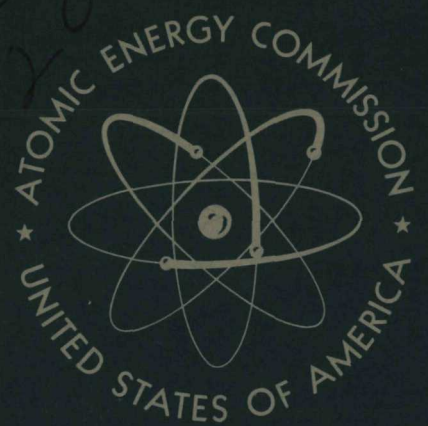

CIVILIAN POWER REACTOR PROGRAM

TID-8518(1)
Book 1

PART III

Status Report on Fast Reactors as of 1959

UNITED STATES ATOMIC ENERGY COMMISSION

DISCLAIMER

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Published : 1960

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SUMMARY

Description and Advantages of Fast Reactors

In the fast reactor, no attempt is made to slow down the neutrons. The neutrons are only slowed down by inelastic collision with structural and fissile materials to about 0.1 to 0.2 Mev. The characteristics obtained by operation at high neutron energy are (1) a small core with high power density, (2) a high breeding ratio, (3) low parasitic absorption of neutrons by structural materials and fission products, (4) freedom from large hot spot effects existent in some thermal reactors, (5) small reactivity requirements for control, and (6) a fuel element lifetime limited only by irradiation damage considerations. Because sodium is used as the coolant, a high thermal efficiency is attainable with low operating pressure for the primary system.

Objectives

Simply stated, the objective of the fast reactor program is to achieve economic power while burning a large fraction of the source uranium.

The reactor would operate on a closed fuel cycle to assure high utilization of uranium. Because Pu^{240} and Pu^{241} will build up in this operation, the fuel material to be handled will be radioactive with α, β, and neutron particles.

A major goal is the development of a low-cost fuel cycle. Because the raw material costs for the fuel cycle are small, the basic operations required to fabricate and reprocess the fuel elements need not be complicated, and the volume of material to be handled is small, low fuel cycle cost should be achievable. The following fuel cycle cost targets are set: (1) 2 mills/kw hr by 1968, and (2) less than 1 mill/kw hr by 1975. To achieve these objectives fuel elements capable of attaining high burnup and also possibly with low fabrication cost should be developed. In order to achieve a high utilization of uranium, the plutonium buildup to cycle loss ratio in the blanket should be above about 4:1. To accomplish this it appears that blanket elements capable of 4 percent Pu buildup per cycle should be developed.

Another major goal is the achievement of a low capital cost for the plant. For the 1968 plant, the following targets are set: (1) Produce steam at conditions achievable in a conventional plant, (2) more fully utilize cheap, high strength materials of construction in the plant, (3) cheapen the heat transport equipment by utilizing the higher rise and greater temperature differences which can be used with sodium, (4) simplify the plant by taking advantage of better design understanding, and (5) improve mechanism by developing better design data and by simplifying designs.

For 1975 plants, develop new reactor concepts such as paste fueled and binary fluid direct cycle plants; investigate major developments for components, such as use of liquid lithium–6 for control; and simplify blanket design.

General Research and Development

Sufficient work has been done to confirm the characteristics of the fast reactor.

Physics.—Critical experiments have been run for moderate size U^{235} fuel reactors, and calculated critical masses have been checked within a few percent. Important reactivity coefficients have been checked by differential experiments. Worth of the control materials for the Experimental Breeder Reactor No. 2 EBR–II and the Enrico Fermi Atomic Power Plant reactors have been checked. These data are sufficient for reactors being constructed. However, much information is re-

quired for future fast reactors. Primarily, these reactors will be plutonium fueled and will be larger. Probably the blanket arrangement and possibly its neutron energy will be different than the present reactors. The following important types of deficiencies exist in the physics area:

(1) Alpha for plutonium in the 0.1 Mev range is not known within a factor of two. This is important to both breeding ratio and control.

(2) Delayed neutron fractions for plutonium-fueled reactors are not well established.

(3) The reactivity coefficients for large Pu fast reactors will not be well known until certain special critical experiments are run. Additional codes may be necessary to achieve satisfactory methods of calculating these coefficients.

Fuels and materials.—The fast reactor is capable of operating at high temperature and to high burnup. It therefore has an excellent economic potential. Little has been done to fully exploit this potential. Data are available on the irradiation stability of alloy systems of the 10 w/o molybdenum class. Perhaps $20 million has been spent developing this class of materials for uranium fuels. The data indicate a moderate degree of success at 2 percent burnup and a temperature up to 1,100° F. The objective is about 5 percent burnup and 1,400° F. for this system. Further work should be concentrated on plutonium alloys, methods of accommodating growth, and on cladding materials capable of high temperature operations.

The AEC has carried out extensive work on cermet fuel elements, but with inert matrix material; 25 percent burnup of the uranium in the dispersed phase has been successfully achieved. This type of fuel element can be adopted to give an economic fuel cycle for the fast reactor, but its breeding ratio will be only about 1.0. Therefore, fuel elements with fertile matrix material, which are capable of breeding ratios above 1.4, must be developed for a fuel temperature of 1,400° F.

The AEC has done extensive work on ceramic fuel systems, particularly the oxide systems. The results are encouraging but do not correspond to the conditions in a fast reactor. For this type of fuel it is desirable to achieve 25 percent burnup of the uranium and plutonium atoms present. The fuel elements should be easy to fabricate and should be high power density elements. Most present programs do not appear to be directed toward these objectives.

Heat transfer experiments.—The basic heat transfer characteristics of liquid metal coolants, particularly sodium, are very good. Because the heat transfer film coefficients are so good, they are not controlling and present heat transfer correlations are adequate. More data are needed in some specific areas such as the effect of thermal shock on fatigue strength of materials and the effect of oxide and other films on the behavior of some fuel elements. The performance of the steam generator is discussed in the section Components and Systems.

Fluid flow.—Because the fluid flow behavior can be correlated with that for other fluids, such as water, the general information available is adequate. Special problems need further investigation. Some of them are: (1) The nature of fog formation in the inert cover gas, (2) removal of vapor in vapor traps to prevent plugging of gas lines, and (3) flow distribution for unique geometries.

Coolant chemistry.—As a result of the operation of large system performances, the general behavior of sodium coolant systems has been determined and is considered to be excellent. This plant operated with no sign of sodium corrosion or mass transport in the sodium system, and it is felt that operation below 1,000° F. has been satisfactorily demonstrated. Further information is desired on some special materials and on operation with a cover gas such as nitrogen. Extensive operation at temperatures of 1,200° F. to 1,300° F. is desired.

Mass transport of carbon is of concern if certain materials like 2½ percent Cr, 1¼ percent Mo steel is used. This is not a problem asso-

ciated with the coolant but with diffusion of carbon out of the chrome-Mo steel and absorption by stainless steel.

More refined methods of purifying sodium, particularly of oxygen, are required. Improved purity inspection is required. An AEC program is under way for oxygen removal. The problem of detecting ruptured fuel elements has not been solved and requires attention.

In general, the basic behavior of sodium has been excellent. It is a rather new coolant; therefore, the broad chemical analyses which have been developed for water have not yet been developed for sodium. Although excellent work has been done on Na-H_2O and Na-Air reactions, further tests should be carried to a more quantitative conclusion.

Reactor safety.—Questions have been raised concerning the inherent safety characteristics of the fast reactor because of the behavior and meltdown of the Experimental Breeder Reactor No. 1 (EBR–I) during special reactor physics tests. For this reason, the many features of fast power reactors related to safety and control have been extensively studied both in the United States and in England. In the United States the bulk of this work has been done by Argonne National Laboratory, Los Alamos Scientific Laboratory, and by Atomic Power Development Associates, Inc. (APDA) and its contractors. The positive temperature coefficient once connected with the EBR–I has been eliminated, and it has thus been established that there are no inherent nuclear characteristics in fast reactors which make them less safe than other types. Consequently, no special or extra costs are involved in providing for the fast reactor the safety features required to contain or otherwise control the hazards of radiation.

Extensive work is being carried out on reactor safety associated with characteristics attributed to fast reactors. The problem receiving most attention is supercriticality and consequent energy release associated with nonmoderated systems with high concentration of fissionable material. The results to date have been encouraging, but analyses have not been extended to plutonium systems nor to the larger reactors of the 10-year program. Safety programs should be reviewed after reference reactor systems are established. In the meantime, generalized studies should be carried out. In addition to the studies on cross sections, alpha, beta, and temperature coefficients of reactivity, further work should be done on bowing of fuel subassemblies and on the nature of plutonium system meltdowns.

Component and auxiliary systems.—Operation of EBR–I, the one-tenth-scale model of EBR–II, and individual component tests have demonstrated that high temperature, compact reactors cooled with a coolant which reacts with air can be successfully operated. The remarkable situation is that the sodium-cooled reactors designed to date have been designed on the basis of "make it work" rather than for economy. In spite of this, sodium-cooled reactors are comparable in first cost with other reactor types. What has been demonstrated is that reactor mechanisms can be operated in sodium and sodium vapor and that heat exchangers can be designed for radioactive sodium service. Preliminary studies have been made which show that substantial improvements can be made in fuel handling equipment and other mechanisms if a better understanding of the performance of these mechanisms in high-temperature sodium can be obtained. One example will illustrate the point. Because of early experiments, it was considered that bearings operated in sodium should be loaded to only one one-hundredth the load for a bearing operated in oil. The British found they had to operate at 10 times this load; and their experiments showed that as long as sodium oxide (particles) was kept out of the bearing, the bearing worked fine at the higher load. The way to keep the sodium oxide out of the bearing was to examine it infrequently, thereby minimizing exposing the sodium to air.

An illustration of the improvement that can be made in the intermediate heat exchangers is the comparison of the heat exchanger design used for the Enrico Fermi plant with the type which resulted from the AEC high-temperature

components study. The price can be cut almost in half, and a savings of over $1 million can be effected in a 300-Mwe plant. Two things were done: A better tube header arrangement was designed and the shielding was located so as to simplify the heat exchanger.

This type of improvement is possible in other areas. Design data on methods of removing heat from fuel elements can greatly simplify the fuel handling and decay system.

For the Enrico Fermi plant, APDA developed a single-walled, once-through-type steam generator. Although close to $1 million was spent on this development, more than this amount was saved by its use in the Enrico Fermi plant. Long-term performance tests are desired on any steam generator used. Water side corrosion information is particularly desired.

The maximum operating temperature to which sodium can be successfully handled without corrosion in standard materials of construction has not been established: it is believed to be about 1,200°–1,300° F. Reactor systems which have been operated or which are under construction have been limited to 900° F., except for the short-time Sodium Reactor Experiment (SRE) demonstration at 1,000° F. The potential of sodium systems is, therefore, not being fully utilized at this time.

Reactors

EBR–I.—Operation of this reactor, which was the first reactor to produce power, began in December 1951. As a result of its operation, the principles of operation of a fast reactor cooled with NaK were successfully demonstrated. It fully achieved its objective. Considerable effort was expended to obtain refined physics information for which it was not designed; however, it essentially has been done. Plans are now under way to load the reactor with plutonium.

EBR–II.—This reactor is a 60-Mwt, 20-Mwe fast reactor with an integrated pyrometallurgical processing plant. It is now being constructed at the National Reactor Testing Station (NRTS) and is expected to go critical in December 1960. The design and construction is progressing satisfactorily, and no unforeseen problems exist. The fuel elements are one-sixth-m.-diameter fissium alloy which is sodium-bonded to stainless steel tubes. Control is by fuel movement. The major components and the fuel elements have been proved by separate tests. This plant will be operated on a recycle basis and ultimately with plutonium fuel.

Enrico Fermi reactor.—This developmental reactor is designed for initial operation at 300 Mwt and 100 Mwe. The sodium temperatures are 550°–800° F. for initial operation and 600°–900° F. for final operation at high power outputs. The fuel elements are one-sixth-inch-diameter 10 w/o Mo-U alloy which is metallurgically clad to zirconium. Offsite aqueous reprocessing will be used. The primary system of the plant is essentially complete, the preliminary operation of mechanisms has begun. The erection and preliminary operation has proven to be satisfactory. The primary system will be operated with dummy fuel elements as a nonnuclear test facility for 1 year with sodium at plant temperatures. This test is expected to confirm mechanical and hydraulic operation of the primary system and to demonstrate its integrity. The plant is expected to go critical in the fall of 1960. The operation of this plant and the EBR–II plant should demonstrate the feasibility and practicality of these plants operating on U^{235}.

Other reactors.—The British are constructing a reactor of the same size as EBR–II. It attained criticality in November 1959. The Russians are operating one 5-Mw reactor for fuel development and have started the design of a larger reactor.

DESCRIPTION OF FAST REACTOR TYPE

In the fast reactor, no attempt is made to slow down the neutrons. The neutrons are only slowed down by inelastic collision with structural and fissile materials to about 0.1 to 0.2 Mev. The advantages of operation at higher energy are the small neutron absorption of structural material and fission products, the large number of neutrons produced per absorption in fuel, and the high fast fission effect. These result in the ability to attain breeding ratios of the order of 1.2 in U^{235} fueled reactors, 1.4 in plutonium fueled reactors, and about 1.3 in U^{233} fueled reactors. The small cross sections of structural materials make it possible to choose materials independent of their cross sections; therefore, stainless steel and molybdenum are used without restriction. The small cross section of fission products and the modest reactivity coefficients make possible the operation of a fact reactor with only a few percent excess reactivity. This fact, plus the large mean free path for absorption, permits such a reactor to operate with only two control rods. To maintain a high neutron energy, moderating materials are avoided as the coolant; therefore, water and normal organic coolants are not used. Sodium is used, although other liquid metals might be considered, as is indicated in Report ANL-4312. Because sodium boils at 1,620° F., a reactor can be operated at high temperature, with low pressure, thereby providing a means of minimizing the cost of the heat transfer system. The high coolant temperature that can be used provides a means of attaining a high cycle efficiency. Its high conductivity, reasonable specific heat, and its low cost of only 40 cents per quart make sodium a good heat transport medium. It does react with moist air and water, and these factors must be taken into account in design. Sodium becomes radioactive and, therefore, several feet of shielding is required for the primary system. In order to prevent the release of radioactivity, in case of a rupture of a steam generator tube, an intermediate link is used.

Because the cross section of fissionable material is only several times that of fertile material, the equivalent enrichment of the core material of 10 percent must be used. In order to minimize inventory charges and to attain a short doubling time, fast reactor cores are usually made small. The initial fuel loading for the Enrico Fermi reactor is only 2.5 feet in diameter by 2.5 feet high. To achieve the high power density which results from the use of small cores, the fuel elements are finely subdivided. The size is the equivalent of about one-sixth-inch diameter rods. In this type of reactor, the fuel element lifetime is only limited by the effect of irradiation damage or swelling on heat transfer and by the release of radioactivity to the coolant. As a result, high burnup is sought. When high burnup or cheap reprocessing and refabrication is achieved, very low fuel cycle costs can result. For example, the achievement of a burnup [1] of 25 percent in a ceramic system makes a fuel cycle cost of around 1 mill/kw hr attainable.

To simplify the presentation on the fast reactor, a fuel cycle using uranium fertile material and plutonium fuel is described. The fast reactor can be operated so that it will breed using U^{235}, Pu or U^{233} fuel, and using either U^{238} or thorium fertile material. The mechanical design, thermal operating conditions and heat transport systems for these different fuel and fertile material are the same. The fuel elements may be different.

[1] Burnup as used here means the percent of total Pu and U atoms in the fuel which have fissioned.

OBJECTIVES

Simply stated, the objective in developing the fast reactor is to achieve economic power by burning a large fraction of the source uranium.

Closed Fuel Cycle

The reactor will be operated on a closed fuel cycle to assure high utilization of uranium. Therefore, the fuel feed is assumed to be natural (or depleted) uranium. This material is continuously recycled until it is essentially all burned. The equilibrium fuel fed to the core is alpha and gamma active because of the Pu^{240} and Pu^{241} present, even if complete decontamination is used. Therefore, at least glove boxes are required in handling fuel elements, and shielding is needed for handling subassemblies. Beyond the material required to get started, the fast reactor is free from the need of a separate source of plutonium or U^{235} from an isotope separations plant and requires only a very small feed of uranium.

Low Fuel Cycle Cost

For fuel cycle cost, the target is 2 mills/kw hr by 1968 and under 1 mill/kw hr by 1975. The potential cost is lower than this. Such low-cost fuel would make possible the use of nuclear energy as the prime supply of energy, whether it be for lighting, heating homes, or making steel. It is pointed out that the burnups required to achieve low fuel cycle costs have been achieved; however, they must be demonstrated under fast reactor operating conditions.

The significant characteristics of the fuel and blanket elements being considered and the objectives in their development are summarized:

(1) Uranium-plutonium alloy fuel elements similar to the pins used in EBR–II. The alloying material would be nonvolatile fission products that would remain as a result of partial decontamination, and other additives, such as molybdenum, added to improve irradiative stability at elevated temperature. This type of fuel element is of interest because it can be operated at high power density, giving a good breeding ratio and short doubling time. Furthermore, the fuel can be reprocessed by simple pyrometallurgical techniques, holding promise of a cheap fuel cycle.

(2) Ceramic fuel element, such as mixed oxides of uranium and plutonium and also separate oxides of uranium and plutonium. These are of interest because they can be operated at high temperature and reasonable power density and because of their potential to accommodate local fission product damage. Because of the possible difficulty in fabricating the small diameter pin elements required for acceptable doubling time of the oxide element, alternate designs, such as "radiator-type," should be considered.

(3) Cermet fuel elements of a dispersion of plutonium oxide or other ceramic body in a matrix of fissile alloy material. The matrix should be capable of operating at high temperature and of restraining the growth of the dispersed material.

(4) Blanket elements. In general the power density in blanket elements is much lower than in the fuel elements, and the concentration of fuel is low. Blanket elements are, therefore, much cheaper to fabricate than fuel elements. Development work on fuel elements for thermal reactors may only require slight extension to apply to these elements. Designs capable of operating to a buildup of the order of 4 percent of fuel atoms is desired to minimize cycle losses.

Low Capital Cost

For the 1968 target, the features of the sodium-cooled fast reactors that characterize its economic potential must be more fully exploited. The major objective to date has been to demonstrate the satisfactory operation of prototype plants rather than the full utilization of their economic features. The short-range objective is to evolve plants whose capital cost will not be more than 15 percent higher than conventional coal plants by 1968. Preliminary studies indicate that this is possible because of some of the features of sodium-cooled fast reactors; some of these are:

(1) The use of sodium as a coolant makes possible the production of steam at, or higher than, the conditions achievable in a conventional plant. Thus, a high electrical output is achievable for a given thermal reactor rating.

(2) Cheap materials can be used: The small absorption of structural material gives a wide selection of reactor materials. The noncorrosive nature of sodium also permits the choice of system materials based on their high temperature mechanical properties.

(3) Low pressure operation of the primary and secondary loops makes possible the use of thin-walled vessels.

(4) The ability to operate with a high temperature rise and with a high temperature drop through exchangers and with high heat transfer coefficients minimizes the heat transport equipment, in spite of the fact that an intermediate link system is used.

For the 1971 objective, new reactor concepts should be developed. The objective is to develop plants whose capital costs are equal to or less than conventional plants. To do this, reactors, such as a paste-fuel reactor or an evaporatively cooled, binary fluid reactor, should be investigated.

GENERAL RESEARCH AND DEVELOPMENT

(Completed and Under Way)

General Physics Features of Fast Reactors

The neutron energy range of greatest interest to the fast reactor designer is a very wide one, extending from about 50 Kev to several Mev. In this energy range absorption cross sections for almost all materials of interest are about two orders of magnitude smaller than in the case of thermal neutrons. For example, the fission cross section of U^{235} is 582 barns for thermal neutrons but only 1.5 barns for 0.2 Mev neutrons. Likewise a comparison of the fission product cross-section for the two energies shows 77 barns per fission for the thermal case but only 0.07 barn per fission for the 0.2 Mev case.

The value of α (alpha), the capture-to-fission ratio of fissile material is of great importance to breeders. The following table illustrates approximately some of the important features of the variation of alpha with energy:

Energy	U^{235}	Pu^{239}	U^{233}
Thermal	0.18	0.42	0.12
10^2 ev	.52	.72	.12
10^3 ev	.43	.6	---
10^5 ev	.23	.28	.05
10^6 ev	.10	.05	---

The value of alpha for Pu^{239} is not well known below 0.2 Mev and reported values differ by factors of two for energy regions where this property is of great importance. The flat and low characteristic of alpha for U^{233} over a wide range of energy is of considerable interest.

With appropriate design a very appreciable fraction of the fissions in fast reactors can be made to occur in U^{238} or in Th^{232}; for instance, in the Enrico Fermi reactor about 15 percent of all fissions occur in U^{238}. This feature is of importance since it permits the direct use of fertile material as well as producing neutrons to enhance the breeding characteristics of the reactor.

In fast reactors fueled with U^{235} the effective delayed neutron fraction is slightly less than in U^{235} fueled thermal reactors since the delayed neutrons have somewhat less reactivity importance due to their inability to cause fission of U^{238}. The large number of fissions which can be caused to occur in Th^{232} or in U^{238}, which have larger delayed neutron fractions than U^{233} or Pu^{239}, result in an effective delay fraction in fast reactors larger than that in thermal reactors in cases where either Pu^{239} or U^{233} is the fuel.

The average neutron lifetime for all the fast reactors is not appreciably different than that for thermal reactors, being about 0.1 second. The prompt lifetime, however, is appreciably less, being of the order of 10^{-7} second as compared to values in thermal reactors ranging from 10^{-3} to 10^{-5} second.

In fast reactors local inhomogeneities are not important due to the long mean free path of fast neutrons as compared to that in certain thermal reactors.

Physics Experiments and Calculations

Cross section data.—Total cross sections are generally based on those reported in BNL–325 but modified by transport approximations. Scattered neutron angular distributions, elastic and inelastic, are obtained from various sources (1–15, –33, –67, –100, –109).

Fission cross sections are usually based on BNL–325. Neutron yields are a modification of measurements (1–111) and calculation (1–112).

Parasitic capture cross sections are obtained from BNL–325, recent work by Diven (1–53)

and from Zero Power Reactor III (ZPR–III) experiments (1–45). Radiative capture cross sections of primary fissionable material are based on EBR–I experiments (1–102) and other measurements (1–53).

Inelastic cross sections for lighter elements are generally based on known energy level measurements (1–115). Some inelastic cross section data are obtained from Los Alamos experiments (1–68).

The major problems associated with our current knowledge of fast neutron cross sections are those of precision and difficulty of experimental determination. Many pertinent cross sections have been measured but not with the precision needed by reactor design groups. Other cross sections have defied experimental attempts at determination. Theoretical methods give qualitative results but generally with appreciably less precision than successful experiments.

Papers submitted to the 1958 Geneva Conference more or less summarize the better known pertinent experimental results. Since the conference, most of the pertinent activity in this field may be summed up by more capture cross section determinations, some additional angular distributions and some efforts at determining neutron yield per fission, all as a function of neutron energy.

Important cross sections requiring improved precision or a first measurement include:

(1) Low energy inelastic scattering in U^{238}.

(2) Inelastic scattering in thorium.

(3) Alpha of Pu^{239}, U^{235} and U^{233}, particularly below 200 kev.

(4) Capture in U^{238} below 500 kev.

(5) Fission in Pu^{241}.

(6) Capture in Pu^{240}, Pu^{241}, Pu^{242}, U^{234}.

(7) $\nu(E)$ for all fissionable isotopes.

(8) Capture and inelastic scattering in potential structural, alloying and coolant materials, where not fully measured.

Critical experiments.—A large number of elementary fast critical assemblies with a high core density have been operated at the Los Alamos Scientific Laboratory (LASL) (1–44, –62, –63, –64, –65).

(1) Godiva — sphere — unreflected — 93.5 percent enriched with 49 kg of U^{235}.

(2) Topsy—sphere—21 cm natural uranium reflector—93.5 percent enriched core—16 kg U^{235}.

(3) Jezebel—sphere—unreflected plutonium core—16.45 kg. Pu.

(4) Popsy—sphere—21 cm natural uranium reflector—plutonium core—5.79 kg Pu.

(5) Jemima—L/D 0.58—unreflected—53.6 percent enriched with 89.5 kg U^{235}.

(6) Jemima—L/D 1.00—unreflected—37.7 percent enriched with 101.0 kg U^{235}.

(7) Jemima—L/D 1.08—unreflected—29 percent enriched with 123 kg U^{235}.

(8) U^{233}—sphere—unreflected core with 16.2 kg U^{233}.

The data from these assemblies furnish a foundation for predicting behavior of more complex systems. Absorption and transport cross section data, U^{235} to U^{238} fission rates, delayed neutron yields, influence of shape or size, effect of composition perturbations on critical size, reflector savings, and prompt neutron lifetimes, are some of the useful data obtained.

The British fast critical assemblies are Zephyr and Zeus (1–3 through 1–7). Zephyr is a plutonium fueled assembly used to extend existing data and to obtain more knowledge of plutonium phenomena. Zeus is a U^{235} fueled mock-up of the Dounreay fast reactor.

The ZPR–III critical assembly at National Reactor Testing Station, Idaho (1–44 through 1–46) is a flexible dilute core assembly permitting investigation of a wide range of sodium-cooled fast reactor designs. At least 12 ZPR–III assemblies with a U^{238} to U^{235} ratio ranging from 0 to 7, L/D of 0.88 to 1.04, graphite composition from 0 to 0.734, and various critical masses have been constructed.

Comparison of several different means of calculating critical masses for various assemblies of different U^{238} to U^{235} dilution ratios indicates that though the predictions throughout the entire range may not be of great accuracy

the correlation is good and useful in obtaining a realistic critical mass estimate. For a 3 to 1 ratio and a 0.88 L/D the measured critical mass of 151.9 kg U^{235} compares with 155 kg calculated, but with a 7 to 1 ratio and a 0.88 L/D the measured 240.6 kg compares to 292 kg calculated. The experiments indicate that predictions of fission ratios, in general, are accurate.

Central material replacement experiments utilizing Pu and U^{233} indicate that calculated reactivity effects of replacement check well with the experimental results, considering the lack of detailed cross-section information. These replacement experiments provide nuclear behavior data of Pu and U^{233} since dilute critical assemblies of these two materials are not available. For example, the experiments indicate that the calculated quantity $(\nu-1-\alpha)\sigma$, the relative effectiveness, in barns of fissionable material, averages 3.34 for Pu, 2.06 for U^{235} and 3.39 for U^{233}, and is in good agreement with experimental results.

Calculated prompt neutron lifetimes compare well with experimental results using Rossi α measurements except for more dilute systems with softer spectra.

Argonne National Laboratory has done considerable theoretical work, some very preliminary design work, and experimental work to the extent of a series of critical experiments on the fast-thermal coupled reactor. The general objective of such a system is to obtain a prompt neutron lifetime characteristic of a thermal reactor concurrent with the breeding characteristics of a fast reactor. The reactor is essentially a four-region device: A fast core, an inner blanket in which thermal fissions take place, a moderating region surrounding the inner blanket, and an outer blanket surrounding the moderating region.

Engineering of fast-thermal coupled reactor concept has not been carried far enough to allow judgment of its potential, but many of its principal problems are common to those of the simple fast reactor. Others, such as fission product poisoning effect in the thermal region, are unique and as yet not evaluated.

Calculations.—Analysis of fast reactor systems rests on solutions of the Boltzmann equation (1–106). A solution based on a transport approximation, the Sn method (1–54), is useful for spherical geometry. A second method based on diffusion theory (1–14, 1–41) is adequate to describe the nuclear characteristics of large dilute fast breeders. This last method lends itself to separability in simple geometry, and also to digital computation. A third method uses asymptotic calculations (1–41) which can generate multigroup spectra without high speed machines. For a 400-liter metal core the deviation in the critical radius between the Sn and diffusion methods is 0.016. For an 800-liter oxide core this deviation is 0.013. For a similar core but less dense reflector the deviation is 0.020, indicating diffusion theory neutron leakage is not adequately described.

Analyses for EBR–II indicate that one-dimensional analyses give critical masses 10 percent higher than two-dimensional. This is mainly due to reflector nonuniformity and inclusion of nonfertile material. The two-dimensional diffusion calculations are adequate for L/D ratios from 0.213 to 2.13 as determined by the constancy of k_{eff} which stayed in a range 0.947 to 0.951.

Calculated spatial neutron flux variation is well within experimental uncertainties.

Determination of temperature and/or power coefficients must rely first on theory, and finally on the actual reactor, since measurements on zero power reactors are not possible. Some indirect information from material replacement experiments and small perturbations on core geometry can be obtained. Calculation by two-dimensional diffusion theory is expensive. One-dimensional nonspherical calculations are not adequate, but multiregion spherical calculations provide satisfactory results for most nonspherical systems. This has been demonstrated in EBR–I Mark III loading (1–48) and EBR–II (8–8, 8–30).

The type of multigroup code program used depends on the purpose. Engineering design analysis with fixed dimensions requiring varied compositions for fuel alloy enrichment determination can be obtained from a UNIVAC one-dimensional code, with up to 20 groups, and with 6 inelastic transfers. An IBM 704 and a Datatron can also be used. An SNG program (1–54) using an IBM 704 provides for attainment of criticality by varying dimensions, for studies of heterogeneities, and for corrections to diffusion theory calculations (1–45, 1–54).

Two group calculations using a MUG–II UNIVAC program can account for geometric effects. This can also be done by CURE and PDQ programs using IBM 704 machines. The Nick–II–IBM 704 program provides for two-dimensional inelastic scattering to more than one group. It also provides for more accurate power distribution analysis. PROD–II using an IBM 650 is used widely and most one-dimensional diffusion theory programs are based on it. The Hobo–IBM 650 program provides for determination of reflector savings, extrapolation distances, and equilibrium spectral effects.

Fuel and Materials Properties

ANL fuel.—Research at ANL on solid fuel has included investigations on metal and on ceramics (4–56). Most of the irradiations have been on 1.0-inch long and 0.125- to 0.250-inch diameter specimens.

Dimensional stability is of prime importance. The growth coefficient is a good index of relative dimensional stability between different materials. This coefficient is defined as:

$$G_1 = \frac{ln\ L/Lo}{\text{fraction of total atoms fissioned}}$$

Where length changes are small, or where G_1 varies with burnup the approximation

$$G_1 = \frac{\text{percent length change}}{\text{atoms percent burnup}}$$

is used.

The G_1 for unalloyed wrought uranium varies from 25 for 0.5 a/o burnup 300° C. rolled and beta quenched material to 690 for 300° C. rolled material. Values for some alloys are given in table 1.

The tests indicate that unalloyed uranium even with the best known metallurgical treatment is not stable to burnups as low as 2 a/o. The deleterious changes which occur in unalloyed materials at moderate irradiation temperatures are evidenced principally as surface roughening and anisotropic growth. Certain residual alloying additions have been known to greatly refine the grain size in uranium, thus reducing surface roughening. Furthermore, alloying additions can alter the transformation kinetics of uranium so that the effects of preferred orientations resulting from rolling or extruding fabrication procedures may be largely removed by heat treatment. For alpha-phase alloys, these heat treatments usually consist of holding the material briefly in the gamma phase, followed by an extended isothermal anneal to permit the alloy to transform completely to alpha. For alloys which can retain the gamma phase on cooling, the preferred heat treatment is usually a quench from the gamma phase, which may be preceded by a homogenizing anneal at gamma temperatures. Both uranium and uranium-base alloys show definite irradiation temperature limitations, above which swelling occurs. Thorium and thorium-uranium alloys show dimensional stabilities which appear to equal those shown by the metallic uranium fuels. Oxide fuels are generally characterized by relatively good stability under irradiation. Pellets of ThO_2 with additions of UO_2 show negligible dimensional changes after burnups of the order of 1 percent of the metal atoms with central temperatures of 2,000° C. or more during irradiation.

