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(INFORMAL)

Nuclear Materials & Propulsion Operation

INTRODUCTION TO NUCLEAR PROPULSION

Lecture 1 - INTRODUCTION AND BACKGROUND

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Lecture 1 - INTRODUCTION AND BACKGROUND

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CONTENTS

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INTRODUCTION TO NUCLEAR PROPULSION
LECTURE 1 INTRODUCTION AND BACKGROUND

The participants in this lecture series are concerned with the application of nuclear rockets to space vehicles. From the viewpoint of the hardware involved, the propulsion machinery used in a nuclear rocket differs very little from that of a chemical rocket except in the nature of the heat source. Nozzles, pumps, controls, propellant storage and handling equipment, etc. differ only in secondary characteristics, if at all, from those used in chemical systems. It will be assumed in this lecture series that the participants are already familiar with the propulsion machinery common to chemical and nuclear rockets or have access to such information. Therefore, the lectures will concentrate on those aspects of nuclear propulsion systems which differ from chemical systems. This means that we will be concerned mostly with the nuclear reactor and with the environmental problems caused by the presence of nuclear radiation. The specific topics to be covered in the lectures are listed below.

1. Introduction
2. Basic Physics
3. Reactor Physics
4. Reactor Physics
5. Shield Physics
6. Shield Physics
7. System Design
8. System Design
9. Thermal Design
10. Thermal Design
11. Materials
12. Materials
13. Mechanical Design
14. Mechanical Design
15. Control
16. Control
17. Operation
18. Operation
19. Radiation Effects
20. Nuclear Space Systems
21. Nuclear Rocket

1. HISTORICAL SUMMARY OF GAS COOLED REACTOR DEVELOPMENT

With the exception of nuclear reactors, present day propulsion machinery is the outgrowth of a relatively long period of development. This history is fairly familiar to most of us and in any event is too long for us to recount at this time. However, the history of nuclear propulsion reactor development is still so short that it is feasible to summarize it virtually in its entirety before we proceed further. Tracing this historical development will enable us to become familiar with what a nuclear reactor looks like. In addition, it will help us identify the types of problems which one encounters in reactor development.

Since the nuclear rocket reactor is one in which a gas, hydrogen, is heated by passage through the reactor, we will restrict our historical summary to so called "direct open cycle-gas cooled propulsion reactors". The words "direct open cycle" signify that the working fluid is heated by direct contact with the surface of the reactor fuel elements, and is exhausted to the atmosphere or to space as the propellant rather than recirculated in a closed loop. The use of the words "gas-cooled reactor" is a holdover from the early days of reactor development when the concern was to get rid of reactor heat rather than to utilize it. For example, there are direct open cycle gas cooled reactors in existence at Oak Ridge and Brookhaven National Laboratories which operate at low temperatures and which are used primarily as sources of neutrons for research purposes. A better expression for our purposes would be "gas heating reactor". However, since the words "gas cooled" have become jargon in the nuclear industry we can use it as long as we understand what it means.

Active development of direct cycle gas cooled nuclear propulsion reactors started in 1951 following feasibility studies performed primarily by the NEPA Project of the Fairchild Engine and Airplane Company. The reactor development work has been performed for three major propulsion systems, the nuclear turbojet, ramjet, and rocket. These propulsion systems are shown schematically in Figures 1.0.1, 1.0.2 and 1.0.3.

Nuclear Turbojet Reactors

Active development of the direct air cycle nuclear turbojet started in 1951 under simultaneous contracts between the General Electric Company and United States Air Force and United States Atomic Energy Commission. The nuclear turbojet program was terminated in 1961. Because of the highly developed status at the time of termination, the

