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A PRELIMINARY STUDY OF THE TURRET EXPERIMENT

AN OPERATING TEST OF UNCLAD FUEL AT HIGH TEMPERATURE

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ABSTRACT

A preliminary study has been made of an impregnated graphite reactor to produce helium at 2400°F for process heat uses. The proposed facility will determine the characteristics and problems associated with operation of unclad graphite fuel. Volatile fission products will escape from the fuel elements into the helium stream, which will be continuously purified. Laboratory studies have shown that impregnated graphite will perform satisfactorily at the proposed conditions, and a simple fuel cycle has been developed.

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1. SUMMARY

Lowering the cost of reactor fuel cycles is an essential step in attainment of economic nuclear power or process heat. A large increase in allowable temperature, very high burnup, and cheaper fuel fabrication and recovery are all possible through the use of unclad fuel if the accompanying contamination of the coolant system is tolerable in a reactor. There is at present no experimental information as to the feasibility of such a drastic change in reactor philosophy. To obtain such information, a high temperature experimental reactor facility is proposed for the study of fission product release from an unclad ceramic fuel, and of the problems imposed by the operation of a contaminated coolant system. The 3 Mw helium-cooled reactor, the fuel elements, and the high temperature heat exchanger are constructed principally of carbon or graphite, and from laboratory tests are expected to operate satisfactorily at the proposed design temperature of 2400°F. Fission products released by the fuel elements to the helium coolant are to be continuously removed by a charcoal trap system, so that only very short-lived radioactive materials are present in the gas during operation. Some active materials of long half-life are expected to deposit on the inside surfaces of the system, however. The study of the rate of formation and decay of such deposits at various fuel element temperatures, and means of removal, will be an important objective of the project.

The rotating turret design of the core allows the reactor to be fueled continuously at full power. The expected burnup of 10-50% means that only one to three elements per day need to be reprocessed. Uranium is recovered by burning the spent elements, leaving uranium oxide, which

is dissolved in acid and purified. A small cave using laboratory-scale equipment and hand manipulators is proposed to recover the fuel and to study the types of fission products remaining in the fuel elements.

The reactor will have safety rods capable of shutting it down at any time, but will not have shim or control rods. Slow injection of fuel elements will be used to bring the reactor up to the desired temperature, and the power level will be regulated by varying the mass flow of the coolant. A preliminary operating program proposes that initial testing be devoted to careful study of the nature and quantity of fission product boil-out and deposit at slowly increasing temperatures, followed by a long period at full power to study burnup limitations.

The chemical compatibility of all the materials proposed for the plant has been demonstrated by laboratory testing in the absence of radiation. Uncertainty exists, however, about the radiation damage suffered by graphite at high temperature. Observations of possible damage, dimensional change, etc., will be made continuously as the reactor is operated. Even fairly extensive damage is not expected to impair the operability of the reactor, since very low structural demands are made upon the graphite, and dimensional tolerances are large.

Preliminary designs have been made of the various mechanical devices required to charge fuel elements, rotate the core, etc. These designs have not been perfected nor tested on operating models. However, no requirements have yet appeared which exceed present-day industrial practice. Helium-circulating blowers designed for the range of temperatures and pressures required for Turret are commercially available, fully sealed and equipped with gas-lubricated bearings. Reliability and maintenance problems associated with such blowers have not yet been assessed. The degree to which radioactive deposits complicate blower maintenance will be one of the studies in the experimental program.

A preliminary analysis of the safety aspects of the Turret project discloses no credible nuclear incidents. The extremely slow rates at which fuel can be added, the absence of excess reactivity in operation, and the very large degree of core overheating which can be tolerated without damage are factors in reaching this conclusion. Definite hazards exist in case of release of the helium coolant, both from fission products in the helium and from the possibility of oxygen reaching the hot carbon in the reactor. These hazards necessitate the use of a containment sphere for the helium and localized containment around the reactor restricting the air available in case of rupture.

As a joint project of the Los Alamos Scientific Laboratory and Sandia Corporation, detailed design of the Turret facility and testing of mechanical prototypes is expected to require about 14 months and cost about $1,000,000. Equipment, site, and erection costs are expected to be about $4,600,000 and require another 12 months. After preliminary testing, start-up would be expected in late 1961.

Following is a list of design data for the Turret reactor.

Summary of Design Data

Reactor power	3 Mw_t
Core	
Configuration	Hollow circular cylinder
Material	Graphite, $\rho = 1.7$
Dimensions	12-1/2 in. i.d. x 74 in. o.d. x 40 in. high
Weight	9260 lb
Number of fuel channels	312
Channel dimensions (nominal)	1.10 in. diam. x 30.75 long
Radial channels per layer	24 @ 15^o
Vertical layers @ 3 in. on centers	13
Critical core fuel loading (approx.)	8 kg U^{235}
Attainable core fuel load	15 kg U^{235}
Fuel Elements	
Configuration	Hollow circular cylinder
Material	Graphite, $\rho = 1.65$ (impregnated with U^{235}
Dimensions	1/2 in. i.d. x 1 in. o.d. x 5.875 in. long

Weight	
Unloaded	93.6 g
Loaded - critical (solution impregnated)	99.43 g
Loaded - maximum (solution impregnated)	104.23 g
Number per channel	5
Total number in core	1560
Heat transfer coefficients	
Hole	118 B/hr ft^2 $^\circ$F
Annular	44 B/hr ft^2 $^\circ$F
Temperature	
Average	2700°F
Maximum	3355°F
Stresses	
Maximum	300 psi tangential tension at inner surface

Power/in.^{2}	Hole	Annulus
Average	119.5 watts/in.^{2}	44.5 watts/in.^{2}
Maximum	153.0 watts/in.^{2}	57.0 watts/in.^{2}

Reflector	
Material	4 in. graphite, $\rho \sim 1.7$ 12 in. carbon, $\rho \sim 1.7$
Insulation	
Material	17 in. (ave.) porous carbon (type 60, NCC)
Pressure Vessel	
Configuration	Spherical
Size	12 ft, 4 in. o.d.
Wall thickness	2 in.
Weight	37,000 lb
Material	Carbon Steel SA-212
Design temperature	400°F
Design pressure	500 psig
Test pressure	750 psig
Safety Rods	
Configuration	Cylindrical rods
Number	
Main	1
Secondary	8
Composition	
Poison	Sintered B_4C
Base Rod	Carbon
Total rod worth	
Main	4-6% in k
Secondary	5% in k (4 rods)

Withdrawal rate	
Main	40 in./min
Secondary	40 in./min (sequenced)
Weight	
Main	30 lb
Secondary	25 lb
Scram time (all)	<1 sec
Coolant	
Type	Helium
Volume (total in primary loop at operation conditions)	770 ft^3
Flow	10,250 lb/hr
Effective active system volume	96.1 ft^3 - 10.14 lb
Particle flow rate	3.546 sec/circuit
Temperatures	
Reactor inlet	1600°F
outlet	2400°F
Recuperator inlet	2400°F
outlet	1600°F
Heat exchanger inlet	1600°F
outlet	800°F
Blower	800°F
2nd pass recuperator inlet	800°F
outlet	1600°F
Operating pressure at rated power	
outlet	500 psi
Pressure drop around loop	4 psi
Pumping power	35 horsepower
Heat flux	
Maximum/average ratio	1.28

2. INTRODUCTION AND OBJECTIVES

2.1 Introduction

There is general agreement that the successful attainment of economic nuclear power must include major reductions in the cost of reactor fuel cycles. Such cost reductions are potentially possible for solid fuel reactors at several points in the cycle. Fuel fabrication, cladding, cladding removal, reprocessing, and reconversion to metallic or oxide form are all expensive operations whose economy may be improved by experience, standardization, and large-scale operation. Improvement in burnup, or allowable heat release per pass through the reactor, is even more important, as it reduces the number of times the above-named operations must be repeated to release all the heating value of the fuel.

At present there is intensive effort upon each of the above factors in fuel cycle cost. The potential gains in each area are small, but a number of such gains added together may effect a fair lowering of overall fuel cycle costs. An alternative method of lowering the fuel cycle cost is the omission of cladding, and operation of porous fuel elements directly in contact with the coolant. The consequences of this concept on the reactor as a whole are almost unknown both as to technical feasibility and economic penalty.

2.2 Objectives of the Turret Project

The potential advantages of unclad fuel systems must be weighed against the disadvantages of operating a contaminated reactor coolant circuit. There is, however, almost no experimental information available to aid in assessing such a task. It is possible to speculate upon the problems which will be encountered, but which of these will turn out to be minor and which may become very difficult remains completely undetermined. This uncertainty can be resolved only by experimental facts--by operating a small reactor with unclad fuel and attempting to solve the resulting problems.

The present situation is very similar to the history of boiling reactors. A considerable uncertainty existed a few years ago as to the feasibility of operating turbines directly on steam from the reactor core. Borax III, in a conclusive test, showed that boiling reactors were feasible, that some anticipated problems were in fact negligible, and that others not previously suspected were important. At the end of the tests, the future of boiling reactors was assured on a sound basis of experimental fact. There is an urgent need for similar factual information on unclad fuel concepts.

The advantages of an unclad fuel system apply to almost all types of solid fuel reactors, but there are strong reasons for selecting a gas-cooled concept for an initial test: (a) with helium coolant there is no problem of compatibility with the fuel elements, (b) a substantially complete experimental study has already been made of unclad uranium-impregnated graphite in high temperature gas, (c) gas-cooled reactors are currently exhibiting considerable economic promise, and this promise is materially enhanced if they can be freed of the temperature and burnup limitations imposed by clad fuel, (d) gas, especially helium, is relatively easy to separate from fission product impurities.

The Turret project proposed herein will undertake as its primary purpose the study of the effects on both the reactor and the heat transfer system of using porous, unclad fuel elements. The following is a listing of specific objectives which will implement the primary purpose.

2.2.1 Study of Gas Cleanup

The experiments will show the effectiveness of continuous purifiers in removing volatile fission products from the helium coolant. If the purification can be made highly efficient at a reasonable cost, the consequences can be very important for gas-cooled reactors. For example, if the fission products can be removed from all the helium every few minutes, then the circulating gas would contain essentially only radioactivities with short half-lives. Accidental release of such gas would present only a minor hazard outside the plant, and would require only a vented delay tank rather than a pressure-tight containment shell.

Although the basic methods of purification by charcoal adsorption are well understood, the detailed engineering of a large system of this type presents several problems, particularly in the method of regenerating the beds and collecting the radioactive wastes. The experimental plant will be an important test of ideas in this field.

2.2.2 Operation and Maintenance of Contaminated Equipment

Regardless of the efficiency of the gas purification system, a portion of the fission products will condense or decay to nonvolatile substances in every part of the gas circuit. There is no way of predicting to what extent these will remain suspended in the gas or deposit on the interior surfaces of equipment. If deposited, it is not known to what extent such deposits are long-lived, difficult to remove prior to servicing, or cause unusual shielding requirements.

A rough estimate of the weight of fission products which might form solid deposits is 10 mg/hr per Mw. Observation of the actual quantity and the location of deposits in the equipment, however, will be an important result of the experiment. The quantity and type of fission products evolved will be strongly dependent on the temperature of the core, which will be the principal variable in the tests. The proposed reactor is designed for exit gas temperatures up to 2400°F, with fuel element temperatures somewhat higher. The blower, heat exchangers, piping, and other components exposed to the contaminated helium will be designed with convenient sample ports and removable sections for examination and testing of deposits on the interior surfaces. Facilities for testing decontamination methods on small samples of representative deposits will be provided instead of elaborate wash-down and decontamination devices.

Since there is no necessity for continuous operation of the experiment, no provision is made for elaborate remote maintenance or replacement features. Major equipment replacement would require time for decay, and installation of moveable temporary shielding. Essential moving parts, however, such as the blower seal and bearings, will be made accessible upon reactor shutdown.

2.2.3 Fuel Manufacturing Process

Since it is essential for the Turret project that fission products escape from the fuel elements, impregnated graphite forms an ideal medium. Methods of introducing uranium oxide into porous graphite by solution impregnation have been fully developed at Los Alamos, but have never been utilized in actual fuel element production. A demonstration of such production will form one objective of the Turret program. The process requires only two simple operations: soaking and baking. The low power level and high burnup of Turret make possible a low refueling rate, so that the demonstration plant will consist of laboratory-size components operated remotely with a hand manipulator. The process is readily adaptable to mechanization.

2.2.4 Fuel Recovery and Reprocessing

Another purpose of the project will be the demonstration of a highly simplified process for the recovery and partial purification of uranium from spent fuel elements. The discharged elements will be ashed in a small furnace, and resulting U_3O_8 dissolved in dilute acid and separated from most of the fission products. There is no need for a particularly high decontamination factor in the separation process since the recovered uranium can be re-used in new fuel elements without complete decontamination. This criterion, therefore, has been subordinated to the goal of extreme simplicity and low cost, particularly capital cost. Although solvent extraction methods could be used, the most promising purification step at present is a simple precipitation of the uranium as peroxide. Laboratory-size equipment will be used in the demonstration plant.

Another important use of the fuel processing laboratory installations will be testing the fission product residues from the reactor. As mentioned above, the nature of these residues will vary considerably with reactor operating temperature, so that the reprocessing laboratory will provide valuable basic data on the volatility of particular species.

2.2.5 Feasibility of Turret for Process Heat Uses

The application of nuclear energy to produce industrial process heat has for some years been an area of investigation by the AEC. As part of this program, the U. S. Bureau of Mines has received support from the AEC in investigating the conditions necessary for gasification of coal preparatory to Fisher-Tropsch synthesis. At present the U. S. rate of consumption of energy from petroleum sources is about four times that from coal, yet the country's reserves of these fuels are in just the opposite proportion. There will come a time when conversion of coal to fluid fuel will be essential. Experiments show the conversion process is most economical if it can be performed at temperatures and pressures above the creep limit of most metals. A high temperature, high pressure, noncorrosive heat source surrounding the gasification tubes gives satisfactory results, however. The Turret reactor is expected to deliver helium at 2400°F and 500 psi pressure to fulfill the required conditions. If the simplified fuel cycle proposed for Turret is successful, nuclear heat delivered to the gasifier tubes would probably cost less than conventional heat produced in pressurized combustors.

At present there is no intention of operating a coal gasification unit in conjunction with the Turret reactor, but the operation of the reactor and fuel system at the design conditions will provide essential data for technical and economic evaluation of using nuclear heat for coal gasification.

2.2.6 Rotating Core

The inward radial movement of fuel at full power in the Turret concept provides important operating advantages which exploit to the greatest extent the simplified control and high fuel burning capabilities of the reactor. Although it is conceivable that radial fuel injection could be applied to a stationary core, much less injection machinery is required if the core rotates. A revolving core is unusual enough to require demonstration on a small scale before it can be seriously considered for large-scale application. Investigation to date shows that for this project, the bearings required are within the range of present industrial practice.

2.2.7 Miscellaneous Technical Data

Although the neutron flux in the Turret reactor will not be high, the combination of this flux with extreme temperatures will be an unusual set of conditions valuable for materials testing. Studies of high temperature irradiation effects on graphite, for example, will be easy to accomplish with Turret. There is no proposal to include special experimental facilities in the project, but a number of useful experiments can be visualized using the reactor without modification. The noncorrosive helium atmosphere makes compatible experiments easier to plan.

In addition to in-pile experiments, considerable miscellaneous technical data will result from the Turret project. One type of data is information on obtainable helium loss rates from a high pressure system. Many present reactor concepts propose such systems, and the project will provide a convenient proof laboratory for helium components.

3. DESCRIPTION OF PROPOSED FACILITIES

3.1 General

The proposed Turret reactor facility is to be located in Technical Area 3 of the Sandia Corporation grounds on Sandia Base, Albuquerque, New Mexico (see map, Fig. 1). Technical Area 3 is remotely located, and it is presently used for experimental work.

The facility consists of a steel containment sphere, service building, stack, water storage tank, water cooling tower, and a structure for housing the heat dump (see Figs. 2, 3, and 4).

The containment sphere is for the purpose of preventing possible escape of fission products from the helium coolant loop to the atmosphere. The sphere has a diameter of 60 ft and it is located partially below ground level to utilize the earth as a biological shielding as much as possible. Inside the containment sphere is a 3 Mw reactor, together with primary coolant loop components.

A single-story service building (Fig. 3) located adjacent to the containment sphere serves to house the fuel processing cave, chemical laboratory, counting laboratory, reactor control room, maintenance shop, health physics room, and locker room.

3.2 Reactor

The proposed Turret reactor is an experimental helium-cooled reactor rated at 3 Mw_t. It uses graphite fuel elements impregnated with uranium oxide. A cross section of the reactor is shown in Fig. 5.

The reactor will operate at an inlet temperature of 1600^oF and an outlet temperature of 2400°F, making a ΔT of 800°F over the core radius (a distance of about 31 in.). The operating pressure of 500 psi will be contained in a spherical vessel with such extensions or thimbles as are necessary for the indexing drive, and the loading, unloading, and safety components, which will also be under operating pressure.

The core material and part of the reflector are graphite. Carbon brick and porous carbon fill the remainder of the pressure vessel as added reflector and as insulating material for the pressure vessel.

The fuel elements will be short unclad tubes of graphite, impregnated with highly enriched uranium oxide. Volatile fission products will be released from these unclad fuel elements to the primary helium coolant stream. A separate gas clean-up system will operate continuously, drawing off a fraction (20% of the circulating volume per minute) of gas from the primary coolant loop for removal of impurities, including fission products.

The reactor will be fueled during operation. Any fuel channel may be loaded by means of a single vertical row of loaders since the core can be rotated and indexed to any one of the 24 radial channel positions. Each time a new fuel element is added, a partially spent fuel element will be ejected from the core at the center, removed from the reactor, and taken to a fuel processing cave where the remaining U^{235} will be recovered from the graphite for reuse in impregnating new fuel elements.

3.2.1 Core

The core of the Turret reactor is cylindrical in shape with dimensions and specifications as follows:

Core dimension = 12.5 in. i.d. x 74 in. o.d. x 40 in. high

Radial fuel channels per layer = 24

Vertical layers = 13

Vertical spacing = 3 in. center to center

Total number of channels = 312

As shown in Fig. 6, the core structure will consist of 24 pie-shaped pieces of graphite pinned to the base reflector graphite to prevent radial "ratcheting" movement of individual pieces as a result of temperature cycling. Each piece will be keyed to the base reflector graphite to maintain accurate azimuthal location for indexing when loading fuel.

The central hole in the rotating core is partially filled with a stationary cylinder of graphite. This core plug provides three advantages: (1) addition of moderator in the under-moderated portion of the core to reduce thermal neutron flux depression induced by the crowding of fuel channels at the center; (2) increased worth of the central safety rod; (3) means for guiding the spent fuel elements during ejection from the core.

An annulus is provided between the core plug and core proper for the inlet helium flow. From this annulus the coolant flow is radially outward, with all channels acting in parallel and emptying into an outer annulus which exhausts into the reactor coolant outlet pipe. The total helium flow through the reactor is 132,700 ft^3/hr.

Since the fuel elements rest on the bottom of the annulus, the coolant flow is distributed between the central hole in the element and the eccentric annulus around the element. Channel flow is divided as follows: 334 ft^3/hr with a Reynolds number of 5830 in the element hole; 91 ft^3/hr with a Reynolds number of 377 in the eccentric annulus. The total pressure drop across the core is approximately 0.08 psi.

3.2.2 Fuel Elements

The fuel elements are hollow cylinders of graphite impregnated with highly enriched uranium oxide. They are 1/2 in. i.d. x 1 in. o.d. x 5-7/8 in. long. There are five elements in each channel or a total of 1560 elements.

A shoulder is provided in each channel of the core to prevent elements from moving radially outward should transient disturbances in the coolant flow produce a large pressure drop across a channel.