The pyrometallurgical processing at ANL builds up a group of fission product elements called "fissium" which are difficult to remove economically. A program of testing the U-Fs alloys indicates that the as-cast alloy gave relatively good stability under irradiation when the fissium was gamma quenched. A fabrication process has been delevoped for a relatively sim-

TABLE 1.—G_i VALUES FOR SOME ALLOYS

	Burnup, a/o	G_i	Note [1]
(1) Alpha phase uranium alloys:			
(a) U-Cr, 0.1 and 0.4 w/o Cr	Up to 0.65	>25	
(b) U-Mo, 1 to 3.5 w/o Mo	Up to 0.5	1.5 to 20	
(c) U-Pu, 3.7 to 18.7 w/o Pu	Up to 0.84	10 to 100	
(d) U_3Si, 3.8 w/o Si	Up to 0.8	−3 to +4 cast 30 extruded	Had large diameter change.
(e) U–1.62 w/o Zr casting	Up to 1.6	3.8	Large diameter changes above 600° C.
(f) U–2 w/o Zr wrought	N.A	470 unquenched. 64 quenched. 3 tempered after quenching.	
(g) U–1 w/o and 2 w/o Zr rolled and annealed.	N.A	−300 with 20 ppm carbon. −200 with 400 ppm carbon. +150 with 4,000 ppm carbon.	
(h) U–5 w/o Zr–1.5 w/o Nb	N.A	150 as swaged. 300 gamma quenched. 5.4 24-hour isothermal transformation at 650° C.	
(2) Gamma-phase uranium alloys:			
(a) U–3.17 w/o Fissium	N.A	50 gamma quenched. −0.1 24-hour isothermal transformation at 650° C.	
(b) U–5 w/o Fissium	N.A	−0.1 gamma quenched	
(c) U–5 w/o Fissium 2.5 and 7.5 w/o Mo.	N.A	2 quenched or slowly cooled	
(d) U–20 w/o Pu–5.4 w/o Fissium and 10.8 w/o Fissium.	N.A	1.8 as cast	
(e) U–20 w/o Pu–5.0 w/o Mo		3.2 as cast	
(3) Thorium:			
(a) Th–0.1 to 5 w/o U–235	Up to 4.4	0.065 to 0.85	Up to 200° C.
(b) ThO_2–2.5 w/o UO_2	Up to 0.75	0.02	Up to 1,500° C.
(c) ThO_2–10 w/o UO_2		0.02	

[1] Unless indicated to the contrary, all irradiations were carried out at temperatures of less than 550° C.

NOTE.—N.A. signifies "not available."

ple injection casting technique for this material. A program of determining the best fissium composition has been in progress. The reference is U–5 w/o Fs. Swelling at 0.5 a/o burnup and 700° C. is small. A Pu–fissium program will be carried out in the same manner as the U–fissium program, beginning in 1960.

APDA-sponsored fuel (4-19, 4-21).—Through 1958 approximately 175 specimens have been irradiated in the Materials Testing Reactor (MTR) and 3 full-length pins in the Argonne Research Reactor (CP–5) in support of the Enrico Fermi plant. The first phase of the program consisted of alloys of U-Cr, U-Zr, U-Mo. The U–5 w/o Cr eutectic alloy did not have radiation stability, and irradiation of U-Zr alloy specimens containing 2, 2.2, 3, 5, 10 and 15 percent by weight of Zr indicated that the radiation stability is inferior to the U-Mo alloys containing 3.5, 5, 7 and 10 percent by weight of Mo. U-Mo alloy containing 10 percent Mo was selected as the reference fuel alloy

for the first two core loadings of the Enrico Fermi reactor. On length Changes, G_1 values range from 0.4 to 1.1 at temperatures below 1,100° F. Test results show that fuel pin swelling is not sensitive to irradiation temperatures below 1,100° F. for U–10 w/o Mo. Post irradiation heating of the reference alloy irradiated below 1,100° F. and heated for 100 hours at 1,300° F. resulted in diameter increase directly proportional to the square of the burnup. Post irradiation measurements of density and other physical properties have been conducted on reference specimens irradiated to beyond 2 total a/o burnup. Transformation kinetics have been extensively studied for the reference alloy and indicate that the retained gamma structure of the reference alloy is more stable than the partially transformed or the transformed alpha plus epsilon structures. The radiation stability of the gamma treated reference alloy is good to 2 a/o burnup below 1,100° F. and to 0.5 a/o burnup below 1,350° F. Large physical changes, such as swelling or ruptures, have occurred at less than 1.5 a/o burnup when irradiated above 1,100° F. Further work is continuing to determine the threshold fission rate necessary to maintain the reference alloy in the gamma phase for temperatures of 800° F. to 900° F.

The fuel fabrication program for the Enrico Fermi reactor has gone through three experimental phases. In the first phase the direct casting of U–5 w/o Cr alloy and the direct casting of U–2 w/o Zr alloy were terminated because of radiation stability, the undesirable restrictive requirements of the direct casting procedure, and lack of reprocessing technology. Sodium bonding of the U-Cr and U-Zr elements was also attempted and discontinued because of the poor irradiation instability of these alloys. Low alloys of molybdenum and niobium were included as well as eutectic alloys of chromium. Fabrication of core elements included rolling, extrusion, drawing, hot mold casting, lost wax casting, and hot pressing. The irradiation results on these alloys were not satisfactory.

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In the second phase of the program heat fluxes were reduced and more emphasis placed on radiation stability. A U–2 w/o Zr alloy was fabricated as a metallurgically bonded coextruded pin and irradiated. Dimensional stability was poor.

The third phase tests indicated that stabilized alloys of the U-Mo system of 9½ to 13½ percent w/o Mo had radiation stability superior to previous alloys. The decision was made in December of 1955 to proceed with a U–10 w/o Mo metallurgically bonded pin. Extensive fabrication experimentation was conducted to determine the most feasible method of fabricating the present reference pin.

Work has been done on uranium-molybdenum alloy-UO_2 dispersions. Fabrication procedures, including the production of base-alloy powder by a hydriding technique, were established and tensile-strength and thermal-expansion data were obtained for U–3.5 w/o Mo plus 27 w/o UO_2 dispersion plates. A technique for powdering U–10 w/o Mo alloy by hydriding was developed and unclad U–10 w/o Mo plus 27 w/o UO_2 dispersion plates were fabricated for irradiation in the MTR. Duplicate sets of unclad specimens were irradiated to approximately 1 and 2 total a/o burnups at calculated center-line temperatures of 650° to 800° F. Neither the UO_2 fuel dispersion nor the U–10 w/o Mo matrix showed any gross damage attributable to irradiation. Density values, calculated from measured dimensions, decreased an average of 1.5 percent per a/o burnup. Dispersion plates of U–10 w/o Mo plus 27 w/o UO_2 were successfully clad by gas-pressure bonding with both molybdenum and niobium; however, no clad specimens were irradiated.

In September 1956, the APDA Core II program to develop and study improved reactor cores was started with Battelle Memorial Institute because it was known that the present core would result in very high operating costs for the Enrico Fermi Atomic Power Plant. The initial aims of the program consisted of three separate phases: (1) design and fuel cycle cost evaluation, (2) research and devel-

opment, and (3) production of one full or partial core loading for test purposes in the Enrico Fermi reactor. Several major revisions to the original ground rules were considered, and this work, constituting Phase I of this program, was completed in July 1958.

The specific objectives of this work were twofold: (1) To develop an inexpensive fuel cycle for advanced fast breeder power reactors and (2) to produce a low-cost fuel cycle for the Enrico Fermi reactor. The scope and schedule of the program were established in order to provide the sixth core loading for the Enrico Fermi reactor. Uranium 235 was assumed to be the fissionable fuel material, the reactor size was equivalent to a Fermi I 139-subassembly core, all reprocessing costs are assumed to be those in the Power Reactor Development Company (PRDC)–AEC contract or similar, and the blanket design was assumed to be the same as the present sodium bonded blanket. The emphasis in Phase I was on fuel element design, materials evaluation and fuel cycle cost studies to determine the optimum economic core design parameters. All core physics calculations were performed by the APDA Nuclear Engineering Section. Three reports were required because of successive changes in the ground rules of fuel cycle comparisons. The first report was prepared for an evaluation of core fuel systems of a second Fermi reactor for 616 Mw core heat output with a 139-subassembly core size limitation, plutonium revenue at $44 a gram and an optimistic estimate of reactor plant costs. The Addendum I report summarized core fuel cycles at 300 Mw electric output with plutonium revenue at $30 per gram. The Addendum II report showed a comparison of fuel cycle costs for a new reactor with core size variable to be economically optimized for each core material considered. The requirement that the reactor have a conversion ratio greater than one was eliminated in the last revision to the ground rules, since for U^{235}-fueled fast reactors, economic considerations would determine what the conversion ratio should be.

While the fuel cycle cost studies were made on the basis of a new, conventionally financed power reactor with 12.83 percent fuel inventory charges, APDA calculations indicate that the core fuel system thus selected would also be the least expensive for Fermi I operation.

Evaluation of potential fuel systems under the first Phase I ground rules led to the following materials and designs as showing the most promise:

(1) Alloy fuels—U–10 w/o Mo or gamma-phase-type alloy clad in zirconium in a flat plate subassembly design.

(2) Ceramic fuel—90 percent dense UO_2 pellets in sodium-filled stainless steel pins. A parallel plate design utilizing a stainless steel radiator-type assembly was economically attractive but would require release of fission gases to the main coolant stream. As a result, this design would require development work beyond the scope of the original program at Battelle.

(3) Dispersion or cermet fuels—A 30 v/o dispersion of UO_2 in U–10 w/o Mo or gamma-phase-type alloy, zirconium clad, in a flat plate subassembly design.

In the portion of the study concerned with revision 2 (addendum I) of the ground rules, the same fuel systems as given above were considered. Uranium monocarbide (UC) was considered both as a ceramic fuel and as a dispersed fuel in a U–10 w/o Mo matrix. Although under the revised ground rules the economics were not as favorable as under the original rules, Battelle Memorial Institute (BMI) felt the cermet system was the most promising.

When core size and conversion ratio limitations were removed, it was possible to consider cermets with nonfuel matrices. The cermet UC dispersed in Inconel X was considered to be very promising. It and UO_2 pins were recommended as the fuel systems for future development.

Table 2 shows a comparison of the six fuel cycles of primary interest. The four ceramic systems using UO_2 or UC in pin or plate geom-

TABLE 2.—APDA CORE II FUEL SYSTEMS

	UO_2 ceramic plates	UC ceramic plates	UC ceramic pins	UO_2 ceramic pins	U–10 w/o Mo alloy plates	UC in Inconel X cermet plates
Reactor power, Mw	281. 2	278. 0	276. 2	277. 2	279. 9	281. 2
Load factor, percent	75	75	75	75	75	75
Core diameter and length, inches	49. 8	49. 8	62. 3	62. 3	43. 5	43. 5
Sodium temp, of—In	450	550	600	600	550	550
Sodium temp, of—Out	800	900	950	950	900	900
Number plates or pins/subassembly	29	10	784	784	27	18
Plate thickness or pin diameter, inches	0. 062	0. 250	0. 146	0. 146	0. 066	. 125
Critical mass, kg–25	871	1, 400	1, 850	1, 533	1, 038	550
Conversion ratio—Core	0. 43	0. 646	0. 649	0. 57	0. 56	. 13
Conversion ratio—Blanket	0. 61	0. 508	0. 446	0. 47	0. 58	. 79
Burnup:						
U^{235} atom percent	[a] 58. 5	94. 2	95. 0	77. 5	23. 4	84. 0
Total atom percent	3. 95	6. 15	6. 95	4. 23	2. 62	3. 0
Core life, days	769	1810	2796	1846	354	793
Costs—Mills/kw hr:						
Fixed [b]	7. 871	7. 970	8. 025	7. 994	7. 911	7. 871
Core fabrication	. 164	. 054	. 064	. 097	. 269	. 136
Fuel cycle	. 449	. 201	. 178	. 363	1. 108	. 231
New fuel	2. 873	2. 607	2. 599	2. 789	2. 774	2. 916
Blanket cycle	. 816	. 796	. 696	. 679	. 825	. 899
Fuel inventory	1. 133	. 926	1. 439	1. 470	2. 046	. 660
Gross costs [b]	13. 306	12. 554	13. 001	13. 392	14. 933	12. 713
Less Pu revenue	5. 231	5. 389	5. 330	5. 165	5. 174	4. 717
Net costs [b]	[a] 8. 08	[c] 7. 17	7. 67	8. 23	9. 76	8. 00

[a] Not an optimized design. Net power costs for this system should be only slightly higher than the UC ceramic plates.
[b] The fixed changes used were for a 300 Mwe plant. Net power costs for each of the cases presented should be reduced by 0.4 to 0.6 mills/kw hr.
[c] Scrap used fuel after Pu removal.

etry have consistently indicated promise of low-cost power production. The alloy plate system is included as a basis for comparison, not because of its economic equality. The UC-Inconel X cermet also gives promise of being a low fuel cycle cost fuel.

Burnup of each fuel material is the most significant parameter in achieving economy of fuel cycle performance. The alloy system burnup is based on current expectations of best performance at 3.0 percent maximum burnup at 1,200° F. The burnup of the UC-Inconel X system is determined from extrapolations of the UO_2–SS cermet irradiation stability assuming that the transverse tensile strength is the limiting factor.

Fixed costs are based on Fermi II studies. Fabrication costs are for the period 1965–85, and are the best estimates of BMI. These costs include UF_6 conversion to fuel material and complete fabrication costs. Fuel cycle costs are those of core reprocessing, core plutonium metal conversion, and conversion of enriched uranyl nitrate to UF_6. Fuel inventory costs are based on 12.83 percent fixed charge for in-pile and 10-month out-of-pile inventory.

Net costs indicated are useful only for comparative purposes at this time, because of the

large number of assumptions made in the study.

Fast oxide breeder program.—The fast oxide breeder (FOB) was initiated by the Knolls Atomic Power Laboratory (KAPL).

The FOB fuel element consisted of a type 347 stainless steel tube 0.200-inch I.D. with a wall thickness of 0.015 inch, 42 inches long enclosing a 65 percent dense pin of mixed oxides of Pu and U. The voids in fuel material and 0.0015 inch annulus between fuel and clad were to be filled with helium during fabrication. The fission gases released by the fuel material were to be contained under pressure in the fuel element voids and inside the cladding tube. There is some question whether the proposed fuel element could operate at the design heat generation rates without melting during rapid reactor transients of small magnitude. The irradiation test work indicates that sintering takes place in the center of the pin when it is raised gradually to operating temperatures, producing a central void, but increasing thermal conductivity of the higher density sintered layer. Since thermal conductivity of this material is not well established, it is not clear if fuel melting would result during a, say, 50 percent rapid reactor overpower transient. The effects of fuel melting in the reactor were not studied, but it is now believed that some can be tolerated.

The cladding thickness was determined to contain 100 percent fission gas released from the fuel up to 50 percent burnup of the original plutonium loaded. Fission gas release rate has not been determined, and it must be known to properly design the fuel elements and reactor cover gas system.

The FOB program did not include such design studies as effect of core composition on power costs, subassembly bowing problems, coolant pressure drop effects on subassembly design or reactor vessel components.

Reactor safety studies and reactivity coefficients apparently were not studied, undoubtedly because of the lack of reliable data to depend on for fuel behavior under irradiation. A simple experiment was devised and completed to predict fuel expansion coefficients under rapid increases in fuel material temperature (500° C. per second). Measurements recorded were indicative of thermal expansion rates, but were accurate to a factor of two or three. Effects of cracking in elements could not be evaluated.

The FOB reactor design studies consisted only of enough information to allow them to identify the development problems with the mixed oxide fuel, and then they concentrated major effort on the fuel cycle development problems.

Fuel cycle development was the major effort of the FOB program. A core fuel cycle process was devised based on a minimum number of operations of a simple nature, and then the group set out to prove that this process would work. Part of the process operations were demonstrated in full size equipment constructed at the laboratory. The dissolution of mixed oxides of Pu and U produced by this process was established after irradiations in MTR.

Full-scale equipment was built to chop fuel elements, dissolve them in HNO_3, and aqueously separate the fission products. This equipment used substitute fuel elements to demonstrate process operation feasibility. Irradiation specimens manufactured were made in hot cell operations in small batches starting with the solvent extraction column products (uranium and plutonium nitrates), precipitated, filtered, dried and reduced to the mixed crystal state. Specimens were formed by compacting and sintering into pellet shapes. The extrusion equipment was built and tested a few times to produce long pins.

Feasibility of the chopping and dissolving operations was established. The radiation chemistry analysis of the solvent extraction operations indicates good promise of success, the only problem remaining being that a second stage extraction process or longer cooling period might be required. Conversion from the nitrate products of solvent extraction to the mixed oxide crystal is a feasible process, subject to investigation of process quality control.

Uncompleted fabrication development problems are mainly concerned with methods of making straight pins and achieving control of density of the finished product. These were not considered insurmountable problems.

Four irradiation experiments have been conducted to date in this program. The fuel elements for these tests (except irradiation test (KAPL 26–3)) were made with 0.250-inch O.D., 0.1875-inch I.D. type 347 stainless steel tubes filled with 1 inch of oxide pellets and one-half-inch long end plugs of MgO. The end closure was welded on with one-half atmosphere of helium in the tube. The elements were inserted in a NaK-filled stainless steel capsule. The capsule was inserted into an Al container with MgO, thermocouples, calrod heaters, and during reactor operation helium surrounding the capsule. The calrod heaters were designed to keep the NaK bath at the desired temperature.

The first irradiation test (KAPL 26–1) was conducted with NaK temperature above 600° C., specific power=3,600 kw/kg of Pu and to a depletion of 5 percent of the Pu (about one-quarter percent total atoms). This test produced a central void, a sintered annulus around the void and some radial cracking.

The second irradiation test (KAPL 26–2) was similar in conditions of operation except Pu depletion was 35 percent (about 2 percent total atoms) and NaK temperature was near 300° C. This test produced about the same results as KAPL 26–1, but more radial cracks were observed.

The third irradiation test (KAPL 26–3) was designed to eliminate the void observed in the first two tests. A three-eighth-inch O.D., 0.030-inch wall stainless steel tube was filled with lead. A 0.1875-inch diameter hole was then drilled in the lead and the oxide pellets loaded as before. As was intended, the lead melted in operation to form a good thermal bond, but the oxide fuel floated partly above the lead surface. This element was irradiated to 5 percent depletion of Pu (about one-quarter percent total atoms). The results were unusual in that no sintering occurred, although pronounced cracking resulted and the oxide was found to be quite friable.

The fourth irradiation test (KAPL 39–1) was conducted with UO_2 substituted for the mixed oxides previously used. The purpose of this test was to determine whether spalling and settling occurred as a result of void formation. The 20 percent enriched, 86 percent dense UO_2 fuel was prepared by extrusion with a Ceramul C binder, dried, fired and sintered for one hour at 1,600° C. in hydrogen. This capsule was cycled in an MTR cycler 50 times in a 3-week cycle. U^{235} depletion was less than 5 percent (about one-quarter percent total atoms); a central void formed equal to about 11 percent of the original fuel volume. No settling of the fuel occurred.

In none of the above experiments was any cladding change observed; fission gas quantities were not measured.

Other materials.[2]—The reactor vessel for the Fermi plant has been designed to withstand an integrated neutron exposure that is within the range for which experimental information exists. The expected flux exposure of the reactor vessel in a 20-year lifetime and the available experimental data are given in table 3. Although these data indicate no serious effects for exposures at ambient temperatures and energies, a continual check on the mechanical and physical properties will be kept by means of surveillance tubes containing control specimens located in areas exposed to high radiation. The specimens were cut and machined from the same sheet, plate, or bar as the section under scrutiny and will be removed at periodic intervals so that a complete and comparative service history will be available.

Intensive tests at ANL, KAPL, Atomics International, and Babcock & Wilcox on type 304 stainless steels and low-alloy steels indicate that, for the uses intended in the Fermi plant, these steels are more than adequate to do their jobs. Mass transport and corrosion loop tests

[2] A discussion of UO_2-stainless steel dispersion fuel elements is given under the heading Fuel and Material Properties.

TABLE 3.—RADIATION OF STABILITY CONSIDERATIONS USED IN THE REACTOR VESSEL DESIGN

Parameter	Integrated flux, nvt	
	Exposure and spectrum	Fast, after 10-year operation
Maximum exposure data available:		
No serious deleterious effects noted: ~5.5 years in MTR [1]	3.8×10^{22} ((>100 ev)*	
EBR–1 core container and flow separator [2]	2.0×10^{21} (370 kev)	
Calculated exposure of vessel wall [3]		6×10^{21}.
Calculated exposure of spot in blanket 12 inches from core (2.6×10^{14} nv).		7×10^{22}.

*The nvt >100 kv is estimated to be 10^{22}.

[1] M. R. Bartz, "Effects of Irradiation on MTR Materials," Proceedings of the Conference on Radiation Effects, TID–7515 (pt. 1), U.S. Atomic Energy Commission, 1956.

[2] R. E. Bailey and M. A. Silliman, "Effect of Fast Neutron Irradiation on the Properties of Stainless Steel," Symposium on Radiation Effects to Materials, vol. III, American Society for Testing Materials, June 1958 (to be published).

[3] Based on half time at 300 Mw and half time at 430 Mw with 75 percent plant factor.

indicate that where low alloy steels of 2¼ percent chromium are used and temperatures are below 850° F. decarburization is of no consequence.

Corrosion tests of stressed and unstressed specimens of type 304 stainless steel and 2¼ percent Cr, 1 percent Mo ferritic steel in static and dynamic sodium with up to 6 percent addition of sodium hydroxide have not shown any evidence of stress corrosion cracking. Proper design and careful control of materials will preclude such cracking.

Intensive tests at KAPL, ANL, Detroit Edison's engineering research department, and Allis-Chalmers on the galling and diffusion bonding of different materials in sodium indicate that hard surfaced materials must be used where undue pressures and high temperatures can exist. Nitrided, stellited or colmonoy surfaces are used in the Fermi system where mating parts are subject to high temperatures and pressures (4–3, 4–20).

Various shielding materials have been investigated. Serpentine rock up to 800° F. has good long-time water retention as determined by tests and is being used in the Fermi plant in the plugs of the intermediate heat exchangers. Irradiation tests made on calcium borate insulation and shielding, used around sodium piping near the reactor, indicate that there are no apparent adverse effects on the load supporting properties due to irradiation to a total of 2.4×10^{20} nvt. Post irradiation heating to 1,000° F. did not cause any breakdown of the material. Calcium borate is an excellent neutron shield as well as possessing thermal insulating properties. Tests run on graphite to determine its sodium retentivity indicate that, while binders using petroleum bases are not adequate for retention of sodium, an anthracite base is adequate. The use of a borated cast carbon steel in the rotating plug required physical testing, particularly of its machinability. Since the tests indicate brittleness, good machinability and poor weldability, the use of such castings in the plug has been based on low strength requirements and no welding (4–1, 4–31).

Various tests have been run with sodium to determine its qualifications as a heat transfer agent in the Enrico Fermi plant. Sodium-water reaction, sodium-air burning, sodium cleaning, vapor trap, compatibility of rubber and plastic in sodium, sodium concrete reaction, displacement of sodium in lead-tin-bismuth alloy, compatibility of non-hydrogenous lubricant with sodium, effect of sodium on protective

clothing, and sodium-insulation reaction are some of the tests which have been conducted (3–12 through 3–18).

Heat Transfer

The use of sodium to remove heat from fast reactors is based on its excellent thermal and physical properties and on its acceptable neutron absorption characteristics. Its use introduces into power plant operation additional new technology and problems. Considerable experience with the handling of sodium in reactors has been accumulated. This experience indicates that the advantages of sodium far outweigh its problems.

ANL.—Sufficient sodium heat transfer data are now available to permit calculation of liquid metal heat transfer coefficients for practical application to sodium-cooled reactors. Experiments with NaK since 1955 at ANL at temperatures 85° to 1,175° F., 4 to 60 feet per second, Reynolds number 13,000 to 466,000, Peclet number 268 to 3,850, average heat flux 28,600 to 3,200,000 Btu/hr-ft^2 (maximum 6,000,000) gave Nusselt numbers of 1.4 at low Peclet number, to 22.4 at high Peclet number. Burn-out occurred at the maximum heat flux section when the copper tube reached the melting point of 1,981° F. (3–58).

Lubarsky & Kaufman have made an excellent summary of experimental heat transfer data (3–37). The Nusselt-Peclet relationship is normally expressed as $Nu=a+bPe^c$. A convenient empirical equation by Monson is

$$Nu=2.3+0.23\ Pe^{1/4}$$

Another is $Nu=0.625\ Pe^{0.4}$, a correlation by Lubarsky and Kaufman. These relationships indicate that for Na the heat transfer coefficient can run higher than 25,000 Btu/hr-ft^2-F. at velocities of 25 feet per second, temperatures of 500° F., and hydraulic diameter of 0.15 inch—typical of sodium-cooled reactor cores.

Enrico Fermi plant.—The APDA steam generator test in support of the Enrico Fermi plant is described in the section Components and System.

Fluid Flow Experiments

ANL.—Reactor transient coolant flow rates were determined by the use of the equations similar to those derived by Acker & Louis and verified experimentally at ANL and elsewhere.

Extensive test programs have been conducted on the hydraulic pressure drop relations of the core subassemblies including the hydraulic holddown effect.

Qualitative hydraulic tests on a fuel subassembly indicate that the spiral fuel pin construction does provide some fluid mixing.

Enrico Fermi plant.—Pressure drop tests on fuel subassemblies indicate that 18 percent of the 90 psi drop through a fuel subassembly occurs in the entrance, nozzle, and exit; 67 percent in the core; and 15 percent in the upper and lower axial blanket.

Hydraulic endurance tests of the core pin bundle assembly reproducing reactor flow conditions indicate that vibration produces failures of the wire spacing system. This required redesign of the wire assembly. Present design has experienced no failures with flow rates 30 percent higher than nominal.

A 0.25-scale model of the 15,000 gallon upper plenum of the reactor vessel was tested with water to determine mixing effects during scram. The tests indicated that mixing is effective in this pool to the extent that a core exit transient of over 300° F. per second is reduced to 300° F. per 20 seconds or 15 degrees per second. The holddown is an effective mixing agent. The pressure through the holddown was simultaneously tested. This pressure drop is less than 4 psi.

The centrifugal sump-type primary coolant pump was successfully tested in water prior to final shipment. Decay flow characteristics necessary for emergency cooling analyses, as well as developed head, efficiency, and shaft seal effectiveness were determined.

Primary system check valve flow characteristics under back pressure were tested in water, indicating less than 100 gpm flow under system pressure.

Prototype safety control rod drop tests in sodium show that safety rods can be designed for scram characteristics of 100 inches per second.

Coolant Chemistry

Large sodium systems are being operated successfully. The sodium coolant system has given excellent trouble-free service. One such system was operated for 2 years without loss of any sodium and with no evidence of corrosion or mass transport. The fuel elements performed perfectly. Early troubles experienced with this system did not result from the sodium coolant and were eliminated by bypassing the superheater section of the boiler. Valuable experience concerning sodium in reactors is being accumulated on the Sodium Reactor Experiment by Atomics International. The operating experience with the Enrico Fermi, Dounreay, and the EBR–II sodium systems and the EBR–I NaK system will provide further information which will be available in time to incorporate into the detailed design of the plutonium-fueled fast breeder reactor (PFFBR).

Based on the results of detailed studies of various gases, argon was selected for the inert gas system of both the EBR–II and Fermi plant. The reason nitrogen, which is inexpensive, was not used in either case was based on a lack of sufficient information regarding the nitriding of exposed component surfaces. The supply system for inert gas supply is maintained by either a gas trailer tank or one of two banks of gas cylinders. All supply sources have pressure-indicator alarms to alert the operator to the gas supply conditions.

Corrosion has already been discussed in the subsection "Other materials," while a discussion of impurities appears in appendix A.

ANL.—The Liquid Metals Handbook and its supplement and other previous data regarding sodium chemistry have been supplemented by extensive experimentation on such items as sampling and analysis for impurities in sodium systems.

A vacuum distillation process has been improved. The vacuum distillation removes sodium from its nonvolatile impurities and the residue is analyzed by standard radiological or chemical methods.

A plugging indicator, developed by KAPL is also used for direct oxide determination for checking efficiency of cold trapping. "Cold trapping" technique for oxide removal, well developed at KAPL for the Submarine Intermediate Reactor (SIG) and the S2G Sodium Reactor (S2G) programs, has been but little improved.

Enrico Fermi plant.—In the Na-air reaction test (2–7, 2–9, 2–10, 2–11), a stoichiometric amount of hot sodium (850° F.) injected as a fine dispersion and at a high rate into a 532-liter pressure vessel developed a maximum pressure of 38 psig in 6 seconds. Further tests simulated large amounts of sodium spread out in a pool exposed to air until all the oxygen is consumed. The initial burning rate of sodium was less than 5 lb/hr-sq ft of burning surface. The tests indicate that the rate is primarily controlled by a diffusion process rather than the velocity of chemical reaction and is proportional to the product of oxygen concentration and the square root of the absolute temperature. These series of tests have shown that containment buildings can be designed in the future with more assurance than at present.

A two-phase NaK-water reaction test program was carried out:

Phase I (3–1)

Injection of a one-half-inch stream of 600–1,200-psig and 200°–400° F. water into 50 psig and 600° to 800° F. NaK, simulating failure of a steam generator tube developed excessive pressures with zero gas volumes. This pressure was considerably reduced by adding gas volume to 92 percent of the NaK volume.

Phase II

Water at 900 psi was introduced through a simulated tube rupture of a $1\frac{3}{16}$-inch-diameter

tube into approximately 250 pounds of sodium heated to 500° F. A relatively large, 26 cu ft, gas surge volume was placed adjacent to the sodium and contained a 30-inch-diameter stainless steel rupture disc to relieve the products of the reactions. Maximum pressures reached were 175 psi in the reaction area, 70 psi at the rupture disc with a maximum temperature of 1,950° F.

These series of tests have been the basis for the relief system provided in the Fermi plant steam generators, as well as the basis for the design pressure in the secondary system.

Reactor Safety

Questions have been raised concerning the inherent safety characteristics of the fast reactor because of the behavior and meltdown of the EBR–I during special reactor physics tests. For this reason, the many features of fast power reactors related to safety and control have been extensively studied both in the United States and in England. In the United States, the bulk of this work has been done by Argonne National Laboratory, Los Alamos Scientific Laboratory, and by APDA and its contractors.

The EBR–I was initially constructed not as a device to prove the kinetic aspects of fast reactor behavior but as an experiment to show the physics practicality of breeding and the engineering practicality of using a liquid metal coolant. The early demonstration of a prompt positive power coefficient in the EBR–I led to speculation that some inherent characteristic of fast reactors was at fault. Although not as yet conclusively proven, recent experiments have yielded circumstantial evidence at least that core distortion was at fault. If this is the cause, then there are no inherent fast reactor characteristics which make it less safe than other power reactors. Consequently, no extra costs are to be expected in providing for the fast reactor the safety features required to control the hazards of radiation.