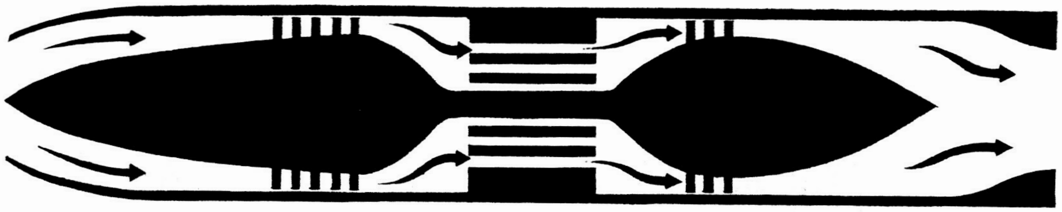

Fig. 1.0.1 - Turbojets

TURBOJET

The basic nuclear turbojet is illustrated schematically in this figure. In this arrangement the nuclear reactor is located between the compressor and turbine. Air is admitted at the forward end, compressed by a turbine driven compressor, heated in the reactor, expanded through the turbine and exhausted from the jet nozzle, thus providing forward thrust. Thrust augmentation for takeoff may be obtained by burning chemical fuel in an afterburner. A chemical interburner may also be utilized ahead of the turbine and in series with the reactor if it is considered desirable to restrict airport operation to chemical fuel utilization.

The applications for which the nuclear turbojet is most useful are those involving extended flight at sea level and intermediate altitudes and at high subsonic and intermediate supersonic Mach numbers.

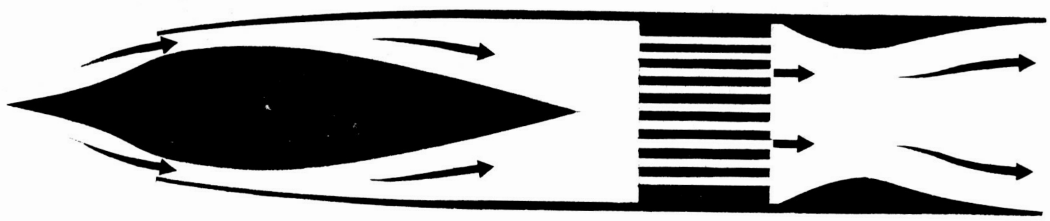

Fig. 1.0.2 - Ramjet

RAMJET

The nuclear ramjet is shown schematically in this figure. The ramjet operates on a thermodynamic cycle which is basically the same as that of the turbojet. Air is admitted at the forward end, compressed by diffusion, heated in the reactor and expanded through the jet nozzle to provide forward thrust. Since sufficient compression is provided by inlet diffusion at high flight speeds, the requirement for rotating turbomachinery is eliminated.

The nuclear ramjet is most useful for applications requiring long endurance at high supersonic flight speeds in the atmosphere.

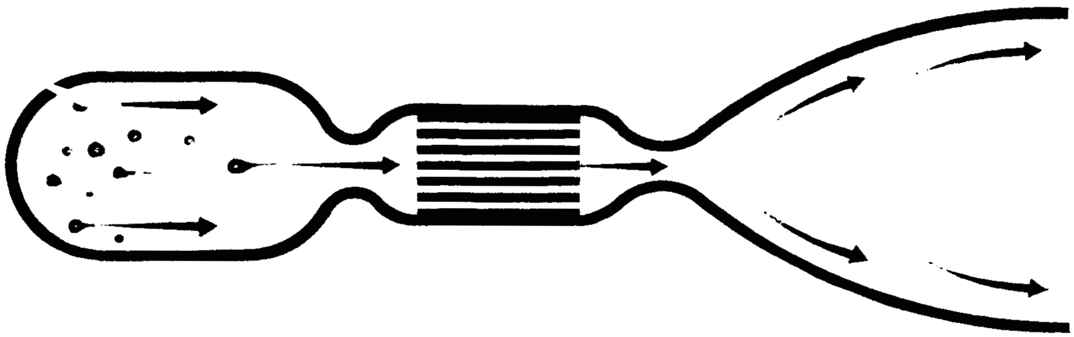

Fig. 1.0.3 - Rocket

ROCKET

A typical nuclear rocket engine is depicted schematically in this figure. Propulsion in the nuclear rocket is attained by heating a propellant in the reactor and discharging the expanding fluid through a jet nozzle.

A noteworthy advantage of the nuclear rocket over its chemical counterpart is the very high specific impulse which is obtainable in the nuclear system. The reason for this is embodied in the fact that the propellant for the nuclear rocket may be exclusively hydrogen which has a low molecular weight, whereas the chemical system requires high molecular weight oxidizer.