Each element can easily be impregnated to contain 10 g of U^{235} or a total core loading of about 15 kg of U^{235}. The expected critical mass is about 8 kg. The maximum power output of a fuel element was calculated to be 2470 watts at a temperature of 3280°F. With average values, the power output per fuel element is 1930 watts at a temperature of 2700°F. The heat transfer coefficient in the hole of the fuel element is 118 B/hr ft^2 °F, and the coefficient at the outside diameter of the fuel element is 44 B/hr ft^2 °F.

3.2.3 Pressure Vessel

The reactor pressure vessel (see Fig. 5) is a high strength, low carbon steel sphere. It encloses the graphite moderator, reflector graphite, and carbon insulation. The pressure vessel provides support for the reactor loading and safety mechanisms, as well as containing the helium gas at 500 psi. The pressure vessel has an outside diameter of approximately 12 ft, 4 in., with a wall thickness of about 2 in. It is externally cooled to less than 500°F.

3.2.4 Turntable Mechanism

The core rests on the core support plate, which in turn is carried by a bearing (see Figs. 5, 6, and 9). The core is rotated and indexed to each of 24 radial fuel loading stations by a motor operating through a Geneva drive system, which provides smooth motion and positive location. The time consumed in rotating the core from one station to the next is about 30 sec. The motor and Geneva drive are accessible for maintenance upon reactor shutdown. The complete mechanism is housed in a pressure-tight container forming part of the main helium system. A small inflow of clean helium at this point serves to keep the drive purged of fission product gases. There are no rotating or sliding seals required since the turntable mechanism is completely canned.

3.2.5 Loading Mechanism

The proposed loading process is carried out by the following sequence of steps, all operated from the control room (see Figs. 5, 7, and 8). A hopper in the fuel processing cell is caused to discharge a new element onto a small conveyor. The element is moved to the reactor loading position, is pushed through two gas locks by piston-operated rams, and enters the loading elevator. The carbon rams at the several loading channels in the reflector are normally left inserted to prevent leakage of heat and neutrons. When an element is ready to be loaded, however, the ram in the desired channel is withdrawn and the elevator lowered. As the elevator passes the desired loading position, the fuel element rolls into the cavity left by the withdrawn ram. Advancing the ram pushes the new element into the core.

3.2.6 Discharge Port

The insertion of a new fuel element into a channel, as described above, moves all the elements in the channel toward the center of the reactor, displacing the innermost, which is discharged into a slot in the central graphite plug. The slot is curved at the bottom, so that the falling element is brought to rest in orientation with the first discharge ram (see Fig. 9). A series of valves and rams serves to pass the element out of the reactor atmosphere and discharge it onto a small conveyor which brings it to the fuel processing cell.

3.2.7 Safety Devices

The main safety rod will be suspended in a hole in the core plug extending up through the top reflector, as shown in Figs. 5 and 10. The rod is constructed of 1.5 in. diameter graphite impregnated with boron. It is supported by a metal extension arranged to travel freely in a nonmagnetic thimble or sleeve extending vertically from the reactor pressure vessel and forming part of the main helium container. An iron armature at the top of the extension can be supported by a solenoid surrounding the thimble but outside the helium system. Slow raising of the solenoid withdraws the rod, which can be dropped at any time by interruption of the solenoid current. A dash-pot action slows the rod when nearly inserted.

The eight secondary safety rods will be constructed in a similar manner but will only be 3/4 in. diameter overall. The suspension system and actuation are also identical with those for the main safety rod. These rods will be suspended over the rotating core, which will be provided with slots to receive the rods. If the rods are dropped while the core is in a loading position, all eight will enter the core. If the rods are dropped while the core is in motion, at least four of the rods will enter the core and the core will be stopped at this point.

3.3 Main Coolant System

A schematic diagram of the coolant system is shown in Fig. 11.

The purpose of the coolant system is to remove heat and fission products from the reactor core, and to transport them to some other point where they are removed from the system. The primary coolant system forms a closed loop inside the secondary containment shell. A secondary cooling loop takes heat from the primary loop through an exchanger located inside the secondary containment and carries it through the secondary containment wall to another exchanger which dissipates the heat to the atmosphere.

In Fig. 11 it will be noticed that the 2400°F gas leaving the reactor is immediately cooled in a "recuperator" and the heat used to preheat the gas entering the reactor. Such a device would, of course, have no use in an actual process heat plant. The high temperature pass of the recuperator similates the coal-gasification unit, high temperature turbine, or other end use for the reactor heat. The low temperature pass represents the product heat recovery unit or the regenerator of a power plant. As far as the reactor cycle is concerned, therefore, the system operates almost identically to a real plant, but the combination of two heat exchangers into one greatly simplifies the Turret system. The development of suitable heat exchangers for coal gasification is now in progress at the U. S. Bureau of Mines Experiment Station, Morgantown, West Virginia.

Any fission products which are in a gaseous form, and other gases which may be given off by materials in the system, are removed by passing about 1% of the primary coolant flow through traps consisting of copper oxide followed by charcoal. The radioactive fission product gases collected by these traps are periodically removed, concentrated, and suitably contained for controlled disposal.

3.3.1 Primary Coolant Loop

Description. Under conditions of full power (3 Mw) operation, helium is circulated around the primary coolant loop at the rate of 10,250 lb/hr and transfers heat from the reactor core to a secondary coolant loop which dissipates the heat to the atmosphere.

Primary coolant helium enters the central plenum hole of the reactor at a temperature of 1600°F. As the coolant passes radially outward through the fuel channels in the core, it gains heat from the core and leaves the reactor at a temperature of 2400°F. The coolant then passes through the recuperator, where it is cooled to a temperature of 1600°F. The coolant next goes through the main heat exchanger, where it is cooled from 1600°F to 800°F in transferring heat to the secondary coolant loop. After leaving the heat exchanger, the coolant goes through the blower, which restores the pressure loss in the loop (about 4 psi). From the blower, the coolant enters the second pass of the recuperator, where it is heated from 800°F to 1600°F, at which temperature it again enters the reactor.

Pipes and Valves. The sections of pipe which carry very high temperature helium, i.e., 2400°F and 1600°F, are internally insulated and externally cooled to allow a maximum metal temperature of 800°F. The sections for 800°F helium are externally insulated.

Welded joints are used wherever possible to approach zero system leakage. Components requiring periodic removal for inspection and/or maintenance, such as the blowers and the instrumentation, are installed with zero leakage mechanical seals.

Thermal expansion compensation is provided through the use of bellows-type expansion joints. All valves have bellows or other zero leakage stem seals and can be operated remotely.

As shown in the schematic drawing (Fig. 11), the primary loop has two valves for isolation of the blower. Not shown are smaller valves that provide pressure regulation and control for the helium coolant within the primary loop and/or auxiliary gas systems.

Blower. Circulation of the helium coolant in the primary loop is maintained at the desired rate by use of one or more blowers located above the main operating floor of the containment shell. Portable local shielding will be placed around the blowers as required.

The blower is a motor-driven unit capable of circulating 10,250 lb/hr of helium at approximately 500 psig and 800°F inlet with a pressure rise of 4 to 5 psi. The motor and blower are completely contained in a pressure envelope, i.e., canned, to insure zero system leakage from the blower.

Preliminary calculations have shown the power required to circulate the coolant in the primary loop is about 35 hp. Commercially produced centrifugal-type units which meet all the above requirements appear to be available. Investigation has shown the use of several units in parallel to be feasible, if desired.

Recuperator. Details of the recuperator are shown in Fig. 12. The heat-exchanging core of the recuperator is formed by a pattern of parallel 1/2 in. diameter holes drilled the length of a graphite cylinder 35 in. in diameter and 54 in. long. Holes for the cooling pass alternate with holes for the heating pass throughout this pattern. Hot coolant from the reactor flows through 225 of the holes and is cooled from a temperature of 2400°F to 1600°F. The heat transfer area of the hot side is 371 ft^2. The remaining 256 holes, having a heat transfer area of 422 ft^2, receive helium from the blower at 800°F. The helium is heated to a temperature of 1600°F in passing through these holes. Counterflow of the two coolant streams is employed. Headers are formed by graphite discs attached to the block by means of hollow graphite bolts which also serve as passages for the cool helium.

The cylindrical graphite exchanger block is contained in a cylindrical graphite sleeve, in which it is free to move axially as required by temperature-induced differential expansion. The graphite sleeve is surrounded by porous carbon brick insulation, and the entire assembly is enclosed by a cylindrical steel pressure vessel. External cooling of the pressure vessel is provided to limit the pressure vessel temperature to 800°F.

Main Heat Exchanger. Heat is transferred from the primary to the secondary coolant loop in a shell-and-tube heat exchanger. Construction of the exchanger is shown in Fig. 13. Primary loop coolant flows in a bank of austenitic stainless steel tubes having an area of 575 ft^2 and is cooled from 1600°F to 800°F. Secondary coolant flows in a combination of counterflow and crossflow in the shell side of the exchanger and is heated from 400°F to 1200°F.

3.3.2 Secondary Coolant Loop

Description. As mentioned in Section 3.3.1, heat is transferred from the primary to the secondary coolant loop in order that the primary coolant, which contains fission products, should not penetrate the containment sphere. The coolant used in the secondary loop is helium at a pressure of 500 psi, circulated at the rate of 10,350 lb/hr by a blower similar to that in the primary loop. The temperature of the helium in the secondary loop is raised from 400°F to 1200°F in its passage through the heat exchanger, and the temperature is again lowered to 400°F in a heat dump exchanger outside the secondary containment. Atmospheric air is passed through the heat dump exchanger by a motor-driven fan and is discharged at about 500°F.

Heat Dump. The heat dump is a finned tube heat exchanger. Secondary coolant loop helium flows inside 452 tubes 15 ft long, having a total heat transfer area of 1100 ft^2. Three fans, of 30 hp each, pass a total of 60,000 ft^3/min of air across the externally finned tubes.

3.3.3 Helium Pressure Control and Storage

The pressure in the main helium system is varied through the use of two sets of storage tanks. One set is kept at low pressure and serves as a receiver for helium drawn from the main system. The second set is kept at high pressure in readiness to supply helium to the system. A transfer pump is provided to move helium to the high pressure tanks, which have sufficient capacity to store all system helium. The low pressure tanks when empty have the capacity to reduce the system pressure to about one-half. Further reduction in pressure is accomplished with the transfer pump. Suitable piping and controls are provided for automatic reduction of helium pressure in case of emergency. Manual control is used for normal operation. Helium from the high pressure reservoir, exhausting into the low pressure tanks, provides a convenient source of power for actuation of fuel handling mechanisms within an enclosed system.

3.4 Helium Purification System

The helium in the primary loop of the Turret reactor is purified by continuously circulating about 6 ft^3/min through a purification system.

3.4.1 Purification Cycle

As shown in Fig. 14, the hot gas from the reactor coolant stream passes first through a copper oxide bed where such impurities as hydrogen and carbon monoxide are oxidized so that they will be more readily adsorbed later. The gas is then cooled to $40^{\circ}F$ or lower and passed through the charcoal adsorption beds where the impurities are removed. Following this, the purified helium flow rate is measured and the gas returned to the reactor system through auxiliary equipment.

3.4.2 Regeneration Cycle

Since the charcoal must be periodically regenerated in order to yield gas of the desired purity, a regeneration loop is provided to remove the adsorbed gases from the primary charcoal beds. This reactivation is brought about by isolating the bed to be regenerated from the system and circulating clean helium at $1100^{\circ}F$ and 1 atm pressure through the charcoal to raise its temperature and to flush away the desorbed impurities.

By selective valving one gas adsorption bed will be separated from the primary coolant system and connected into the regeneration loop shown in Fig. 15. In this system, gas from the primary adsorption bed will pass through a heat recuperator and cooler and then through a series of charcoal beds. These act to bring about a partial chromatographic separation of the impurities and to purify the helium which is circulated through a heater back to the bed being regenerated.

After separation, those chromatographic beds which do not contain radioactive gases can be purged by heating and vacuum pumping the impurities directly to a stack. The beds which hold the radioactive gases can be purged by heating and vacuum pumping to a storage vessel. These partially concentrated fission product gases can be stored for a long decay period and vented as desired, or further concentrated for sale or burial.

3.4.3 Description of Equipment

The purification system consists of a battery of 24 vertical tanks in a shielded pit. Above these tanks is a valve manifold arranged so that various tank connections can be made as the optimum operating system and cycles are established. The tanks are of stainless steel to withstand the regeneration temperature and are 20 in. i.d. and 10 ft high. Four tanks are used as copper oxide containers and are operated at 1000°F and 500 psi. These tanks have auxiliary heating to insure the correct operating temperature for the copper oxide even when the gas from the reactor is at a lower temperature.

The remaining tanks are filled with 6 to 14 mesh activated charcoal and are thermally insulated so that they can be heated for regeneration. These are operated in groups in the various cycles of collection, decay, regeneration, and reserve.

3.5 Auxiliary Cooling

All system components which contain high pressure fluids or gases at temperatures exceeding 800°F are provided with external cooling of their pressure envelopes or are made of stainless steel.

This external cooling, together with the internal insulation, allows the use of ferritic steels in most components. Cooling is required for the following components:

a. Reactor vessel shell

b. Coolant outlet pipe, reactor to recuperator

c. Recuperator shell

d. Coolant pipe, recuperator to main heat exchanger

e. Coolant inlet pipe, recuperator to reactor

f. Gas clean-up pipe, primary loop to gas purification system

The removal of this heat from the containment sphere, along with the removal of stray heat such as that generated by radiation and electric motors, plus the required cooling for the gas in the gas clean-up process, is referred to as auxiliary cooling.

This cooling requires a refrigeration capacity of approximately 200 tons and is provided by natural or forced convection of the air in the containment sphere around the components requiring cooling. The heat is then transferred outside the containment sphere by direct transfer and mechanical refrigeration units.

3.6 Fuel Handling and Processing System

Fuel elements are dispensed to the reactor from hoppers located in the processing cave. The elements are carried into the reactor area through a conveyor tube which is isolated from the containment sphere and are fed to the fuel loading mechanism. In a similar manner, fuel elements are removed from the bottom of the reactor through a system of gas-tight valves as described above, and transferred by another conveyor to the cave for processing.

The multiple gas locks serve to isolate the fuel conveyors from the reactor gas. The conveyors in turn are isolated from the containment sphere and are extensions of the processing cave. In an emergency, however, the conveyor tubes can be isolated from the cave and gas leaking into them relieved to the containment sphere.

In order to recover the fuel for reuse in the reactor, the spent fuel elements are burned in air and the remaining residue dissolved in acid. The uranium is precipitated from this solution to effect a partial separation of fission products. The uranium precipitate is then dissolved and this solution used to impregnate new graphite elements. This process is completed by drying, baking, and inspecting the new elements (Fig. 16).

These processing operations are done in the cave, which is external to the reactor containment sphere but inside the reactor building (see Figs. 3 and 4). Concrete blocks provide biological shielding and the interior of the cave is metal lined to prevent the spread of radioactive material. Operations inside the cave are viewed through a dense window. Two CRL Model 8 mechanical manipulators are used to conduct the processes necessary within the cave.

The cave is nearly airtight, and the inleakage of air is controlled so that there will be enough to permit burning the spent fuel elements. Gases exhausted from the cave are filtered and monitored for radioactivity before being vented to the stack. An adsorption system can be installed if the quantity of radioactive gases passing the filter exceeds the permissible limit.

Within the cave standard types of laboratory equipment are used. Two furnaces are provided to heat spent elements while ashing and to dry and bake the new elements. The chemical separation is made in a centrifuge bowl, and the impregnation of the new elements is done in a stainless steel beaker of eversafe geometry. This is fitted with a remote reading hydrometer for concentration determination and a heater and condenser for concentration adjustment. A shielded waste solution container on wheels is used to remove the solutions containing fission products. A balance is provided to weigh the fuel elements for inventory purposes.

3.7 Containment and Shielding System

The reactor and all components of the primary loop containing high pressure contaminated helium are located within a steel containment sphere to prevent the accidental release of the fission products. The sphere has a diameter of 60 ft and is designed to withstand an internal pressure of 5 psig. In the event of an accident, fission products can be held inside the sphere for as long as required. The design of the sphere and operational procedures preclude the accidental release of fission products to the atmosphere.

Biological shielding is provided by massive concrete shields around the reactor and lesser amounts of concrete around the primary loop components. The containment sphere is located partially underground in order to utilize the earth as much as possible as a shield.

Sufficient shielding is provided to reduce the total dose rate outside the sphere to a maximum of one tenth of the AEC allowable limit during full power operation of the reactor. Internal shielding is adequate to reduce the dose rate in the fuel loader access room inside the sphere to the AEC tolerance level 30 min after reactor shutdown following an extended full power run. The primary loop blower and the manifold system for the gas clean-up units are shielded as required by portable concrete blocks and lead bricks on the operating floor of the containment sphere.

The internal shielding provides sufficient impact resistance to protect the containment shell from metal fragments which may be ejected from primary coolant loop components as a result of rupture. The portable shielding is in large sections arranged to prevent them from becoming missiles in the event of a rupture in the loop.

Personnel may enter the sphere through a set of ground level pressure-tight doors arranged to provide an air lock. A larger welded equipment access panel is removed for major equipment entry, if required.

3.8 Instrumentation and Control

The instrumentation and control system proposed for the Turret facility will permit all normal operating functions of the plant to be conducted from the control room. These include: fuel insertion and discharge, movement of safety rods, addition and removal of helium, operation and regeneration of helium purifiers, starting and control of helium blowers, control of auxiliary cooling system, operation of sampling devices, recording of temperature, pressure, and radiation levels at certain points in the system, and operation of normal and emergency shutdown devices.

3.8.1 Neutron Instrumentation

Startup instruments will consist of a neutron source and three independent counter channels (BF_3 or fission counters). These counters will be located in thimbles through the reactor shielding which bring them close to the reactor and source. When the reactor has been brought critical and to sufficient wattage to make the ion chambers operative, the startup counters are to be moved to a location where they will remain operative on leakage neutrons through the expected power range. At least two of these counters will be kept in operation at all times, so that the registers will provide an audible indication of any change in power level. During shutdowns the counters will be re-inserted to the startup position and kept in operation from the source.

Two intermediate power channels will be provided, consisting of independent compensated ion chambers. These chambers will be used to operate linear and logarithmic neutron level indicators and recorders. The logarithmic channel also operates the period circuit.

A duplicate set of chambers in a lower neutron flux location will perform the same functions at power levels above the range of the intermediate power chambers. The switching for change-over of ion chambers will be arranged so that ranges overlap and operator error cannot leave safety circuits disconnected.

3.8.2 Fuel Charge and Discharge

Fuel elements are prepared and processed in the shielded cave and the new elements inspected and stacked in a dispensing hopper. A second hopper holds dummy elements containing no uranium. Once there is a supply of elements in the hoppers, the rest of the charging operation is carried out from the control room. The conveyors, valves, pneumatic rams, and other parts of the fuel handling system previously described will be actuated by an electric-pneumatic control system. Suitable signal instrumentation will be employed so that positive indication will be transmitted to the control room of each mechanical function. Where valves or rams are to be operated by helium from the main system, air-operated valve positioners will be used, so that primary helium lines will not penetrate the containment shell.

3.8.3 Pressure and Temperature Instrumentation

As mentioned above, primary helium lines will not penetrate the containment shell, so that pressure, flow, and valve position indication in the helium system will be obtained with pneumatic transducers and actuators. Temperatures will be indicated with duplicate thermocouples installed to facilitate replacement.

3.8.4 Safety Rods

The construction of the safety rods has already been described (see Section 3.2.7). The holding solenoids will be supplied with current through a safety circuit using conventional fail-safe design. The actuation of the safety circuit is discussed in Section 3.8.5. The mechanisms for lowering and hoisting the solenoids will be controlled through a sequencing relay system so that only one rod may be moved upward at a time, although all may be moved downward at once. Positive position sensors will be used to indicate the position of the solenoid in its travel. An independent system actuated by the rod itself will be used to indicate when a rod is fully withdrawn or fully inserted.

