surface of the aluminum. The thickness of the oxide varied between 0.001 and 0.003 in.

By far the largest part of the program will consist of a continuation of the studies to determine the effect of heat flux on the corrosion of aluminum. The dependence of corrosion on heat flux, water flow rate and temperature, and system pressure will be established for certain commercial aluminum alloys, and later for recently developed alloys, including those with second-phase additions such as iron and nickel and those with extra-low silicon content. To implement this investigation a second loop will be placed in service in fiscal year 1961.

Heat Transfer. - This program is directed toward obtaining the heat transfer and fluid dynamic data needed for the successful design of HFIR cores. Due to the high-power densities involved, the emphasis in heat removal was placed on techniques for assuring that fuel element temperatures will be maintained at acceptable levels. Friction factors, non-boiling heat transfer coefficients, and burnout surface temperatures and heat fluxes have been determined for aluminum surfaces in a parallel, flat-plate system with geometrical parameters identical to that proposed for HFIR cores. The results show that the correlations used for design to this date give, at worst, a conservative estimate of the heat transfer situation. The magnitude of natural-convection cooling in flat-plate geometries, with and without external recirculation, was determined.

Heat Transfer Coefficients. - Thirty-one experimental determinations of average heat transfer coefficients and 248 determinations of local coefficients with test sections duplicating the HFIR core geometry (Al surfaces and 50-mil water gaps) have shown that an average curve through ETR heat transfer data[12] give safe minimum values for design purposes. A slight adjustment of the presently used correlation[12] will be made for final calculations.

Burnout Fluxes. - Seven forced-convection burnout heat flux tests have been conducted for the HFIR geometry. The experimental burnout fluxes are in good agreement (14.8% avg. deviation) with the Soviet Zenkevich-Subbotin correlation,[13] which is presently being used for design calculations. This correlation may be expressed as follows:

$$(\phi_{bo})_{min} = 283\ G^{0.5}(\Delta t_{sub})^{0.33}\left(\frac{\rho_\ell - \rho_v}{\rho_\ell}\right)^{1.8}$$

where $(\phi_{bo})_{min}$ = min. burnout heat flux, Btu/hr x ft^2 = 0.715 $(\phi_{bo})_{avg.}$

G = coolant mass velocity, lb/hr x ft^2

Δt = degree of subcooling, $(t_{sat} - t_{water})$, °F

ρ_ℓ, ρ_v = liquid and vapor densities, lb/ft^3.

Friction Factors. - Experimental determinations of HFIR friction factors for non-boiling conditions have given results in good agreement with standard

correlations for the relative roughness of the test section used.

Miscellaneous. - A few check tests of the intensity of the Al-H_2O reaction at burnout have indicated that this is not a serious problem.

Temperature distributions and burnout heat fluxes for HFIR test sections operating under natural-circulation conditions have been determined; using these data preliminary estimates indicate that if flow recirculation is not highly restricted, the core may be safely cooled by natural circulation alone within half an hour after reactor shutdown.

The work already completed provides a sound basis for design of the HFIR in regard to heat transfer. It is now planned to continue this work in the investigation of special problems which are presently known or may arise during design and to investigate problems in connection with the design of fuel elements of advanced types. The effect of axially and laterally oriented cylindrical spacers on fuel plate temperature distribution and burnout heat flux will be determined for a variety of conditions. An experiment is planned to determine the magnitude of fuel element temperature changes after pump cut-off toward the end of the flow coast-down period when buoyant forces cause a gross flow reversal of the coolant between fuel plates. These tests along with such others as may be found necessary should provide experimental answers to many of the questions which normally fall into the region of considerable uncertainty.

Fuel Plate Stability Tests. - As part of the investigation of fuel element and reactor core design the problem arises of fuel plate stability under the hydraulic forces caused by the high velocity coolant flow. This problem is one in which difficulty has been encountered in both the MTR and the ETR reactors and it is believed that an experimental approach will be the most rewarding, although some effort is being expended in analytical work also.

An experimental set-up has been designed in which a single curved plate of the HFIR-type can be supported in side plates and subjected to a uniform pressure on the convex or the concave surface. Hydraulic pressure is utilized with the water contained in balloons to minimize leakage. Dial indicators and strain gauges are used to measure the deflection and stress, respectively. This work is in a preliminary stage of obtaining data and no generalizations in regard to the data are presently available.

Studies of flow and pressure distribution between individual channels and across the fuel annuli will be made using typical sections. The information obtained from these tests combined with the results from the plate deflection tests will provide a good understanding of the deflections and stresses which will occur due to hydraulic forces in the HFIR. In addition, the results from the flow distribution tests will provide information which is required to support the engineering hot channel factors used in the design studies on cooling.

Hydraulic and Mechanical Mock-up. A full-scale hydraulic mock-up, except for the system pressure, is proposed to demonstrate the operability of the design. This mock-up will include control drive and control rod components. Dummy fuel plates, target rods, and reflector components will be used. The system will be operated at full flow conditions in order to insure the mechanical adequacy of the design.

An existing pumping system used for other hydraulic studies has been modified for use in the HFIR tests. This system will supply up to 16,000 gpm at 100 psi pressure drop. Initially only dummy mock-ups of the fuel region will be tested in a special spool piece in this loop. Later tests will include the mock-up vessel installed for the full core assembly tests.

6. CONSTRUCTION SCHEDULE

6.1 Schedule

Table 4 presents the present schedule for design and construction of HFIR. The estimated cost of the entire facility, including engineering but excluding research and development work, is $12,000,000.