ANP direct cycle turbojet reactors provide our richest source of gas cooled reactor development technology.

In addition to the direct cycle, an indirect liquid metal cycle reactor for a nuclear turbojet was being developed in the ANP program under contract to the Pratt and Whitney Aircraft Company. Since termination of the ANP program the liquid metal reactor development work has been redirected toward application to a space electric power supply.

Nuclear Ramjet Reactors

Active development of the nuclear ramjet started in the mid 1950's. This program is still active under auspices of the USAF and USAEC. The reactor development work is being performed by the University of California Lawrence Radiation Laboratory. The ramjet reactor is very similar to the nuclear turbojet reactor which was under development at the time of the ANP program. However, since its development has continued after termination of the nuclear turbojet development, it can be expected to provide further insight into high temperature reactor development.

Nuclear Rocket Reactors

Active development of the nuclear rocket started in the mid 1950's. This work is being accomplished under the auspices of the USAEC and NASA. The reactor development work has been performed primarily by Los Alamos Laboratory of the USAEC with the recent participation of Aerojet-Westinghouse. The problems in developing the nuclear rocket differ only in degree from those of the ANP reactors. The big difference is that the ANP reactors were to operate for long periods of time at fairly high temperatures with an oxidizing propellant, whereas a rocket reactor operates for a relatively short period of time at very high temperatures with a non-oxidizing propellant. As we will see later, these two requirements have many features in common.

These developments will be summarized in more detail in the following.

1.1 NUCLEAR TURBOJET REACTOR DEVELOPMENT

1.1.1 Summary

The first operation of an aircraft engine on nuclear-power was achieved on January 31, 1956, using an experimental direct-air-cycle reactor and a modified General Electric J47 turbojet engine. This was followed by a series of additional reactor operations using improved reactor designs and materials. Concurrently, high performance turbomachinery (X211) was under development which could be used for a variety of nuclear propulsion system applications at both subsonic and supersonic speeds.

A series of power plants was designed combining the turbomachinery and the contunually improved reactor materials and components. The power plant under development at program termination, the XNJ140E nuclear turbojet (Figure 1.1.1) was designed in accordance with Department of Defense guidance for a nuclear propulsion system capable of propelling a Convair NX2 (Figure 1.1.2) or equivalent aircraft at high subsonic speeds for 1000 hours before refueling. An aircraft with this capability was believed to be best suited for an airborne alert and counterstrike mission in which it would remain airborne for periods of five days at a time, carrying ballistic missiles with nuclear warheads for air launch from outside the target area. Growth versions of the XNJ140 power plant were in preliminary design for use at supersonic speeds.

The Aircraft Nuclear Propulsion program was terminated following the President's annual budget message to Congress on March 28, 1961, recommending omission of funds for program continuation. The program termination was based primarily on the fact that there was not considered to be a specific military requirement for a manned aircraft with the characteristics of the subsonic, long endurance system that was under development. The work on alternative subsonic missions and on growth versions for supersonic operation was simultaneously discontinued. The work on the unmanned nuclear ramjet and nuclear rocket propulsion continued in the national laboratories.