3.8.5 Automatic Safety Instrumentation

Although the requirements of the safety instrumentation have not been fully analyzed, a preliminary analysis of equipment malfunctions and proposed instrument response has been prepared. Some of the more important responses are as follows:

Excessive neutron power level will actuate a warning gong, followed by a safety rod trip at a slightly higher level. The power to the helium blower will be cut off at the same time.

Excessive positive reactor period will cause safety rod trip and blower shutdown.

Excessive temperature in reactor, blower, heat exchangers, or externally cooled pressure shells will sound an audible warning, followed by shutoff of the helium blower at a slightly higher level.

Mechanical vibration or overspeeding of the helium blower will cause it to be shut off.

Excessive pressure in the helium system will be relieved through a relief valve to the low pressure system.

A sudden drop in helium pressure will cause reactor trip, blower shutoff, closing of blower isolation valves, and rapid relief of helium to the low pressure system.

After experience with the plant has been gained, gamma-ray monitors in certain locations can be equipped with alarms to notify the operator of radiation increase above the normal level. It will be desirable to equip various other parts of the plant with malfunction alarms, but plans for these cannot be made until detailed designs of the equipment and plant are available. In general, it is deemed best to trip the reactor for only a few really necessary events, and to give the operator warning of other developments before scram occurs. Such a procedure saves many false scrams, and often allows corrective action to be taken to avoid real ones.

3.9 Site Location, Buildings, and Utilities

3.9.1 Site Location

The proposed Turret reactor facility is to be located in Technical Area 3 (TA-III) at Sandia Corporation, Albuquerque, New Mexico. TA-III comprises three sections of land located approximately 4 miles due south of TA-I, where the major installations of Sandia Laboratories are located (see map, Fig. 1). At present TA-III is devoted to experimental test facilities required in the Sandia Corporation weapons program. The proposed Turret reactor facility site is approximately 1,000 ft from the Sandia Engineering Reactor Facility and the Centrifuge Facility, which are the nearest working areas. Electric power and water lines are available in the general area but will have to be extended to the Turret site.

3.9.2 Containment Sphere

The containment sphere is 60 ft in diameter with its midplane at ground level. It is designed for an internal pressure of 5 psig. It will serve as the weather protection for the reactor and equipment inside. The reactor and major equipment are located in the bottom half of the building, surrounded by concrete shielding. The midplane of the sphere is the operating floor, which is served by a small jib crane in the center. The main helium blower, gas purifier manifolds, transfer pumps, and other equipment are located above the floor. The building is normally ventilated with outside air, but the ducts are provided with automatic pressure-tight hatches where they penetrate the shell.

3.9.3 Service Building

Adjacent to the sphere is a 60 x 60 ft industrial type building, as shown in Figs. 3 and 4. The maintenance and equipment room is a high bay area served by a 3 ton overhead crane. This room contains the fuel processing cave and chemical laboratory. The rest of the main floor, which has normal headroom, is occupied by instrumentation laboratory, counting room, health physics room, shower and locker room, and office. The control room is located below this part of the building and serves as the evacuation shelter for all personnel if contaminated helium is released to the containment sphere.

3.9.4 Other Structures

As shown on Figs. 2, 3, and 4, other structures on the site include the heat dump and its fans, an evaporative cooler for auxiliary cooling and ventilation air, a small stack, and a water tank.

4. PROPOSED OPERATIONAL PROGRAM

4.1 General

The Turret project is expected to be an important source of experimental data of basic value to designers of gas-cooled reactors. The operational program for the plant is arranged to yield the maximum of such information. The information expected can be grouped into six types: (1) Characterization of fission product release as a function of temperature, (2) Nature of active deposits produced on various types of wall surfaces at different temperatures, and means of removing such deposits, (3) Operational behavior and control properties of gas-cooled reactors, (4) Operational behavior of components, such as fuel handling mechanisms, helium blowers, heat exchangers, and materials of construction, (5) The investigation of minimum-cost fuel processing and fabrication, and (6) Efficiency and cost of gas clean-up methods.

4.2 Pre-assembly Testing

4.2.1 Mockup and Operational Testing

Components or features of untried design will be tested and their reliability demonstrated by mockups and test loops prior to final assembly of the system. Components which at present appear to fit into the category are:

a. The 800°F helium blower
b. The turret mechanism
c. Fuel charging system
d. Fuel removal system

e. Helium lock system

The blower, for example, will be tested in a circulating loop at full temperature, 800°F, in order to insure satisfactory operation of the bearings and the seal, and to determine the helium leakage rate.

4.2.2 Leak Tightness and Pressure Testing

All equipment designed to withstand the operating pressures encountered in the Turret system will, insofar as possible, be designed and tested according to the ASME Code for Unfired Pressure Vessels.

All components, and eventually the entire system, will be tested for leak tightness using standard helium mass-spectrometer techniques.

a. Components, insofar as possible, will be purchased with specifications for helium tightness, and tested for acceptance at the point of manufacture by installing blank flanges, temporary heads, seals, etc., as required. The sealed vessel, pipe section, or other component will then be pressurized with helium and leaks sought by bagging the entire component or portions at a time.

b. Field-assembled joints will be tested by similar techniques as various sections of the system are completed. If leaks are found, vacuum methods will be used to pin-point the locations for repair. The completed helium system will be given additional tests, including a pressure loss test.

4.3 Experimental Program

Although it is not practical at this time to write a detailed operational program, the following represents the expected sequence of tests for the Turret system based on studies to date.

After cold critical tests, the initial operation of the reactor will be done at as low a temperature as is consistent with a power of about 300 kw, probably under 800°F. At this condition, thorough checks of shielding, gas clean-up, and plant integrity will be made. All instruments will be checked, and the plant then shut down for any final adjustments or modifications. At this time samples of the radioactivity in the gas stream will be taken and analyzed by gamma-ray spectrometry combined with chemical separation. Equipment walls will be examined for radioactive deposits. If such deposits are found samples will be removed and analyzed, and the methods of sample removal evaluated. Since the plant will be shut down, this can be done thoroughly, and any desired modifications introduced. One or two fuel elements will be discharged and reprocessed to test the fuel system.

The next phase of operation will be conducted at higher temperature, first at the same power, then at higher power (higher helium pressure). During this run attempts will be made to take the same samples as before but without shutting down the plant, or at least without cooling it off. If entry into the containment sphere is difficult during operation at power, the helium pressure will be reduced so that the reactor maintains temperature at minimum power while the samples are being taken.

If no further modifications seem necessary, the same procedure will be repeated at increasingly higher temperatures and power output without intermediate shutdown. A possible selection of temperatures and times for runs is as follows: (1) 800°F helium outlet temperature for 1 week, (2) 1200°F for 3 weeks, (3) 1800°F for 1 month, (4) 2100°F for 2 months, and (5) indefinite operation at 2400°F.

During the above tests, the transition from one operating temperature to the next would be made by charging fuel elements. This makes the process necessarily slow, but even this would be done intermittently, so that the temperature is raised gradually at something like 200°F per day. Continuous gas sampling and recording of survey instruments will be conducted during this time, so that the appearance of new isotopes in the evaporated fission products may be plotted as a function of temperature.

Allowing some time for shutdown and modification, the above program may occupy approximately 6-12 months. Subsequent experimental tests would be planned around the steady operation of the reactor at full power (not necessarily at the highest temperature), to achieve burnup experience with the fuel. After a year at full power about 10% of the fuel will be consumed, and individual fuel elements discharged from the center of the reactor may have as high as 15% burnup. The programming of fuel movement will be carefully worked out to provide the maximum of information. Some fuel channels will be left undisturbed for the life of the reactor, so that the maximum possible burnup can be achieved and evaluated. Other channels will be left for varying periods of time and then sampled by discharging one or more elements. Since there are 312 channels, the possible variations allow sampling of all available burnups at any time.

The fuel processing laboratory will be operated as needed to recover uranium from discharged fuel elements. Careful data will be obtained also on the character of the fission products retained in the graphite at various temperatures, the effects of varying temperature on graphite properties, radiation damage effects, uranium loss or migration, and process experience with the recovery steps. If other recovery methods show promise of low cost on cold tests, they will be tried in the cave. The equipment in the cave is flexible and easily changed.

Operating procedures for the gas clean-up system will also be established by careful experiments during low power operation. It is expected that the system as proposed represents a considerable overdesign in capacity, which was introduced because of the large degree of extrapolation from laboratory tests and the importance of this part of the plant. The actual capacity of the system, methods of regeneration, fission product species collected, and methods of disposal will be studied extensively at very low power levels and applied to higher power operations.

4.4 Shutdown Procedures

Manual Shutdown. It is proposed that normal shutdown of the plant be accomplished by reducing the power of the reactor slowly by venting helium to the low pressure system. At the same time the heat demand of the secondary coolant system would be decreased by reducing flow and shutting off fans, etc. Operation at minimum power would be continued until the high power afterheat had decayed, whereupon the safety rods would be lowered and helium circulation stopped. Cooling of the entire system to room temperature would then be done by the auxiliary cooling system and would normally take at least 24 hr. The process could be speeded when desired by continuing helium circulation after reactor shut down.

Emergency Shutdown. Emergency shutdown procedures are difficult to specify in detail until after the plant is tested. Emergencies involving leakage of contaminated helium from the primary system are of greatest hazard and concern. The general procedure when such leakage is detected would be to reduce reactor power and helium pressure as rapidly as possible. The low pressure receiver system will allow reduction to half pressure in a few minutes, and the rest of the helium must then be pumped out fairly slowly with the transfer pumps. Since the most probable leaks are extremely small ones, this procedure will usually result in negligible release of radioactivity in the sphere. With major leaks or equipment ruptures, which are most unlikely, the rapid blowdown to half pressure will at least segregate part of the radioactivity and salvage some helium. There would be no time for the transfer pump to be effective. When the system is at low pressure but high temperature, precautions will be taken to prevent admission of air to the hot graphite.

5. FEASIBILITY ANALYSIS

5.1 Reactor

The feasibility of the Turret reactor will be discussed from four standpoints: materials compatibility, mechanical performance, fuel element and heat transfer performance, and nuclear operation and control.

5.1.1 Materials Compatibility

Except for the steel pressure vessel, core baseplate, and bearing, all the components of the Turret reactor are made of graphite or carbon. The core and inner part of the reflector are graphite, the rest of the reflector is made of dense carbon, and these parts are all insulated from the steel pressure vessel by a thick layer of porous carbon. All these materials have well known mechanical and thermal properties.[1] The radiation damage effects on graphite are well understood at lower temperatures,[2] but very little work has been conducted at the maximum temperatures contemplated for Turret. There has been very little work on the properties of carbon in the presence of radiation.

The structural demands placed on the materials in the reactor are extremely low, even with severe thermal gradients and transients. There seems to be little doubt that the proper shapes can be produced with the required tolerances, and that the assembly will perform structurally as expected.

The chemical compatibility of the helium coolant with the carbon and steel components is complete.[3] However, carbon and graphite materials as purchased contain adsorbed atmospheric gases. As the reactor is warmed, these gases are gradually released, and at high temperatures will convert small portions of the carbon to carbon monoxide.[4] From the listed gas contents of the proposed materials as furnished by the manufacturer, it is evident that the amount of carbon thus attacked is negligible, provided that the carbon monoxide gas is not allowed to recirculate in the system, which would cause carbon transport and deposit. The proposed gas cleanup system, however, is sized to provide adequate removal of CO, H_2, H_2O, N_2 and other contaminants on a continuous basis as the system is warmed up. This requirement makes the trap system much larger than would otherwise be required. After the carbon materials are outgassed, the load on the cleanup system will consist almost solely of the radioactive contaminants, for which a very large reserve of capacity will exist. Whenever new fuel elements are charged into the reactor, they will also release small quantities of carbon monoxide as the uranium oxide becomes converted to carbide. This gas is trivial in quantity in comparison to the outgassing of the carbon structure, and can easily be handled by the cleanup system.

The effect of the maximum temperature of about 2500°F on the graphite and carbon materials is certain to be negligible in view of the extensive tests of such materials at even higher temperatures.[1] The effect of radiation damage combined with high temperature, however, is less certain. There is some evidence that a slow shrinkage of graphite occurs under such conditions. Study of this effect will be one of the important objectives of the project. Knowledge of changes in graphite at high temperature under irradiation is, of course, important to other gas-cooled reactor concepts as well. Tolerances and adjustments are built into the fuel loading scheme which will accept more than the expected change in dimensions. If unexpectedly large changes are encountered, they will still be gradual enough that regular inspection of test pieces will signal such a result far ahead. Even if cracking or some other mishap should prevent loading of some channels, the reactor can be kept in full operation for several years with fewer than half of the loading channels operable.

In summary, it is concluded that there is sufficient knowledge of the compatibility of materials used in the reactor and that there is little chance that a materials failure will prevent the project from producing the data for which it is proposed.

5.1.2 Mechanical Performance

The mechanical functions in the operation of the Turret reactor are: rotation of the core, charging and discharging fuel elements, and movement of safety rods.

The internal turntable bearing for the rotary core will be the only inaccessible moving part of the entire plant. A more elaborate design could have made the bearing accessible, but a study of the known industrial art in this field resulted in a conclusion that the extra expense was not justified. The bearing, by intentional overdesign, will be lightly loaded and can run with no lubricant. The entire reactor can be recharged with only one revolution, and the required lifetime is expected to be not over 200 full revolutions. The noncorrosive atmosphere and the inflow of cooling helium give operating conditions well within the known serviceability of materials. The spur gears, bevel gears, bearings and other components of the drive will all be made replaceable. The duty cycle for these parts also is so low that operation without lubricant is entirely feasible. The drive components are canned in an extension of the pressure system so that no moving seals are required.

The fuel element charging device as proposed has been operated only as a very simplified laboratory mockup. There will have to be a full-scale operating mockup and thorough testing and modification before the final installation is made. However, the problem is a purely mechanical one involving ordinary materials at ordinary temperatures. The final machine as installed on the reactor is fully accessible with the plant shut down. It is not therefore believed that the charging offers any threat to the success of the project.

The discharge of fuel from the inner end of the fuel channel is accomplished by charging a new element to the other end of the same channel. On emerging from the channel, the displaced element descends in a vertical slot in the stationary core plug. This slot constrains the falling element to the plane of the slot. At the bottom of the slot a curved landing gently introduces an additional degree of constraint, so that the element comes to rest aligned with the discharge ram with no possibility of jamming. A thorough study of the shape of the discharge channel has yet to be made, but it seems highly probable that a configuration can be found which will deliver even broken fragments of fuel elements successfully in the path of the ram. The ram will normally push the aligned fuel element gently through a gas lock, but it will have sufficient power to crush a broken element and deliver the fragments if necessary. It is concluded that the controlled discharge of fuel elements from the Turret reactor is feasible with the proposed mechanism.

The construction and actuation of the nine safety rods proposed for Turret is conventional and employs thoroughly tested principles. There is no question of feasibility in this component. More detailed flux calculations, now in progress, show promise of allowing simplification of the peripheral rods.

5.1.3 Fuel Element and Heat Transfer Performance

In the section on materials compatibility above it was noted that graphite had been studied thoroughly in the presence of helium at temperatures up to and far exceeding those proposed for Turret. Other studies made at Los Alamos[3] included graphite impregnated with uranium oxide and carbide as well. Except that uranium oxide becomes converted to carbide above about 2200°F, there were no effects noted which could affect the performance of the Turret fuel elements. The tensile strength, creep rate, thermal conductivity, and other properties were measured, with, in general, very little effect from the presence of the uranium in the concentrations required for the reactor. In addition, a few tests were made of the rate at which tracer quantities of fission products were evolved from the graphite at various temperatures, as well as the rate of loss of uranium itself.[5] Uranium migration rates were almost negligible at the temperatures proposed for Turret.[6] The fission product releases were consistent with the predicted groupings given in Section 5.3.

Studies reported in Ref. 5 lead to a proposed alternative method of fuel processing. It seems feasible from the tracer studies that a Turret fuel element could be heated to 2500°C for five minutes, lose over 90% of the fission products, and suffer no loss of uranium in the process. The element could then be cooled and reimpregnated with additional uranium if necessary. Further investigation of this possibility is under way.

The fuel element has almost no structural requirements except to remain more or less in one piece and to withstand the thermal gradients set up by the heat release within it. At the low specific power of the reactor experiment (400 watts per gram of uranium) these gradients are very small and the rate of radiation exposure fairly low.

From the standpoint of heat transfer also the Turret fuel elements operate at a fraction of their capability. The difficulty of designing a very small graphite reactor has resulted for Turret in a reactor readily capable of 30 Mw being used for 3 Mw service. One factor in scaling down a high temperature graphite reactor is the temperature rise per foot of coolant channel. To design a very small reactor, only a small ΔT could be used to limit radial heat conduction. Other limiting factors are the carbon-uranium ratio and the impregnation level of the fuel.[7]

5.1.4 Nuclear Operation and Control

The feasibility of testing, starting up, and operating the reactor as desired for the experimental program listed in Section 4.3 is discussed here. Nuclear safety is discussed in Section 6.

The safety rods are normally either in the fully up or fully down position. For trial loadings of the reactor prior to startup, however, it is proposed to use the central rod as a shim. Multiplication tests can then be made by moving the central rod to various positions.

It is presently proposed that the reactor be placed into operation with a "tapered" fuel loading, having higher concentration of uranium in the fuel elements near the periphery of the core, lower concentration near the middle of the channel, and unloaded dummy elements near the center. This method requires the minimum number of additional fuel elements to attain operating temperature, and gives a final loading approximating a highly burned-out fuel near the center.

When the desired initial loading is in place and the reactor sealed and tested, the safety rods will all be fully withdrawn. All further manipulation of the reactor in the direction of increasing temperature will be done by charging fuel. This is completely feasible because of the small increase of reactivity per element. Even with a tapered fuel loading which discharges only a dummy element, the maximum effect of charging one element is estimated to be not over $10^{\circ}F$. Only about one element can be charged per minute. Thus the charging method of control is feasible. This method has the great advantage that no other variables are being changed and there is no excess reactivity present.

When a desired operating temperature is reached, power will be controlled by gradually changing the helium pressure. The rate of such changes is easily subject to any desired degree of control. Unless it is desired to cool the reactor to room temperature, the safety rods will not normally be used. To return to a lower operating temperature some of the fuel must be discharged. Although not as convenient as increases in temperature, the fuel handling system can readily effect a complete discharge of all the fuel, sorting of fuel elements, and recharge for a lower temperature. The time required would be about two days.

In summary it may be concluded that the proposed methods of operational control of Turret are adequate for any likely program of tests.

5.1.5 Neutron Calculations for Turret

An 18 group one-dimensional diffusion code[8] for the IBM 704 computer is being used to make the preliminary calculations for Turret.

The radial variation in effective fuel concentration across the Turret core leads to some complication in utilizing a one-dimensional code, however. The C/Oy ratio versus the core radius is shown in Fig. 17 for three different uniform oralloy concentrations in the fuel slugs. (So far no calculations have been done using the proposed tapered loading.) Since the oralloy concentration in the fuel slugs is small ($\leq$ 0.15 g/cc), fuel slug self-shielding effects may be neglected. In the calculations a one-dimensional cylinder is used with the core divided into five homogeneous radial regions of equal width corresponding to the length of a fuel slug (5-7/8 in.). Vertical group leakages are taken into account by using fictitious group absorption cross sections. The latter are determined from separate one-dimensional plane calculations using

$$\Sigma_{ag} = \frac{-2D_g \frac{d\varphi_g}{dz} \text{ core boundary}}{\int_{\text{core}} \varphi_g \, dz}$$

where D_g and φ_g are the group diffusion coefficients and fluxes, respectively. Critical masses obtained by this procedure give good agreement with experimental masses obtained for graphite-moderated, graphite-reflected critical assemblies at the LASL Pajarito critical facility.[9]

It is estimated the Turret critical mass will be about 8 kg, or that a fuel slug oralloy impregnation of about 0.08 g/cc will be required. Thus the proposed Turret design allows a considerable margin for error since the fuel elements may be impregnated up to 0.15 g/cc. This conservative design approach was adopted, in part, to avoid construction of a separate critical mockup of Turret.