Extensive work is being carried out on reactor safety associated with characteristics attributed to fast reactors. The problem receiving most attention is supercriticality and consequent energy release association with the agglomeration of a high concentration of fissionable material. The results to date have been encouraging, but analyses have not been extended to plutonium systems nor to the larger reactor of the 10-year program. Safety programs should be reviewed after reference reactor systems are established. In the meantime generalized studies, some of which are discussed, should be carried out. In addition to the studies on cross sections, alpha, beta, and temperature coefficients of reactivity, further work should be done on bowing of fuel subassemblies and on the nature of plutonium system meltdowns with and without sodium.

Past status (1–20).—The past several years have seen major advances made in the field of fast reactor safety. Four years ago, the major items under discussion at an APDA-sponsored meeting on fast reactor safety and controls (1–28) included the following:

(1) A first theoretical effort at the Doppler effect in fast reactors.

(2) A report on a prompt positive reactivity coefficient and some tendencies toward spontaneous oscillation of power in the Mark I or Mark II cores of EBR–I. The meeting produced the suggestion for oscillating experiments as a means for investigating these undesirable dynamic characteristics. It was suspected that the Doppler effect or rod bowing played a major role therein.

(3) Parameter studies on the effects of step and ramp insertions of reactivity. Performed with assumptions only approximating a real reactor, these calculations, nevertheless, provided some insight into the size of insertion which would cause trouble, without defining the manner of reactivity introduction.

(4) Some very qualitative discussions of hypothetical accidents which might lead to a core meltdown or a nuclear burst. Also, some very brief discussions of the manner of fuel element failure upon overheating, including

some speculation on the possible melting of fuel pins in a Godiva burst.

Present status (1–20).—The present status of the work was reviewed in two papers (1–48, 1–50) and brought up to date at a second APDA-sponsored meeting on fast reactor safety in March 1959. The situation has changed considerably in the 4 intervening years, as follows:

(1) Refined measurements of the Doppler effect have been made both in the United States and England, showing it is too small to have played a major role in EBR–I. The experiments are in semiquantitative agreement with recent theoretical predictions, but more careful work may be in order.

(2) The Mark II core of the EBR–I was oscillated, showing a large resonance for certain ratios of power-to-flow. This core was partially melted during the series of kinetics investigations, and has been replaced by a new "rigid" Mark III core, wherein fuel element bowing was prevented. The Mark III core is highly stable and shows no sign of a positive reactivity coefficient. Considerable theoretical progress in the understanding of the dynamics of fast reactors has been achieved. However, only partial success has rewarded the attempts to calculate quantitatively from first principles the detailed behavior of the Mark II or Mark III cores.

(3) Accident studies have shifted from ramp insertions of reactivity to meltdown studies and explosion calculations. Reactors under design have been instrumented and designed to make the likelihood of "startup" accidents and loading mishaps remote. Similar precautions have been taken against the loss of coolant accident, etc., so that it becomes difficult to define a "credible" accident which will lead to core meltdown and the possible subsequent reassembly in a more reactive form.

(4) Only partial success has rewarded the theoretical efforts thus far on the difficult problems of prescribing the course of a meltdown. Some out-of-pile experiments have been performed and more are in preparation. A new Transient Reactor Test Facility, TREAT, a source of neutron bursts for engineering tests, has been designed and constructed especially to permit controlled studies of the modes of failure of fast reactor fuel elements upon overheating. The mechanisms important to dispersal or reassembly will be studied herein to provide a guide for further theoretical studies.

(5) New methods of computing the energies and pressures developed in hypothetical nuclear accidents have been devised. Uncertainties still remain in the specification of the equation of state for temperatures from 10^4 to 10^5 degrees Kelvin, in complications of geometry, and especially in the determination of the threshold energy for pressure buildup, as influenced by voids initially present in the exploding medium.

(6) Experience with ZPR–III provides reasonable confidence in the prediction of most static physics properties of fast reactors. On the other hand phenomena such as fuel element swelling due to fission products gases, density changes due to phase changes and the effects of boiling sodium on a meltdown have risen into some prominence. Furthermore, bowing of fuel elements and subassemblies thereof remains a very real and practical problem, and the reactor designer must exercise considerable care to control its effects in each new design.

EBR–I kinetic studies (1–21).—One of the main features incorporated into the design of the Mark III loading of EBR–I is a tightening rod for varying the clearance between neighboring fuel and blanket rods.

With the tightening rod in the expanded condition, the fuel rods are forced outward against the inside of the hex can. With the tightening rod in the closed condition a total clearance of 0.070 inch between rods is allowed.

This design makes possible a direct investigation of possible rod bowing effects which have long been suggested as the source of the positive temperature coefficient of reactivity observed in the Mark II loading.

The results of transfer function measurements and flow change tests carried out with all fuel tightening rods loose failed to reveal any significant evidence of a positive temperature coefficient of reactivity. Within the limits of experimental accuracy no change was noted in the phase and amplitude of the transfer function. Sudden decreases in coolant flow rate were accompanied by an immediate reduction in power. Conversely, flow increases caused an immediate increase in power. Such behavior contrasts sharply with the results of similar tests conducted on the Mark II reactor. In this case a flow descrease resulted in a sudden increase in power, the result of a prompt positive temperature coefficient, followed by a slower decrease to some lower equilibrium power level.

The Mark III results should not be interpreted as a rejection of rod bowing as a possible mechanism for temperature dependent reactivity effects. The fuel rods were not rigidly fixed above and below the fuel section. Further tests will be conducted to establish the existence or absence of rod bowing effects. Transfer function and power coefficient measurements augmented by flow change tests will be conducted. The results will be compared with the rigid reference core. Any difference caused in the reactivity feedback will then be subject to interpretation in terms of material displacement.

TREAT (1-22, -23).—The TREAT reactor is a graphite-moderated system with the fuel uniformly distributed as uranium oxide particles in the micron-size range. It has been built as a versatile transient testing facility with primary emphasis on meltdown studies of fast reactor fuel elements. The reactor building was completed in October 1958 and the reactor went critical in February 1959. It is now undergoing shakedown and physics experiments prior to actual test runs.

Provisions for experimental facilities are of two primary types: (1) vertical holes for capsule tests with attendant external holes to provide for loop installations, and (2) horizontal access holes through shield reflector and core to allow visual and photographic observation of experiments in progress. There is also a thermal column.

The primary value of TREAT will lie in the study of fuel element failure and meltdown product motion. The early experimental meltdown work with TREAT will be concentrated on experiments with single fast reactor fuel elements, designed to explore element failure and disassociation mechanisms. These experiments will form the basis for investigations into meltdowns of clusters of pins and the subsequent "melt movement."

A program of initial TREAT meltdown experiments—for convenience divided into series SI and SII—has been planned.

SI and SII will bridge the gap between tests designed to check the characteristics of TREAT as a neutron source and tests on fast reactor fuel elements utilizing TREAT as a source of neutrons for nuclear heating. These series should serve as "proof tests" on TREAT operations and yield information concerning the experimenters' ability to predict occurrences during TREAT bursts and provide actual meltdown data for EBR–II Mark I type of fuel elements.

Both series are experiments on single, uncooled, unirradiated EBR–II Mark I type of pins. SI will use opaque graphite-lined capsules; SII, transparent capsules with fused silica windows. Variables whose effect on sample meltdown characteristics will be studied are total energy input, rate of energy input, and possible perturbations caused by instruments. Data which are desired from the experiments includes pin elongation, fuel-clad eutectic formation, pin hot spots, and types and extent of clad failure.

SI consists of four experiments in which the sample energy input will be sufficient to raise the average pin temperature as follows: SI–1, ~500° C.; SI–2, ~900° C.; SI–3, ~1,130° C. (but not melt the fuel), and SI–4, ~1,130° C. (including enough energy to melt the fuel).

SII consists of six experiments during which extensive pin failure and eutectic formation are expected; burst durations and total energy

inputs of SII–1 through SII–6 are to be partially dependent on the results of those experiments already completed. Optical motion picture photography, both high speed and normal, will be used with SII. Later tests will include (1) pin immersion in stagnant sodium, and (2) previous irradiation of sample pins.

Experiments with clusters of pins and with pins immersed in flowing sodium are being planned, but for much farther in the future.

Instruments are being developed now for tests on pins in a liquid metal environment. These include special thermocouples, pressure measuring devices and a γ-ray camera.

Containment (2–19).—The two potential sources of energy that are germane to discussions of containment for fast reactors are (1) possible nuclear energy release following a core meltdown and (2) sodium-air reactions following a sodium leak.

Studies indicate a maximum fission energy release of 5×10^8 calories for any fast reactor system and probably less. Sodium-air reactions are probably lower by a factor of 10 or 100, and heat stored in the fluid systems is not an important factor for sodium cooled reactors.

These energies do not all become available at the same time in the event of an incident. When released, a large portion is used up in heat capacity of available sinks.

The released energy can develop into pressure and temperature increases, either prompt or delayed.

Containment is necessary with the present state of knowledge. The problem may be eliminated or greatly reduced by:

(1) Increasingly accurate determination of the maximum possible initiating energy release (nuclear and sodium-air reaction), where one can hope to pick up reduction factors of 10 or 100 for the current pessimistic maximums.

(2) Determination of a reasonable division of energy into that soaked up and that which goes to physically damage the system.

(3) Accumulation of damage experience for systems having energy releases of the form encountered, and comparison with currently used assumptions of release as from propellants and high explosives, hopefully only reducing current effects by a factor of 2 or so.

(4) More accurate appraisal of the release of stored energy with respect to the effect on the pressures and temperatures of its time history and removal of energy to existing sinks which may introduce reductions in loading by factors of 10.

(5) A better determination of pressure loads from materials burning after ejection into the space proximate to the barrier, which may also lead to loading reductions by a factor of 10.

(6) An attack of the problem of reasonably estimating the motion of radioactive materials to a position of availability for discharge from the system, and determining the time rate of change of this loading as a result of removal through fallout, precipitation, condensation, and radioactive decay to stabler states, all of which could yield reductions in radioactive loading assumptions of 100 or 1,000 initially, and perhaps an additional 10 or 100 subsequently.

Components and Systems

ANL.—Liquid metal pumps: A general program of liquid metal pump development was initiated in 1956 to build and test a 5,000-gpm mechanical centrifugal pump, a 5,000-gpm a-c linear induction pump, and a 10,000-gpm d-c electromagnetic pump. Inspection of the mechanical pump after 6,300 hours of successful operation revealed no measurable wear. Its efficiency is 85 percent overall. The a-c induction pump has operated successfully for over 7,000 hours at 850° F. Its overall efficiency is 43 percent. Flow control of increments as small as 1 gpm have been demonstrated.

The d-c pump has had difficulty in meeting design head. The pump and its supply of 300,000 amperes require a large physical plant. The mechanical and a-c pumps were chosen for EBR–II.

Instrumentation: Magnesium oxide insulated thermocouples in stainless steel sheaths have demonstrated in-pile reliability in EBR–I core. The operation of EM flowmeters while submerged in sodium has been demonstrated in the EBR–II model. Commercially available pressure transmitters submerged in the model tank have proved reliable. A self-seeking sodium level probe has been developed for use in remote locations. The use of 60-cycle induction heating of the EBR–II model tank, and of the large 12-inch sodium pump loops has been demonstrated using a thin carbon steel sheath around the stainless steel pipe.

EBR–II model (3–6): The EBR–II 5,000-gallon model operated at 700° F. duplicated the salient features of EBR–II, has been operated since 1955, has demonstrated fuel handling components as well as instrumentation, cold traps, plugging indicators, pumps, heating, and the handling of sodium.

KAPL S1G and S2G.—Extensive research and development at KAPL has undertaken the design of mechanical pumps, electromagnetic pumps, valves, instrumentation, cold traps, plugging indicators, analytical devices, steam generators for use in sodium systems in the S1G and S2G plants. Much of this experience has been useful in designs of fast breeder sodium systems.

Enrico Fermi Atomic Power Plant.—Reactor: Extensive experimental stress studies of the effects of piping reactions on the 30-inch reactor outlet nozzles were made on a one-quarter-scale model. The resultant stresses are lower than estimates made in Bijlaard's analysis.

Experimental verification of the bowing of fuel subassemblies was made, with and without mechanical restraints, at the core center. Core restraint can be designed to give zero net inward movement due to bowing.

Bases for stress analyses have been determined for all possible loadings including effects of thermal shock. Plastic deformation theory is used in setting up stress allowables based on a modified Goodman diagram and S–N fatigue curves.

Pumps: Extensive development tests were conducted on pump, shaft seals for the mechanical sodium pumps. These are in an inert gas phase above sodium and use a fluorocarbon oil containing no hydrogen. Tests have been successful and the results incorporated into the pump design.

Once-through steam generators: The practicability of once-through steam generators has been proven by a series of tests on two prototype units.

Starting on September 17, 1956, and continuing for 15 months, a 7-tube horizontal U-shell unit once-through steam generator was tested for steady state, transient, and water fouling conditions under various operating conditions. This model contained full-length tubes with as much geometric similarity to a typical full scale unit as possible. NaK 56 alloy was used as the coolant. The unit was made of type 304 stainless steel. Inlet feedwater temperatures were varied between 175° and 200° F. Over 16 steady state tests were completed to determine the operating performance, temperature gradients, temperature profiles, and heat balance. Twenty-one transient tests were run to determine temperature response, rate of change, total change, and control response. The unit withstood over 40 normal shutdowns and over 25 forced shutdowns. The liquid metal system and the once-through steam generator were not responsible for any of the unscheduled outages. The NaK temperatures into the steam generator were 950° F. maximum, and 450° F. at the outlet. Steam temperatures ran from over 900° F. down to 680° F. depending upon the load. Transient changes from 50 to 75 percent load were made in 30 seconds.

Once-through steam generator operation has been completely satisfactory in all respects. The tests indicate that a single wall tube design of a once-through unit is quite practical. There is no evidence to date of either water side or liquid metal side deposits extensive enough to cause measurable losses, heat transfer or increases in pressure drop.

A second 7-tube bayonet type vertical once-through steam generator was installed after the completion of the first test and operated for a total of 472 hours. The steam generator tubes in this unit were made of 2¼ percent Cr, 1 percent Mo ferritic steel. The operating experience and results of tests indicate that this type unit could be readily adaptable for plant service and again proves that a once-through steam generator for sodium use is quite practical.

Vapor traps: The test of a sodium vapor trap using various meshes is still proceeding.

Reactor components test: All major components in the primary system for Fermi will be mechanically and hydraulically tested in hot gas and sodium in a test facility comprising the reactor vessel, rotating plug and drive, fuel handling mechanisms, holddown, two safety rod drives, two operating rod drives, oscillator drive, one piping loop with pump, and dummy fuel and blanket subassemblies. The tests are scheduled to begin July 1959, and will continue until summer 1960 prior to completion of the plant.

Fuel and Materials Properties

Uranium oxide-stainless steel dispersion fuel elements.—Stainless steel clad fuel elements containing a core of UO_2 dispersed in a stainless steel matrix have been extensively studied, the largest scale application being a core loading of plate-type elements for the Army Package Power Reactor (SM–1). For high power density, enriched uranium-fueled reactors, the stainless steel dispersion or cermet combines the advantages of excellent high temperature stability and ability to accommodate fission products of a stable ceramic, with the high strength, relatively high thermal conductivity of stainless steel. Irradiation tests have shown that through proper choice of UO_2 particle size, most of the fission products can be retained in the UO_2, minimizing damage to the steel matrix and avoiding the extensive embrittlement that occurs in alloy systems.

Fabrication studies have been carried to the point that some types of elements are essentially commercially available items. Based on powder metallurgy techniques for forming UO_2-stainless steel compacts, flat plates have been fabricated by the "picture frame" hot rolling process, externally and internally clad tubes by powder rolling, and cylinders have been fabricated by hot swaging, hot extrusion, and a "cold binder" process.

Irradiation and metallographic data indicate that the performance capability of a UO_2-stainless steel dispersion fuel element is a definite function of the shape, core thickness, and temperature, and that the thermal stress level may also be important. The performance limits have not been completely defined for this type element, particularly the effect of lowering the temperature of the cylindrical elements. The SM–1 type of elements has been successfully irradiated to 50 percent burnup of the uranium in the dispersed phase at temperatures below 800° F.

REACTORS COMPLETED OR UNDERWAY

Experimental Breeder Reactor No. 1

Mark-I and Mark-II cores.—Refer to figures 4 and 5, appendix D. The reactor consists of three principal parts (Fig. 4). The core of the reactor is an assemblage of small diameter cylindrical fuel elements. These contain U^{235} enriched to about 90 percent. Surrounding the core there is an inner blanket composed of larger diameter rods, each containing natural uranium. The core and the inner blanket are cooled by a sodium-potassium alloy. The coolant is contained by a double-walled tank which fits closely around the inner blanket. The third part of the reactor consists of an outer blanket constructed from natural uranium and having a high uranium density. The outer blanket is air-cooled and is movable. It also contains the control rods. By restricting all movable parts of the reactor to the air-cooled outer blanket, it was possible to avoid bearings and other moving machinery inside the tank containing liquid metal. At the time the reactor was designed, the techniques of operating moving parts in a liquid metal system were not as well developed as they are now, and the expedient of putting all moving parts in an air-cooled blanket was recognized as being suitable only for this experimental application. Ultimately, it turned out that the air-cooling of the blanket was the limitation on the operating power level of the reactor.

In the Mark–I loading the fuel bearing section of each element had two spacer ribs 0.042-inch high located so that when the triangular positioning tips of the elements were engaged in the matching holes in the tube plate, the ribs were brought in line with neighboring rods.

In the Mark–II loading the spacer ribs were omitted and the positioning tip was made cylindrical, thus permitting 0.046-inch separation of rods.

In the Mark–I core four slugs 0.364 inch in diameter and 1⅞ inches in length were loaded in most rods. Some were loaded with slugs 0.384 inch in diameter by 2.5 inches long. Below the fuel, a 4.5-inch-long natural uranium slug was loaded as part of the lower blanket, and above the fuel an 8-inch natural uranium slug was loaded as the top blanket. In the Mark–II loading the fuel slugs were U–2 percent Zr alloy, 4¼ inches long by 0.384 inch in diameter, with a lower blanket slug 4¼ inches long and an upper blanket slug 8 inches long. The annulus between slug and fuel tube was filled with NaK as a heat transfer bond.

EBR–I was designed in 1948–50 with two fundamental objectives in mind. The first was a demonstration of the ability of a fast neutron reactor to breed, and the second was a demonstration of the feasibility of the use of liquid-metal cooling systems in power producing reactors.

EBR–I was based on concepts proposed by Enrico Fermi and W. H. Zinn in 1945. The research and development program was approved in 1945, and the AEC approved construction in November 1947. Design ran from 1948 to 1951. Criticality was reached in August 1951, with electricity generated on December 22, 1951. Mark–II core was installed in 1954. A series of kinetic experiments in November 1955, led to a core meltdown on November 29, 1955.

The EBR–I was operated over 4 years, or longer than any other AEC power reactor has operated with the exception of the Submarine Thermal Reactor. During this period it gave essentially trouble-free operation. The NaK cooling system performed successfully. Under

normal operating conditions the reactor was very stable and did not exhibit either a net positive temperature coefficient or a resonance. Under purposely imposed and drastically abnormal operating conditions anomalies were observed: resonance consisting of oscillations in power level appeared during experiments in which the coolant flow rate was drastically reduced; a net positive temperature coefficient appeared during startups undertaken with reduced coolant flow. Even under conditions where the net positive coefficient appeared, the reactor could be operated safely. Oscillator tests were conducted on the reactor and successfully demonstrated the presence of instability.

The operation of EBR–I demonstrated among other things that (1) breeding was a technically achievable objective and (2) that the use of liquid-metal coolant (sodium potassium alloy in this case) was compatible with breeding economy as well as metallurgically and mechanically feasible. EBR–I's operation together with that of Clementine provided experimental corroboration of previous theoretical determinations that neutron behavior below prompt critical is the same in fast and thermal reactors.

In November 1955, the Mark II core of the EBR–I partially melted down during the last of a series of experiments designed to study its behavior when put on positive periods with reduced or zero coolant flow, conditions which are set up only for experimental purposes. The accident occurred under extremely abnormal operating conditions purposely imposed on the reactor for the experiment and recognized to involve a risk of causing melting. Two of the normally operative safety mechanisms—the flow interlock (which automatically shuts down the reactor if substantially full coolant flow is not maintained) and the period scram meter interlock (which automatically shuts down the reactor if the period becomes too short)—were purposely disconnected. The coolant flow was stopped completely. A certain fixed amount of reactivity was put into the reactor with the control rods, and the reactor was started up on a short enough period so that temperature differentials would be established in the fuel slugs. The net positive temperature coefficient previously observed appeared and, as the power increased, the reactivity increased, thus further increasing the power. It was planned to scram the reactor when the period reached 1 second or the power reached 1,500 kilowatts. When the period reached 1 second, the operator mistakenly activated the slow acting control rods instead of the faster acting scram mechanism. By the time the scram was used, the period had reached 0.3 second. The temperature overshot so that the uranium became heated above 1,328° F., roughly the temperature at which the uranium-iron eutectic forms. The center of the core melted, forming the eutectic. After the manually operated scram button was pressed, the reactor shut down and the meltdown stopped. The automatic power limitation circuits also operated.

As a result of the accident, the EBR–I core partially melted. No explosive force developed. None of the remainder of the reactor, including the inner blanket and the reactor vessel, was damaged. A negligible amount of radioactive material reached the atmosphere through the stack and through temporary thermocouple connections. Neither the operating personnel nor any other persons were injured in any way and, after evacuation to enable precise measurements, the operating personnel returned immediately to the reactor building.

Figures 4 and 6 show the significant cross sections and flow diagrams.

Some of the design parameters are tabulated below:

Fuel	U^{235}
Coolant (primary and secondary)	NaK
Moderator	None
Thermal power rating, kw	1, 400
Electrical power rating, kw	240
Thermal efficiency, percent	17

Temperatures, ° F.:	
Reactor inlet	442
Reactor outlet	600
Superheater NaK inlet	583
Superheater NaK outlet	572
Steam generator NaK inlet	572
Steam generator NaK outlet	455
Feedwater temperature	214
Water heater outlet temperature	446
Steam generator steam outlet temperature	446
Superheater steam outlet temperature	529
Steam pressure, psia	405
Core size:	
Equivalent diameter, inches (approximate)	7. 6
Length, inches	8. 5
Volume, cu ft	0. 22
Core power, kwt	960
Core power density average, kw_t/ft^3	4, 400
Fuel enrichment, percent (approximate)	90
Conversion ratio:	
Core	0. 004
Inner blanket	. 331
Outer blanket	. 639
Control rods	. 035
Total	1. 009
Core loading U–235, kg	52
Specific power, kwt/kg fuel	18. 1
Fuel temperature, ° F.—center of fuel rod adjacent to center rod	675
Average heat flux, Btu/ft^2-hr	209, 000
ΔT across core, ° F	158

Mark–III core.—Refer to figures 6, 7, and 8, appendix D. After the November 1955, incident, a Mark–III core loading was designed in order to continue the important investigation of fast reactor stability.

The fuel rod shown in figure 6 consists of a highly enriched uranium 2 percent Zr alloy slug coextruded with a 0.020-inch zirconium jacket. Similarly extruded natural uranium 2 percent Zr sections are welded on above and below the fuel to provide the upper and lower blanket. This construction prevents bowing of fuel slugs inside the jacket. Inner blanket rods are made similarly of one coextruded natural uranium 2 percent Zr section. Three ribs of 0.046-inch zirconium wire are welded on with 120° spacing. The rods are fitted with triangular tips to so orient them that each rib contacts the adjacent fuel element.

The central rod is a nonfuel bearing rod capable of being expanded after insertion into the assembly. When expanded, the tightening rod forces the fuel elements against each other and against the hexagonal subassembly can, thus preventing bowing of fuel elements inside the assembly.

Figure 7 shows a plan view of the reactor core and inner blanket at the reactor centerline. Notice particularly, the clamping arrangement to insure tightness of the subassemblies. Figure 8 is a cutaway view of the inner tank assembly. Notice the fuel subassemblies are positioned at the lower tube plate, clamped at the reactor midplane as mentioned above, and clamped just above the coolant inlet at the seal plate, thus eliminating the possibility of subassembly bowing.

Coolant flow is directed in series or parallel by means of the inlet and throttle valves. The reactor has been thoroughly instrumented with thermocouples in fuel and blanket elements, coolant passages, and structural members.

The reactor system outside the double walled reactor tank is unchanged, i.e., just the inner core structure has been modified.

Tests have been conducted on Mark–III at powers up to 1,200 kw and flows as low as 110 gpm.

Flow changes at power reveal no indication of a prompt positive reactivity coefficient.

Measurements of the transfer function have been made by the null-balance method. It is believed that the accuracy of this system is ± 1 percent on amplitude and $\pm \frac{1}{2}°$ on phase, thus permitting fairly accurate separation of the feedback.

Experience with EBR–I has shown that:

(1) Instability of reactors is not a function of the neutron energy spectrum.

(2) Instability is created by delayed negative power coefficients.

(3) Prompt positive coefficients tend to lower the power at which instabilities will be noticed.

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Clementine

The Los Alamos fast plutonium reactor was a low power experiment built to demonstrate the feasibility of reactor operation with plutonium fuel and fast neutrons and to serve as an experimental neutron facility. The fuel was fully-enriched plutonium contained in steel cans and cooled by circulating mercury. Construction of the reactor was approved in December 1945, and the reactor operated until June 1953, at which time it was dismantled as a result of a ruptured fuel element.

The construction of Clementine was initiated to explore the uses of plutonium as a fuel for small reactors and for future power reactor studies. Construction of the reactor provided the Laboratory with a high-intensity source of fast neutrons for nuclear research as well as a device for studying methods and ease of control of a fast reactor.

Actual construction of the reactor was started in September 1946. The first critical assembly of the reactor was made at an incomplete stage of construction on November 21, 1946, and nuclear measurements were performed at approximately 1 watt of power level without further construction until February 1947. During this period the reactor was used as a critical assembly and measurements were made concerning the critical mass vs core configuration, effectiveness of reactor control, temperature coefficient of the reactor, and spectrum of the neutrons. Work of this nature generally continued until January 1949, when the reactor was prepared for final assembly. In March 1949, the reactor was brought to full power.

Of the 3½ years spent in assembly of the reactor, approximately 21 months were spent doing low-power critical experiments. In December 1952, it became evident that a plutonium fuel rod had ruptured thereby releasing plutonium into the mercury coolant. Inasmuch as the primary objectives of the experiment had been realized, it was decided to dismantle the reactor. The disassembly was completed by June 1953. Details are given in LA–1575.

Significant design parameters are shown below:

Coolant	Mercury.
Moderator	None (blanket of uranium metal).
Power rating	Nominal 25 kw thermal.
Electric power rating	0.
Thermal efficiency	Not available.
Temperature and pressure.	Mercury outlet temperature 252° F., 5 psig helium overpressure.
Core size	5.90 inches diameter x 5.95 inches high.
Power density	~250 kw/ft³.
Peak-to-average power ratio.	1.1.
Fuel enrichment	~100 percent.
Conversion ratio	Not measured.
Specific power	1.5 kw/kg.
Central fuel temperature	284° F.
Core inventory	~16 kg Pu.
Temperature across core	80° C.
Average heat flux	NA.
Maximum heat flux	NA.

Experimental Breeder Reactor No. 2

Refer to figures 9 through 15, appendix D. The Experimental Breeder Reactor II (EBR–II) is under development as one part of the U.S. Atomic Energy Commission's experimental power reactor program. The EBR–II is an integral nuclear power plant; it includes a complete fuel processing and fabrication facility in addition to the reactor, heat transfer system, and steam-electric plant. The thermal power rating of the reactor is 62,500 kw. Gross electrical power output rating is 20,000 kw. Engineering design and component development are presently nearing completion. Construction of the plant is in progress at the National Reactor Testing Station; operation will begin in 1960.

Exclusive of fuel recycle aspects, the design objectives of the EBR–II are: to attain high thermal performance (high core power density and high coolant temperature level); to achieve efficient breeding (large breeding ratio); to

utilize insofar as possible prototype components (components of such size, design and cost as to permit their use in central station plants with little or no modification); and, to provide a highly flexible, experimental reactor facility for investigation of reactor configuration parameters, advanced fuel element designs, and improved fuel alloys.

The design objectives of the fuel recycle facility are: to determine the technical feasibility of pyrometallurgical processing and remote fabrication of spent fuel elements; and, to provide a versatile, experimental facility for investigation of new or improved processing techniques and remote refabrication methods.

The power cycle is comprised of three major systems: (1) the primary system, consisting of the reactor and the primary sodium cooling system; (2) the secondary system, or the intermediate sodium heat transfer system; and (3) the steam-electric system.

The complete primary system is contained in a single vessel, or "primary tank," 26 feet in diameter and 26 feet in depth. All of the primary system components, including the reactor, sodium pumps, primary piping, heat exchanger, and fuel transfer and storage system, are submerged in the large bulk volume (80,000 gallons) of sodium within this tank. This arrangement provides a number of advantages. Among these are: loss of coolant becomes virtually impossible; rapid changes in load demand or secondary system conditions are prevented from being reflected as temperature changes in the coolant entering the reactor; practically all radioactivity associated with the power cycle is confined to this one location, thus minimizing greatly the probability of a radioactive leak or spill; and, high integrity of the primary sodium piping is not required, since all leakage is into the bulk sodium. The two main pumps provided operate in parallel and supply 8,600 gpm primary sodium flow. Both pumps are of the motor-driven, centrifugal type, the heat exchanger, and back to the bulk sodium, in that order. The temperature of the sodium at inlet to the reactor is 700° F., and at outlet from the reactor 890° F.

The reactor is of the heterogeneous type, unmoderated and sodium-cooled. Control is effected by movement of fuel into and out of the reactor core. A total of 12 peripherally located control rods are employed, of which one is used for regulation. The core is of the shape of an approximate right circular cylinder, with L/D of about 0.8. Maximum core power density is about 1,300 kw/liter of core volume, or about 4,000 kw/liter of fuel alloy volume. Breeding blankets of depleted uranium surround the core on top, bottom, and sides. The major fraction of breeding takes place in these blankets; the remainder occurs in the core. For initial reactor loadings, the fissionable material to be employed is U^{235}, and for subsequent loadings, Pu^{239}. The approximate conversion (breeding) ratios expected with these loadings are 1.2 and 1.6, respectively.

The secondary system transfers the heat from the heat exchanger in the primary tank to the steam generator. This system is nonradioactive and serves to isolate the steam generator from the radioactive sodium of the primary system. A single a-c electromagnetic pump provides a sodium flow rate of 6,300 gpm. Sodium temperature at inlet to the heat exchanger is 610° F., and at outlet from the exchanger, 870° F.

The steam-electric system receives the heat from the secondary system and converts it to electric power. This system is of essentially conventional design, employing a standard, extracting, condensing, single flow type turbine. An automatic, full capacity, steam bypass system for dumping excess steam directly to the condenser is incorporated to prevent major load changes from effecting changes in secondary system conditions. The condenser circulating water is cooled in a forced convection cooling tower. Turbine throttle flow is 192,200 lb/hr; steam rate is 8.04 lb/kwh. Turbine throttle steam temperature and pressure are 840° F. and 1,250 psig, respectively.