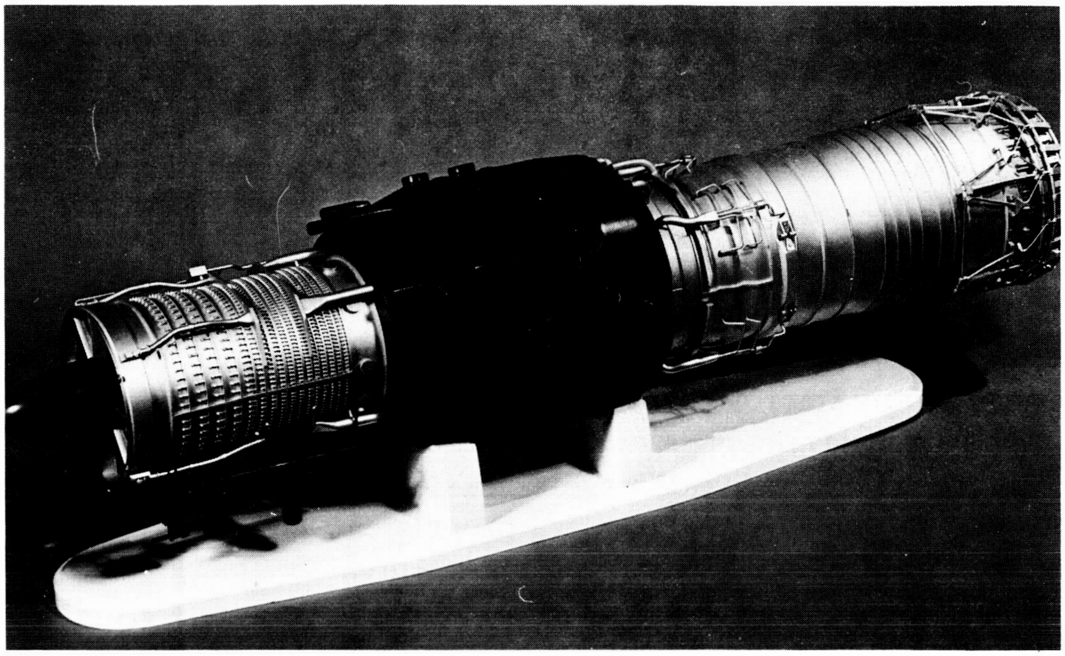

Fig. 1.1.1 -Model of the XNJ140E nuclear turbojet engine

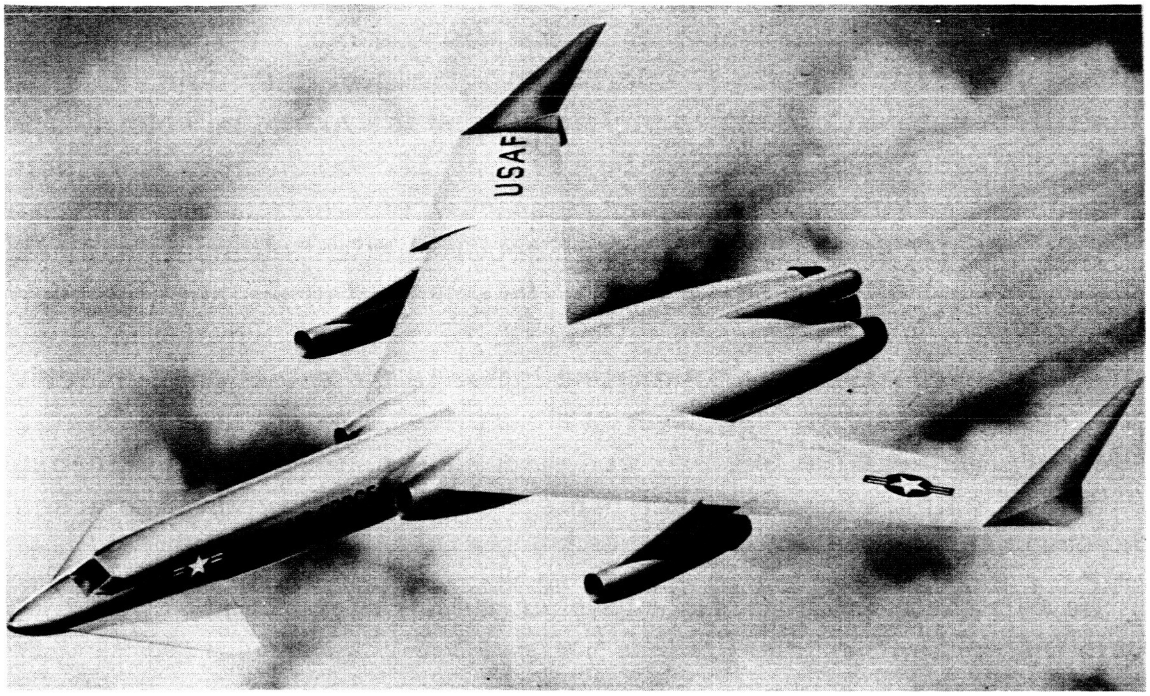

Fig. 1.1.2 - Convair NX2

1.1.2 NEPA Reactors and Shields

Reactor Materials and Design

Early NEPA studies were concentrated on reactors using uranium-bearing ceramic materials, specifically graphite, beryllium oxide, and beryllium carbide. The ceramic fuel elements were to be held in a "mosaic" pattern by means of an external structure. These ceramics were selected because, in addition to their high temperature potential, they were good neutron moderators since they were relatively light elements and had low neutron absorption cross sections. The supply of uranium at the beginning of the NEPA studies was so limited that the achievement of a minimum uranium inventory was a dominating factor in the selection of materials for nuclear reactors. The neutron absorption of most potential high temperature metallic fuel element and structural materials was rather high. The moderating ceramics were preferred in order to minimize uranium investment.

Despite their apparent suitability, a number of problems were foreseen with the ceramic reactors.

1. The high power densities would produce high thermal stresses within the fuel elements and the possibility of breakage, particularly under transient conditions.
2. Extensive development would be required to protect the fuel elements against water-vapor corrosion and fission product leakage.
3. It would be difficult to constrain a matrix of ceramic elements, while allowing for thermal expansion and aerodynamic and maneuvering loads, without the use of a metallic structure.
4. Reactors using beryllium or carbon moderators have a high nuclear sensitivity to even a small amount of foreign materials.

For these reasons, although the ceramic reactors were used as a basis for NEPA power plant design studies, a search continued for other suitable high temperature reactor materials and concepts.

In studying alternative reactors, the primary effort was devoted to hydrogenous systems. Hydrogen-moderated reactors have a low sensitivity to the presence of foreign materials because hydrogen itself has a relatively high neutron absorption cross section. Nevertheless, hydrogen is an excellent moderator because of the large energy degradation in each neutron collision. However, the temperature capability of hydrogenous materials available at the time was relatively low. NEPA's solution to this problem was a new reactor concept which consisted of a cylindrical