A radial thermal flux plot is shown in Fig. 18 for the case of 0.75 g/cc oralloy impregnation. It is seen the flux, and consequently the power density, is fairly flat across the core. The high flux peak in the central graphite region indicates that it may be possible to obtain sufficient shutdown control from absorbing rods placed in this region alone although this has not yet been verified by calculations. Calculations to determine control rod worths and the temperature coefficients are being performed.

5.2 Gas Cleanup

5.2.1 Oxidation of Impurities

Of the expected gaseous impurities shown below, hydrogen, hydrocarbons, and carbon monoxide can be adsorbed on charcoal more readily if they have been oxidized to water and carbon dioxide. Oxidation of these by hot copper oxide is a standard practice. In Los Alamos Laboratory tests[10] of copper oxide at 930°F in beds 7/8 in. in diameter and 4.5 in. long contained in 347 stainless steel, it has been demonstrated that traces of these gases in the few ppm range are oxidized. The hot copper oxide did not add detectable impurities to the gas. The bed was operated at a flow of 500 ml/min, but its full flow capacity and operating life were not determined. Tests will be made to determine these values in the laboratory. Copper oxide can be regenerated if it becomes necessary by blowing air or oxygen through the hot bed.

TABLE 5.2.1

Impurities Expected in the Helium

Nonradioactive	Radioactive
Hydrogen	Dust
Hydrocarbons	Bromine
Carbon monoxide	Iodine
Carbon dioxide	Krypton
Moisture	Xenon
Nitrogen	Others in traces
Oxygen	
Sulfur	
Hydrogen sulfide	
Sulfur dioxide	

5.2.2 Adsorption of Impurities

Use of activated charcoal to remove impurities from gas streams is a standard practice. For example, the recovery of gasoline from natural gas is routine. In the laboratory, adsorption is one method used for gas purification, generally at very low temperatures to increase the bed capacity. However, in gas chromatography, where traces of gases are concentrated from a carrier stream, the bed can be used at room temperature.

The demonstrated ability[11] of adsorption methods for collection of the fission product gases gives confidence to the use of activated charcoal beds. Bed capacity and regeneration cycles are to be studies in the Turret experiment.

The beds are to be piped in such a manner that they can be connected in various arrangements of parallel or series flow so that studies may be made on the optimum system. The total quantity of charcoal needed has not been determined, but an extrapolation of data obtained on very small traps indicate that the area of charcoal need be no larger than 3000 in.^{2}, and that the length of the bed needed to retain the impurities for a day will not be longer than 18 ft. The tests will allow a determination of these parameters.

5.3 Contaminated Loop

The feasibility of operating a reactor system with a contaminated coolant loop is the principal question to be resolved by the Turret project. There is, however, considerable evidence that supports the technical side of the question. The economic evaluation of feasibility must await quantitative answers from Turret.

If the reactor is operating at its full rated power of 3 Mw and 2400°F, a total of about 3 g/day of fission products will be formed. Of this quantity it is believed that at least 50% and probably more will consist of elements of low vapor pressure or forming refractory carbides.[12] These are: Ga, Ge, Y, Zr, Nb, Mo, Tc, Ru, Rh, Pd, Ag, In, Sn, and rare earths. Such elements will combine with the graphite of the fuel element and remain fixed therein. Of the remaining fission products the majority are expected to consist of elements which have an appreciable vapor pressure at even the lowest temperature of the primary system, 800°F. Such elements are: Zn, As, Se, Br, Kr, Rb, Cd, Te, I, Xe, and Cs. These elements will be rapidly collected in the gas adsorption beds. The remaining elements formed in fission, Sr, Sb, and Ba, can be expected to be potentially capable of leaving the fuel element and condensing in a cooler part of the Turret system. The total quantity of such material is not expected to amount to more than 10% of the original fission products, or about 0.3 g/day. If the average density of the deposit were 10 g/cc and the deposit all lodged evenly in the recuperator (which has a surface of 371 ft^2), the thickness would be about 0.00002 in. after 1 year at full power. Of course, such deposits would not occur evenly, but even a hundred- or thousand-fold increase in thickness would not be a matter of operational concern for Turret. All the other heat exchange surfaces have even greater areas. All heat exchangers and pipes in the primary system will have sampling points so that the existence of deposits can be detected, samples taken, and removal methods studied.

The helium blower is not likely to become a condensation point for volatiles, since it will introduce a slight temperature rise into the gas. However, suspended particulate matter in the helium might be collected by impingement on the blades. Such matter might consist of condensed fission products, graphite dust, or uranium carbide abraded from the fuel elements. There is no way of accurately estimating the quantity of such dust. Graphite elements have been routinely handled in the laboratory through heating, cooling, loading and unloading operations with total weight losses of less than 1 mg.[7] If conditions in passage through the reactor are 100-fold more severe, the dust lost by three elements per day might amount to 0.3 g. The fraction of this dust which might occur in the particle-size range (about 1-5 microns) suitable for impingement on the blower but not on other parts of the system is of course unknown, but cannot be high. It is highly probable that invisible but highly radioactive deposits will cause more operating problems than bulky deposits which could affect aerodynamic efficiency of the blades. For this reason the blower is arranged to be isolated from the system by valves, it is located above the operating floor for ready accessibility, and it will be shielded by portable shielding to the extent necessary. The degree to which particulate activity is building up in the blower can be monitored at all times by means of gamma survey instruments arranged to look through slits at particular parts of the machine. With the blower isolated from the reactor during a shutdown, the interior will be readily accessible through inspection ports. Periscope examination, collection of samples, and internal decontamination and wash-down operations will all be made convenient by the above-floor location of the blower. Since it is the only moving part of the system besides the reactor, the blower is potentially vulnerable to bearing or motor failure. The degree to which access to and reliability of the blower can be maintained will constitute an important experimental result of the project.

All other parts of the primary system present much less probability of trouble from the presence of contaminated gas in the loop. In summary, therefore, it may be concluded that the feasibility of operating the Turret reactor with such a loop as proposed is certainly assured for low powers and low temperatures. At higher powers and temperatures it is to be expected that some problems will appear. Since the program calls for a gradual approach to design conditions, with continual observation of performance and collection of samples and data, these problems will be recognized at an early stage and appropriate corrective measures developed, if possible. The internal core bearing is not subject to much correction after operation, but the plan is to develop confidence in the reliability of this item by exhaustive pre-operational testing.

5.4 Heat Exchangers

The recuperator and the primary to secondary loop heat exchanger appear to be the only items of heat exchange equipment which may require an extension of existing technology. Therefore, only these two items will be discussed.

Recuperator. The high temperature involved, ranging from 800°F to 2400°F, precluded the use of an all metal construction for the recuperator. Graphite was chosen as the structural material because it is easily machined, is available in large sizes, and has excellent physical properties at high temperatures. It was determined that holes up to 6 ft in length could be drilled in graphite, and that a counterflow exchanger of holes drilled in a block was more compact than a shell and tube exchanger. The required hole pattern can be fitted into a standard 35 in. diameter electric furnace electrode.

Using a block type of construction results in a stronger unit than the conventional shell and tube construction. The hollow bolt and spacer method of attaching the headers utilizes no new methods of construction, and should introduce no fabrication problems. By mounting the exchanger in a vertical position and surrounding the exchanger with a graphite sleeve, the core of the exchanger is free to expand and contract under conditions of changing temperature, thus minimizing temperature-induced stresses.

Heat Exchanger. Although the heat exchanger will operate with an inlet gas temperature of 1600°F, this device represents no advance of technology for two reasons. First, the exchanger will be constructed of stainless steel which has useable strength at these temperatures. Second, the tubes, which are the hottest portion of the exchanger, will be stressed very little because of the equalized pressures of the primary and secondary loop coolants. The shell side of the exchanger, which will be required to withstand 500 psi internal pressure, will be at a maximum temperature of

$800^{\circ}F$. At this temperature, stainless steel has considerable strength. Thermal gradients in the tube sheets will be reduced by baffles.

5.5 Fuel Processing

At the 3 Mw power level of the reactor about 3 g of uranium will be fissioned per day. If the fuel elements can be used only to 10% burnup, then about 30 g of uranium, or 4-6 fuel elements, must be processed per day to operate at equilibrium with the reactor. This rate is a laboratory-scale operation and is of a convenient size for study of processing methods and properties of fission product residues.

All of the operations of impregnating and reprocessing fuel elements have been tested at Los Alamos using nonradioactive materials and normal uranium.[7] Some reprocessing tests with radioactive tracers have also been done. Results indicate that the proposed steps (see Fig. 16) are easy to carry out and accomplish desired results. The degree of decontamination achieved in actual practice, however, may differ considerably from that obtained with tracer techniques.

The mechanical apparatus used in the reprocessing cave is based on standard laboratory apparatus and well known remote-control techniques. These techniques have been in use in the Los Alamos Scientific Laboratory for the past ten years.

6. PRELIMINARY EVALUATION OF HAZARDS ASSOCIATED WITH REACTOR AND EQUIPMENT MALFUNCTION

6.1 Safety Features of Nuclear and Control System Design

Control of reactivity in the Turret reactor will be by adjustment of the total quantity of fuel loaded. After making an initial loading to produce a critical configuration at room temperature, additional fuel must be added to bring the reactor up to operating temperature. It is estimated that the temperature coefficient of Turret is about $\Delta k/k = -2.5 \times 10^{-5}/^{\circ}F$. In Turret, the net negative temperature coefficient is largely associated with moderator temperature. The relation between fuel temperature and reactivity has not been examined in detail at this time. A change in reactivity of about 6% must be made after cold critical before the operating temperature of approximately $2600^{\circ}F$ can be achieved. The reactor is to be operated at power with all safety rods fully withdrawn. Therefore, when an equilibrium temperature corresponding to a particular loading is reached, no excess reactivity is "held up" in any control devices.

Experimental data from similar graphite-uranium systems[9] allows the estimate that the relationship between fuel mass and k is approximately $\Delta k/k \leq 1/2\ \Delta m/m$. At operating temperature, 1560 fuel elements will be in the reactor. On the average then, the insertion of a single fuel element increases k by 3×10^{-4}. At room temperature, the Δk per fuel element would be perhaps twice this value, or between 5 and 10¢. The design of the fuel element loading mechanism precludes loading more than one element at a time, and about 1 min of "cycle time" is required between injections of fuel elements. Variations in uranium concentration in fuel elements cannot more than double the reactivity worth per element noted above. It is therefore concluded that reactor excursions cannot be brought about during fuel loading by operator error, nor by equipment malfunction.

Nine safety rods (see Section 3.2.7) are provided in the Turret reactor; one rod may be dropped into the central graphite island, and the remaining eight enter the core at approximately mid-radius. All rods are gravity operated, and are released by interruption of current in a holding magnet. The function of the rods is twofold; first, they permit rapid shutdown of the reactor in the event of system failure. Second, they insure that the reactor can be shut down to room temperature in case trouble with the fuel loading mechanism temporarily precludes unloading the fuel. The central rod, and at least four of the eight core rods, may be dropped in regardless of the rotational position of the core. The central rod is estimated to be worth 4-6% in k, and four of the outer rods about another 5%. More detailed calculations are in progress for estimation of rod worth, but there does not seem to be any doubt that adequate shutdown capacity is available.

Once the outer rods have been dropped in, it is intended that interlocks prevent further rotation of the core or charging of fuel. After a scram from any normal operating condition, the central rod alone will be sufficient for shutdown before cooling has occurred. Therefore if it appears desirable to remove fuel after finding the cause of the scram, the outer rods may be raised again and dummy fuel elements charged until sufficient fuel is removed.

Safety rod drop can be initiated by the operator at his discretion, and by various signals from the reactor system itself. Among the system conditions which may be used to produce a scram are the following: excessive coolant temperature, blower power failure, excessive coolant pressure, too-high neutron level, reactor period shorter than 5 sec, sudden loss of helium pressure, and excessive radiation levels at selected locations in the reactor system and building.

In the event that a spurious (or real) scram signal causes the rods to drop, the withdrawal rate during the subsequent startup is limited to 40 in./min or about 2-3¢/sec. Control system sequencing interlocks will preclude the withdrawal of more than one rod at a time. In the event of interlock circuit failure, the operator could not conveniently activate more than two rods simultaneously.

It may be noted that the control rod worth is adequate to compensate for the increase in reactivity produced by cooling to room temperature, as well as for any increase due to xenon decay during shutdown. Presumably, equilibrium xenon poisoning will be much less than the value characteristic of the flux, since xenon and its iodine precursor will be lost from the fuel elements.

It appears that there is no criticality hazard accompanying a major disturbance of the core which somehow managed to concentrate the fuel elements into one region surrounded by graphite. Estimates show that for cubical configurations of fuel elements placed side by side and surrounded by one foot of graphite the critical mass is of the order of 50 kg, or more than six complete loadings. On the basis of this estimate, there is also no criticality hazard associated with the handling of fuel elements external to the reactor (i.e., in reprocessing operations). However, uranium solutions used in impregnating the fuel elements should be subject to the usual restrictions applied to the handling of solutions of fissionable materials.

6.2 Safety Features of the Turret Mechanical Design

The Turret design is such that a loss-of-coolant accident is unlikely to initiate a chain of events leading to catastrophe. In case of coolant flow stoppage, a rod scram will rapidly shut the reactor down to a low multiplication. The rods, since they are not to be used as control devices and hence need not be very precisely or rigidly located in the core, can be constructed with relatively large clearances between them and their guide channels. It is most unlikely that the rods will fail to be released from their electromagnetic holders when a scram signal is given, or that once released, they will not fall into place in the core.

No pumped-flow auxiliary cooling for removing residual fission product power has been included in the Turret design. A simple alternative, that of designing into the reactor proper a heat leakage approximately equal to the residual power dissipation requirements, will be adopted. It is estimated that the heat leakage from the reactor pressure shell will be at least 100 kw if the shell temperature is about 200°F above ambient air temperature, and if free convection of surrounding air is allowed. This level of heat leakage will match or exceed the residual power 100 sec after shutdown following a 1 year, 3 Mw operation of the reactor. No detailed examination of the situation during the first 100 sec on shutdown has been made; however, it may be pointed out that heat capacity of the core region alone--roughly 9000 lb of graphite--would absorb 100 sec of 1 Mw operation (~600 w/fuel element) with a temperature rise of about 20°F. In the practical case, the temperature rise in the fuel elements themselves would probably be many times the average observed for the core structure as a whole; however, there is no reason to expect fuel element damage even if the temperature rise in them was as much as 1000°F.

During the first day of the shutdown situation noted above, from 0.5 to 1 Mw-day of energy would be dumped to the air of the containment sphere. Under wintertime climatic conditions, this heat load (25 kw average value) would be dissipated by leakage to the out-of-doors without difficulty. During the summer months, some dependence would be placed upon the refrigeration units which are presently thought to be required if the containment sphere is to be kept at a reasonable temperature during normal, full power operation. It is assumed that these refrigeration units can be operated from "standby" power sources. More detailed consideration of the shutdown situation will probably confirm that if reactor shell temperature is allowed to go to 450°F--a reasonable value, particularily if the coolant pressure has been reduced by the flow stoppage--and the containment sphere permitted to reach 200°F, shutdown power could be dissipated from the system by leakage alone even if out-of-doors ambient temperatures were in the 100-120°F region.

Localized coolant flow stoppage appears improbable, and if it did occur in a few channels, would probably not have any dangerous consequences. In order for a coolant channel to become plugged, it would be necessary not only for a fuel element to fragment, but for the fragments to somehow pile up and plug the channel in question. In view of the small stresses expected in the fuel elements such breakage does not seem likely. Since each channel produces only 0.003 of the total power, the "hotspot" problem should not be severe even if isolated coolant channels were plugged or otherwise "starved."

Since Turret fuel elements are initially unclad, no abnormal release of fission products should accompany the breakup of individual fuel elements in the reactor. It may be noted once again that the design temperature of the core components is quite conservative in view of the known high temperature structural properties of graphite.

The Turret system utilizes "double containment" to reduce the possibility that the reactor core and reflector can be exposed to large quantities of air (oxygen); i.e., the core is contained in the pressure shell, which is in turn enclosed by the containment sphere. The primary coolant system is completely enclosed by the containment sphere as well, dumping its heat to the secondary loop which, of course, penetrates the sphere walls in order to connect with the final heat dump system. Two simultaneous failures, one in the primary loop and one in the secondary, must occur before the reactor core would be exposed.

In the event of a major failure in the primary loop (for instance, overspeeding and consequent disintegration of the blower and its casing) provision will be made for minimizing the rate at which the air in the containment sphere could enter the core region. This can be effected by closing valves or "dampers" in the inlet and outlet coolant pipes at a location near the reactor shell. The valves, and the piping between them and the reactor shell, will be protected (by massive concrete) from physical damage in the event of blower disintegration.

The containment sphere will safely contain all the primary loop coolant gas. Should the accompanying "shutdown" heat dissipation requirements raise the pressure in the sphere to an unsafe value, the excess pressure will be bled off to the gas disposal system. Pressure in the sphere is to be prevented from falling below atmospheric by a conventional type of check valve which will allow air to flow only into the sphere.

6.3 Possible Accidents

In view of the mechanical and nuclear design features previously noted, it appears impossible to imagine a nuclear excursion which would damage the reactor to such an extent that the pressure shell could rupture or that the core would be explosively disrupted.

There remains the possibility of oxidation of the reactor core following rupture of the coolant system. The rate at which air from the containment sphere could enter the core region of the reactor can certainly be made quite small if the shut-off valves described earlier remain operative. About 4% of the reactor system graphite could be oxidized to CO_2 with the air available in the containment sphere; the energy release would be about 25 Mw-hr. No estimate as to the average rate of energy release has been made. However, it may be noted that laboratory experiments carried out with graphite at 1600°F show a very slow (and certainly not explosive) rate of oxidation in an atmosphere of O_2. There is no obvious means of absolutely preventing such a reactor incident, short of filling the containment sphere with nitrogen, for instance. (This greatly complicates the matter of allowing access to the sphere region for blower maintenance, etc.)

No attempt has been made to estimate the rate at which accumulated fission products would be released into the containment sphere under the conditions postulated above. It is certainly unrealistic to assume that all fission products immediately are dispersed throughout the sphere. While the problem will be studied in more detail, it can be said that:

a. Personnel can be adequately shielded and protected in some area of the reactor building while evacuation procedures are implemented, and

b. Evacuation of personnel from the reactor area can be accomplished in a short enough interval so that integrated dosage from the contaminated sphere is not a matter of grave concern.

7. COST ESTIMATES AND SCHEDULES

The cost estimates and time schedules presented here are based on the Turret reactor system design described in detail in Section 3 of this report and shown in the attached drawings.

7.1 Cost Summary

The estimated costs for the Turret reactor project are summarized in Tables 7.2.1, 7.2.2 and 7.2.3. Table 7.2.1 provides a breakdown of the estimated costs into the categories of Engineering Cost, Construction Cost, and Operational Cost. The cost for each of these categories is further broken down by fiscal year. The fiscal year breakdown is based on the planned schedule for the entire project, which is discussed in detail in Section 7.2.

The overall costs were established by estimating the construction and development costs for each major component of the system, as shown in Table 7.2.3, and summarized in Table 7.2.2. This breakdown does not include engineering salaries at Sandia Corporation or Los Alamos.