The main unit of the fuel recycle facility is a 16-sided hot cell. This shielded shop contains the equipment for the decanning step (mechanically removing the fuel jackets), melt refining of the spent fuel, the fabrication of new fuel pins, and the assembly of new fuel elements. An ancillary shielded cell is employed for remote assembly of completed fuel subassemblies and for the servicing of main cell equipment.

The large cell is an annular structure 72 feet across flats which contains a shielded central control room 20 feet in diameter. The in-cell atmosphere is high purity argon; pyrophoric materials such as plutonium, uranium, magnesium, and sodium can be handled without special precautions and without the problems associated with "oxide skins." The cell is equipped with manipulators, cranes, shielding windows and services of such design and arrangement that modifications to any processing step or to any of the equipment involved in processing can be readily accomplished.

Enrico Fermi Plant

Refer to figures 16 through 26, appendix D. In the design of this plant, particular attention has been given to safety and to achieving reliable operation.

It is being built in accordance with ASME and other codes, wherever such codes apply. The schedule for construction provides about 1 year for nonradioactive testing of all of the key components of the plant. A 9-month preoperational program is planned for the period immediately following initial criticality. During this program the expected operating characteristics will be checked in a range of tests beginning at very low power and proceeding upwards in power as confirmation of expected behavior is gained.

A perspective view of the reactor is shown in figure 16. The core and blanket consists of an assembly of square core and blanket subassemblies arranged to approximate a right circular cylinder about 80 inches in diameter and 70 inches high overall. The core, containing the enriched fuel alloy, approximates a right circular cylinder 30.5 inches in diameter and 30.5 inches high; it is completely surrounded by the breeder blanket.

The reactor core, shown diagrammatically in figure 16, is made up of the central portions of 101 subassemblies, 91 of which contain fuel, the remaining 10 being control elements. Fuel is subdivided into a large number of partially enriched uranium alloy pins. The end portions of these 91 subassemblies (the axial blanket), and all the 572 radial blanket subassemblies, consist of uranium alloy that has been depleted in U^{235} and fabricated into cylindrical rods. Plutonium is produced both in the core and in the blanket.

Insertion of boron-carbide poison rods in the core provides regulating and safety control. Regulating control is by two boron carbide rods located near the center of the reactor. Eight safety (shutdown) boron rods are situated at about the half-radius of the core. Both core and blanket are cooled by sodium that is pumped into the bottom of the reactor vessel, goes upward through these sections into a large sodium pool, and flows out near the top of the pool.

Core and blanket subassemblies are loaded and unloaded by an offset handling mechanism mounted in a rotating shield plug, both shown in figure 16. A holddown plate below the plug holds the core subassemblies against the pressure drop forces caused by coolant flow through the subassemblies. This plate and the holddown drive shaft also guide the control element drives. The offset handling mechanism transfers the spent subassemblies to sodium-filled pots in the transfer rotor container, where they decay during the next cycle of operation at power. During the next plant shutdown the spent subassemblies, with their pots, are lifted through the exit pipe into a cask car. The car then carries the spent subassemblies from the reactor building to a decay storage building. In the car, decay heat from the subassemblies and pots is transferred to an inert gas atmosphere which will be circulated through an external heat exchanger that is integral with the

cask car. New subassemblies are inserted by the reverse procedure.

The heat transport system is shown schematically in figure 18. Heat is removed from the reactor core and blanket by the primary sodium coolant, transferred to the secondary sodium coolant in three parallel intermediate heat exchangers, and finally is transferred to water and steam in three once-through type of steam generators. There are three primary coolant loops and three secondary coolant loops.

In both primary and secondary systems, the sodium coolant flow rate is 13,200,000 lb/hr, resulting in an average coolant temperature rise across the reactor or intermediate heat exchanger of 250° F. at a thermal power level of 300 Mw. The primary sodium enters the reactor at 550° F. and leaves at 800° F.

Extensive steps have been taken to assure that sodium will not be lost from the system. These include syphon breaks, secondary containment of the primary sodium system where a single failure would cause loss of coolant, and enclosure of the reactor vessel in a leak-tight primary shield tank so sized and constructed that adequate cooling can be maintained even if the reactor vessel fails.

Failure of the primary system due to thermal shock has been guarded against by extensive use of thermal baffles and bypass flow. The primary shield consists of a 12-inch stainless steel thermal shield inside the reactor vessel and a 30-inch partially borated graphite shield between the reactor vessel and the primary shield tank. The thermal shield, positioned against the inner wall of the reactor vessel, protects the vessel from radiation damage due to fast neutrons and also absorbs gamma rays, thus reducing heat generation within the vessel walls and the borated graphite.

The partially borated graphite shield is designed to moderate and absorb enough neutrons to avoid serious heating within the steel-lined concrete shield wall (2.6 feet thick) that completely surrounds the primary shield tank. This shield wall divides the lower part of the reactor building into an inner reactor compartment and an outer equipment compartment. The latter contains all primary coolant system pumps and heat exchangers as well as decay tanks and other equipment. The shield is designed to reduce the neutron flux in the equipment compartment to less than 10^4 n/sq cm-sec in order to prevent significant activation of the secondary coolant and the equipment in the outer compartment. The steel lining on both faces of the concrete prevents heating within the concrete due to intense Na^{24} gamma rays from the primary coolant, whose activity level is about 0.05 curie/cc.

Neutrons are kept from streaming along the large sodium pipes and into the equipment compartment by installing the greater part of the pipe length within the reactor compartment (i.e., inside the secondary shield), and by enclosing the pipes in a neutron shield. A concrete biological shield wall 7 feet thick is outside the reactor containment building, and a steel and concrete operating floor shield 5 feet thick is above the reactor and equipment compartment to reduce radiation levels to one-third AEC tolerance.

An airtight steel cylindrical reactor building, shown in figure 19, encloses the reactor, the fuel-handling mechanism, the intermediate heat exchangers, and the sodium pumps, piping, and storage tanks. It is 72 feet in diameter and has a wall thickness of 1.125 inches. The purpose of this building is to contain radioactivity from any reactor accident that might release fission products and radioactive sodium. Air in the reactor and equipment compartments below the operating floor is depleted of oxygen and dehumidified to prevent fires in these compartments in case of a sodium leak.

Flexibility has been designed into the system. All mechanisms, including the rotating plug, the subassembly handling mechanism, and the holddown mechanism, can be removed, as can the control drive, rods, guide tubes, and core subassembly support plates. The core subassemblies and the first row of blanket subassemblies are the same size and, with minor modifications, are interchangeable. Conse-

quently, core size can be adjusted if necessary during initial startup to achieve criticality. Furthermore, after some years of successful operation at design power, the core size can be increased to augment the power output.

British Fast Reactor Effort

A following description of the British fast reactor, now going into operation at Dounreay, Scotland, is excerpted from the Second Geneva Conference Paper (15/P/274) by H. Cartwright, J. Tatlock, and R. R. Matthews. This reactor is expected to go critical on August 15, 1959.

The Dounreay fast breeder reactor was first considered as a probable project in 1952, and the early decisions in design were therefore taken 5 and 6 years ago. Although the present layout of the plant is not intended to be a prototype for future systems of this type, it is useful at this stage to note and understand the engineering design philosophy behind some of the principal features of the Dounreay reactor.

The majority of these early decisions of principle tended toward caution and conservatism as the need to achieve complete safety during normal and possible abnormal operating conditions was, and has remained, the feature dominating all design thinking.

In the general approach to the overall design it was considered that the most important objective was to build and operate at power a fast breeder reactor. It was also recognized that the core design with which the reactor would start might well have to be modified in the light of experience.

As well as being a fast fission system, this reactor is the first in the United Kingdom to use a liquid metal coolant on a large scale. Such a circuit has its own problems, and in the Dounreay reactor it was decided not to provide facilities for circuit component testing, nor to attempt to make an economic heat rejection system the prime objective. The philosophy was that the experimental part of the reactor system should be the part in the reactor vessel, and outside this every effort should be made to reduce to a minimum the risk of breakdown of the cooling system. This has in certain instances meant that some of the design problems that will need to be solved in order to build fast reactors at low capital cost, e.g., economic liquid metal/water heat exchangers, have been avoided in the interests of soundness and reliability on this particular project.

The fuel element consists essentially of a tube of enriched uranium, clad on the outside with nobium and on the inside with vanadium. The length of the fuel section is 21 inches, above which is a 6-inch breeder piece of natural uranium. The top and bottom end pieces are of stainless steel. With the high thermal ratings required for economy in fast reactors, upwards of 100 watts/gram of fuel, a large surface-to-volume ratio is required to give reasonable thermal fluxes, and thin sections are needed if maximum fuel temperatures are not to be excessive.

At an early stage of design it was decided to have coolant flow down through the core.

Three basic methods of giving control were originally considered, i.e., movement of the reflector, movement of the fuel or movement of an absorber. The total amount of control available from movement of the reflector was limited and as no effective absorber such as boron–10 appeared likely to be available in sufficient quantity, it was decided to control by movement of the fuel. It is probable that if the choice were made today, the preference would be for boron–10 absorbers as these require less space, the cooling problem is not so severe, and they would not require discharge as frequently as fuel element control rods. The capital cost of boron–10, however, is quite high.

The control of the Dounreay fast breeder is thus achieved by moving 12 groups of 10 fuel elements each, which are situated around the edge of the core. These groups are split up into two safety rods, six control rods and four shutoff rods. The mechanism for operation of the control gear is one of the more complicated features of the Dounreay reactor and illustrates the difficulties of implementing what appeared

at first to be a fairly simple design requirement.

Movement of the control rod mechanism is obtained by the use of a vertical ball-nut and screw actuator, driven through an electromagnetic clutch from an electric motor and gearbox outside the reactor vessel. The use of the electromagnetic clutch eliminates the problem of sealing glands.

The properties and price of sodium make it the most suitable coolant for a fast reactor. It does, however, have the disadvantage that it is solid at room temperatures and consideration was given to the use of sodium potassium alloys for cooling the Dounreay fast reactor. It was recognized that the use of sodium potassium would be an advantage in the early stages of commissioning and operation when liquid metal could be circulated under varying temperature conditions without fear of its freezing. It was eventually decided that the reactor should be started up with the sodium potassium alloy as the coolant and a change eventually made to sodium. This change is unlikely to be made until about 18 months after the reactor has been commissioned.

The decision to have a large number of independent cooling circuits has led to a fairly complicated heat rejection system. To maintain independence each pair of primary pumps and their associated secondary pumps are fed from independent electrical supplies. The power supplies for operation of the Dounreay reactor are obtained from 12 diesel-electric sets with suitable standby, each one of which supplies the power for a pair of secondary circuits and the corresponding primary circuits. Care has had to be taken in the design to insure that the principle of independence has been maintained throughout; for example, in providing standby electrical supplies it was important that these should not be coupled-in to the switchboard in such a way as to link two diesel-electric sets. The principle of separation has meant that each pair of primary coolant circuits is connected to a heat transfer unit which is completely independent of all others right through the steam-raising plant as far as the sea-water cooling system. The resulting steamplant is elaborate and expensive and should not be taken as an indication of the type of steamplant that would be adopted for future fast reactors. It does, however, illustrate very clearly the way in which detailed design can be markedly affected by an overall principle.

The economics of power from fast breeder reactors are materially affected by the costs of fuel cycling and the capital cost of the plant. Because the fast reactor is a highly rated system, there is a distinct possibility of getting a low capital cost installation. Much, however, will depend on developments in the engineering of reliable large-scale liquid metal circuits, and in the skill of the designer in meeting the problems in and around the reactor core, where the very compactness of the system introduces difficulties of design. A great deal of valuable engineering knowledge has been acquired by designing and building the Dounreay reactor and more will be learned from its operation. Experience has underlined the importance of this project as a stage in the development of economic fast breeder reactors in the United Kingdom.

Significant design parameters are shown below:

Design heat output (core)	60 Mw.
Number of fuel elements	367.
Heat transfer surface	215 sq ft.
Average heat flux	531,000 CHU/hr-sq ft.
Design heat output (blanket)	12 Mw.
Number of blanket elements	1,872.
Heat transfer surface	4,690 sq ft.
Average heat flux	4,880 CHU/hr-sq ft.
Primary coolant inlet temperature.	200° C.
Primary coolant outlet temperature.	350° C.
Secondary coolant inlet temperature.	175° C.
Secondary coolant outlet temperature.	325° C.
Primary coolant flow rate	52,000 lb/minute.
Secondary coolant flow rate	52,000 lb/minute.
Steam pressure	200 psia.
Steam temperature	274° C.
Feed water temperature	194° C.
Steam flow rate	3,400 lb/minute.

To support their fast reactor program, the British have considerable facilities. They have extensive fuel fabrication facilities and reprocessing facilities at Dounreay for the fast reactor as well as similar facilities elsewhere. They have critical experiments, such as Zeus and Pluto, at Harwell. There is a sizable effort on metallurgy and fuel development, as well as on physics, being carried on at Harwell and Culcheth. The British are now starting a new study at Risley of a large-scale fast reactor and presently are concentrating on core studies.

U.S.S.R. Fast Reactor Effort

The U.S.S.R. has built three experimental reactors: BR–1, BR–2, and BR–5. Their investigations began in 1949. BR–1 was placed in operation in 1955, BR–2 in 1956, and BR–5 in 1957 or 1958.

BR–1 had a core of Pu and U rods, with a uranium reflector, and a maximum power level of 100 watts, with ambient cooling. Its main purpose was to carry out measurements of neutron cross sections at various energies, and to determine breeding ratios.

BR–2 was a 100 kwt, 10^{14} flux, 140° outlet temperature, mercury cooled reactor fueled with plutonium. Its purpose was (1) to determine neutron cross sections, including elastic and inelastic scattering of materials, (2) capture cross sections of Pu, (3) to obtain operating experience with a liquid metal, (4) to determine stability and reactivity characteristics, and (5) to determine breeding ratios.

BR–5 is a transitional reactor between BR–2, the mercury cooled experimental reactor, and BN–50, a power reactor. BR–5 is a 5,000 kwt, 10^{15} flux, sodium-cooled reactor operating at 932° F. outlet temperature, fueled with UO_2, two loops, one sodium to air, the other sodium-to-water with heat dumped to a condenser. The blanket is part uranium, part nickel. BR–5 has been used to test fuel shielding and components for BN–50, to familiarize personnel with sodium, and to carry out physics experiments at a high neutron flux.

Two power reactors have been designed, BN–50 and BN–250. BN–50 is presumably in a construction phase. It is a 50 Mwe, sodium-cooled, Pu-U fueled reactor, with 10^{16} flux, 900° F. outlet temperature, and a core 26 inches in diameter, L/D equal to 1. BN–250, presumably in a design stage is a 250 Mwe sodium-cooled Pu fueled reactor, with 10^{16} flux, 1000° F. outlet temperature, core 42 inches in diameter, L/D equal to 1, with a specific power of 1,000 kw per liter.

The U.S.S.R. is intensely interested in high breeding ratios using a closed cycle Pu-U system, with some interest in a U^{233}-Th system.

Other Fast Reactor Effort

In addition to the British and Russian efforts there are others outside the USA interested in fast reactors. The Belgians have been actively interested for a number of years. Seven reactor scientists have worked at APDA for 3 years. This group represents a well-rounded nucleus for a design team. The French and Germans are also factoring fast reactors in their programs. The French intend to send personnel to APDA in the near future to develop background for the design of fast reactors. The Japanese have completed some preliminary designs of fast reactors, one of them was designed for an internal breeding ratio of one. This was done to provide a system in which the fuel need never be replaced in the lifetime of the reactor. The cores of these reactors are large and the power output is also. The materials problems associated with maintaining the fuel clad integrity was recognized.

Data

Data for reactors underway are contained in appendix A.

REACTORS UNDER STUDY

Plutonium-Fueled Fast Breeder Reactor

Refer to figures 27 through 31, appendix D. The Plutonium-Fueled Fast Breeder Reactor (PFFBR) under study is an unmoderated, heterogeneous, sodium-cooled fast reactor with a power output of 775 Mw thermal and 300 Mw electrical, fueled with plutonium.

PFFBR is another essential step in the commercial development of economical nuclear power utilizing a full-scale power breeder reactor, designed to operate at substantially less cost than Fermi utilizing Fermi knowledge and experience, and new technology available from research and development programs for Fermi, EBR–I, EBR–II, and Dournreay.

The 300-Mw gross electric output is obtained from heat produced in the reactor which, in turn, produces steam at 870° F. and 1,450 psig to operate a conventional turbine generator. The net plant heat rate is 9,300 Btu/kw hr, resulting in a net thermal efficiency of 37 percent, which is comparable to that obtained in a modern conventional steam plant.

The reactor consists of an assembly of hexagonal shaped core and blanket subassemblies arranged to approximate a right circular cylinder about 6 feet in diameter and 8 feet in height. The core, about 5 feet in diameter and 3 feet in height, is completely surrounded by blanket material; therefore, core subassemblies have both an upper and a lower axial blanket section. The lower axial blanket section is composed of depleted uranium 2¾ w/o molybdenum alloy in the form of plates, and the upper axial blanket is composed of depleted uranium dioxide. The fuel section, located between the upper and lower axial blankets, contains about 400 thin stainless steel pins having a section of mixed oxide fuel at about 89 percent theoretical density. The uppermost 22 inches of the pins are void to provide space for fission gases released from the fuel. The radial blanket subassemblies contain a number of U 2¾ w/o Mo alloy rods bonded by sodium in stainless steel tubes. Altogether there are 188 core subassemblies, 4 control rods, 6 safety rods, 248 radial blanket subassemblies and 46 void spaces in which spent fuel subassemblies can be stored for decay heat removal. All subassemblies are held down against the lifting force of the upflowing sodium by holddown devices located in the bottom of the subassemblies. (Figs. 29 and 30.)

The reactor vessel is a cylindrical stainless steel tank, 9 feet in diameter and 37 feet high having one-half-inch-thick walls. The walls are protected from thermal shock by a single thin steel baffle. A fixed shield plug seals off the reactor vessel. The plug has a central penetration which contains the control rod drives during operation and the fuel handling mechanism during fuel reloading. A second penetration in the plug is for the fuel exit elevator. (Fig. 29.)

Heat is removed from the core and blanket by the circulation of sodium coolant slightly pressurized in three coolant loops. (Fig. 31.) Sodium at 650° F. enters at the bottom of the reactor vessel, flows upward, through and around the subassemblies to a sodium pool above the reactor. It leaves the reactor at 1,000° F., flowing to three intermediate heat exchangers, then through the pumps and back into the reactor vessel. To assure against loss of coolant, the pipes and reactor vessel are provided with double containment. The primary sodium system components are located around the reactor vessel but separated from it by the main shielding wall consisting of alternate layers of iron oxide, serpentine and concrete, each encased in steel. (Figs. 27 and 28.) The shield is cooled

by the forced flow of the below-floor atmosphere (air depleted in oxygen) between the layers. The shield wall is located at a distance of 8 feet from the reactor vessel. Within this space, the primary sodium coolant pipes are so arranged that streaming neutrons are attenuated before the pipes enter the equipment compartment. Beyond the shield wall, the neutron flux is at a biologically safe level.

The reactor and the associated equipment of the primary sodium system are located in the lower section of the 85-foot-diameter containment building. The shielded operating floor over the reactor divides the containment building into two sections. The primary sodium pump motors and the control rod drives are the only pieces of reactor equipment in the upper section. Access to equipment in the lower section is provided by removable plugs in the floor.

The reactor heat is transferred to the secondary sodium coolant system in the intermediate heat exchangers. The heated secondary sodium flows through pipes penetrating the containment building into three once-through sodium-to-water steam generators located in the adjacent steam generator building. From the steam generators, sodium is pumped back to the intermediate heat exchangers. A rupture disc installed on each steam generator protects the equipment from damage from a sodium-water reaction. Feedwater for the steam generators is provided by facilities located in the turbine portion of the plant.

At 3-month intervals, the reactor is shut down for a 3-day period in order to replace spent subassemblies with fresh ones. After shutdown and an initial decay period, the control rod drives are removed remotely by overhead cranes and the fuel handling mechanism is placed into the central penetration through the plug. A spent subassembly is lifted out of the core by the remotely operated handling mechanism and placed into a vacant storage space in the outer blanket where it is allowed to decay for 3 months. A fresh subassembly is brought from the transfer rotor, located at the edge of the containment building, in a sodium filled finned pot, and lowered through the fuel exit elevator port into the reactor vessel. The fuel handling mechanism then places the fresh subassembly into the core. Next a spent and decayed subassembly from the previous unloading is taken from its storage space, transferred to the empty finned pot. The finned pot is then returned to the transfer rotor. The sequence is repeated until the 27 core subassemblies scheduled for replacement have been handled.

The transfer rotor accommodates 45 subassemblies. Exit ports are located on each side of the containment building wall, one being located in the fuel handling and repair building and the other inside the reactor building. Subassemblies may be transferred into or out of the containment building without sacrificing containment integrity and without affecting plant operation. Spent subassemblies are removed from the transfer rotor to the cleanup, storage, and shipping facility.

Considerable flexibility is designed into the plant. All mechanisms can be removed; all equipment, such as pump rotors, IHX tube bundles, etc., that may require maintenance and repair, are removable. Core size can be adjusted for initial startup to achieve criticality or to change power output subsequently, as desired.

The PFFBR design parameters are tabulated in appendix B.

CONSTRUCTION AND OPERATING SCHEDULES

Experimental Breeder Reactor No. 2

The schedule for EBR–II, shown in figure 1 in appendix C, indicates criticality in December 1960. Time lost during construction of the containment building has been made up by expediting the remaining schedule.

Enrico Fermi Atomic Power Plant

The Enrico Fermi schedule, shown in figure 2 in appendix C, can be met if the remaining critical items, such as steam generators and fuel subassemblies, are shipped on schedule. The test facility operation (reactor vessel, plug, handling mechanisms, two safety rod drives, one primary loop) will go into operation in July 1959, and continue operation to the summer of 1960. Remaining construction of EFAPP will be concurrent with test facility operation. The test facility operation will eliminate the need for a shakedown operation after start of criticality operation.

Plutonium-Fueled Fast Breeder Reactor

The schedule for this plant, shown in figure 3 in appendix C, has been based on a design of a second-round fast reactor plant that could be built and operated in 1965. To meet the 1965 schedule the fuel element design should be fixed in 1962. Burnup data could be available by 1962 to determine validity of basic assumptions. A reference and two alternate fuels have been evaluated, each of these being currently considered in the AEC fuel development program. The rest of the reactor plant technology is based on Fermi, EBR–I, EBR–II, and Dounreay with improvements which do not affect feasibility. Prior to 1965 there will also be available the operating experience of these plants.

INHERENT PROBLEMS AND LIMITATIONS

Reactivity and Physics

The requirement for high volumetric concentration of fissionable material in the core is an important limitation on fast breeder reactors. This limitation results in a number of problems:

(1) To hold the critical mass to reasonable limits, fairly small cores are required, resulting in high values of power density.

(2) The high power density requires small subdivision of the fuel which increases manufacturing costs.

(3) The small core size requires a high coolant temperature rise, necessitating protection against thermal shock.

(4) For any given permissible total atom burnup in the fuel, the high fuel concentration increases the number of times the fissionable isotope must go through reprocessing per unit of heat removed.

(5) The high coolant temperature rise results in steep temperature gradients in the core. This produces deformations which may require restraint to avoid undue reactivity changes.

(6) The high fast neutron flux introduces the problem of neutron damage of permanent structural material in or near the core.

In addition to the foregoing problems raised by the requirement of the high fuel consideration, there are a number of other problems:

(1) Details of the energy dependence of capture, fission, and scattering cross sections for fuel, fertile, coolant and structural materials and fission products are not as well known as would be desirable over the wide range of neutron energies present in a fast reactor. At present, this is particularly true for the value of alpha, the capture-to-fission ratio, for the three nuclear fuels. The value of $\bar{\nu}$, particularly for U^{233} and Pu, is not well known.

(2) The large ratio of maximum-to-average heat generation in the radial blanket of a fast reactor gives rise to difficult coolant distribution problems.

Fuel and Materials

Use of PuO_2–UO_2 fuel has uncertainties needing resolution. The thermal expansion properties of the compact need to be known better to determine core stability. The relatively low β for Pu fuels requires careful evaluation. Gas evolution of PuO_2–UO_2 compacts requires further study, test and evaluation. Due to low thermal conductivity of oxides, heat transfer problems of such fuels become aggravated and require extensive research and development. Dilute Pu-fueled critical assemblies have not been operated, requiring a new series of ZPR-type assemblies for Pu fuels. Burnup data is meager.

Fuel cycle costs, including fabrication and processing, have a declining exponential relationship between burnup and costs per kilowatt-hour. For example, at 2 percent burnup a fuel cycle may cost 8 mills per kw hr, while at 14 percent burnup the cost would be down to 2.5 mills per kw hr. High burnups are needed.

Stainless clad for plutonium fuels is adequate up to 1,200° F. For service above this temperature, the development of cladding materials, such as molybdenum, is needed.

Economics (inventory-wise) and heat generation problems dictate that higher burnup of blanket elements become a prime objective. A program should be initiated to study partial thermalization of the blanket, and the use of

larger cores to achieve higher internal breeding ratio.

A method to detect and locate a leaking fuel element is needed.

Fuel Handling

To reduce cost of fuel handling and fuel decay facilities, the following studies and developments are needed:

Operation of mechanisms in sodium at refueling temperatures 400° F. or greater.

Use of heavier loads for mechanisms in sodium.

Inexpensive means to either live with oxide and fog formation or to eliminate them.

Fuel decay facilities less expensive than those now based on water technology.

Simplified design of subassemblies to improve disassembly.

Improved shipping methods including shipping cask standardization.

Scanning under sodium to facilitate inspection, minor repair, and location.

Improved methods of cooling subassemblies on removal from the reactor.

Improved remote maintenance techniques or elimination of need for remote maintenance.

Heat Transport

Present better understanding of thermal transients effects and resultant stresses has reduced this problem considerably but further research and development in this area is needed.

Use of greater than 1,000° F. components may reduce capital costs. This requires mass transport and corrosion material data above 1,000° F., and testing of such items as heat exchangers and steam generators above this temperature.

Elimination of secondary sodium systems is desirable costwise.

Better understanding of reactor shutdown cooling by analytical means, supplemented by tests, is necessary to reduce the shutdown cooling system costs.

A better understanding of sodium vapor behavior at various temperatures and flow conditions can only be obtained by tests.

Safety

Very large increases in reactivity may result if a fast reactor core were compacted due to melting the agglomeration. This is the largest safety limitation on fast reactors. This limitation results in problems in three areas, viz: that of design measures taken to prevent melting, that of design measures taken to prevent agglomeration, and that of design measures taken to contain the nuclear energy release which might occur should steps taken in the first two areas prove inadequate. Some of these problems which pertain to any reactor system are:

(1) The heat removal system must have high integrity to prevent loss of cooling ability.

(2) Experimental and theoretical investigations need to be made of the probable behavior of the fuel upon melting, and the possibility considered of providing design features to prevent agglomeration in case of melting.

(3) Careful and conservative estimates of the maximum credible energy release in case of melting and agglomeration are necessary.

(4) Due to the high coolant activity present when sodium is the coolant, detection in situ of a leaking fuel or blanket element is difficult.

(5) The large amount of fuel normally present in a core subassembly requires that shipping and handling procedures for core subassemblies be carefully studied to prevent criticality accidents during manufacture, shipping, or storage.

(6) The high power density requires that great care be taken to prevent melting of core subassemblies due to decay heating during transfer and shipping operations.

Some problems that pertain particularly to fast reactors are:

(1) Due to the small values of the prompt neutron lifetime, and small temperature coefficients of reactivity, and, for the case of plutonium or U^{233}, the small value of the delayed neutron fraction, great care must be taken in the design to assure that no credible control mal-operation or oscillatory instability can result in fuel melting.

(2) The possibility of positive temperature or power coefficients due to core thermal distortion or due to nuclear effects must be rigorously investigated for each design. Thermal testing is necessary.

Some problems that pertain to all sodium-cooled reactors are:

(1) Cooling system must be designed to prevent inleakage of hydrogeneous materials which could cause large reactivity increases. The fuel loading scheme must be designed to prevent accidents caused by safety rod override.

(2) The use of sodium for cooling requires that adequate provisions be taken to prevent and contain fires or reactions with the thermodynamic fluids.

Appendix A

DATA ON REACTORS UNDERWAY

Description

Experimental Breeder Reactor II.—The EBR–II is an unmoderated, heterogeneous, sodium-cooled fast breeder reactor and power-plant with a power output of 62.5 Mw of heat, and 20 Mw gross of electricity, fueled with U^{235} or plutonium, to be constructed at the National Reactor Testing Station in Idaho.

Enrico Fermi Atomic Power Plant.—The Fermi plant is an unmoderated, heterogeneous, sodium-cooled fast breeder reactor and power plant designed for 430 Mw of heat and 156 Mw gross of electricity (type B core), with an initial power output of 300 Mw of heat and 104 Mw gross of electricity (type A core), fueled with U^{235}, to be constructed on Lake Erie at Lagoona Beach in Frenchtown Township, Mich., 30 miles southeast of Detroit, Mich.

Purpose

EBR–II.—This reactor is primarily a flexible experimental engineering facility to determine the feasibility of this type of reactor for central station power plant application, with major emphasis on achieving high thermal performance at high temperatures, high fuel burn-up with a fast and economical fuel cycle, efficient breeding, pyrometallurgical processing, and remote fabrication.

Enrico Fermi.—This plant is a developmental, full-size power breeder reactor being built as an essential step in the commercial development of economical nuclear power.

Design Parameters

	EBR–II	Enrico Fermi	
		Type A core [1]	Type B core
Fuel and coolant:			
Fuel	U^{235}	U^{235}	U^{235}
Primary system coolant	Sodium	Sodium	Sodium
Secondary system coolant	Sodium	Sodium	Sodium
Turbine system coolant	Water/steam	Water/steam	Water/steam
Moderator	None	None	None
Thermal power rating, kwt	62, 500	300, 000	430, 000
Electrical power rating, kwe:			
Gross	20, 000	104, 000	156, 000
Net	17, 400	94, 000	146, 000
Thermal efficiency, percent:			
Gross	32. 0	34. 7	36. 3
Net	27. 9	31. 3	33. 9

See footnotes at end of table.

Design Parameters—Continued

	EBR-II	EFAPP	
		Type A core [1]	Type B core
Temperatures, ° F.:			
Reactor inlet	700	550	600
Reactor outlet	890	800	900
Steam generator Na inlet	870	750	820
Steam generator Na outlet	610	500	520
Feedwarer temperature	550	340	380
Saturated steam temperature	580	486	532
Superheated steam temperature	840	742	780
Steam pressure, psig	1, 250	600	900
Core size:			
Equiv. diameter, inches	19. 04	29	36
Length, inches	14. 22	30. 5	30. 5
Volume, cu ft	2. 32	11. 65	17. 75
Power density, kwt/ft³ core:			
Maximum	35, 000	44, 170	31, 000
Average	22, 800	23, 000	21, 500
Core power, kwt	53, 000	268, 500	384, 000
Power density maximum to average ratio	1. 53	1. 79	1. 44
Initial fuel enrichment, percent U^{235}	49. 4	25. 6	93
Conversion ratio:			
Core	0. 30	0. 30	0. 40
Blanket	0. 90	0. 90	0. 70
Total	1. 20	1. 20	1. 10
Specific power, kwt/kg U^{235}	314	605	894
Maximum fuel temperature, ° F. (including uncertainty factors)	1, 320	1, 235	1, 325
Core inventory, kg U^{235}	170	444	430
ΔT across core, ° F	190	250	300
Average heat flux, Btu/ft²-hr	680, 000	652, 000	675, 000
Maximum heat flux, Btu/ft²-hr	1, 030, 000	1, 166, 000	1, 167, 000
Control:			
Number of control safety rods	2	8	8
Number of control operating rods	12	2	2
Type of control	Fuel	[2]	[2]
Reactivity worth:			
	∂K/K ¢ INHR	∂K/K ¢ INHR	
Operating	0. 046 6. 30 1, 530	0. 0067 0. 92 255	
Safety	0. 014 1. 92 535	0. 0584 8. 00 2, 230	[3]
Total	. 060 8. 22 2, 065	. 0651 8. 92 2, 485	
Reactivity effects:			
Burnup	0. 011 1. 51 420	0. 0024 0. 33 92	
Fission product buildup	0. 002 0. 27 75	0. 0002 0. 02 6	
Growth	0. 011 1. 51 420	0. 0005 0. 07 20	[3]
Temperature override	0. 003 0. 41 115	0. 0015 0. 20 56	
Total	0. 027 3. 70 1, 030	0. 0046 0. 62 174	

[1] Initial loading.