Table 4

Design and Construction Schedule

Description	By	Start	Completion
Title I Design	ORNL & A-E	March 1960	Nov. 1960
Title II Design	ORNL & A-E	August 1960	May 1961*
Advance Procurement	ORNL	Sept. 1960	January 1963
Site Preparation	CPFF	March 1961	June 1961
Construction & Installation	Lump Sum	July 1961	July 1963
Testing and Initial Operation	ORNL	July 1963	Dec. 1963
Power Operation	ORNL	January 1964	

*Exclusive of certain details of core construction by ORNL which will be completed and ready for initial operation, July 1963.

7. REACTOR DATA TABLE

General

Name of Facility -	High Flux Isotope Reactor (HFIR)
Reactor Location -	Oak Ridge National Laboratory Oak Ridge, Tennessee
Owned by -	U. S. Atomic Energy Commission
Operated by -	Oak Ridge National Laboratory
Reactor Type -	Heterogeneous, flux-trap, pressurized water
Designed by:	
Reactor core, control and instrumentation -	Oak Ridge National Laboratory
Buildings, cooling system, etc.-	Architect-Engineer
Design Power -	100 Mw, thermal
Normal Operating Power -	100 Mw, thermal
Average Power Density -	2.2 Mw/ℓ
Average Specific Power -	16.7 Mw/kg
Operating Schedule -	Continuous, with shutdown every 10 days for refueling
Principal use of reactor -	Production of Cf^{252} and other isotopes
To whom available for research -	USAEC Designees

Reactor

Over-all active core dimensions	41 in. OD, 24 in. high including Be reflector
Core-containing vessel	
Over-all dimensions	8 ft OD x 20 ft high
Material	Carbon steel clad with stainless steel on both sides
Design pressure	1000 psig
Operating Pressure	600-900 psig
Mean operating temperature	140°F
Moderator	H_2O
Reflector	Be metal, 1 ft

Reactor (continued)

Thermal shield		Water
Biological shield		Water followed by concrete
Reactor control		
Shim, Safety, Regulating	combined	Thin concentric cylinders around fuel

Nuclear

Reactor power, Mw	100
Maximum unperturbed thermal flux in island, nv	6.6×10^{15}
Maximum fast flux in fuel region	5.7×10^{15}
Maximum unperturbed thermal flux in beryllium side reflector,	
Beginning of cycle	1.0×10^{15}
End of cycle	2.0×10^{15}
Average thermal flux in island experiment (200 g Pu^{242})	3.7×10^{15}
Temperature coefficient of reactivity, $\Delta k/k/°F$	
Water island (no experiment)	$\sim + 1 \times 10^{-4}$
Fuel region	$\sim - 1 \times 10^{-4}$
Void coefficient of reactivity (uniform voids), $\Delta k/k/\Delta\rho/\rho$	
Water island	+ 0.02 for $\Delta\rho/\rho = 0$ 0.0 for $\Delta\rho/\rho = 0.25$ max + $\Delta k/k = 0.003$
Fuel region:	- 0.25 for $\Delta\rho/\rho = 0$ - 0.41 for $\Delta\rho/\rho = 0.35$
Prompt neutron lifetime, μ sec	
Beginning of cycle	50
End of cycle	100
Length of fuel cycle, days	10
Excess k, fresh fuel	~13%
Burnable poison	Boron for obtaining appropriate power distribution

Heat Removal

Fuel plate thickness, in.	0.040	0.050
Power density, Mw/ℓ		
Average	2.2	2.2
Maximum	4.0	4.0
Heat flux (95 Mw removed from plates), $\frac{Btu}{hr\ ft^2}$		
Average	7.7×10^5	8.5×10^5
Maximum	1.4×10^6	1.5×10^6
Ratio of maximum-to-average power density	1.8	1.8
Burnout heat flux, Btu/hr·ft^2		
Nominal	4.7×10^6	4.7×10^6
Minimum	3.5×10^6	3.5×10^6
Burnout heat flux reduction factor	0.745	0.745
Temperatures, °F		
Inlet water	120	120
Outlet water (from fuel region)	183	199
Maximum surface	362	388
Maximum drop across Al oxide film	190	211
Maximum metal	567	614
Coolant velocity, ft/sec	40	40
Total pressure drop across core, psi	57	57
System pressure at pump discharge, psi	1000	1000

Core Design

Type of core	cylindrical annulus, flux-trap
Type of fuel elements	aluminum plates, involute geometry
Fuel plate thickness, in.	0.040 or 0.050
Fuel plate clad thickness, in.	.01 in. aluminum, min.
Coolant channel thickness, in.	0.050
Length of fuel plates, in.	24
Length of active core, in.	18

Core Design (continued)

Inside diameter of fuel annulus, in.	5.5
Outside diameter of fuel annulus, in.	14.9
Volume in fuel annulus (active portion), ℓ	44.8
Outside diameter of outer reflector, in.	41

Core Materials

Fuel	Enriched Uranium (~ 90%)
Fuel loading, kg of U^{235}	~6
Coolant	H_2O
Island moderator-reflector	H_2O
Side reflector	Be + 5% H_2O

Coolant System

Fluid	H_2O
Circulation	
Direction of flow	Downward
Flow induced by	Centrifugal pumps
Normal total flow rate	~ 15,000 gpm
Mean velocity through fuel	40 fps
Inlet temperature to reactor	120°F
Temperature rise, avg.	~ 46°F
Heat Dissipation method	Intermediate heat exchangers and cooling tower
Means of purification	Filtration and demineralization

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