As shown in Table 7.2.1, the total estimated engineering cost including all salaries is $2,057,000, and the total estimated construction cost is $2,871,000. Allowing 15% for contingency, the total engineering and construction cost is estimated to be $5,667,000.

Operational costs have been estimated only for the remaining eight months of FY-1962. Including the cost of operating personnel and the cost of maintenance and developmental modifications, the operational cost for this period is estimated as $675,000. This operational period includes final check out of the completed reactor system, non-nuclear testing, and loop operation prior to cold critical and full operational tests.

Thus, including the 15% contingency, the required funds by fiscal years may be summarized as follows: for FY-1960, $1,035,000; for FY-1961, $3,419,000; and for FY-1962, $1,888,000, including eight months of reactor operation.

These figures represent the expenditures by fiscal year. However, reference to the time schedule shown in Fig. 19 indicates that it will be necessary to commit funds for the main construction contract during FY-1960 if the schedules are to be met. In addition, purchase orders must be placed for a large percentage of the Sandia-furnished equipment. It is estimated that, in addition to the budgeted funds listed above, carry-over commitments must be allowed at the end of FY-1960 of $2,000,000, and at the end of FY-1961 of $700,000. These funds have been included in the budgeted expenditures for the respective following fiscal years since the actual expenditure will occur then. Permission to commit these funds during the year prior to their expenditure is, therefore, assumed in the time schedules proposed.

The estimated total project cost through FY-1962 is $6,342,000.

7.2 Schedule

The time schedule for the Turret reactor project is summarized in Fig. 19.

The feasibility phase is scheduled for completion by the end of April 1959, and is concluded by the publication of this report. Design drawings and specifications for all system components are scheduled to be completed by December 1959 in order to allow a sufficient time interval for preparation of bid requests, bid evaluation, and contract placement. All components are scheduled to be fabricated, tested, and available for installation by the time of beneficial occupancy.

The building and containment line drawings are scheduled for completion by August 1959, at which time the negotiations for an Architect and Engineering firm will be completed and a contract placed with the selected A and E. Title I is scheduled for completion in 60 days, and Title II in four months thereafter. Three months have been allowed for construction bid evaluation. Construction should be completed in twelve months, including Title III. Thus, beneficial occupancy is scheduled for May 1961.

An additional four months is scheduled to allow installation of additional equipment and instrumentation. Final closure of both the reactor pressure vessel and the containment sphere are scheduled for completion by October 1961.

Final tests and inspection of all equipment will then be followed by cold critical operation of the reactor system, scheduled for January 1962. Full-scale operational tests are scheduled immediately following the cold critical operation and will continue to the conclusion of the project.

TABLE 7.2.1

Turret Reactor Cost Summary (Thousands of Dollars)

		FY-60	FY-61	FY-62	TOTAL
A.	Engineering and Development				
	1. Sandia and LASL Engineering	$650	$760	$287	$1697
	2. Development Hardware	95	49		144
	3. Development Testing	35	30	15	80
	4. A and E Cost	120	10	6	136
	Total Engineering and Development Cost	$900	$849	$308	$2057
B.	Construction Cost				
	1. Construction Contract		$1000	$ 364	$1364
	2. Sandia Purchased Equipment		1094	200	1294
	3. SC Installation and Inspection		30	183	213
	Total Construction Cost		2124	747	2871
	Total Engineering and Construction Cost	900	2973	1055	4928
	Contingency at 15%	135	446	158	739
	Grand Total	$1035	$3419	$1213	$5667
C.	Operational Cost (8 months)				
	1. Personnel			475	475
	2. Maintenance and Developmental Modifications			200	200
	Total Operating Cost			$675	$675
	PROJECT GRAND TOTAL				$6342

TABLE 7.2.2

Turret Reactor Cost Summary

Construction and Development Costs*

1.	Building and Containment	$ 618,000
2.	Site Utilities	280,000
3.	Reactor and Fuel Loader	595,000
4.	Fuel Reprocessing	50,000
5.	Heat Exchangers	137,000
6.	Auxiliaries	331,000
7.	Blowers, Pipes, and Valves	535,000
8.	Gas Make-up and Removal	171,000
9.	Gas Clean-up	274,000
10.	Non-nuclear Instrumentation	85,000
11.	Nuclear Instrumentation	155,000
	GRAND TOTAL	$3,231,000

*LASL and Sandia Engineering Salaries not included.

TABLE 7.2.3

Cost Breakdown

1. Building and Containment	
Sphere	$132,000
Concrete in Sphere	77,000
Excavation	11,000
Service Building	167,000
Holding Tank	5,000
Stack	10,000
Crane, inside Building	20,000
Crane, inside Sphere	60,000
Total	$482,000
A and E	136,000
TOTAL	$618,000
2. Site Utilities	
Roads	3,000
Water Lines	18,000
Septic System	2,000
Electric Lines	150,000
Distribution	70,000
Water Storage	25,000
Back Fill and Level	7,000
Fence	5,000
TOTAL	$280,000

Table 7.2.3 (Continued)

3. Reactor and Fuel Loader Cost	
Pressure Vessel	$ 60,000
Core, Core Plug and Reflector	45,000
Carbon Reflector and Insulation	50,000
Thermal Shield	25,000
Indexing, Plate and Bearing	50,000
Elevating Mechanism	19,000
Loading Mechanism	58,000
Safety Rod, Main	25,000
Safety Rods, Secondary	84,000
Feeder and Discharge	52,000
Handling Elements to Hot Cell	20,000
Primary Loading and Unloading Control System	40,000
Primary Pilot System	43,500
Additional	
Development, Hardware and Testing	23,500
TOTAL	$595,000

4. Fuel Reprocessing Cave	
Structure and Shielding	$ 10,000
Two Windows	4,000
Two Manipulators	20,000
Two Furnaces	1,000
Centrifuge	2,000
Transport Tubes, Cart and Drive	5,000
Cell Air Treatment	2,000
Waste Cart with Bottles and Pig	1,000
Miscellaneous Equipment	5,000
TOTAL	$ 50,000

Table 7.2.3 (Continued)

5. Heat Exchangers	
Recuperator	
Engineering	$ 2,000
Fabrication	35,000
Manufacturer's Test, Adjustment and Retest	5,000
Installation	2,000
Inspection	2,000
Development Tests	5,000
Thermal Shock, Flow and Heat Transfer, etc.	
TOTAL	$ 51,000
Main Heat Exchanger	
Fabrication	$ 24,000
Manufacturer's Test, Adjustment and Retest	5,000
Installation	2,000
Inspection	2,000
TOTAL	$ 33,000
Heat Dump and Air Blower	
Procurement of Exchanger	$ 25,000
Procurement of Air Blower	5,000
Installation	5,000
TOTAL	35,000
Development Hardware	8,000
Development Testing	10,000
TOTAL HEAT EXCHANGERS	$137,000

Table 7.2.3 (Continued)

6. Auxiliaries	
Refrigeration	$113,000
Accessory Pumps	25,000
Installation	80,000
Heat Dumping Exchanger	30,000
Internal Distribution System and Air Circulators to Cool 8 Major Components	40,000
Development Testing	5,000
Auxiliary Power Supply	38,000
TOTAL	$331,000
7. Blowers, Pipes and Valves	
Primary Loop Blowers	$150,000
Blower Development and Test	10,000
Auxiliary Loop Blowers	150,000
Primary Loop Piping	30,000
Development and Test	10,000
Bellows Expansion Joints	30,000
Expansion Joint Development and Test	30,000
Blowdown - Piping and Regulation	6,000
Make-up - Piping and Regulation	3,000
Miscellaneous (Control Piping for Valves)	10,000
Valves, Seals and Actuators	106,000
TOTAL	$535,000

Table 7.2.3 (Continued)

8. Gas Make-up and Removal	
High Pressure Storage for Helium	$ 15,000
Blowdown Tanks	24,000
Pump - Blowdown to Gas Clean-up	53,000
Pump - Gas Clean-up to H. P. Storage	53,000
Helium Storage System	15,000
Secondary Loop Storage and Control	11,000
TOTAL	$171,000

9. Gas Clean-up	
Engineering	$ 3,000
Tanks	72,000
Valves, including Control System	50,000
Piping	25,000
Heat Exchangers Heaters and Coolers	20,000
Regeneration Blower	10,000
Vacuum Pump	5,000
Charcoal	12,000
Copper Oxide	10,000
Development Hardware	10,000
Development Testing	5,000
Installation	50,000
Inspection	2,000
TOTAL	$274,000

Table 7.2.3 (Continued)

10. Non-nuclear Instrumentation

(Pressure, temperature, flow, helium detection)

Engineering	$ 3,000
Procurement	75,000
Test	2,000
Installation	5,000
	$ 85,000

11. Nuclear Instrumentation

Gamma Monitoring Instruments	$ 25,000
Reactor Instruments	110,000
Development Hardware and Testing	20,000
TOTAL	$155,000
GRAND TOTAL	$3,231,000

REFERENCES

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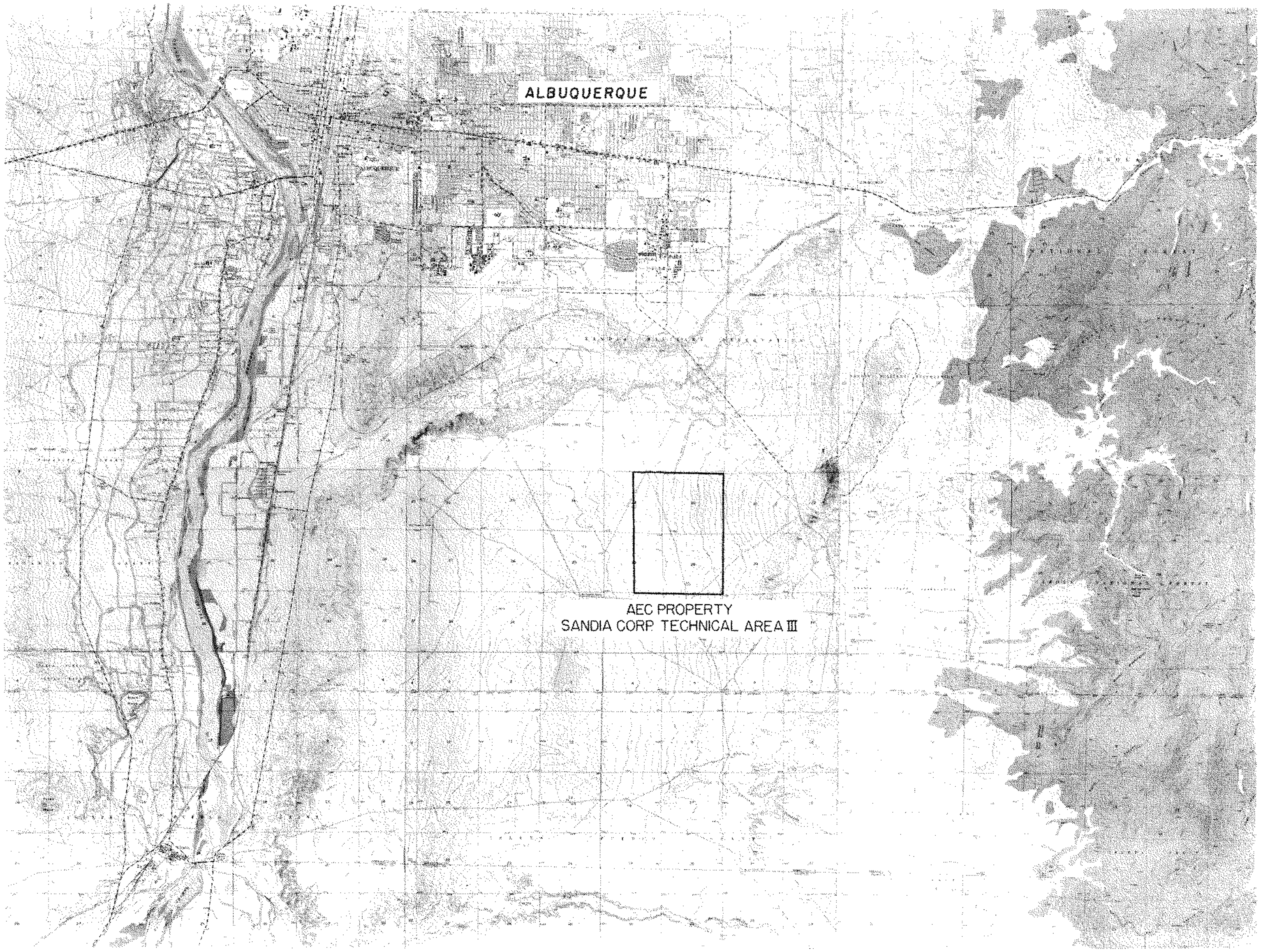