[2] B-10 poison.

[3] Same as Type A core.

Cross-Section and Views

EBR–II—Figures 9 through 15.
EFAPP—Figures 16 through 26.

Operating Considerations

Activity levels in Fermi

Primary sodium coolant	Na^{24}, Na^{22}.
At full power	Na^{24}, 0.043 curie/cc. Na^{22}, 1.7 microcuries/cc.
After shutdown:	
For 1 day	0.015 curie/cc.
For 7 days	19 microcuries/cc.
For 10 days	1.7 microcuries/cc.
Secondary sodium coolant (Na^{24}):	
At full power	1.5×10^{-4} μc/cc.
After shutdown:	
For 1 day	5.3×10^{-5} μc/cc.
For 7 days	6.7×10^{-8} μc/cc.
For 10 days	2.5×10^{-9} μc/cc.
Primary cover gas A^{41} (at STP):	
Over reactor at full power	146 μc/cc.
After shutdown:	
For 1 hour	98 μc/cc.
For 12 hours	1.5 μc/cc.
For 1 day	0.015 μc/cc.
For 2 days	2.3×10^{-4} μc/cc.
In exit port:	
At full power	0.95 μc/cc.
After 1 hour	0.65 μc/cc.
After 12 hours	0.01 μc/cc.
Nitrogen gas coolant in lower compartment:	
Highest inner compartment activity.	3×10^{-7} μc/cc.
Average activity at full power (STP) in lower building atmosphere.	5.8×10^{-8} μc/cc.
Activity at full power at heat exchangers: Outside building (STP).	2.5×10^{-10} μc/cc.
Waste gases before disposal (from subassemblies cleaning operation):	
Before dilution	
Highest expected contamination	0.006 μc/cc.
Normally expected: no higher than—	1×10^{-5} μc/cc.
Discharge concentration	2×10^{-8} μc/cc.
Waste liquids before disposal (from subassembly cleaning operations):	
Greatest expected contamination of Na clinging to fuel subassemblies.	As NaOH 12μc/cc.
Discharge concentration	8×10^{-9}μc/cc.
Storage pool: Highest water activity expected due to leak.	8×10^{-4}μc/cc.
Miscellaneous cleaning operations estimated contamination highest expected:	
Decontamination operations—liquid.	0.3 μc/cc.
Hot laboratory wastes	2×10^{-5} μc/cc.
Hot laundry wastes	3×10^{-5} μc/cc.
Solid hardware activity (from subassemblies, etc.):	
Stainless steel from core and blanket regions.	0.1 to 20 curies/cc.
Stainless steel within vessel	0.1 to 0.3 curie/cc.

Activity of steel on primary sodium

Components in equipment compartment.	1×10^{-6} μc/cc.
After operation at full power:	
Fuel subassemblies γ=activity per subassembly:	
Inner row (1 percent) at 300 BU Mw.	1×10^{5} curies.
Outer row	8×10^{4} curies.
Axial blanket subassembly, activity per subassembly.	700 curies.
Radical blanket subassembly, activity per subassembly.	1.4×10^{4} curies.

Equipment Problems

Mechanical effects.—Galling of threads, sliding and mating parts presents some mechanical problems. Sodium is a poor lubricant and prevents formation of oxide films that could inhibit galling. Loose running fits, tapered threads, high finish surfaces, and surface treatment, such as nitriding, are design methods to eliminate galling. More development of nitriding for application to sodium components is necessary. Diffusion bonding is also a problem and is treated in like manner.

Remote maintenance equipment.—Remote maintenance of radioactive equipment is a problem requiring considerable development. Remote handling without shielding is a must,

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requiring ingenious designs. Prevention of oxide formation is important if steam cleaning is to be used and the equipment is to be placed back in service.

Fuel handling mechanisms.—Bearing loads in sodium will be evaluated under test. Present load limitations are low and need to be increased.

Foolproof devices to prevent hangup of fuel on transfer are essential and require considerable preoperational testing.

Calibration of handling devices in hot gas is necessary since there is no mechanism available to locate fuel under sodium.

Radioactive inert gas handling.—The Fermi gas-handling system is quite complex and will require considerable testing and possible redesign to simplify, prior to radioactive operation.

Cover gas.—The cover gas is argon. Nitrogen is less expensive, and fission gases can more easily be separated from it, but nitriding problems at interfaces require further research and development.

Steam generator — Once-through. — The Fermi once-through steam generator is the first of its kind in nuclear use. Considerable testing has been done on models, but only actual practice will prove its ability in nuclear plants to effectively produce superheated steam in one unit, particularly with the possible rapid temperature transients.

Heating.—Induction heating and resistance heating are used. These are expensive but are used relatively little and provide an impediment to maintenance. Actual practice will indicate the need for extensive heating.

Vapor traps.—An effective vapor trap—small in size per volume of vapor is needed. Existing units are too large.

Oxide analysis equipment.—Existing equipment for analysis below 0.002 to 0.003 w/o O_2 has not been proven. Plugging indicators and analytical devices exist and are adequate above this range.

Tube sheets.—Heat exchanger and steam generator tube sheet designs, particularly the tube attachments, still present a problem and require considerable effort to improve them at low cost.

Tubes.—Methods of inspecting heat exchanger and steam generator tubes in place is needed. Inspection during construction is not adequate. Internal visual inspection of 80-foot-long tubes is not possible. Ultrasonic testing is limited. More development is needed on inspection.

Nuclear instruments.—High temperature neutron counters are being used in Fermi. Only a few sources are available in industry for such counters. Cable connections are inadequate.

System Problems

Impurities.—Oxygen, hydrogen, carbon and calcium are the prime system corrosion agents. The need to keep these down to a minimum at time of system filling and prevention of their entrance during operation presents many problems. Hydrogen is also a moderator and must be kept out of the system in any form.

System cleanliness, cold trapping, NaK bubblers, effective seals, minimum of lubricants, nonhydrogen containing lubricants, intermediate link between water and reactor are some of the means used to reduce or keep out impurities. Further work is necessary.

Hydrodynamic.—Hydraulic flow problems, such as mixing, oscillatory transients, and system sodium levels, require actual system operation to check out analysis and tests.

Control.—Complex water-steam flow problems intermixed with reactivity control, sodium flow control, and temperature transients require considerable analogue simulation, and finally actual system testing. Better understanding of multiloop system interactions is needed. Start-up systems are also inadequately covered by today's technology. The problem of reactor prime control or steam prime control is always present. Fermi has the reactor as the prime control.

General Maintenance

The largest technical problem is cleaning, the most difficult, remote handling, followed closely by remote maintenance. Research and development on various cleaning agents is imperative. Too little work has been done on cleaning. Steam cleaning is predominantly used—requires extensive cycling and has the danger of chloride corrosion present. Remote maintenance tools have to be adapted for use on maintaining equipment. Extensive monitoring systems and rinsing apparatus require development for specific uses.

Appendix B

PLUTONIUM-FUELED FAST BREEDER REACTOR DESIGN PARAMETERS

Fuel	Pu^{239}
Primary system coolant	Sodium
Secondary system coolant	Sodium
Turbine system coolant	Water/steam
Moderator	None
Thermal power rating, kwt	775, 000
Electrical power rating, kwe:	
Gross	300, 000
Net	283, 000
Thermal efficiency, percent:	
Gross	37. 7
Net	36. 5
Temperatures, ° F.:	
Reactor inlet temperature	650
Reactor outlet temperature	1, 000
Steam generator Na inlet temperature	920
Steam generator Na outlet temperature	570
Feedwater temperature	382
Saturated steam temperature	592
Superheated steam temperature	870
Steam pressure, psia	1, 450

Core size:	
Equivalent diameter, inches	60
Length, inches	36
Volume, ft^3	54. 2
Power density, kwt/ft^3 of core:	
Maximum	25, 350
Average	13, 000
Core power, kwt	705, 000
Power density ratio, maximum to average	1. 95
Initial fuel enrichment, percent Pu^{239}	26
Conversion ratio:	
Core	0. 30
Blanket	1. 04
Total	1. 34
Specific power, kwt/kg Pu^{239}	1, 150
Maximum fuel temperature, ° F. (including uncertainty factors)	4, 250
Core inventory (kg Pu^{239})	674
ΔT across core, ° F	350
Average heat flux, Btu/ft^2-hr	337, 000
Maximum heat flux, Btu/ft^2-hr	657, 000

Appendix C

SCHEDULES

WORK UNDER WAY

PER CENT COMPLETED

Shaded lines indicate official construction schedule

Black lines indicate construction progress to date (February 15, 1959) and current estimated completion dates

UNITED ENGINEERS & CONSTRUCTORS INC.
PHILADELPHIA

SUMMARY CONSTRUCTION SCHEDULE AND PROGRESS CHART

POWER REACTOR DEVELOPMENT COMPANY
ENRICO FERMI ATOMIC POWER PLANT
LAGOONA BEACH, MICHIGAN

DESCRIPTION OF WORK	MONTHS
STRUCTURES	
Office Building	
Control Building	
Sodium Building & Tunnel	
Containment Vessel	
Steam Generator Building	
Ventilation Building	
Fuel & Repair Building	
Health Physics Building	
Gas Handling Building	
SYSTEMS	
Reactor Vessel & Shielding	
Reactor Vessel Mechanical Equipt.	
Primary Sodium System	
Steam Generators & Aux. Equipt.	
Secondary Sodium Piping	
Steam & Feedwater Piping	
Sodium Service System	
Instrumentation & Controls	
Steam Cleaning & Fuel Transfer Sys.	
Detroit Edison Co. Generating Sta.	

1957 J F M A M J J A S O N D
1958 J F M A M J J A S O N D
1959 J F M A M J J A S O N D
1960 J F M A M J J A S O N D
1961 J F M A M J J A S O N D

DATE OF REPORT

INITIAL OPERATION - LOW LOAD

FORM 191 *Date of Original Issue* February 15, 1959

FIGURE 2.—Summary construction schedule and progress chart for Enrico Fermi Atomic Power Plant (EFAPP).

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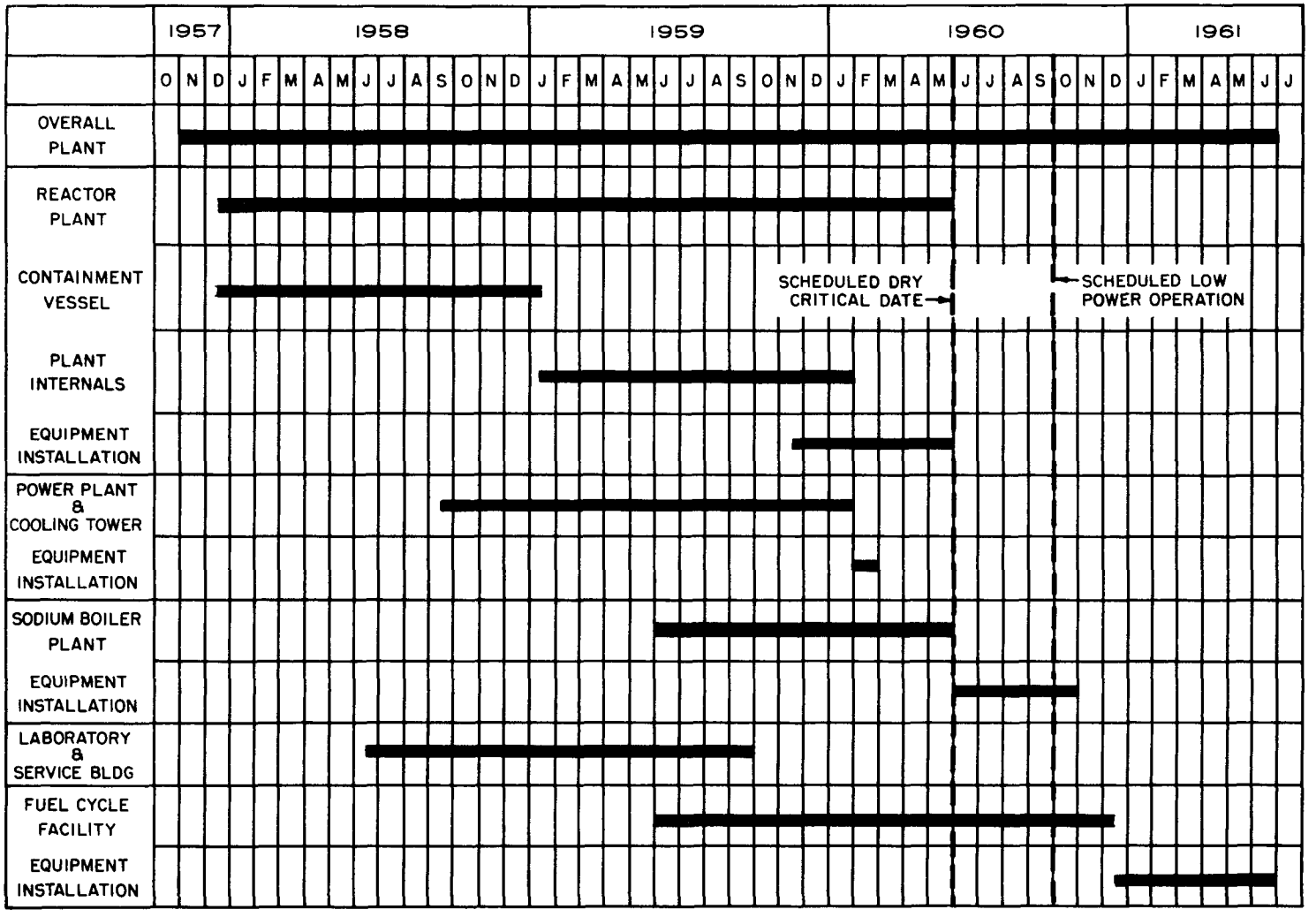

FIGURE 1.—EBR–II construction and operation schedule.

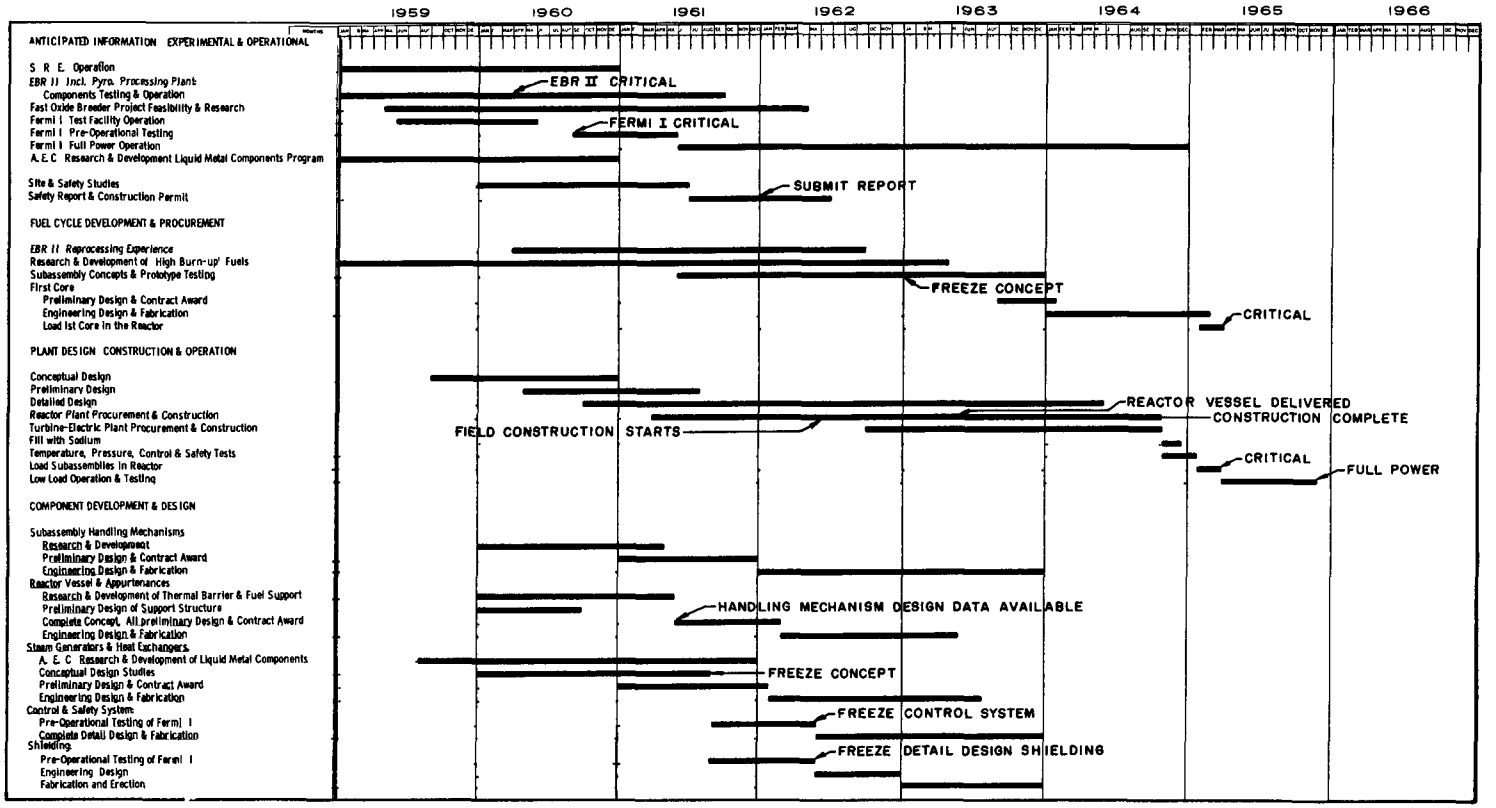

FIGURE 3.—Engineering design, procurement, and construction schedule for the Plutonium-fueled Fast Breeder Reactor (PFFBR).

Appendix D

DRAWINGS

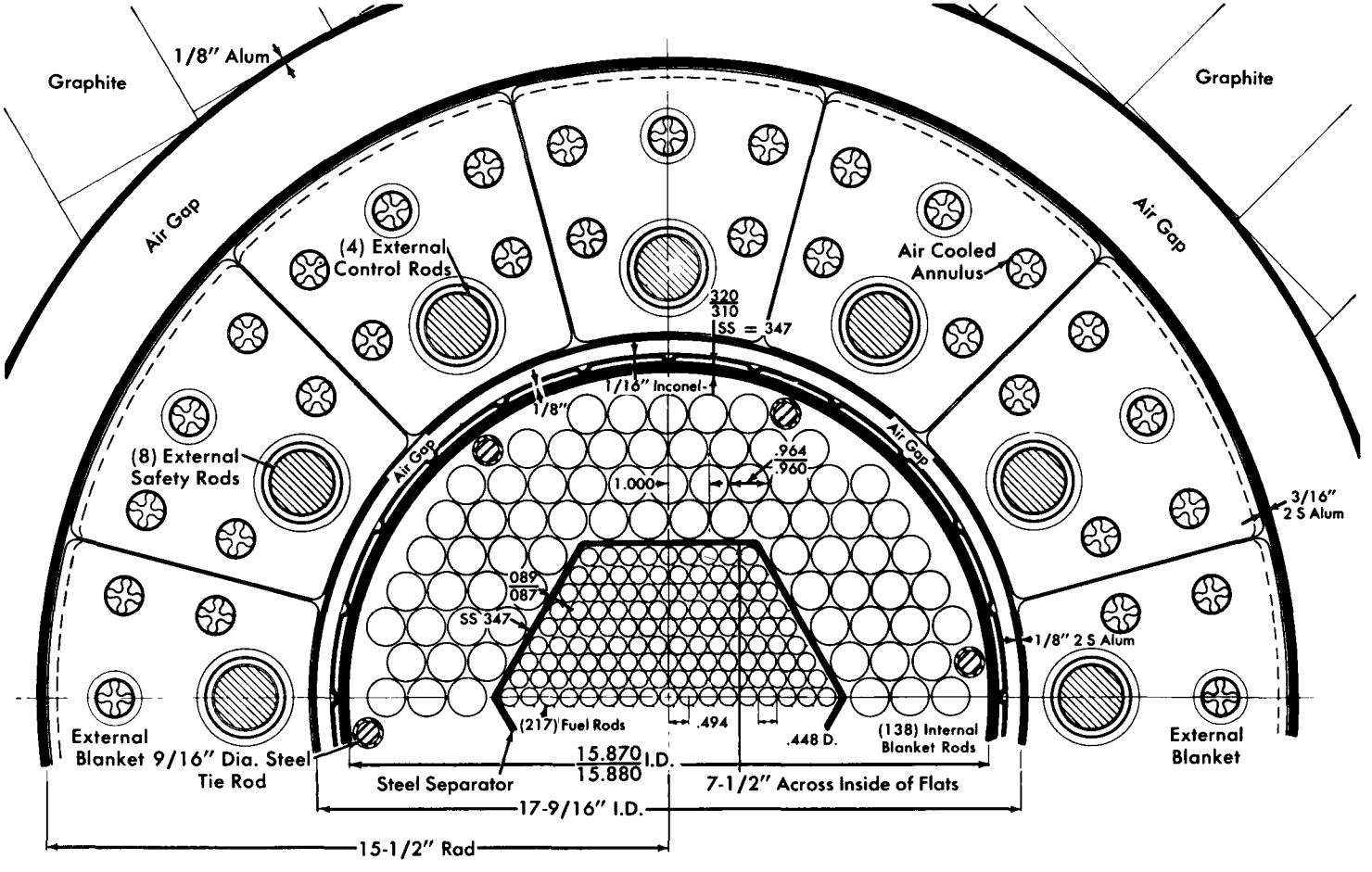

FIGURE 4.—Horizontal cross section at midplane of Mark II core for EBR–I.

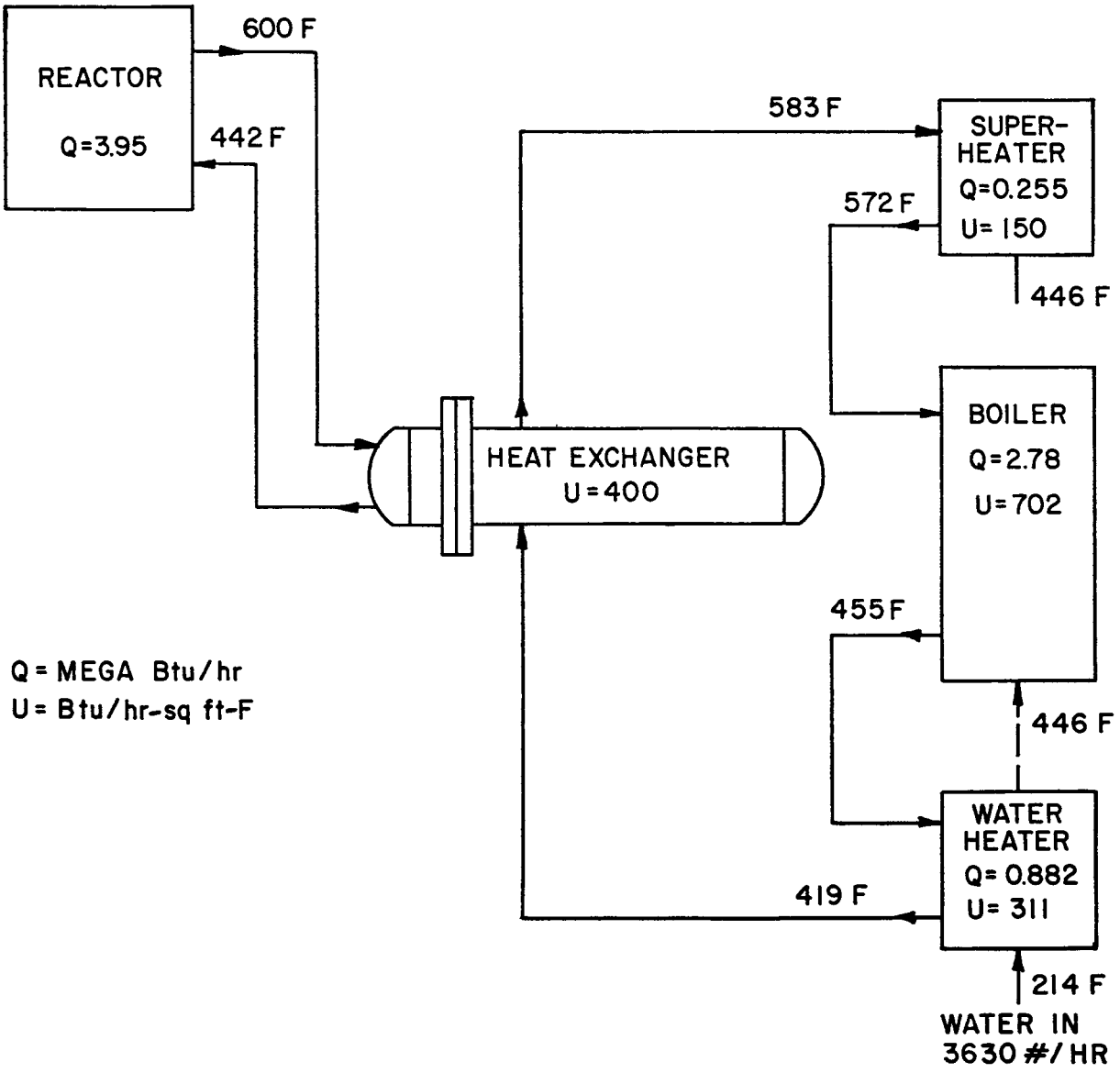

FIGURE 5.—EBR–I heat flow diagram.

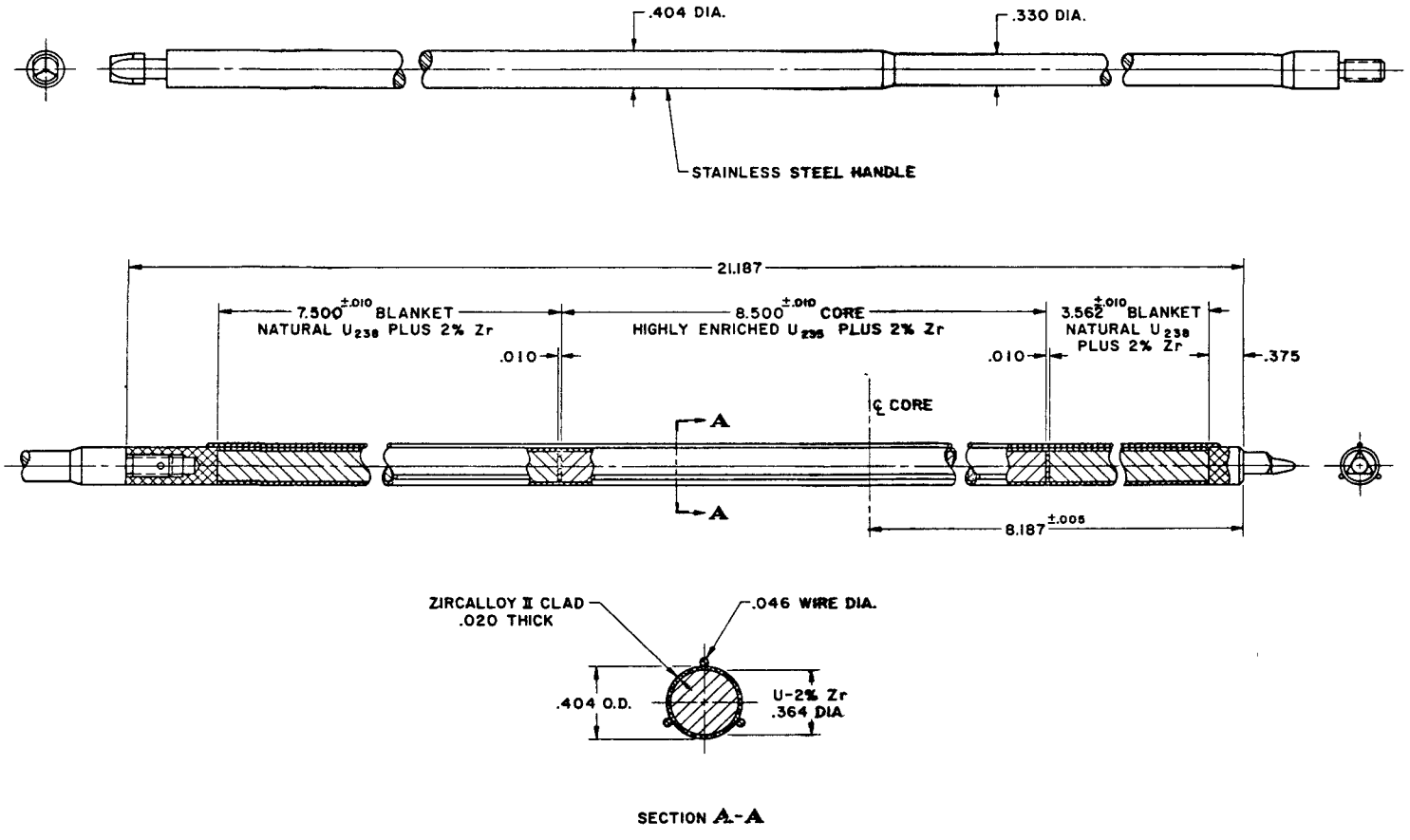

FIGURE 6.—EBR–I Mark III fuel rod.