Fig. 1 Map of Site and Surrounding Area

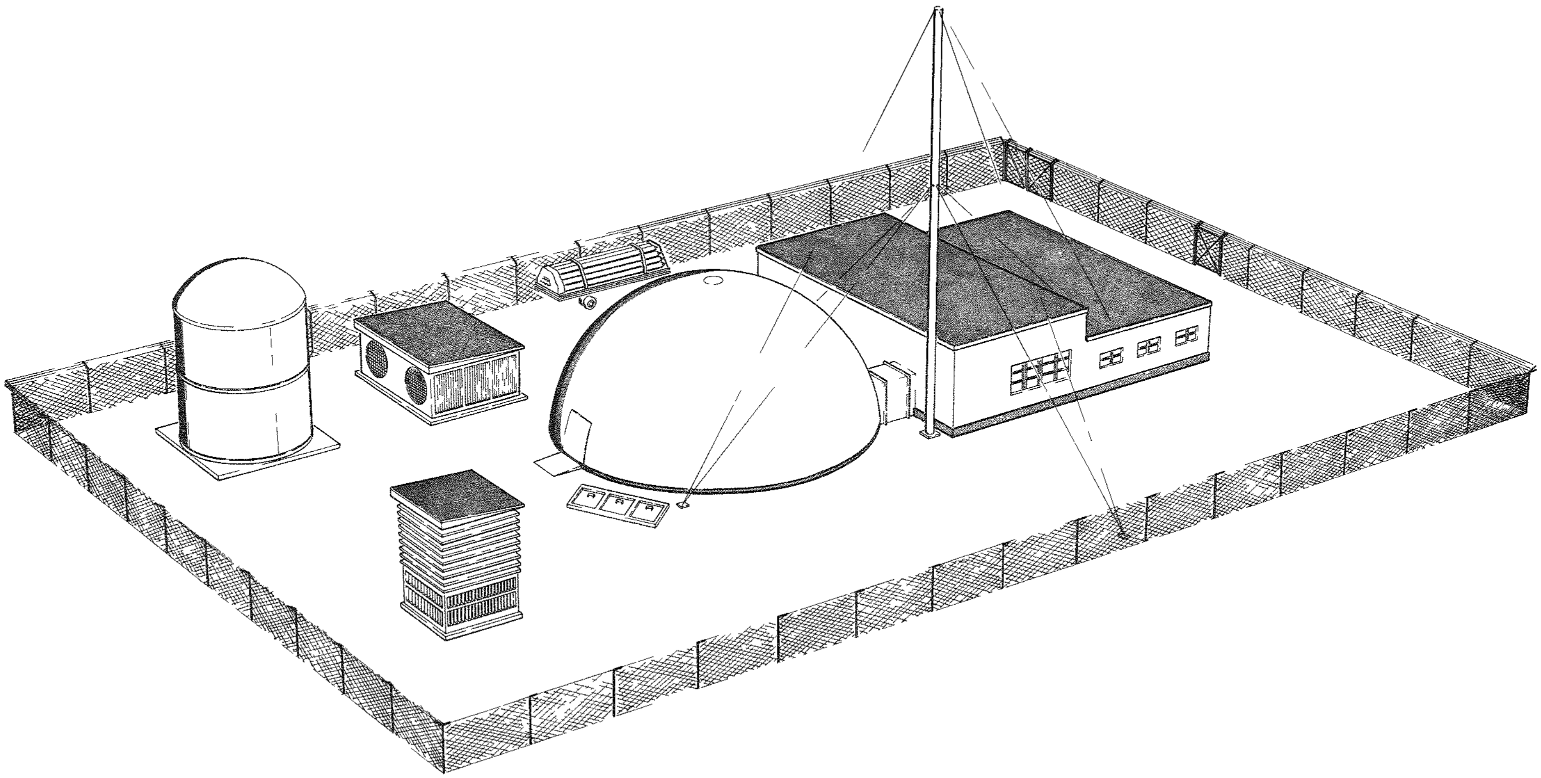

Fig. 2 General View of Turret Facility

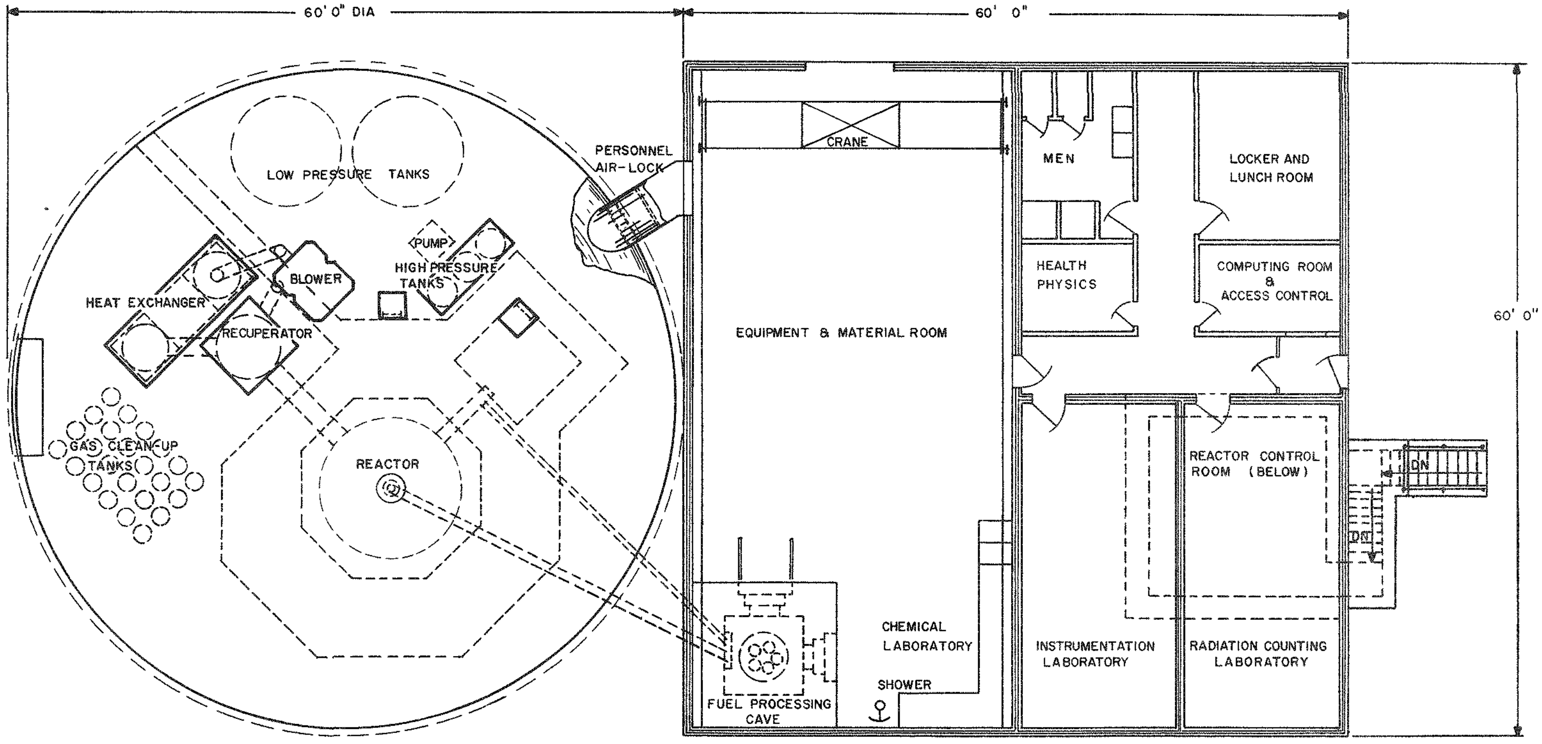

Fig. 3 Floor Plan of Buildings

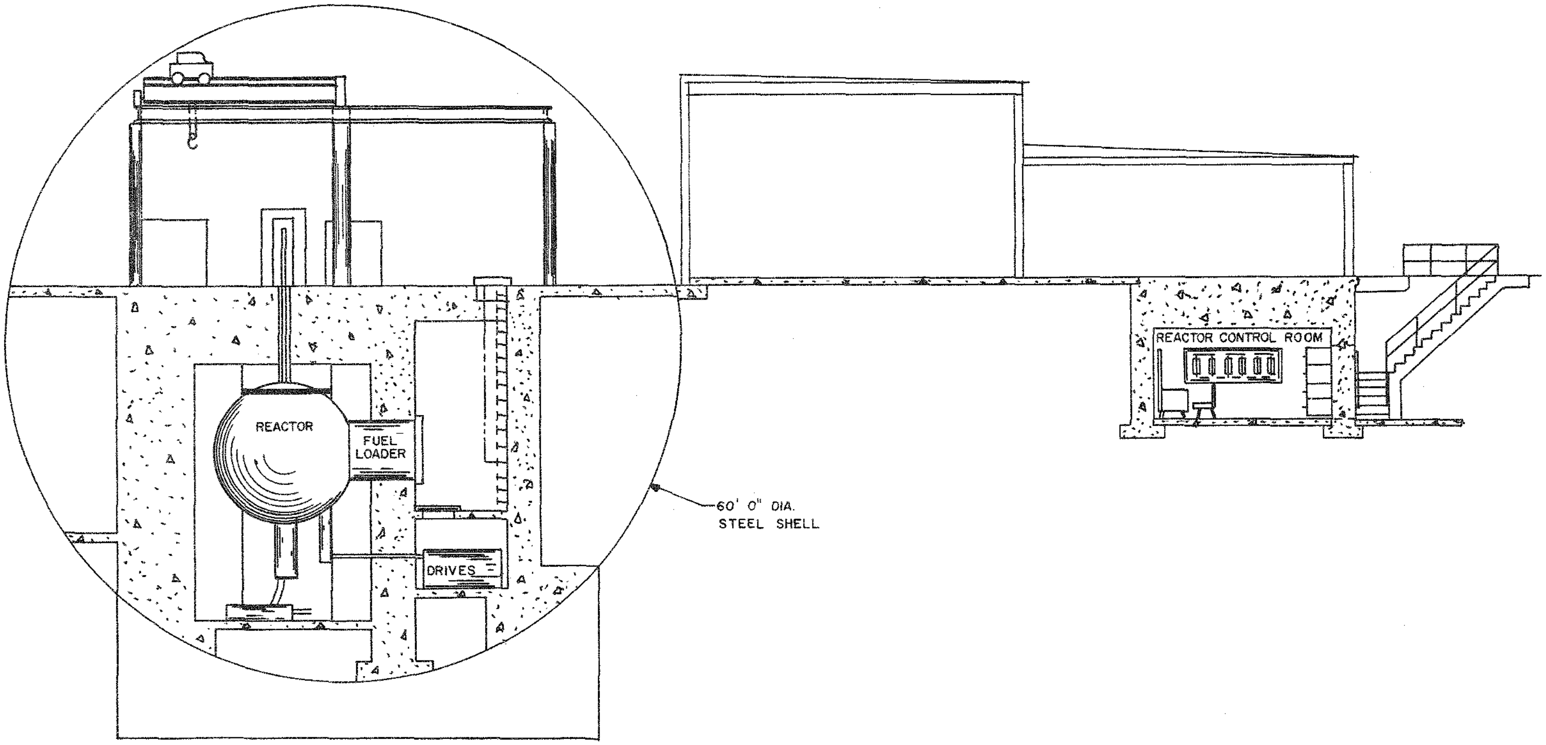

Fig. 4 Section of Buildings

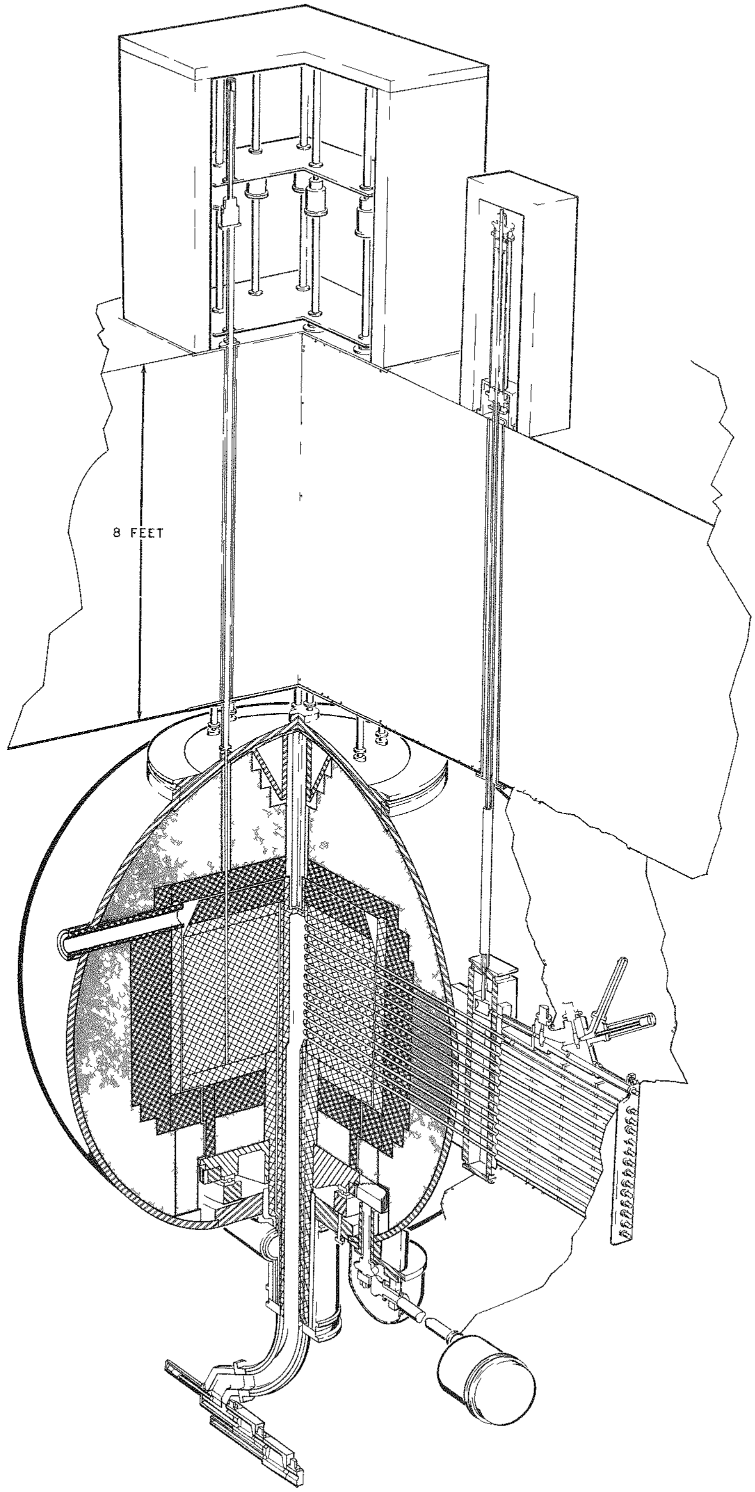

Fig. 5 Vertical Section of Turret Reactor

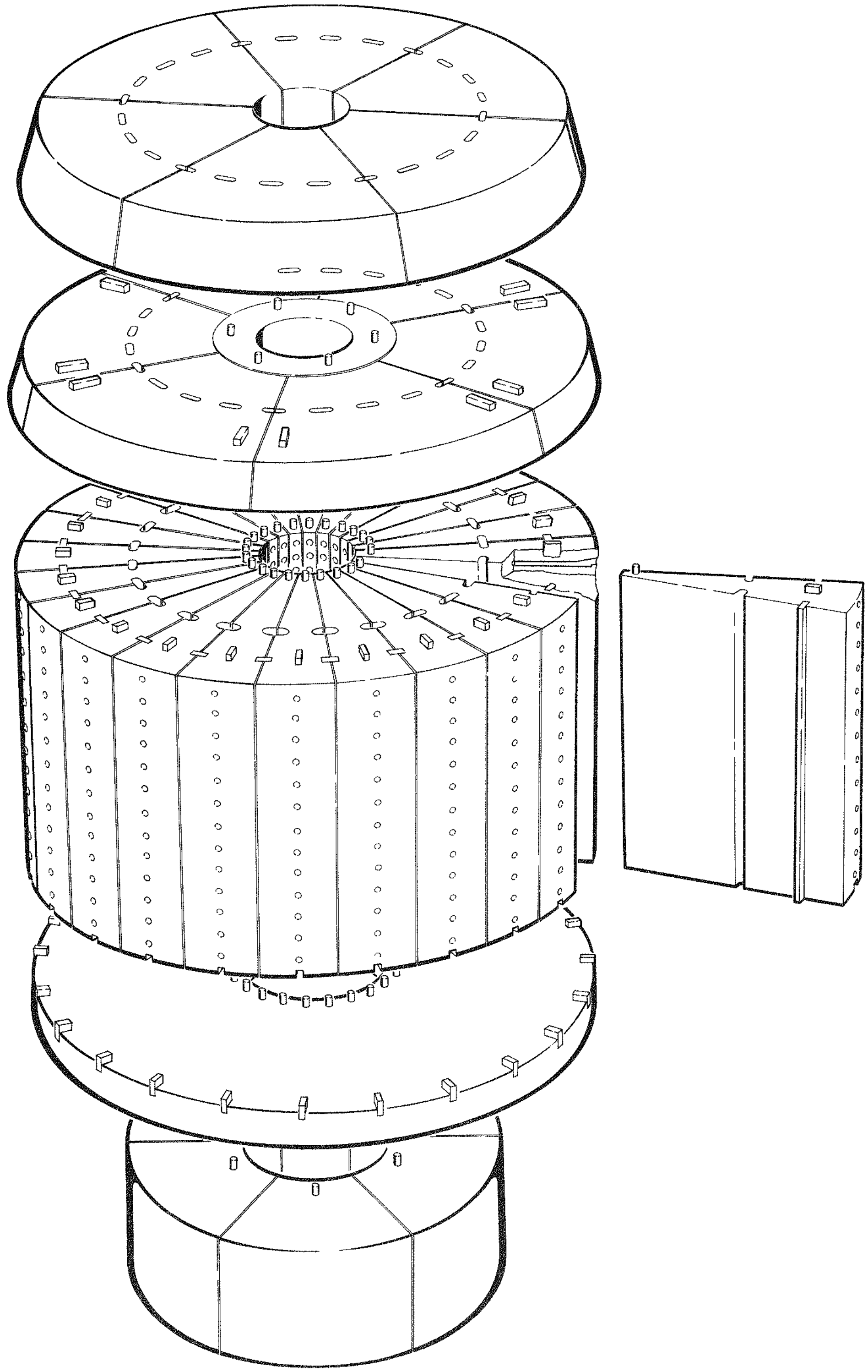

Fig. 6 Detail of Core

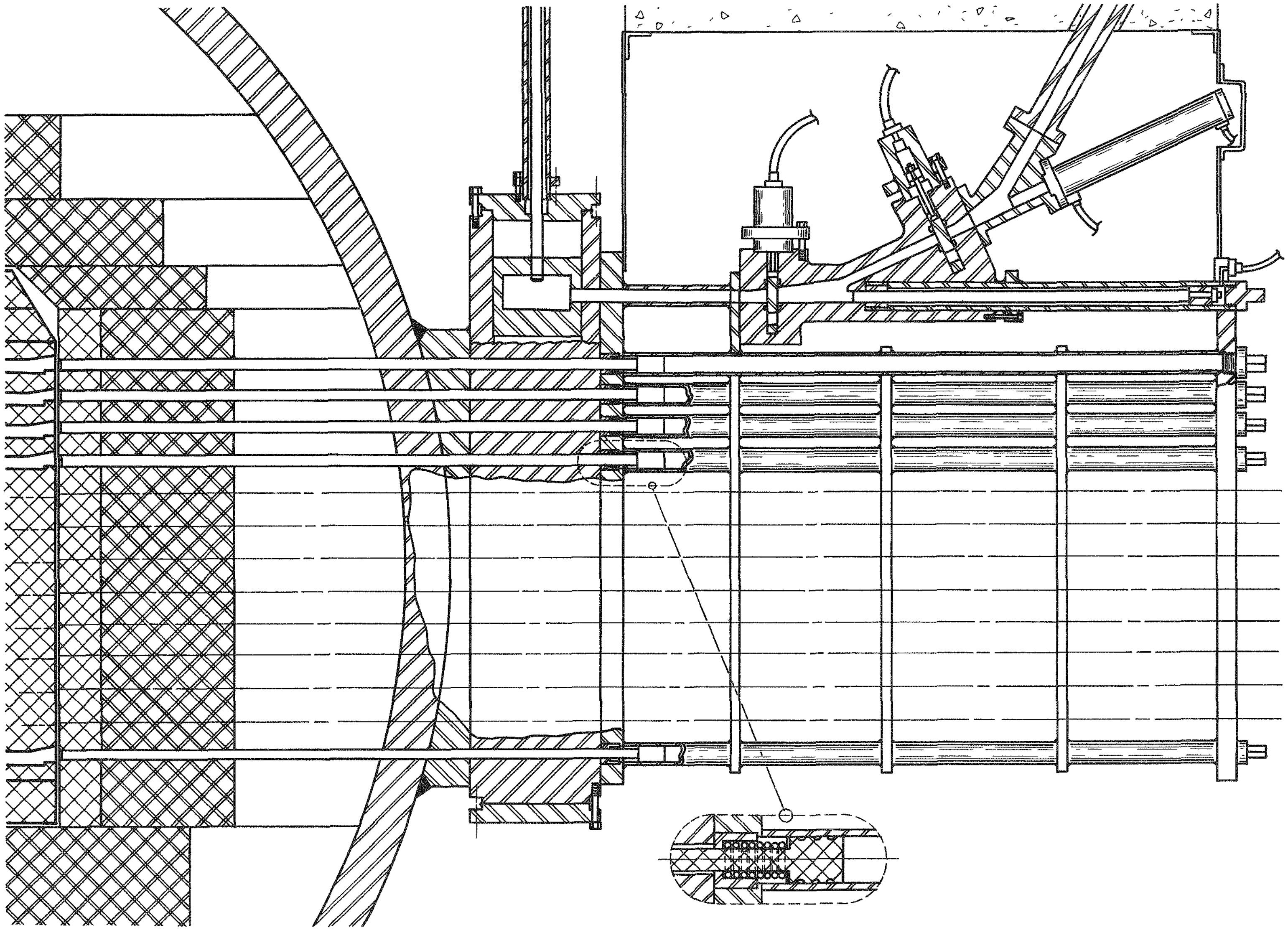

Fig. 7 Section through Fuel Loader

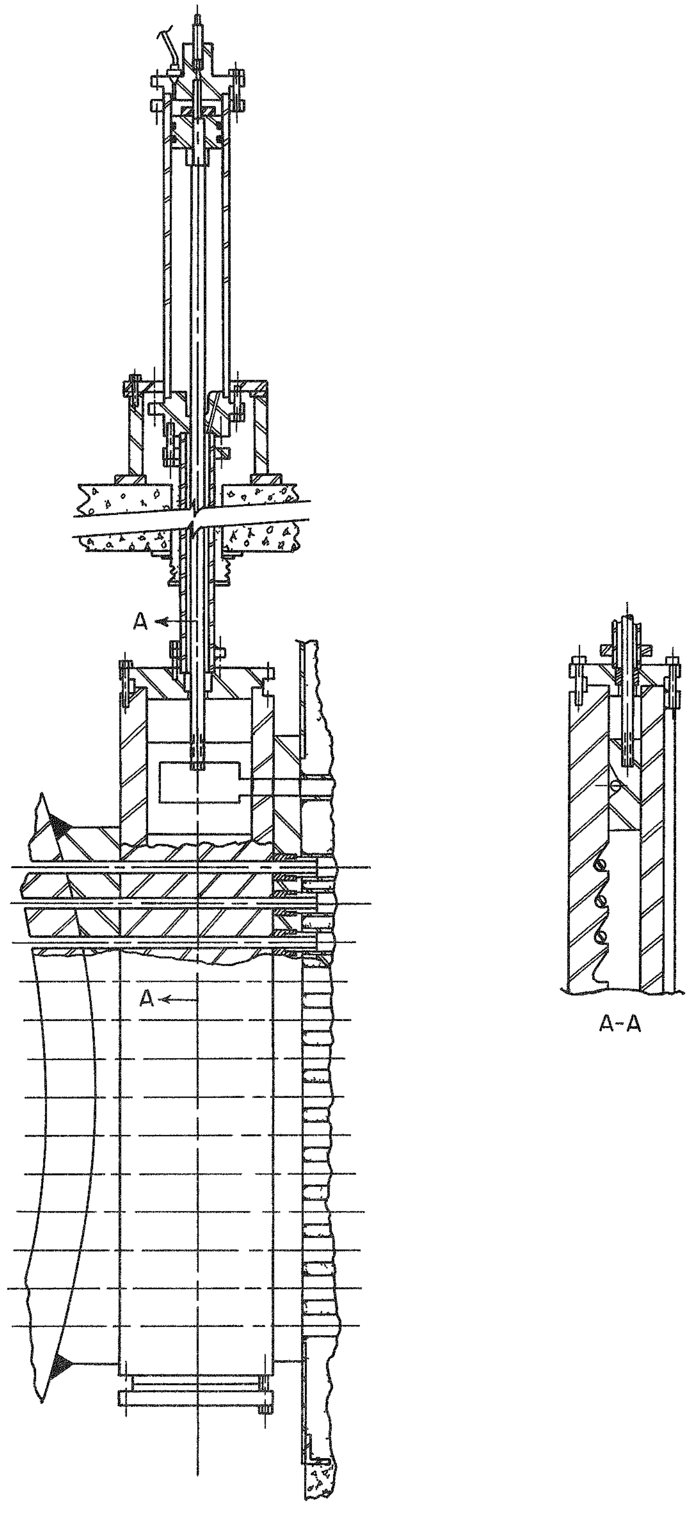

Fig. 8 Detail of Fuel Loading Elevator

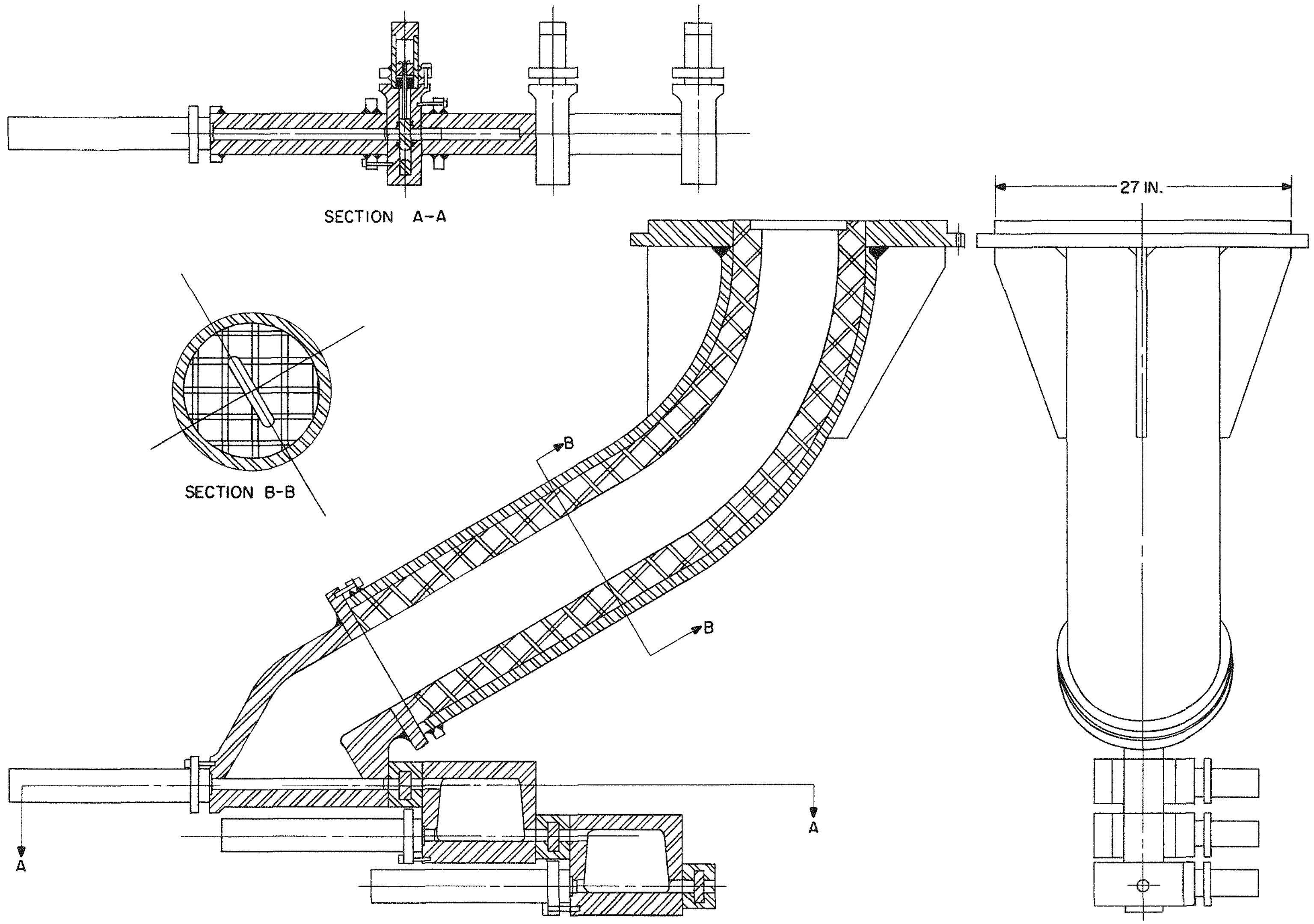

Fig. 9 Detail of Discharge Port

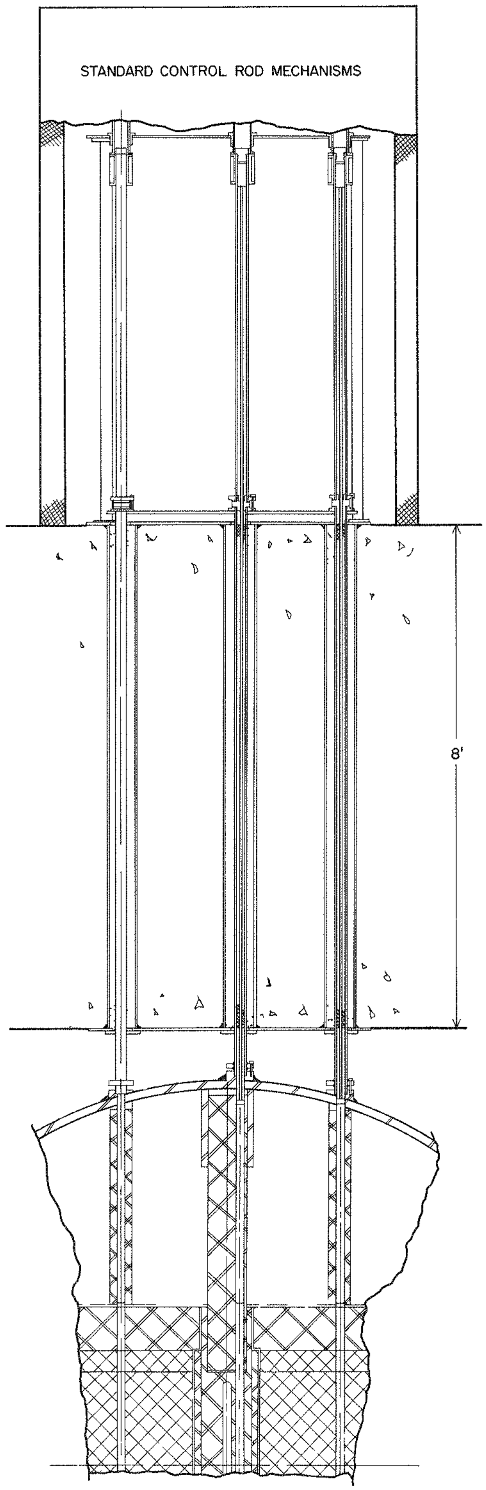

Fig. 10 Detail of Safety Rods

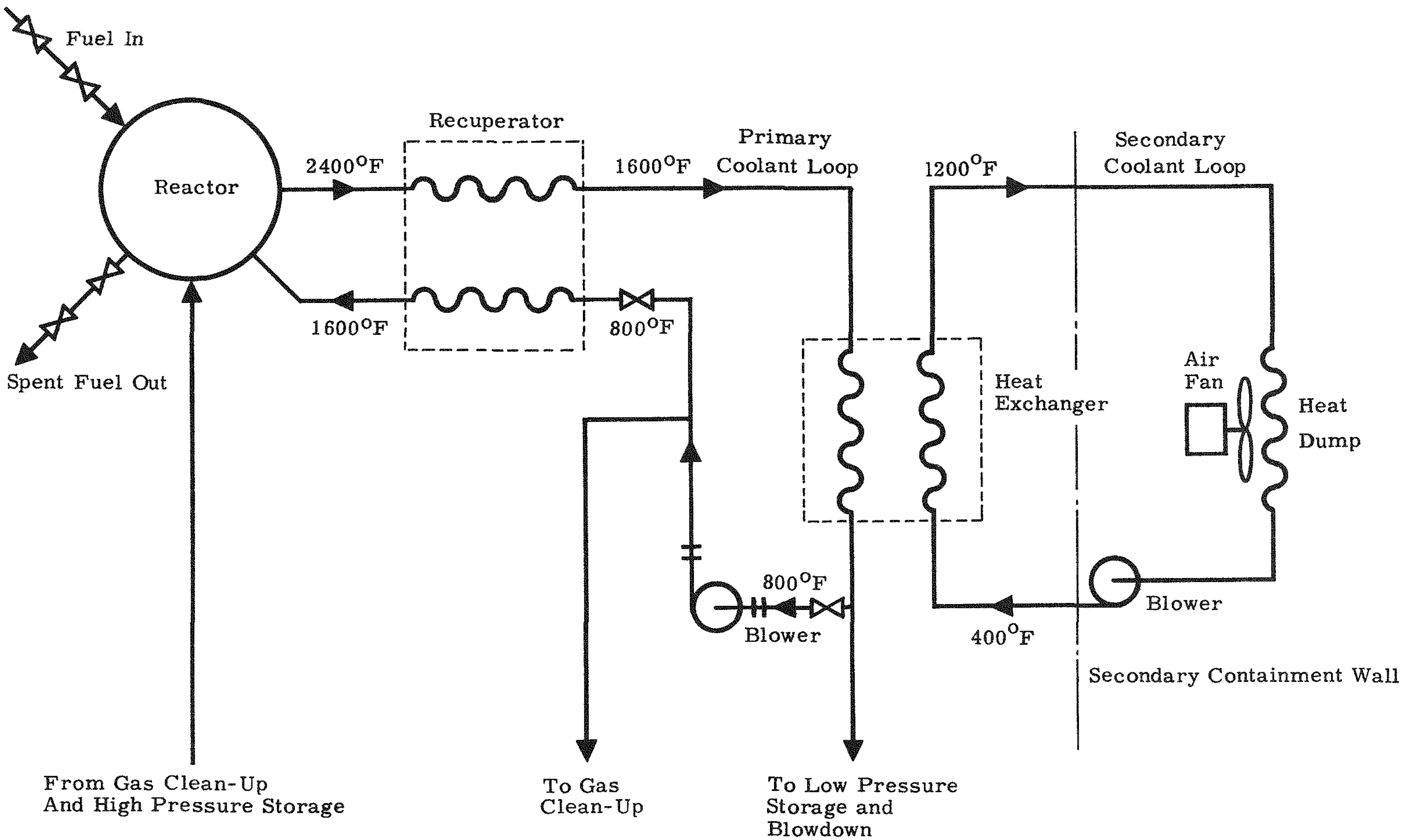

Fig. 11 Schematic Flow Diagram

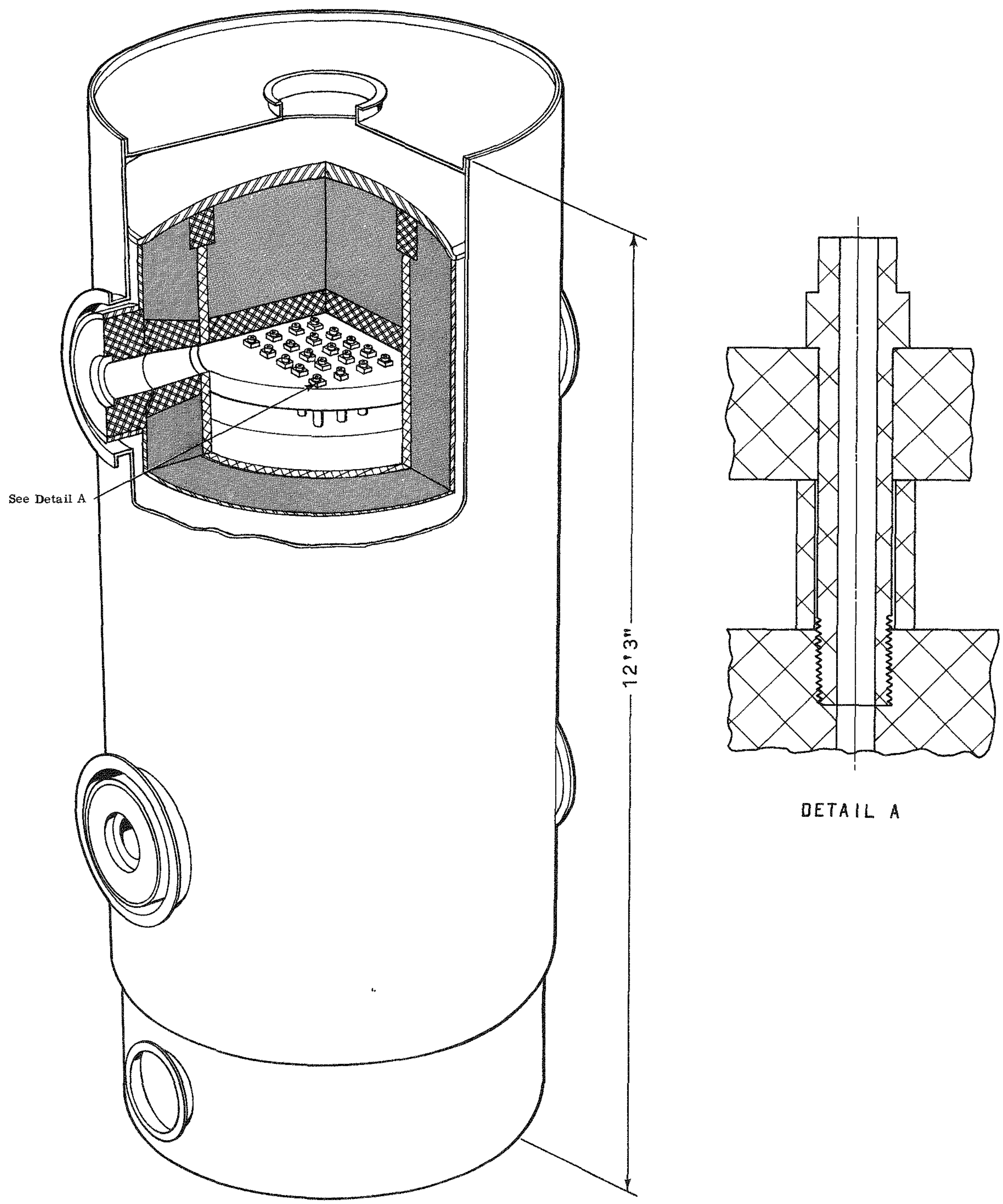

Fig. 12 Recuperator

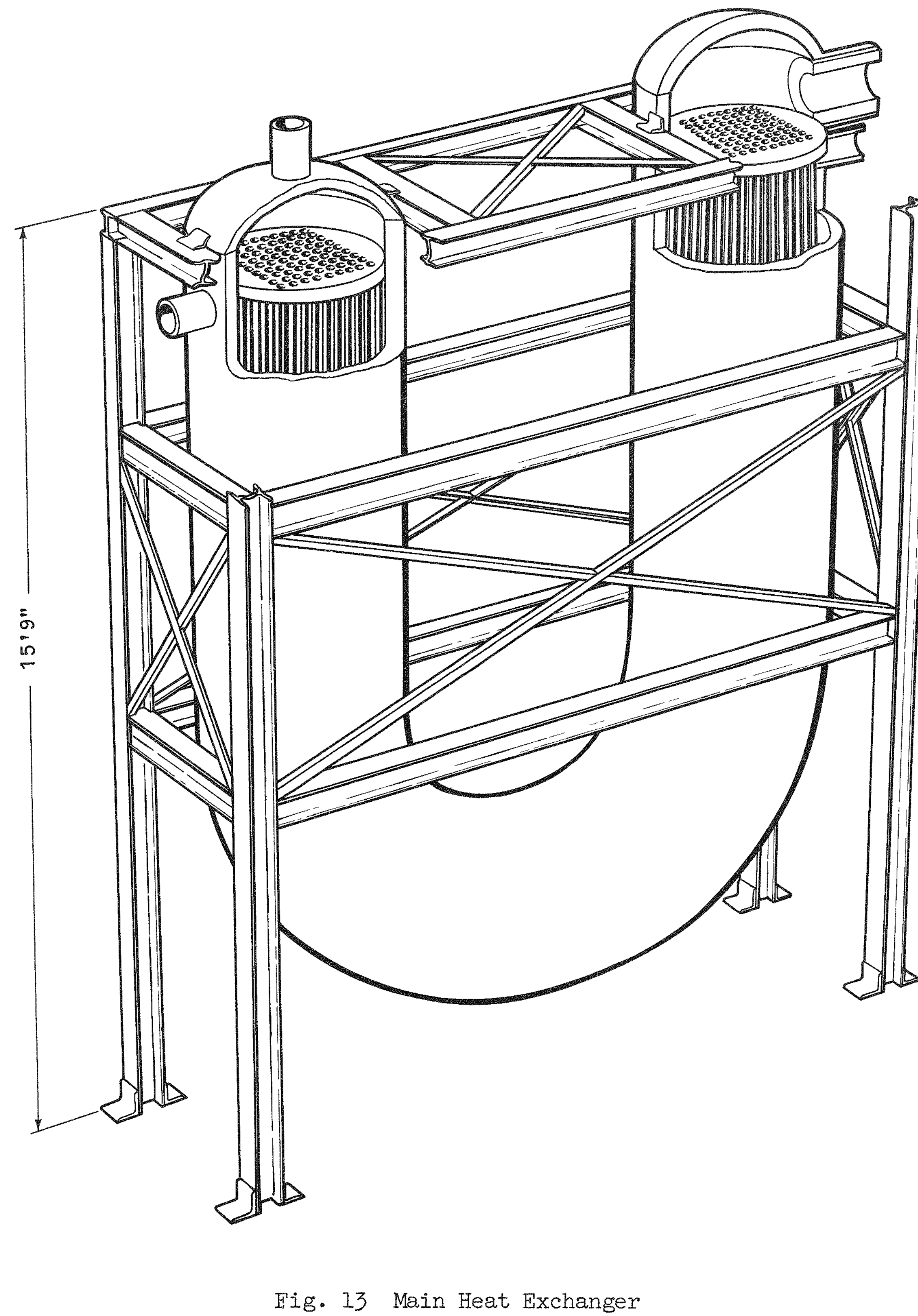

Fig. 13 Main Heat Exchanger

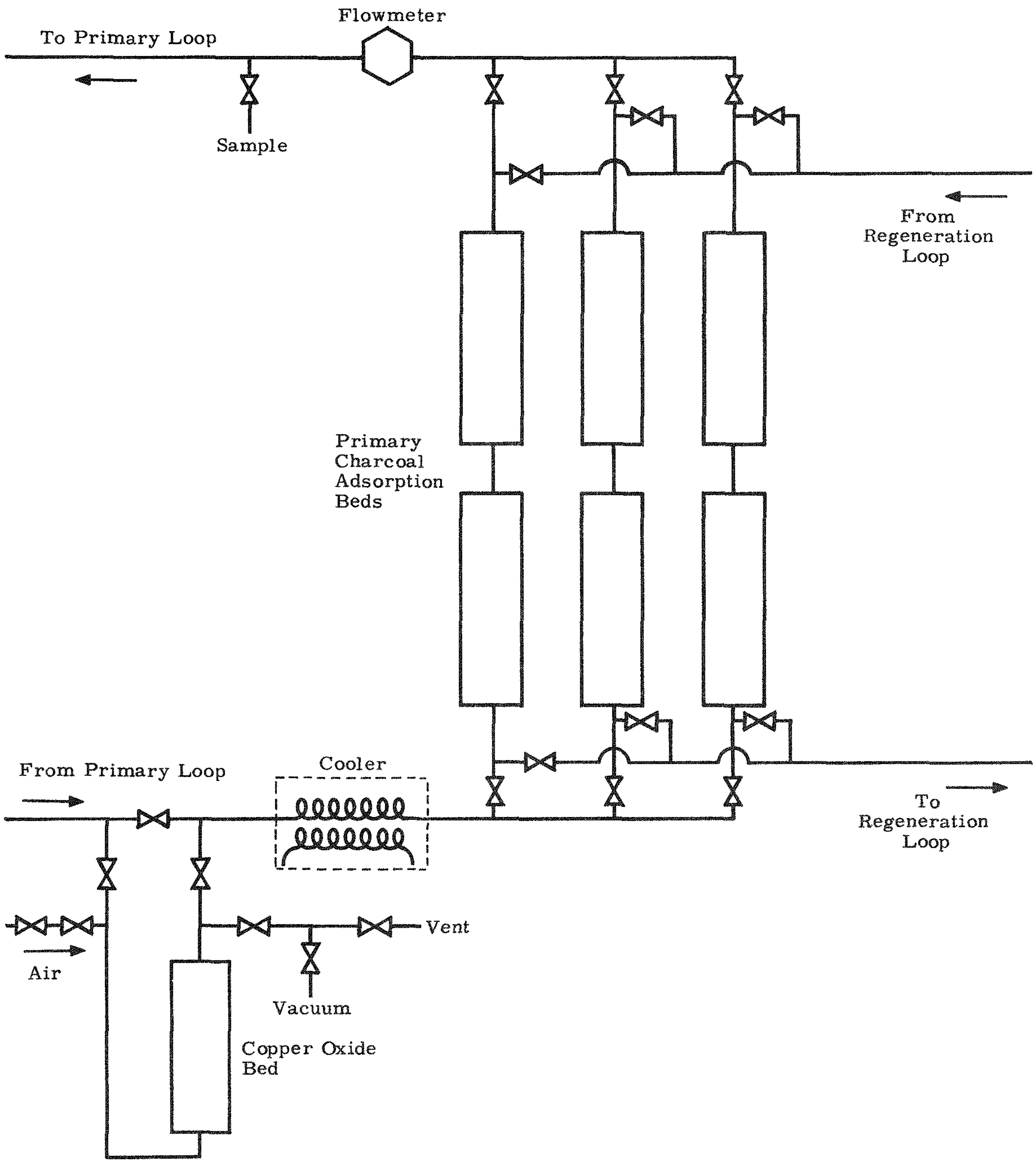