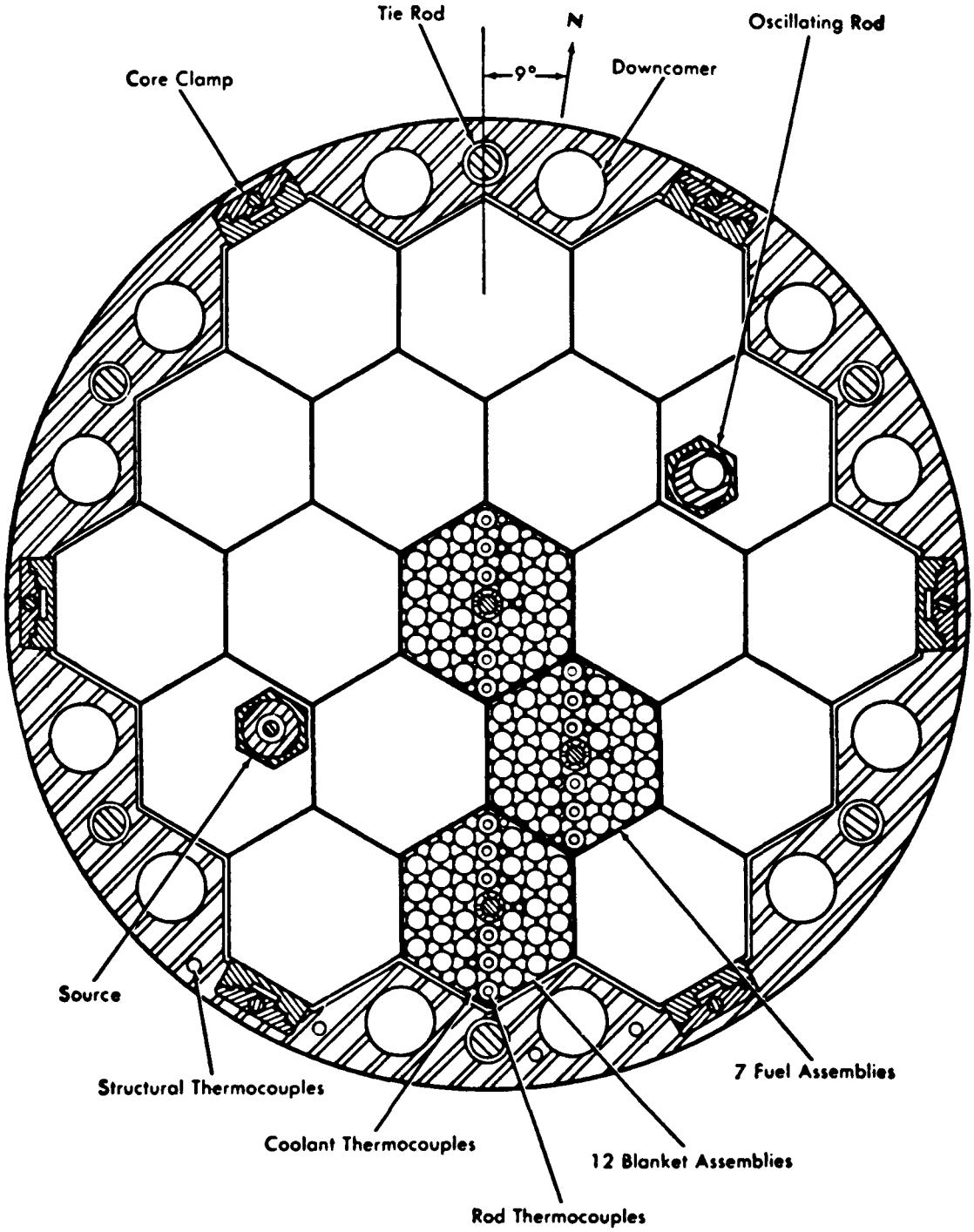

FIGURE 7.—EBR–I Mark III reactor cross section.

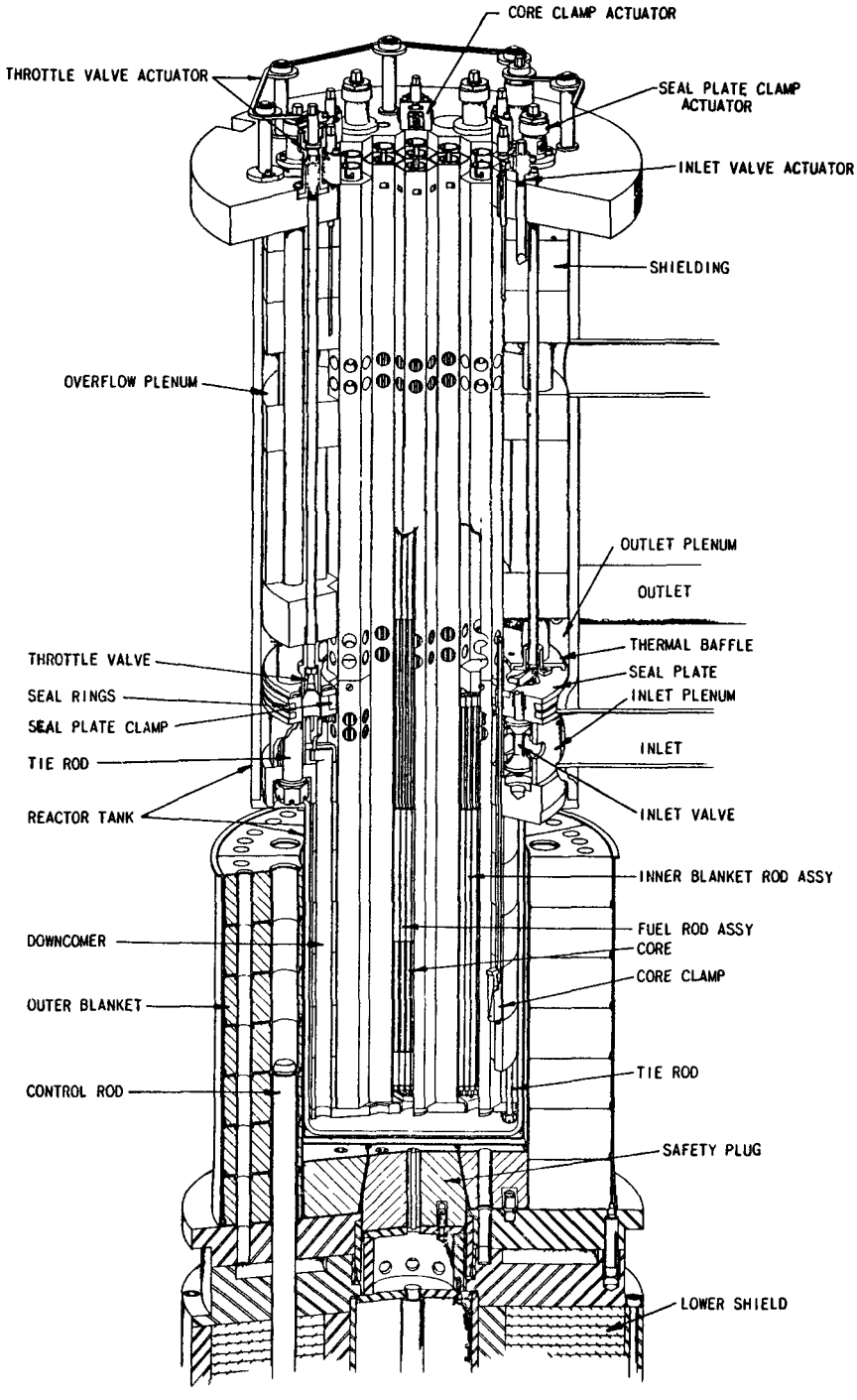

FIGURE 8.—EBR–I Mark III inner tank assembly.

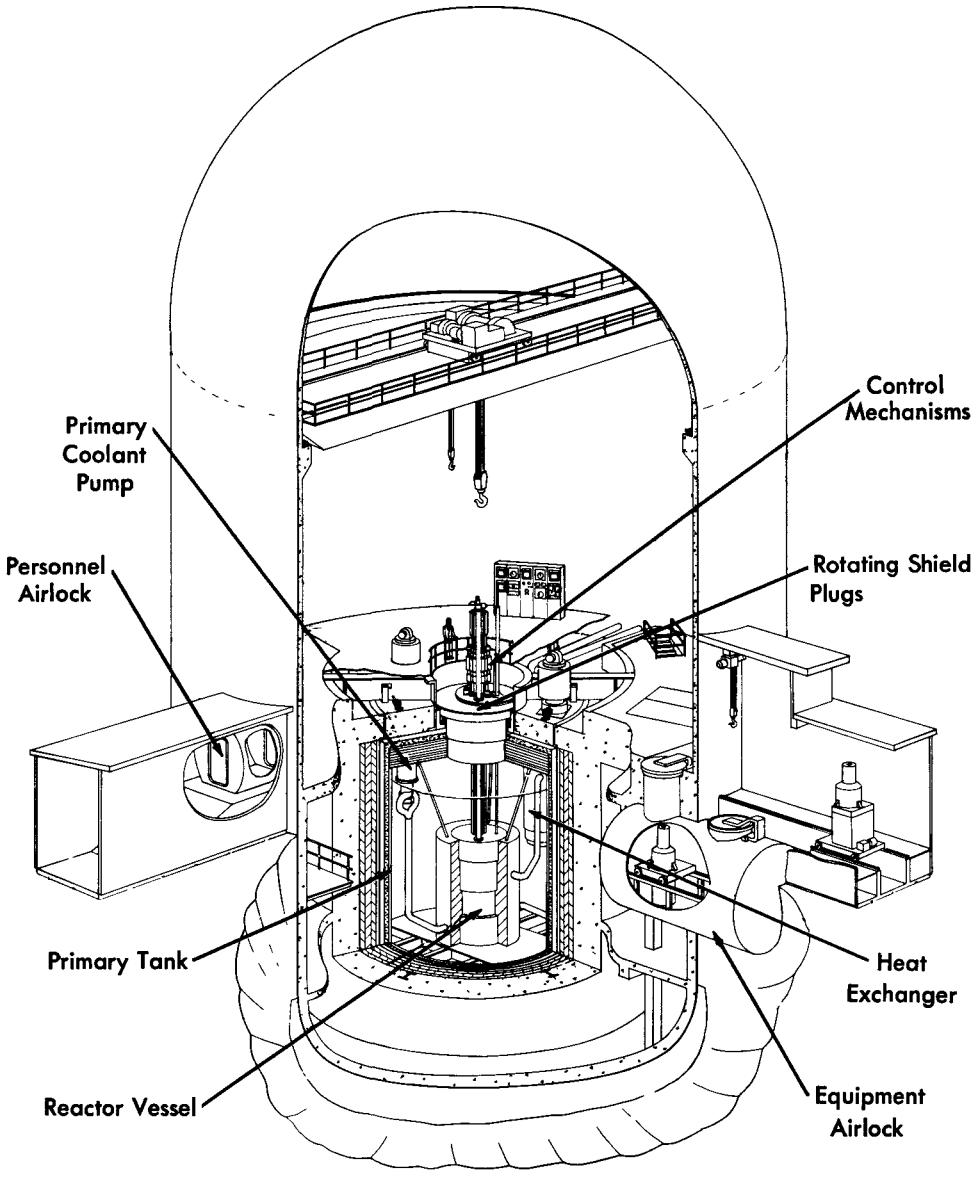

FIGURE 9.—Cutaway drawing of EBR–II plant.

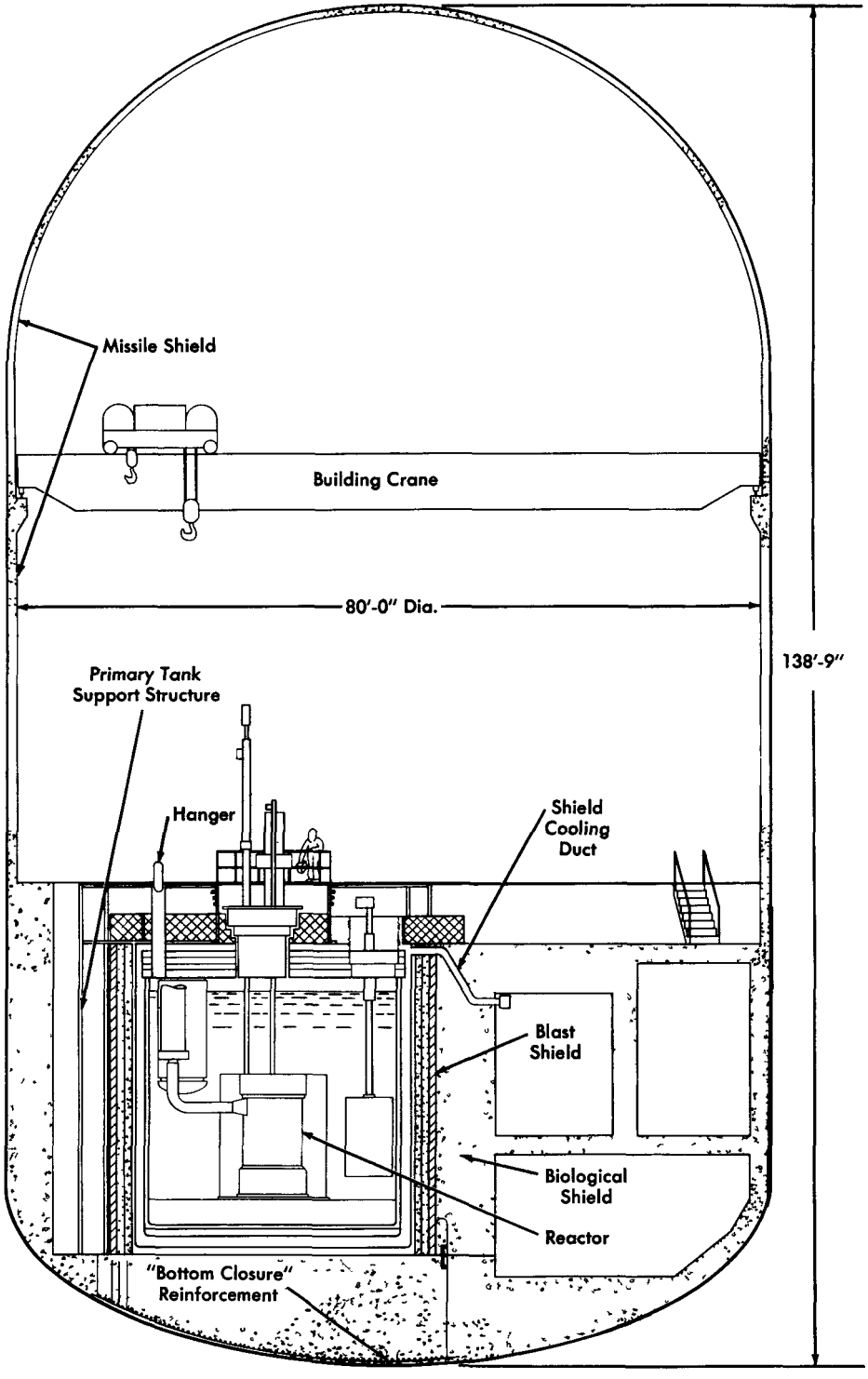

FIGURE 10.—Vertical section of EBR–II plant.

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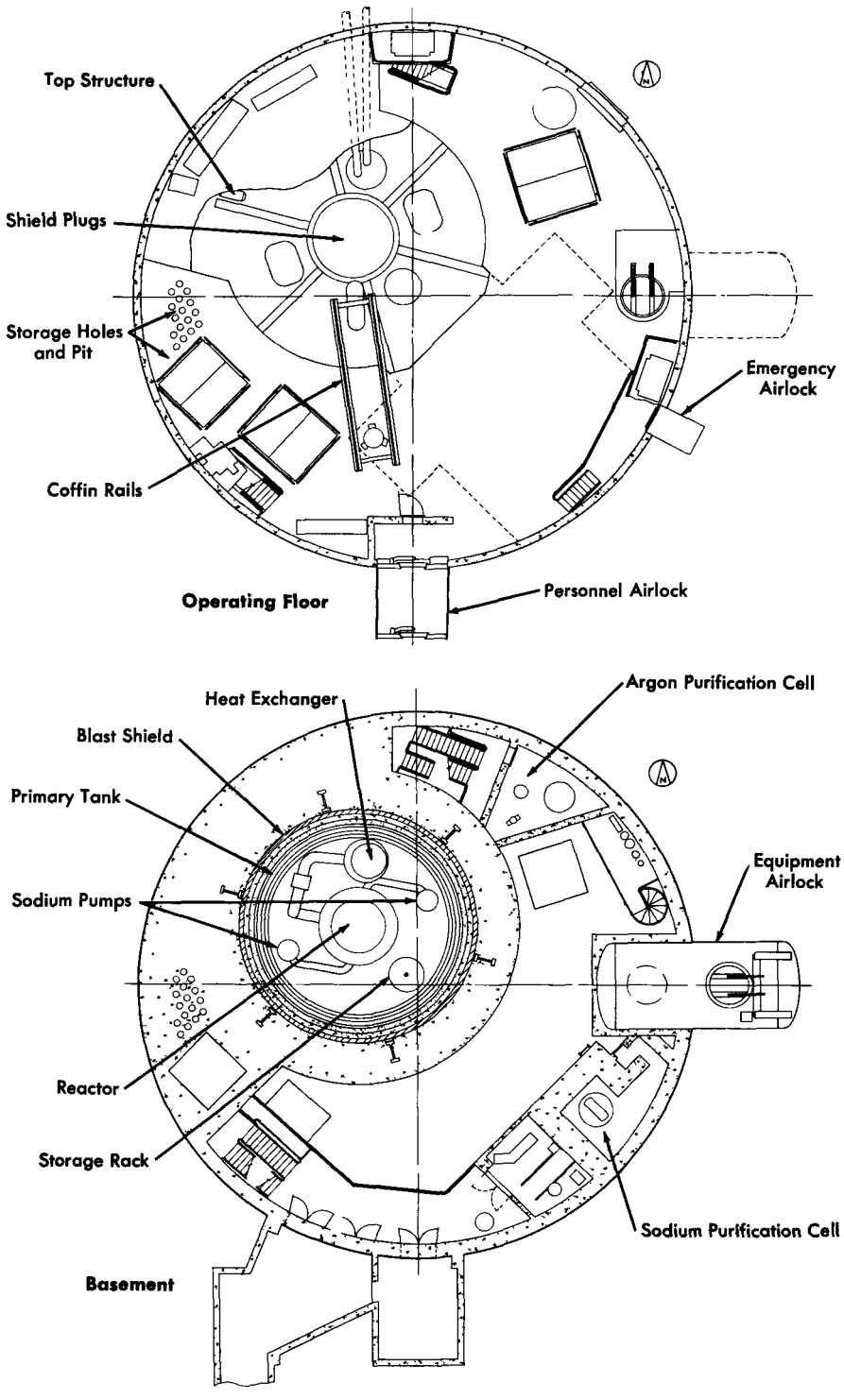

FIGURE 11.—Horizontal sections of EBR–II plant.

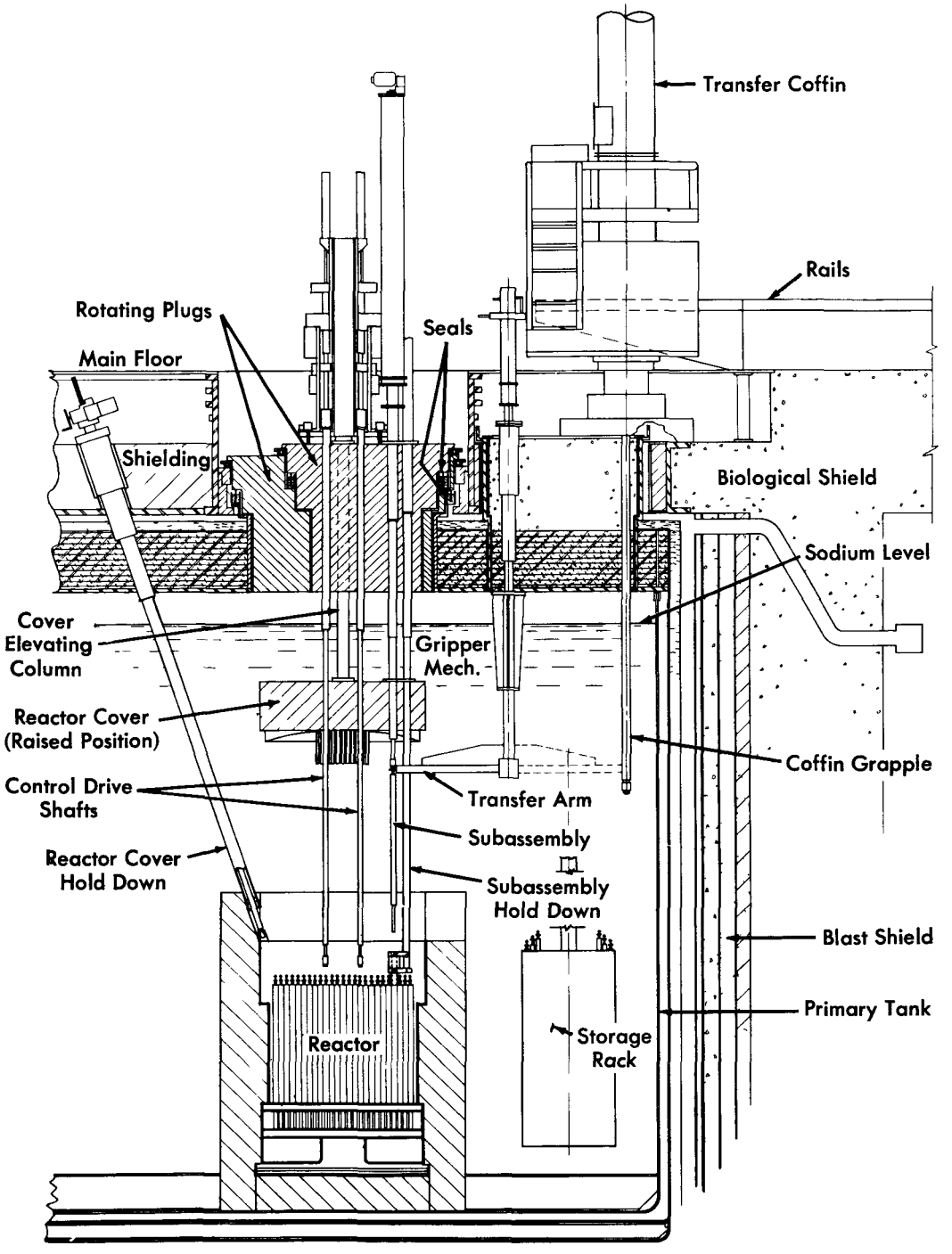

FIGURE 12.—EBR–II fuel handling system.

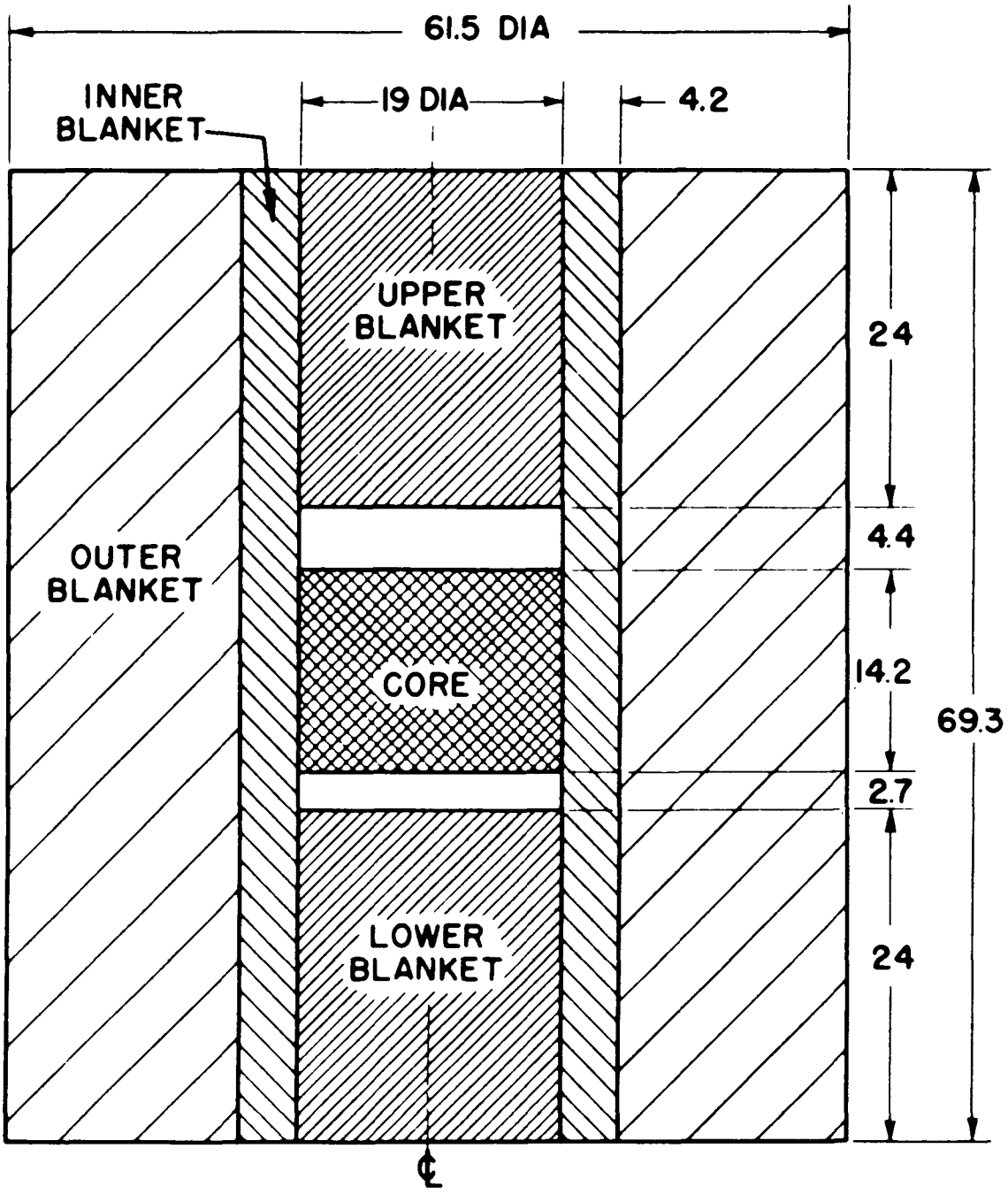

FIGURE 13.—EBR–II reactor elevation.

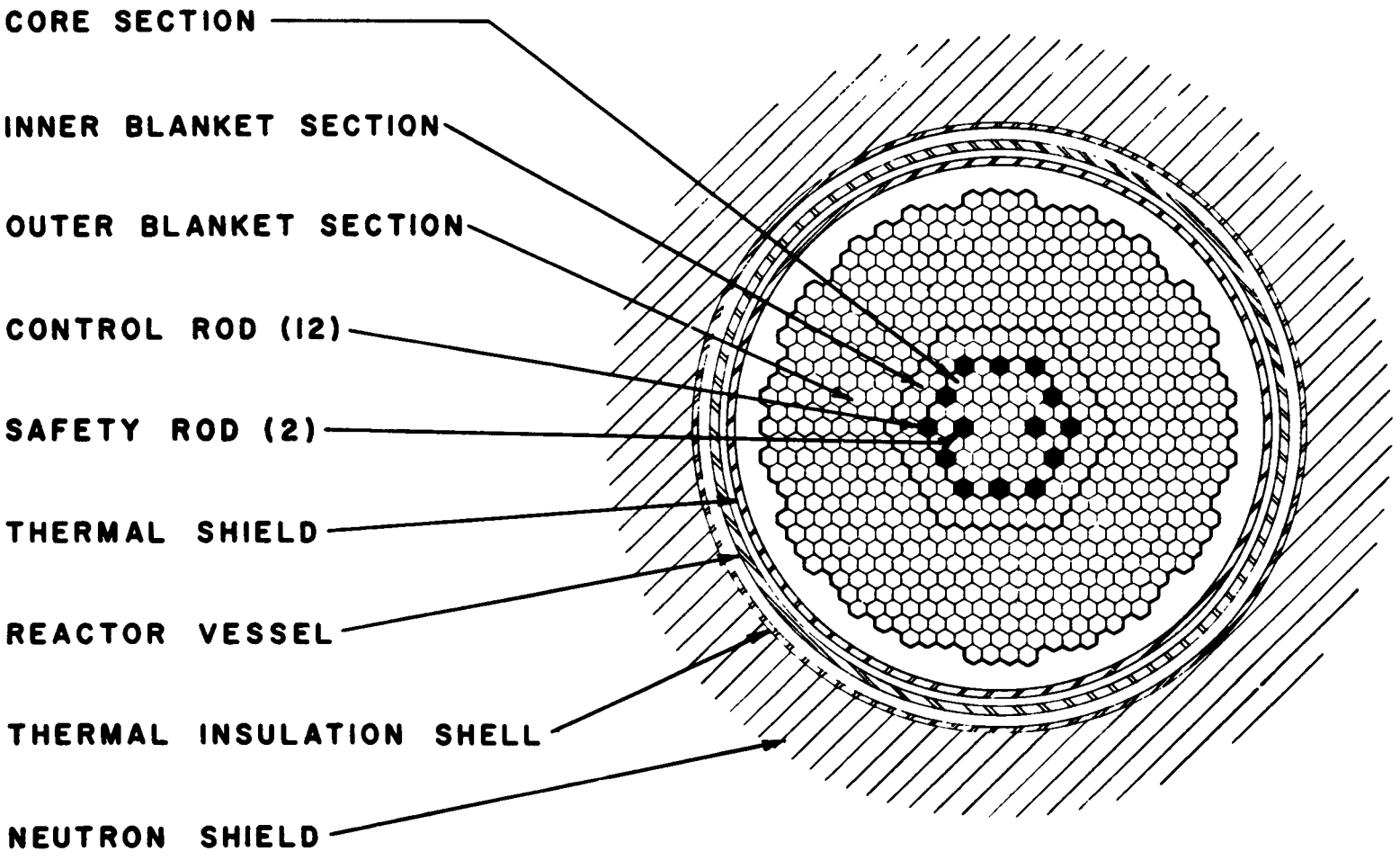

FIGURE 14.—Plan view of the EBR–II reactor arrangement.

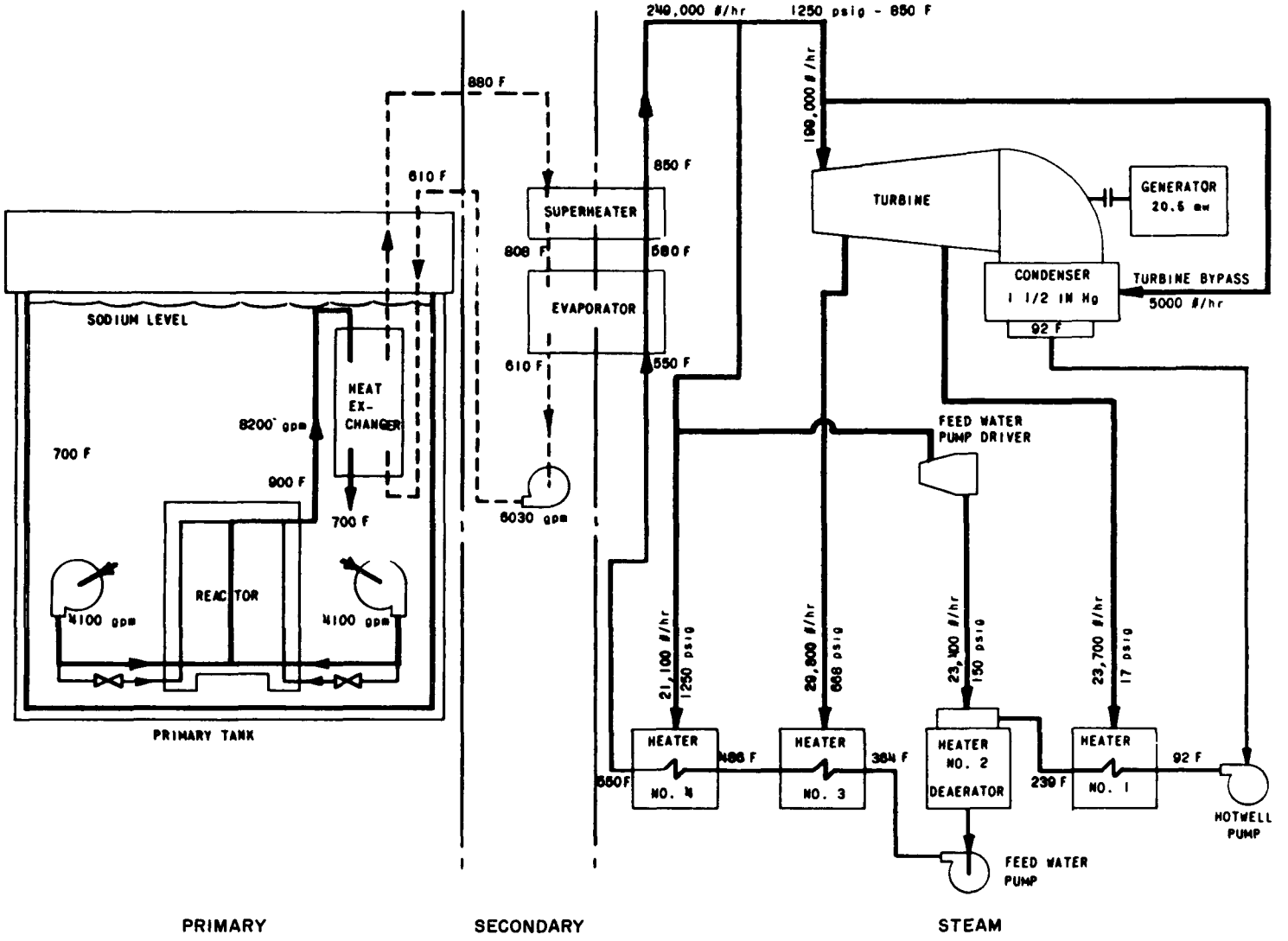

FIGURE 15.—EBR–II flow diagram.

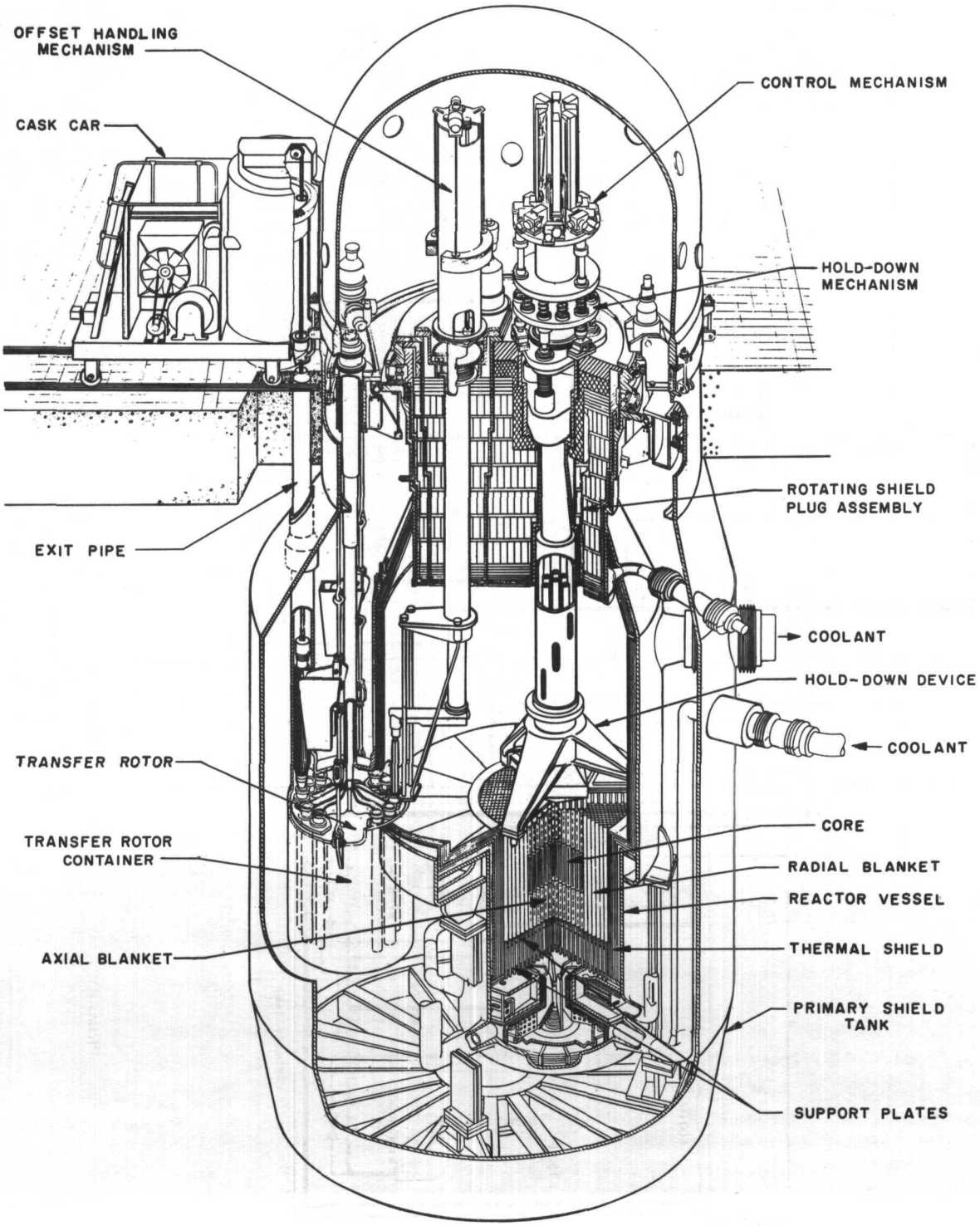

FIGURE 16.—Perspective view of the EFAPP reactor.

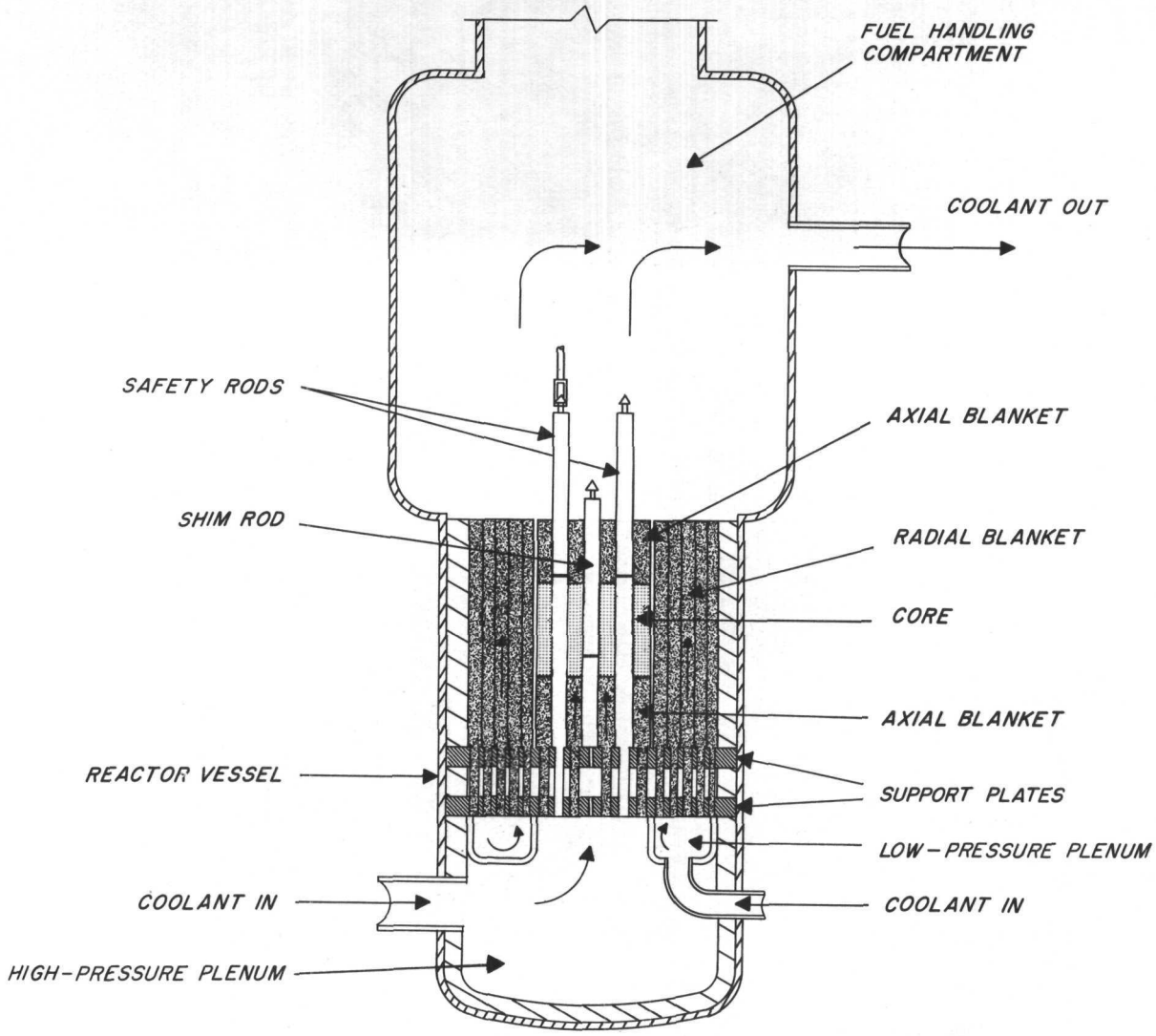

FIGURE 17.—EFAPP reactor arrangement.

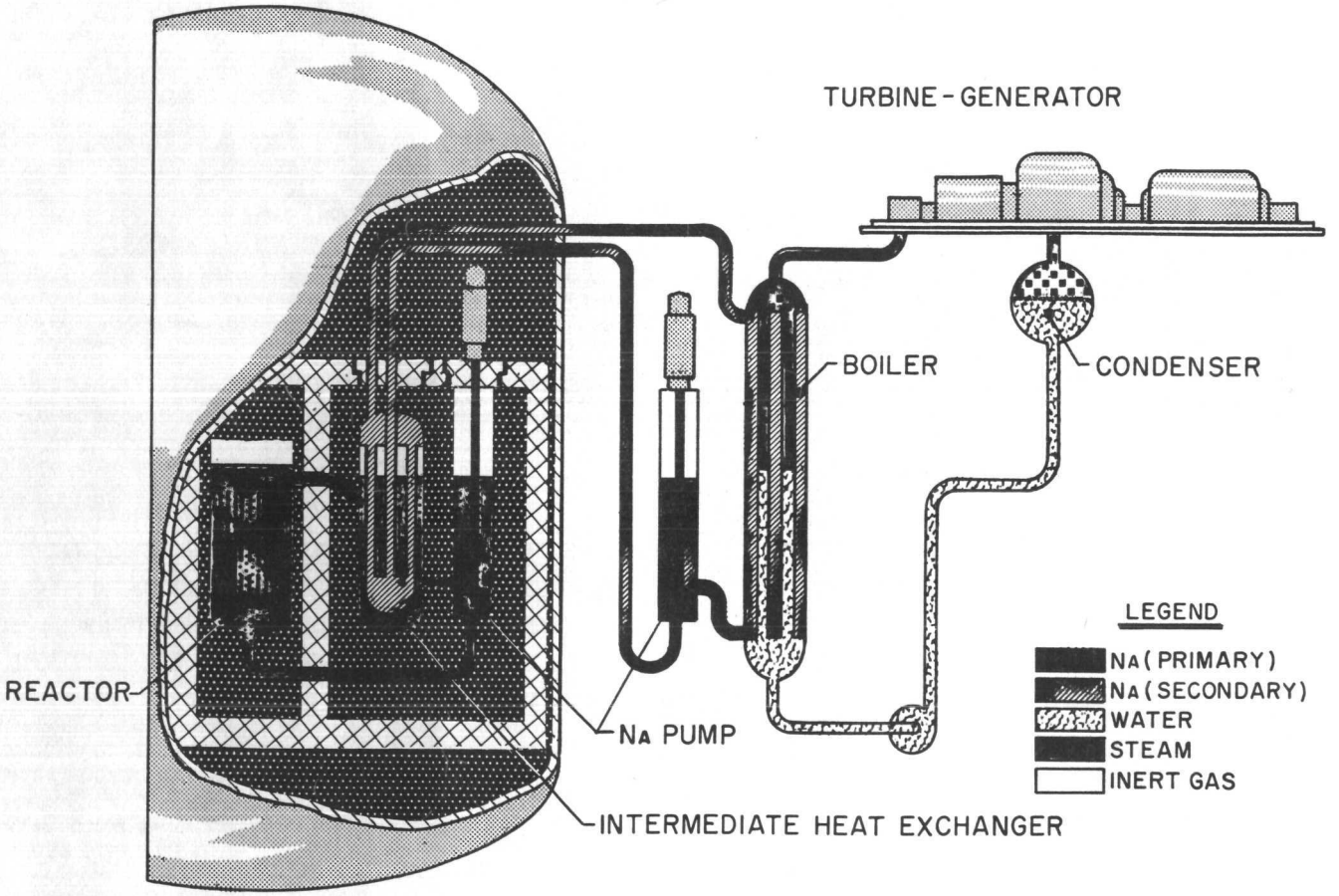

FIGURE 18.—EFAPP plant arrangement.

1. STEAM GENERATOR HOUSE
2. GASTIGHT BUILDING
3. TRANSFER CASK CAR
4. PRIMARY SODIUM OVERFLOW TANK
5. REACTOR
6. PRIMARY SODIUM PUMP
7. INTERMEDIATE HEAT EXCHANGER
8. SECONDARY SODIUM PUMP
9. STEAM GENERATOR
10. SECONDARY SODIUM DUMP TANK
11. CONTROL ROOM
12. TURBINE-GENERATOR

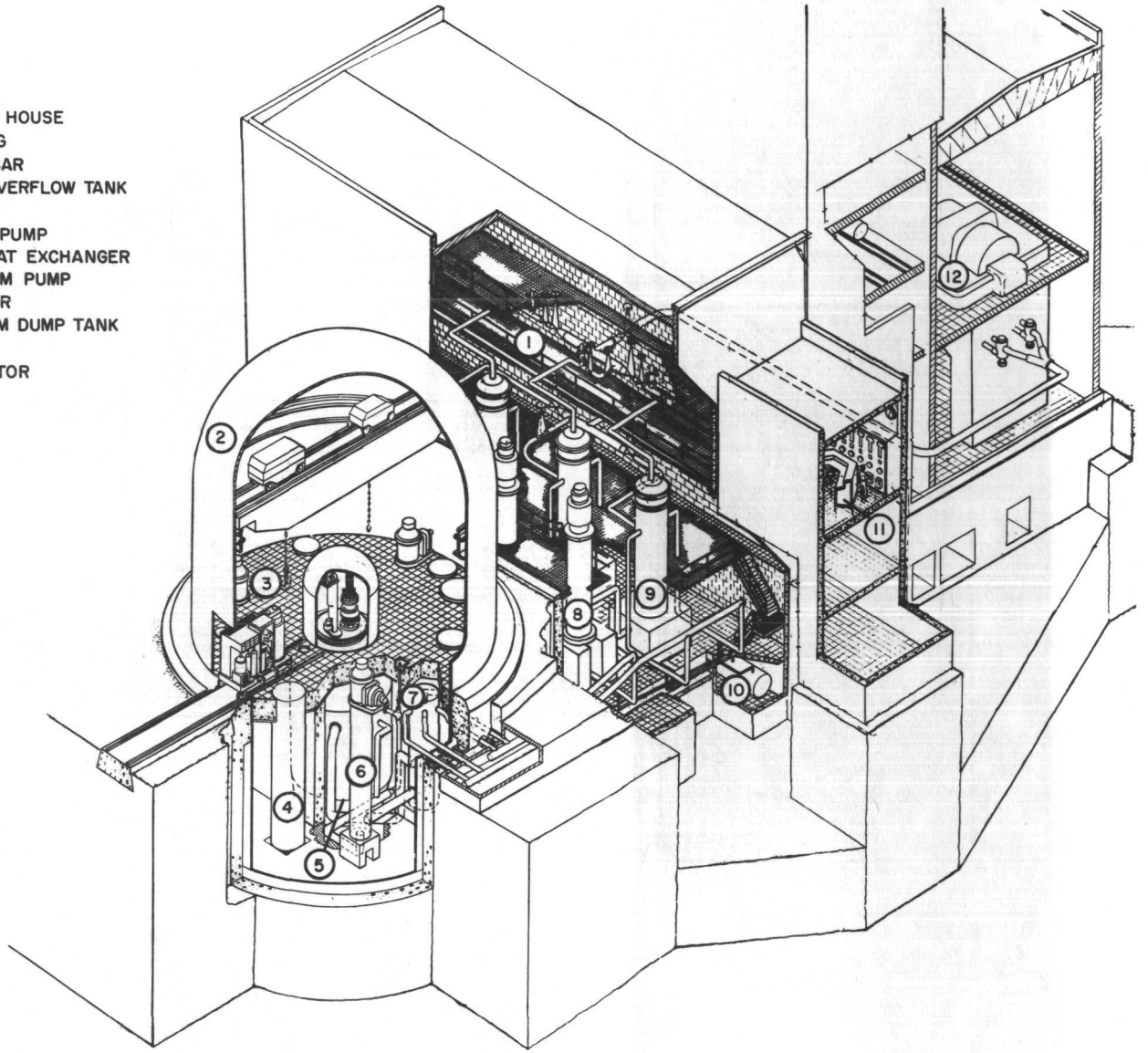

FIGURE 19.—Location of equipment in the EFAPP.

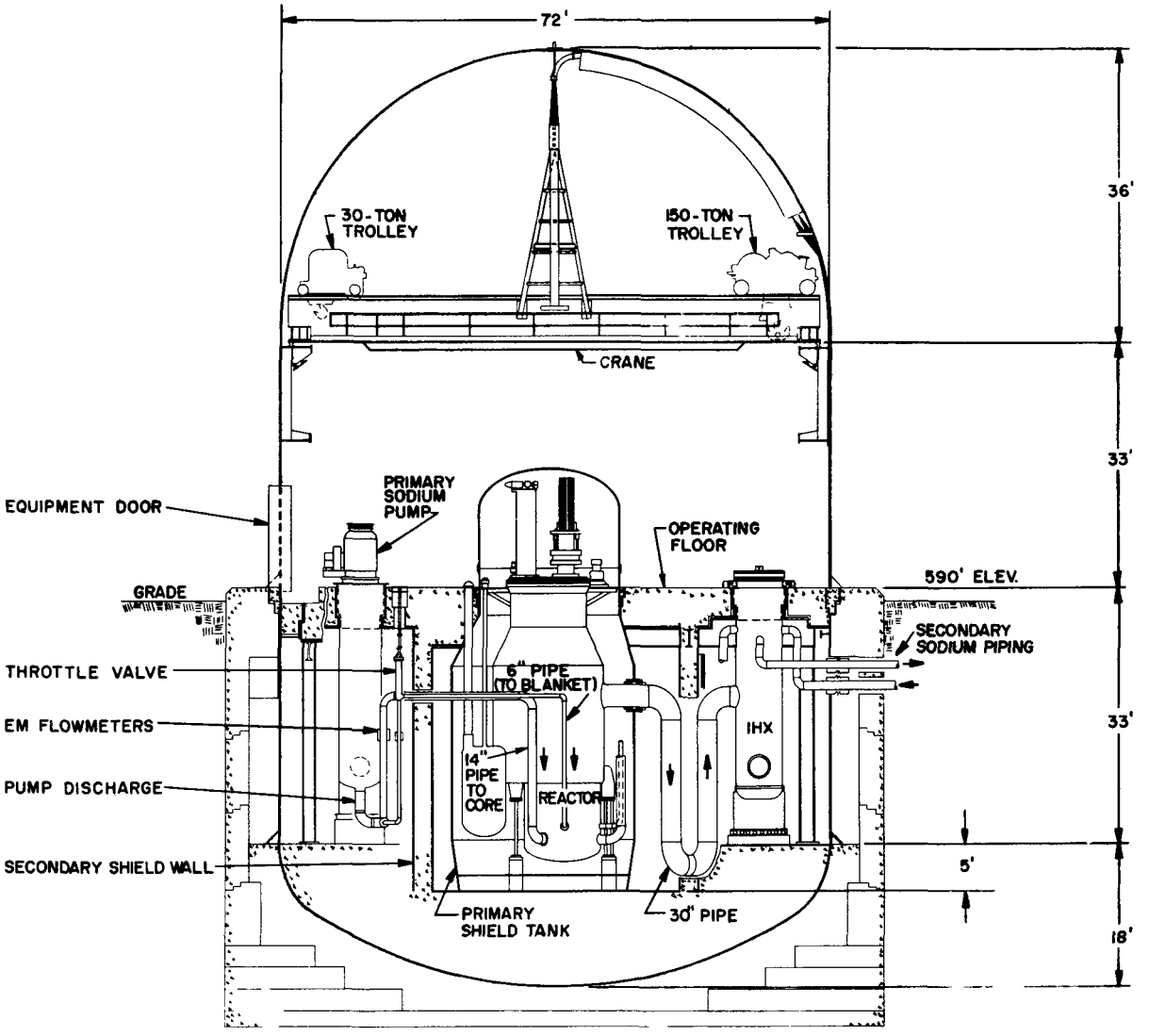

FIGURE 20.—Elevation of reactor plant (EFAPP).

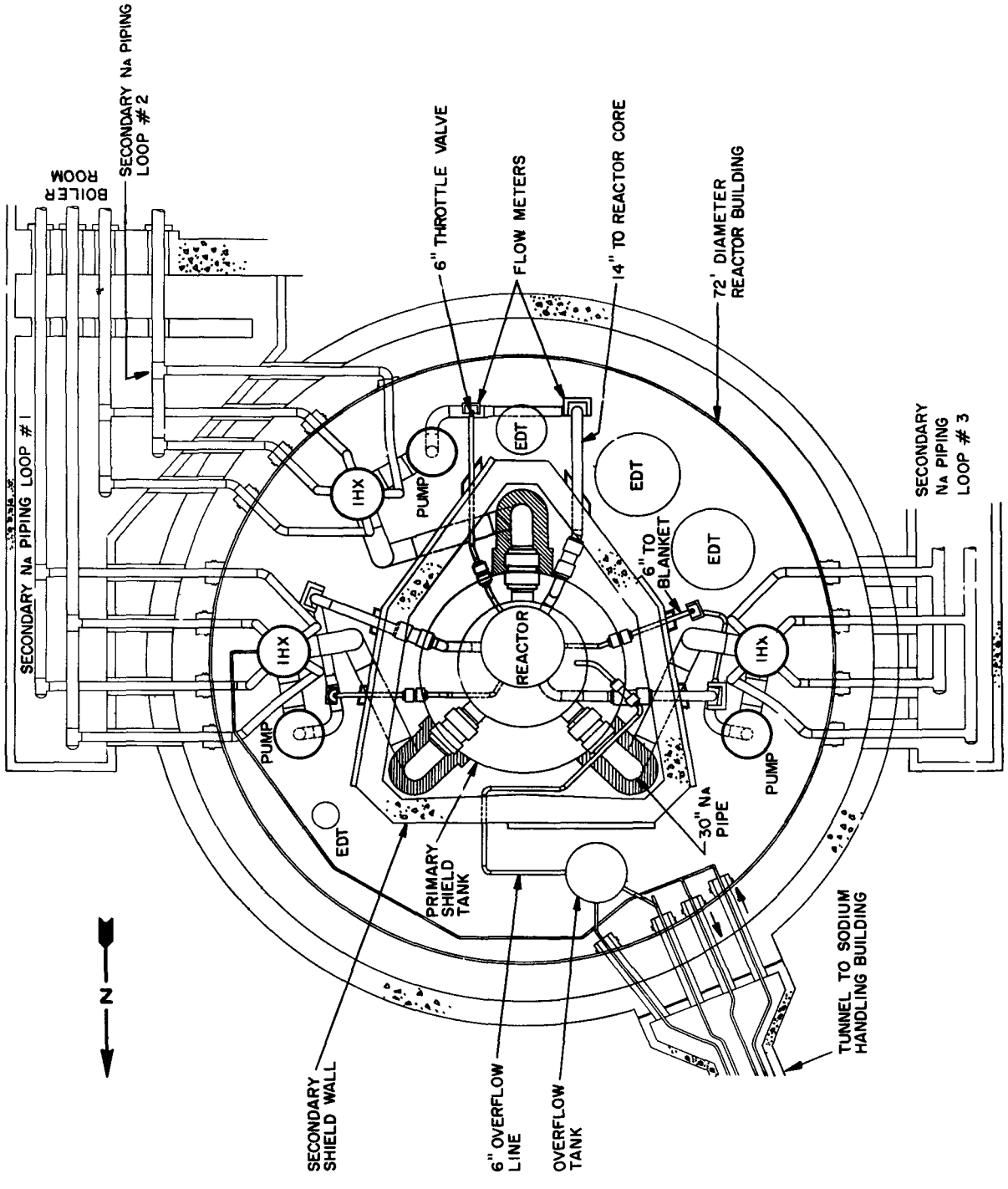

FIGURE 21.—Plan view of reactor plant below operating floor (EFAPP).

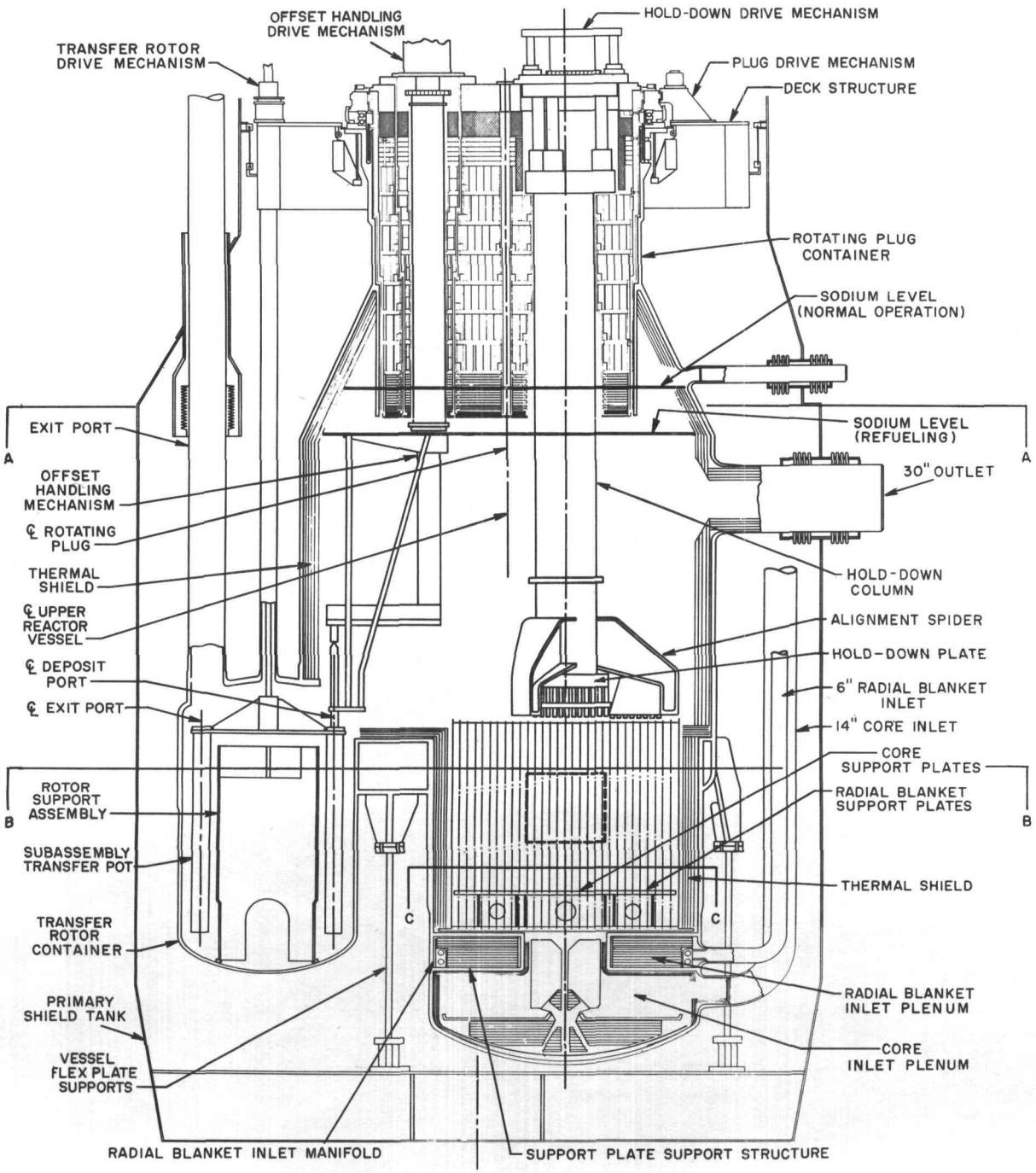

FIGURE 22.—EFAPP reactor vessel (elevation).

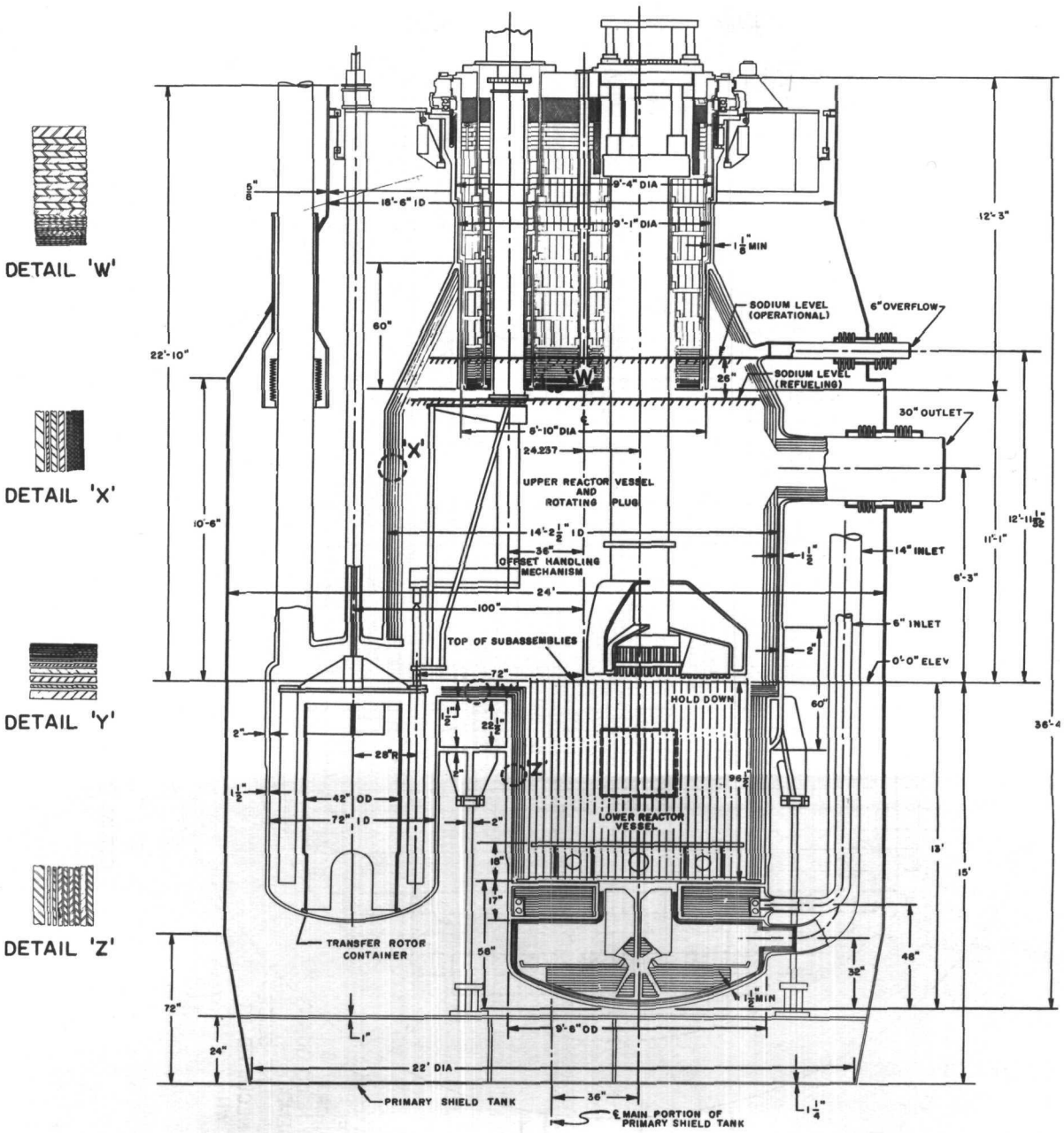

FIGURE 23.—EFAPP reactor vessel (dimensional).