Fig. 14 Gas Purifier Schematic

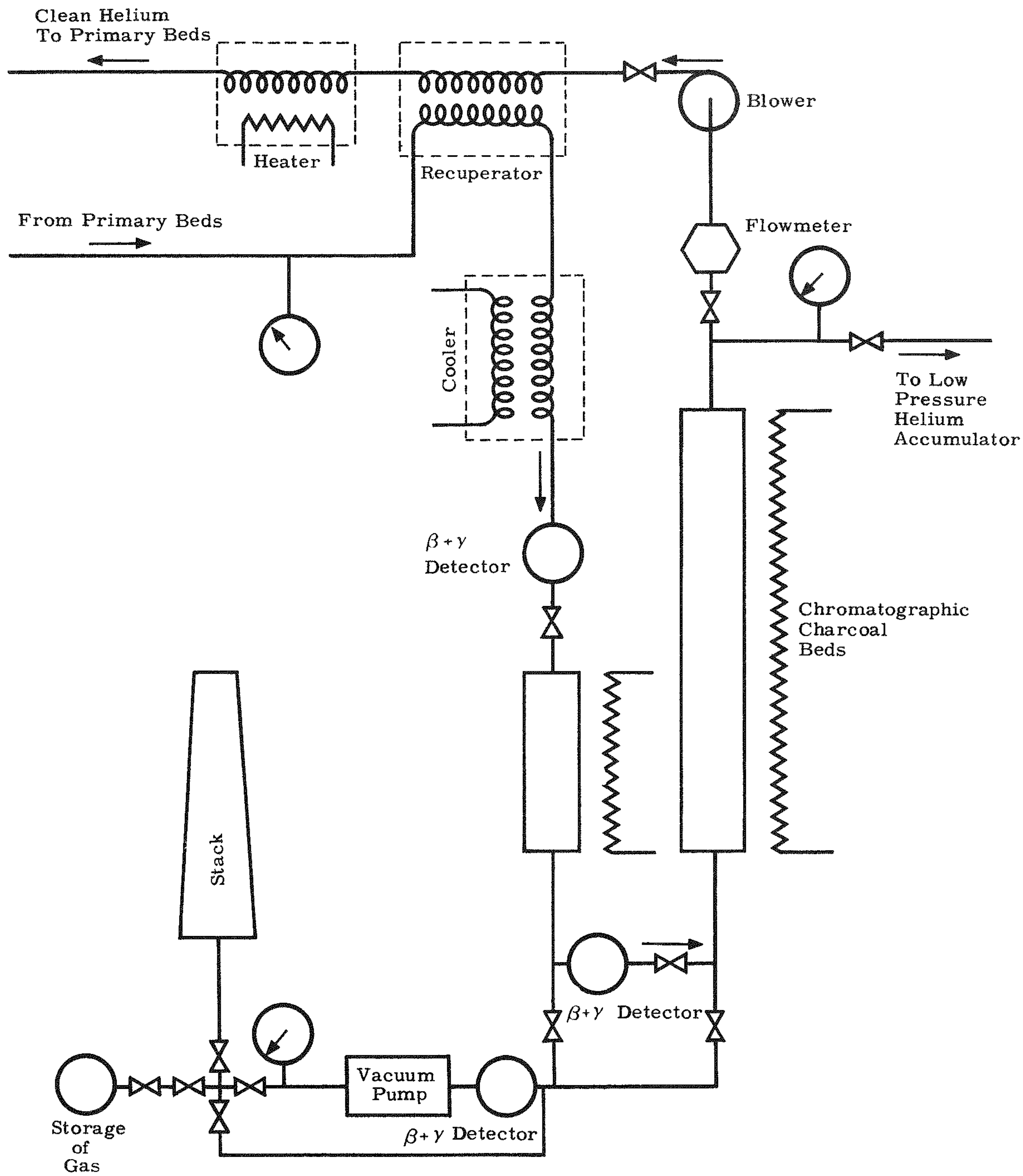

Fig. 15 Regeneration Cycle Schematic

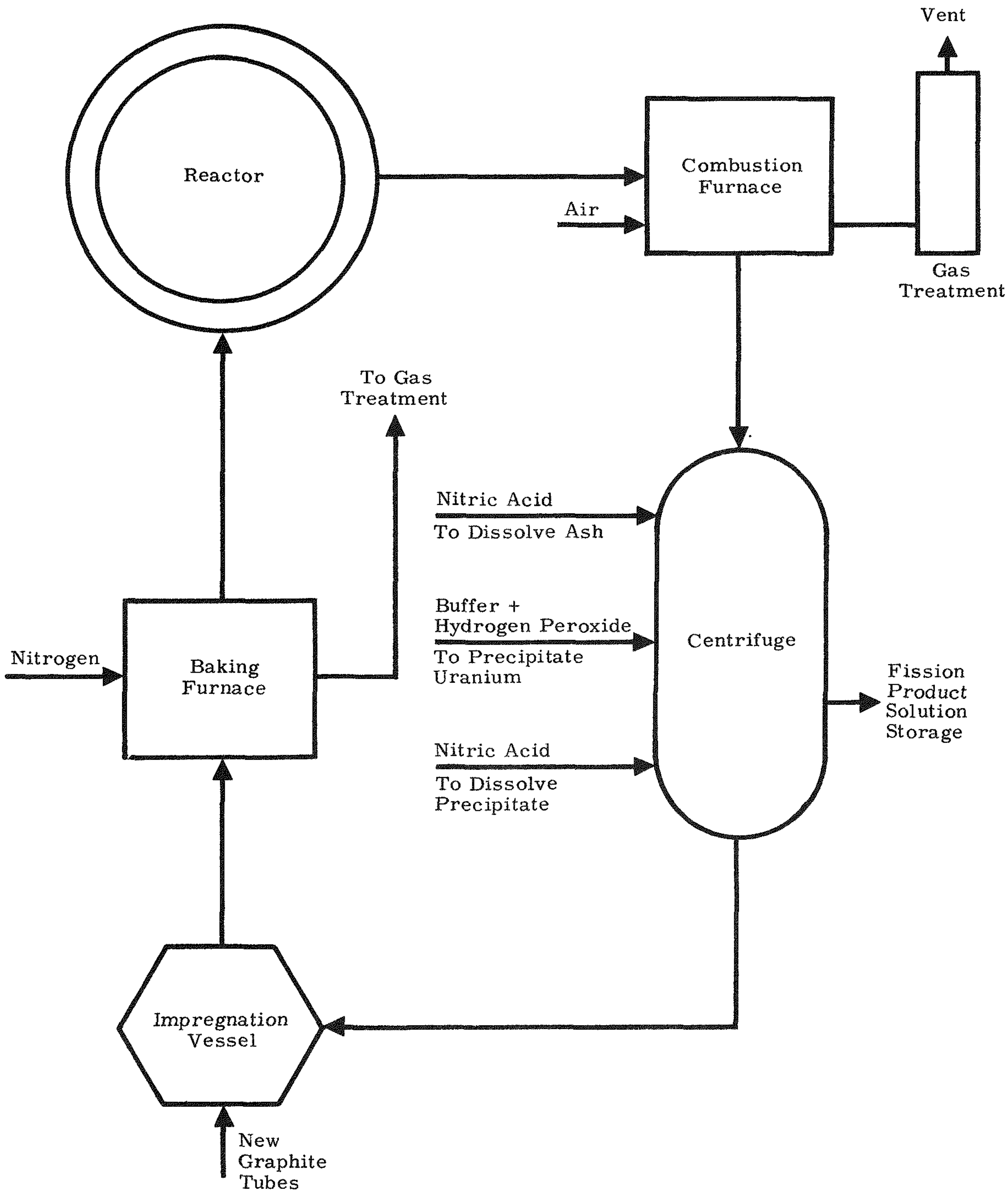

Fig. 16 Fuel Processing Schematic

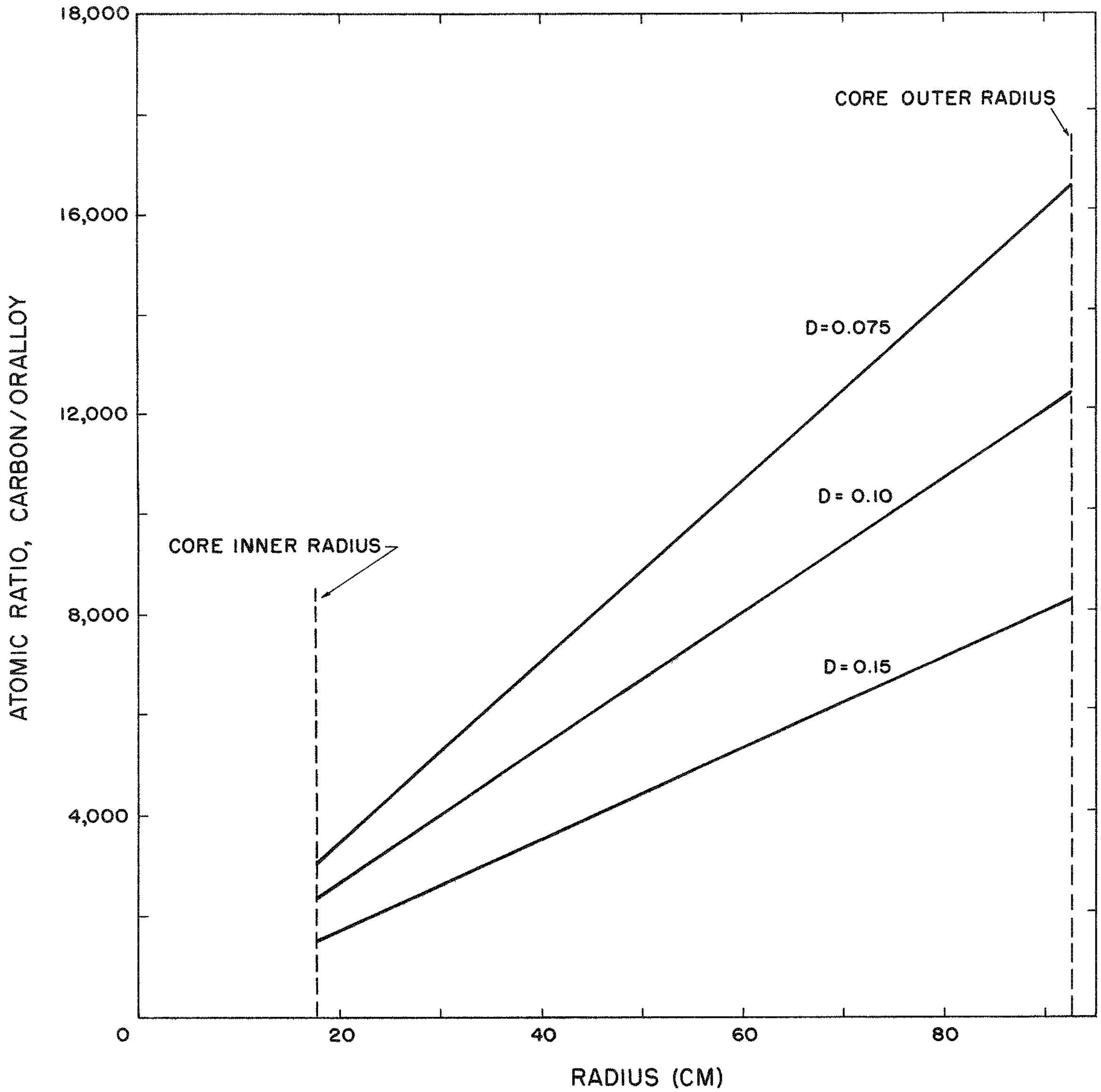

Fig. 17 Atomic Ratio of Carbon to Fuel in Core (D = grams per cubic centimeter of oralloy in fuel elements)

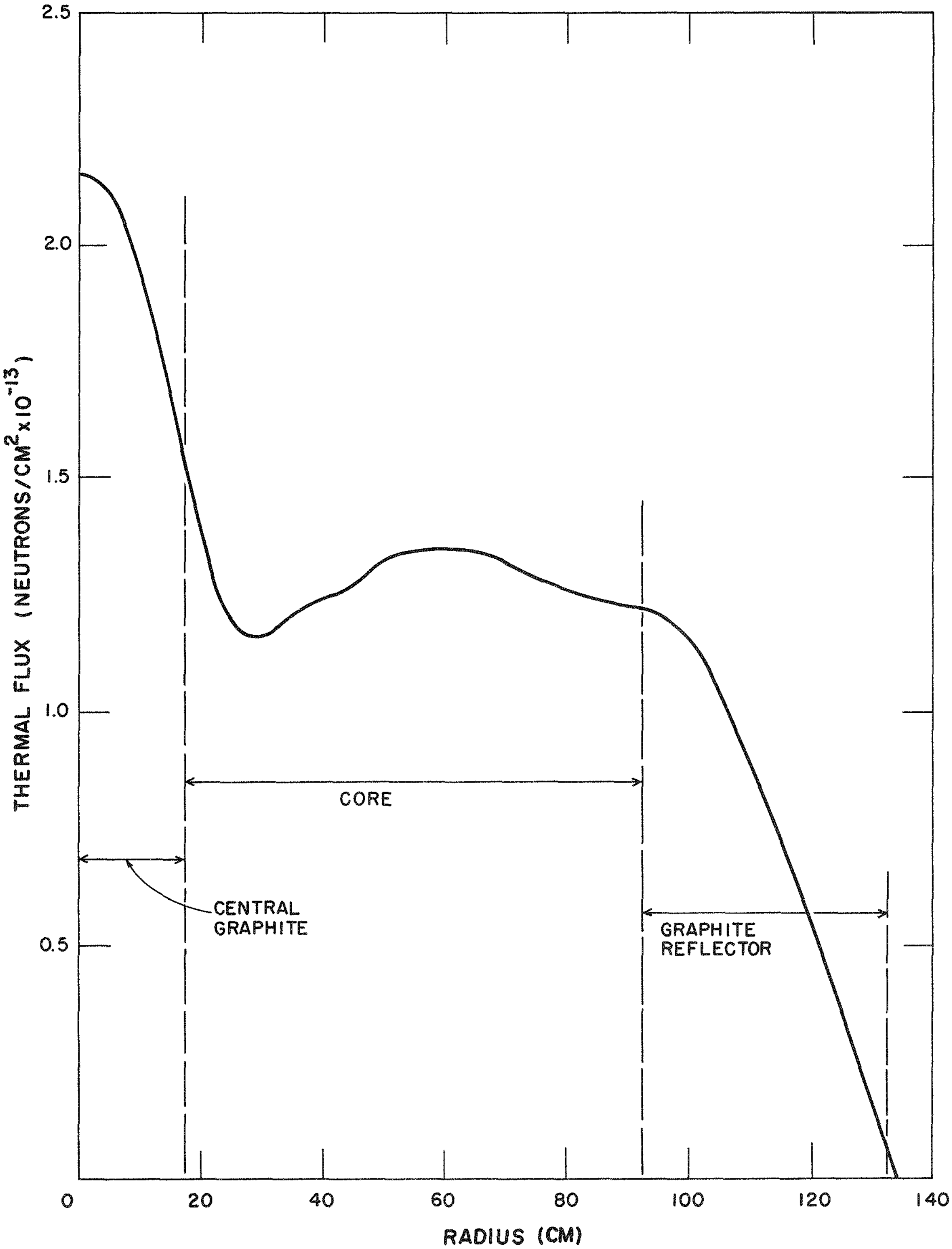

Fig. 18 Thermal Flux Profile

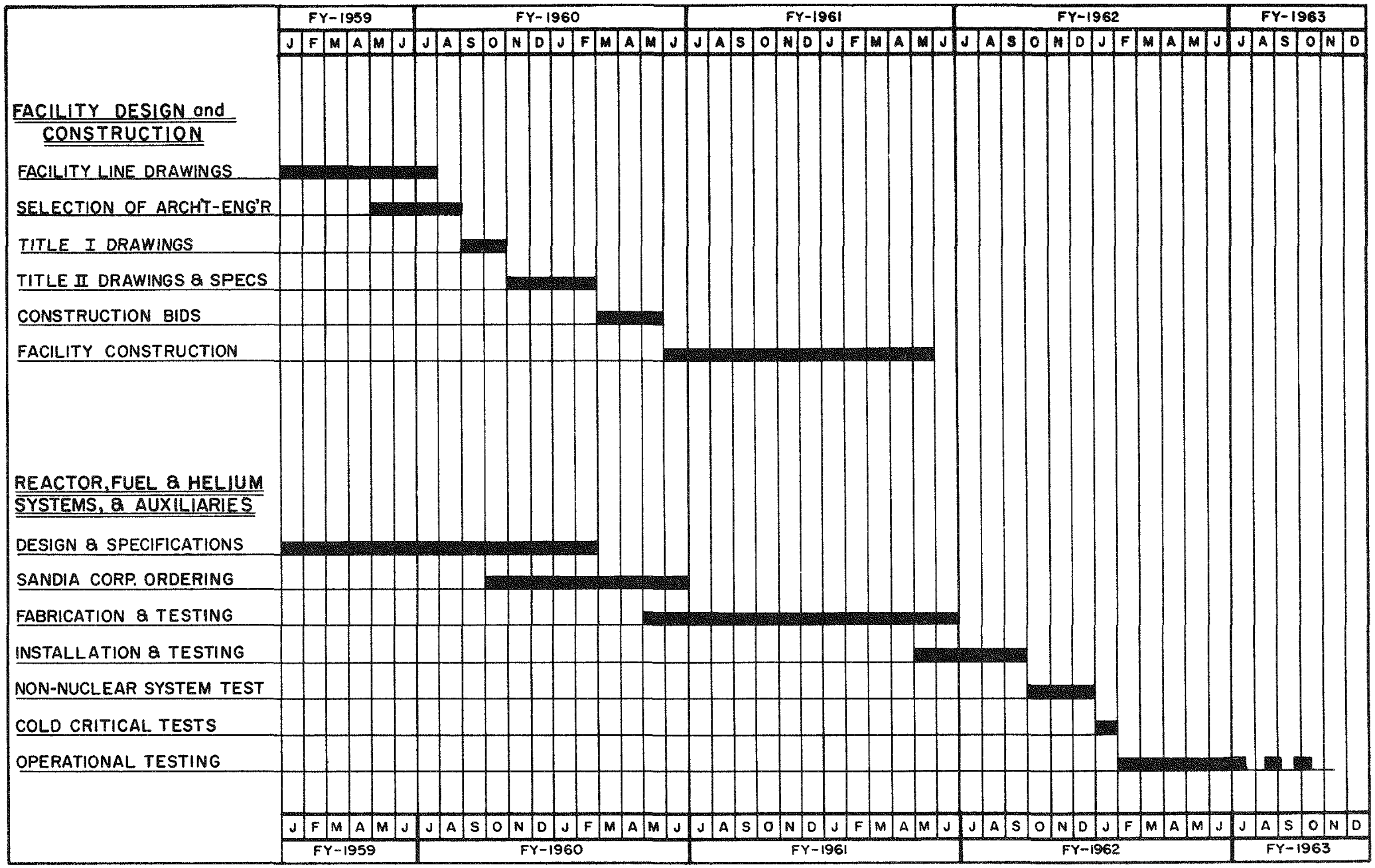

Fig. 19 Time Schedule