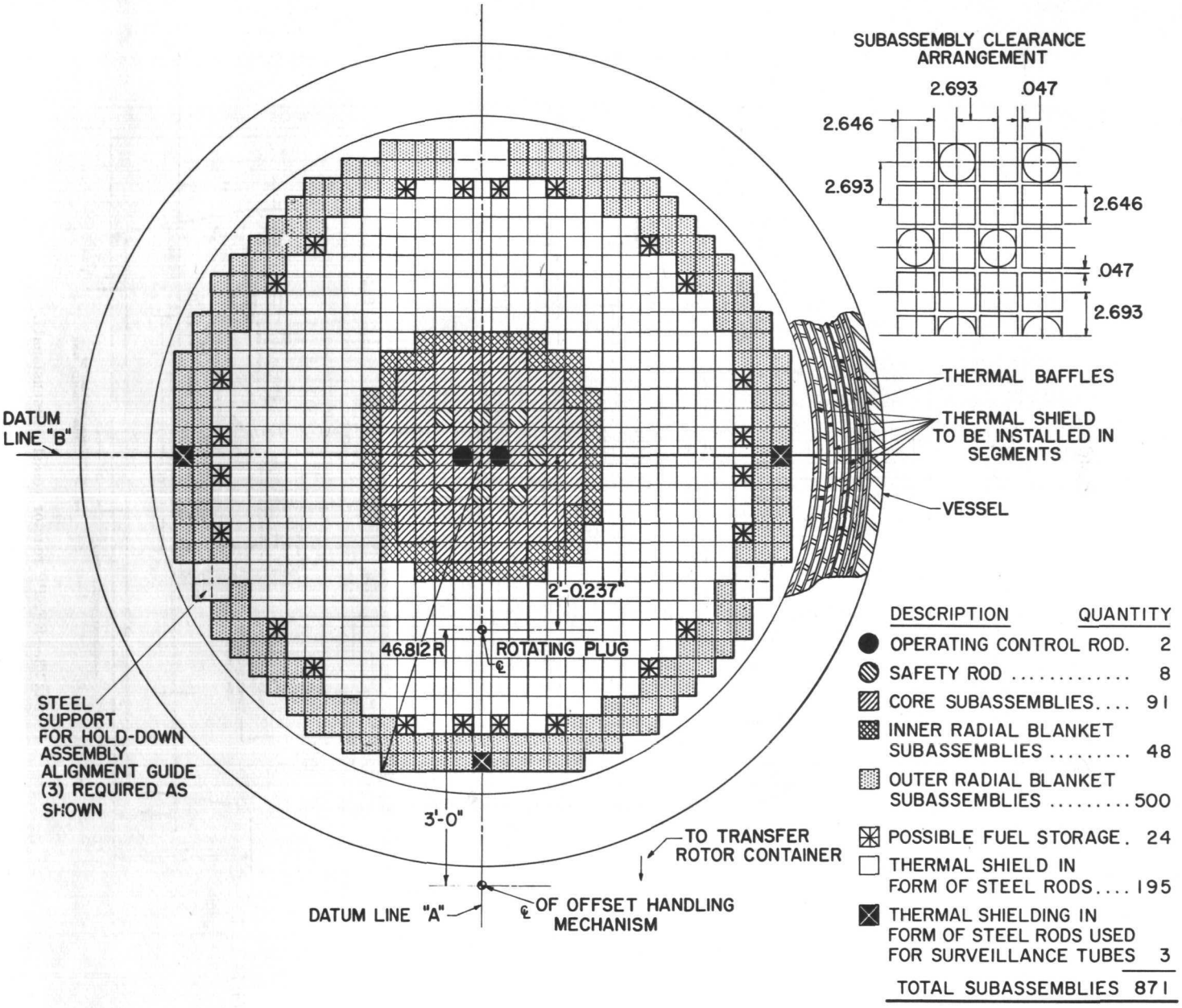

FIGURE 24.—EFAPP reactor cross section.

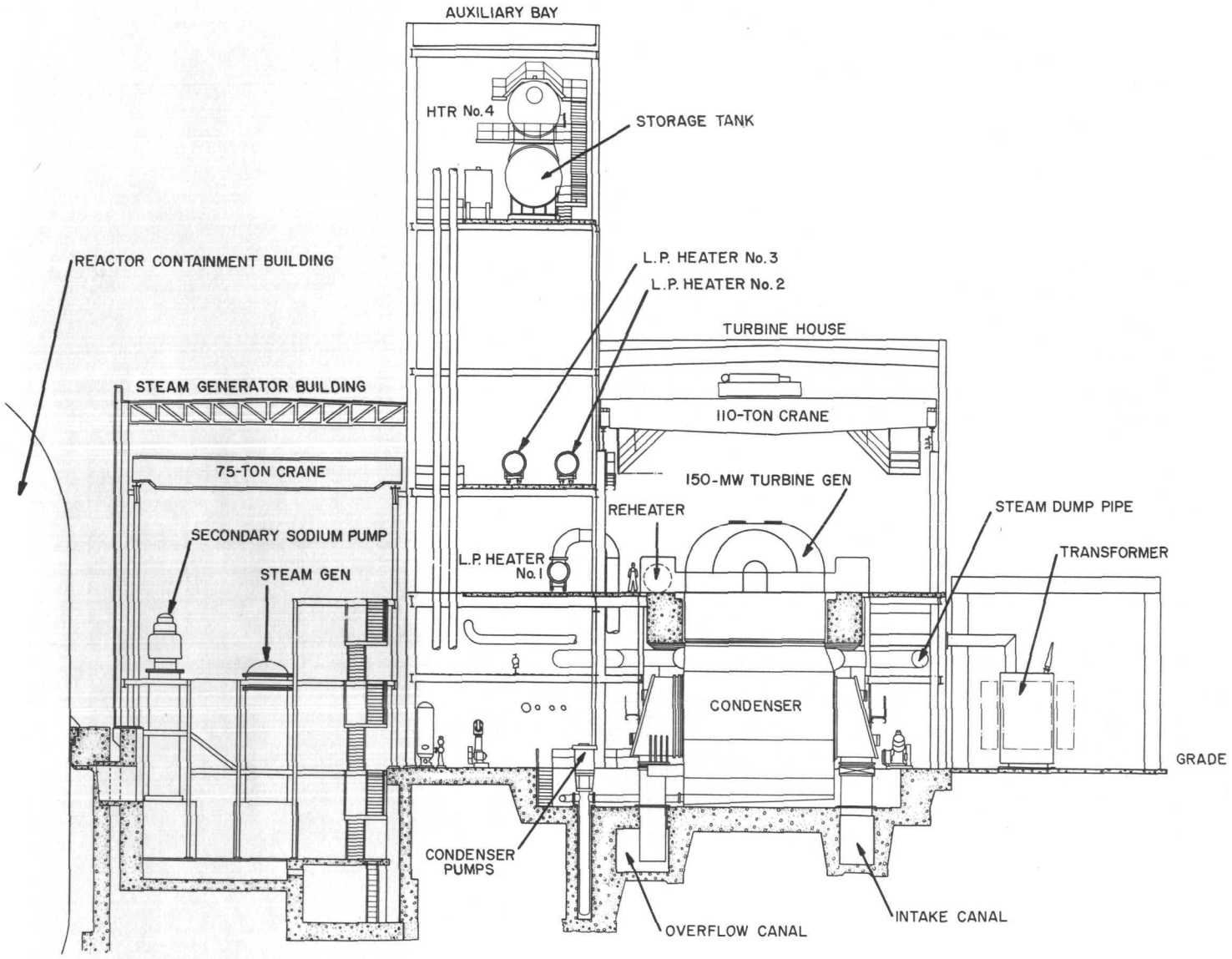

FIGURE 25.—Cross section of steam generator building, auxiliary bay, and turbine house (EFAPP).

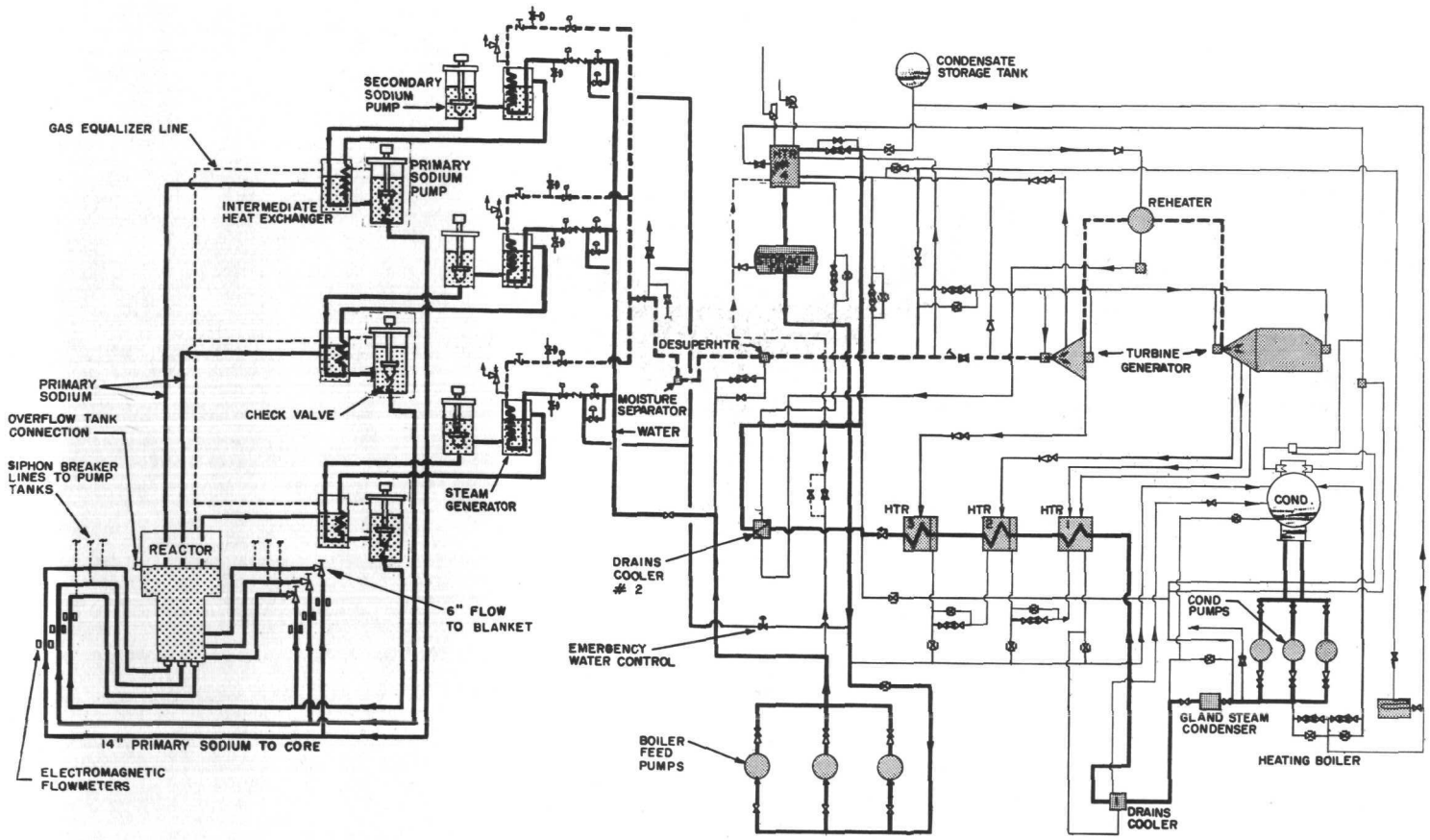

FIGURE 26.—Flow diagram of the nuclear power plant (EFAPP).

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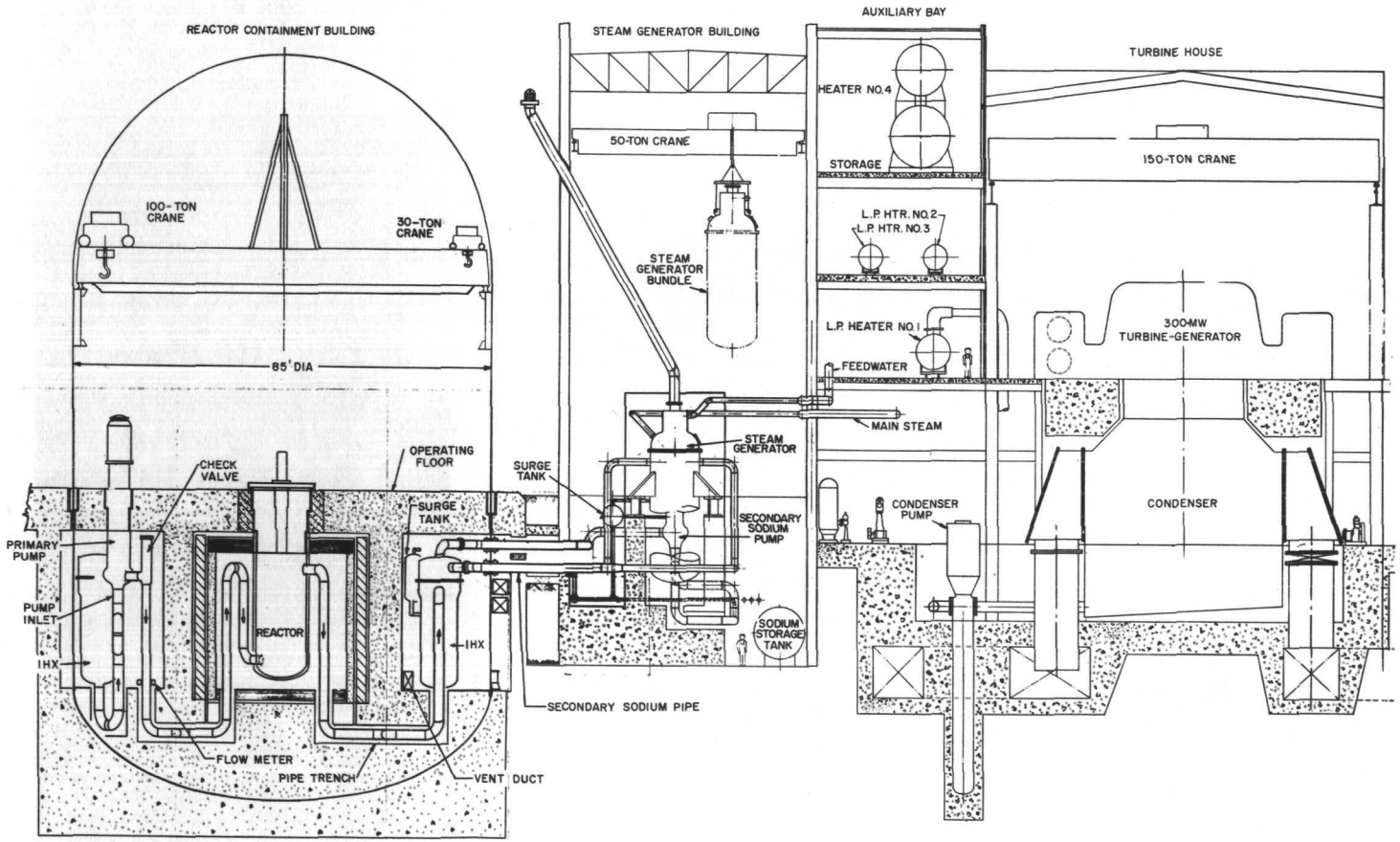

FIGURE 27.—PFFBR plant layout.

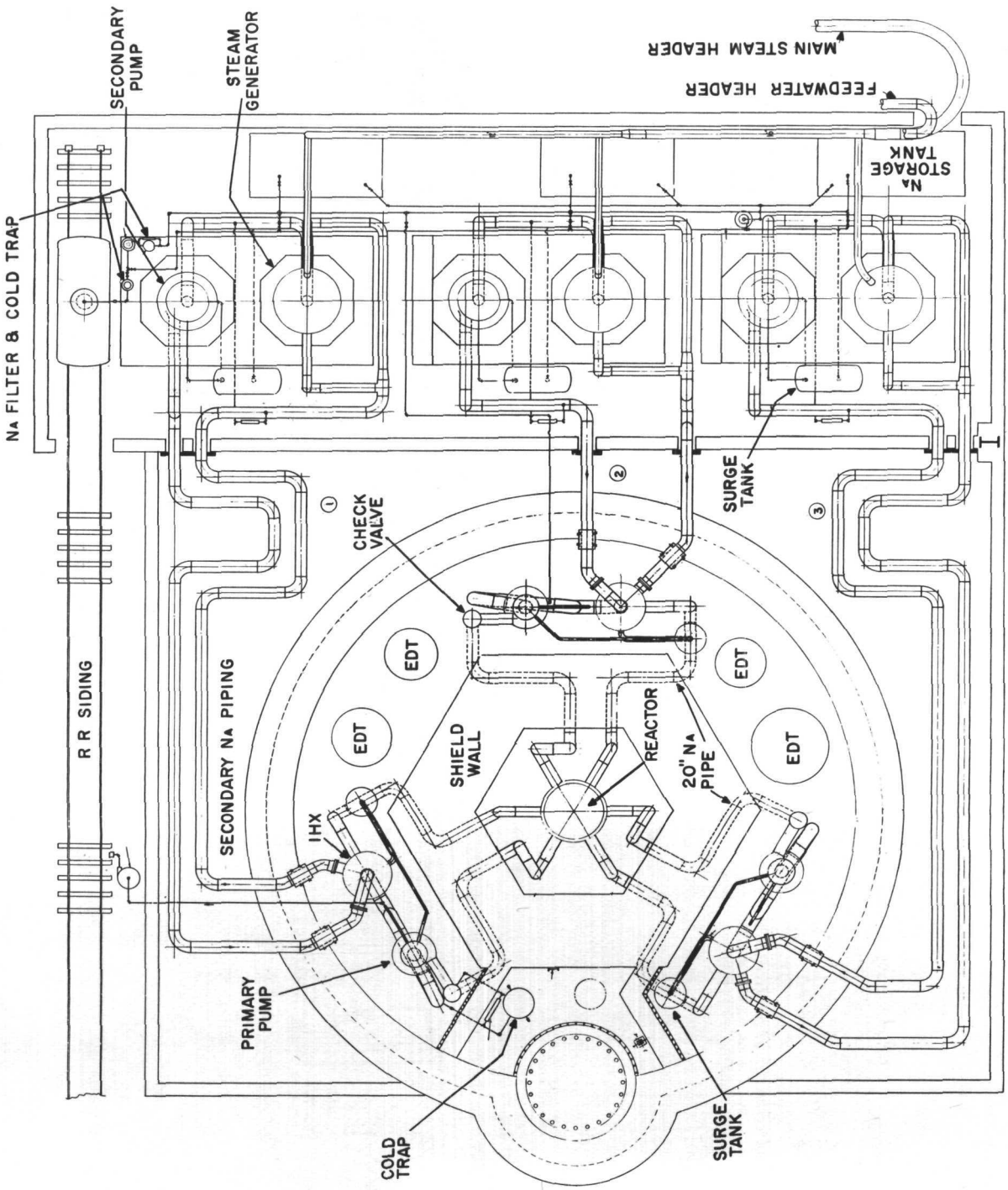

FIGURE 28.—PFFBR plan view of primary and secondary systems.

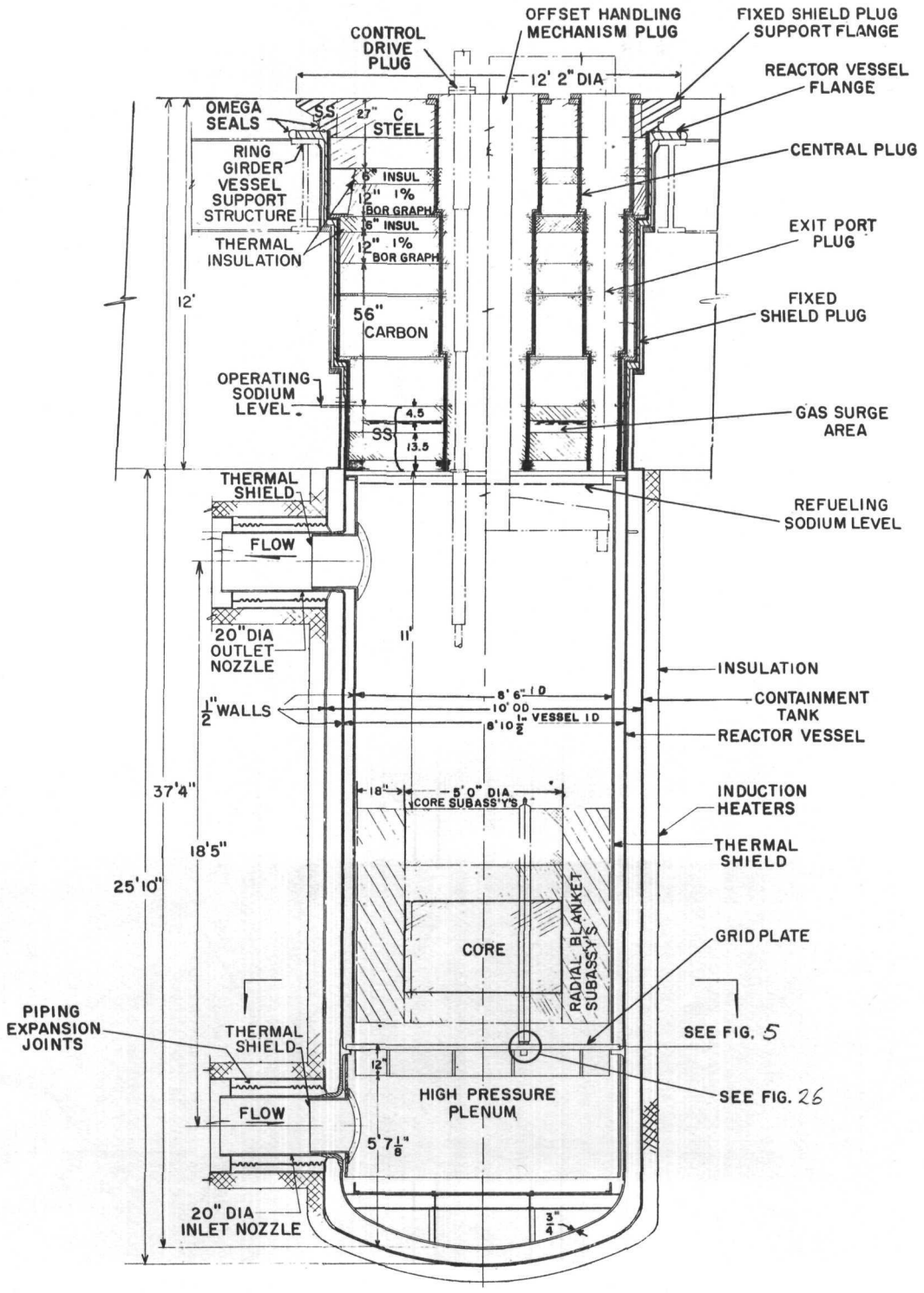

FIGURE 29.—PFFBR reactor vessel.

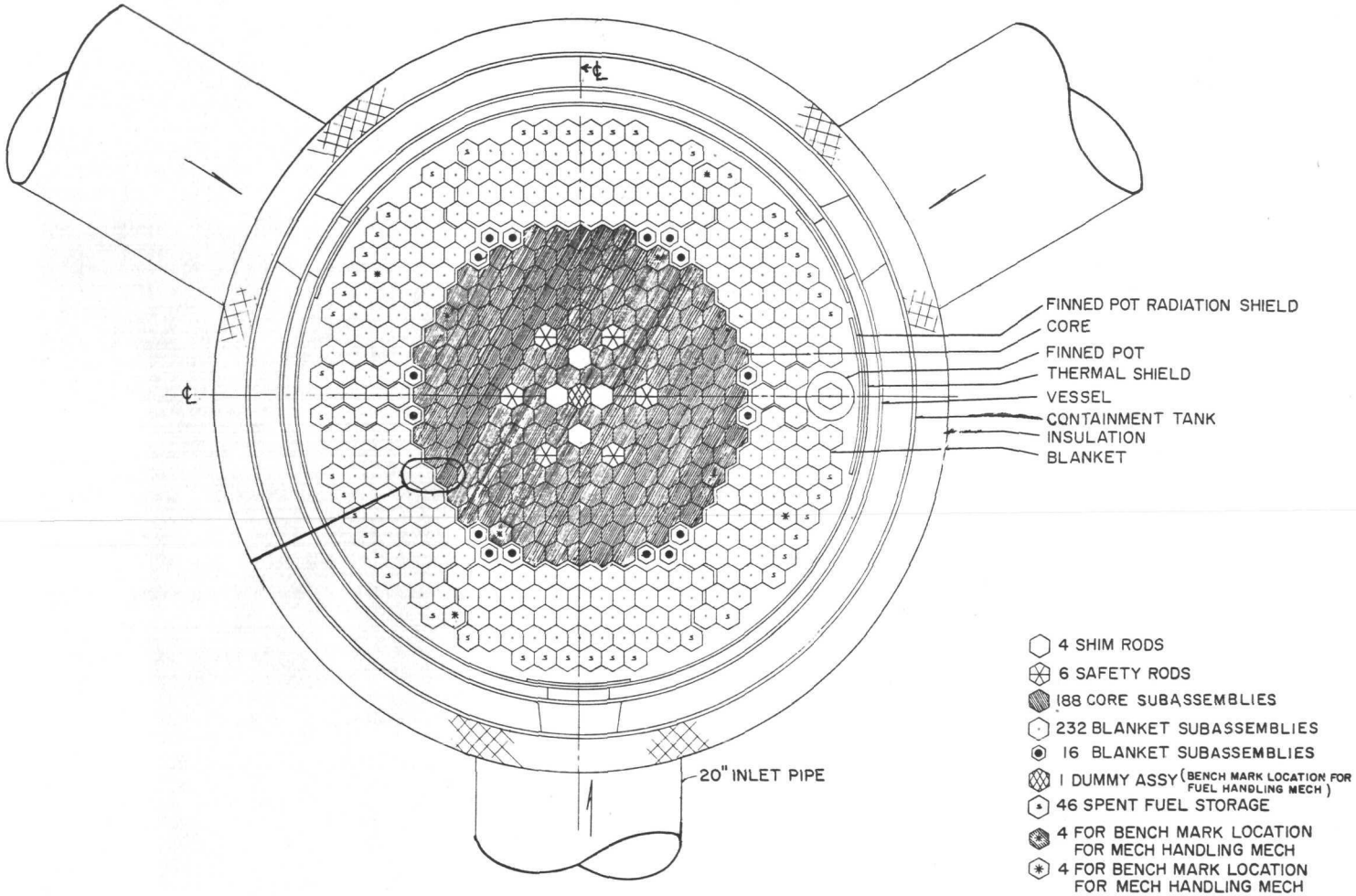

FIGURE 30.—PFFBR reactor cross section.

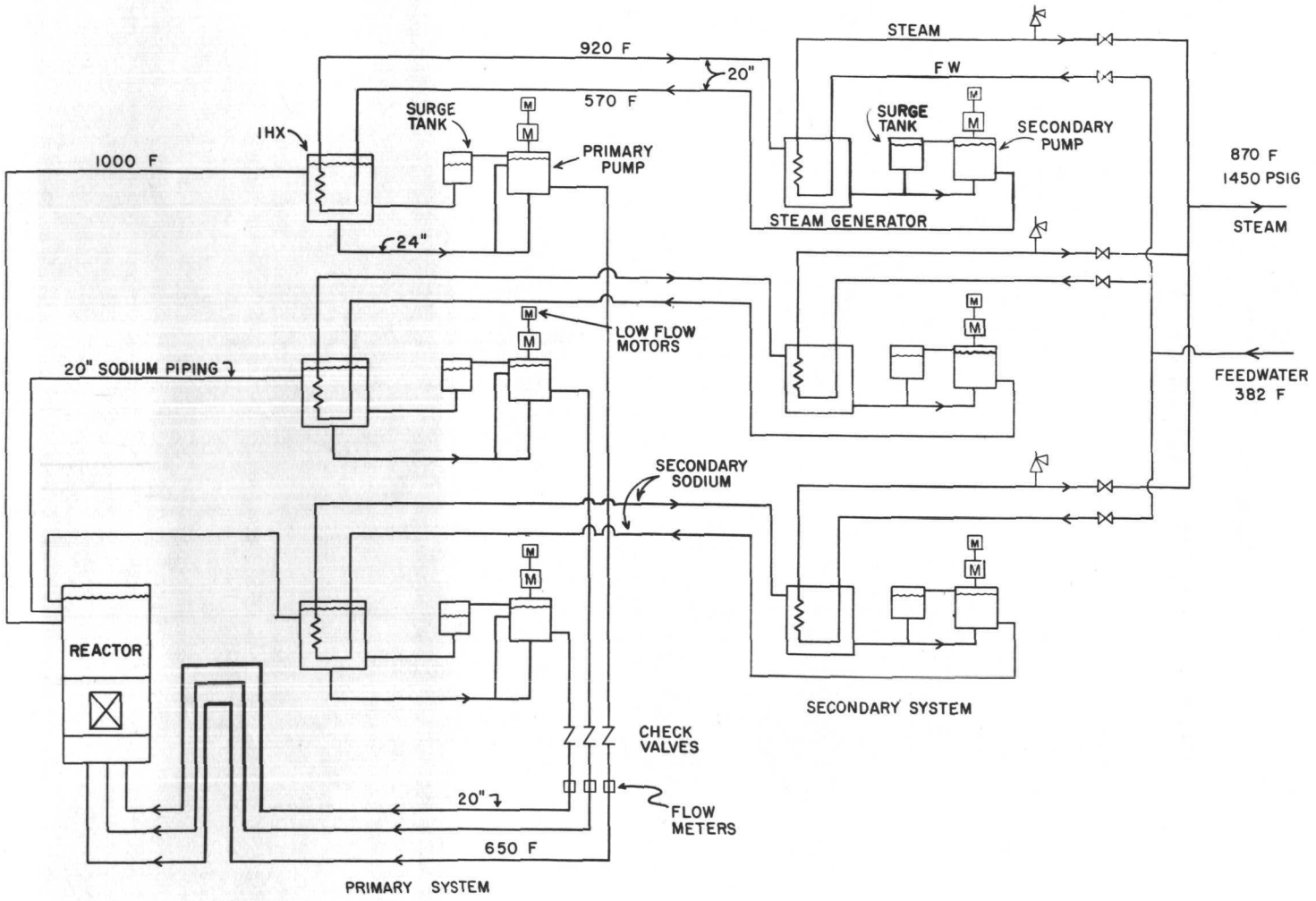

FIGURE 31.—PFFBR flow diagram for primary and secondary sodium systems.

Appendix E

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