water vessel penetrated by many air passages, each of which contained air-cooled, uranium-bearing fuel elements. Water, or a liquid hydrocarbon, filled the interstices between the air passages and served both as moderator and structural coolant. A thin layer of insulation between the fuel elements and the walls of the air passages minimized heat transmission to the water. The small amount of heat lost to the water was removed by circulation to an external radiator. Although still a "heat transfer" rather than an "internal combustion" system, this reactor concept was similar in one respect to the automobile engine - even though the temperature of the working fluid would be high, the structural materials could be kept at relatively low temperatures.

This concept offered the prospect of achieving early development of a reactor which could produce high air temperatures while using readily available structural materials. Thermal expansion and other problems could be localized within each individual fuel cartridge and air passage. Furthermore, the hydrogenous moderator made possible the use of either metallic fuel elements or those ceramic elements with good mechanical properties but with less attractive neutron moderating properties.

Therefore, NEPA concluded that if an early flight program were to be adopted with a direct air cycle propulsion system, the hydrogenous moderated, air-cooled reactor could be developed most rapidly. It was recognized, however, that at high flight speeds, heat rejection from a liquid moderator would be difficult because of the high ram-air temperature. For high speed nuclear flight, higher temperature hydrogenous moderator materials or ceramics would be required.

Shielding

Early shielding studies were directed toward the use of "unit" shielding, placed only around the reactor. The shield could be thinner on the sides and rear, because the radiation from these regions could reach the crew only by scattering from the air or from the aircraft fuselage. It was soon recognized, however, that a lower total shield weight could be achieved by dividing the shield between the crew compartment and the reactor. The combined shield thickness directly in line between the reactor and crew was about the same in either arrangement. However, shielding on the side of the crew compartment was more effective than an equivalent thickness on the side of the reactor because the scattering process reduced the energy of the radiation reaching the crew compartment. Hence, a thinner shield could be used, resulting in less weight.

The divided shield concept was recommended by NEPA. The optimum placement of shielding required further study since this is determined both by the radiation tolerance and induced activities in the airframe as well as by biological considerations.

1.1.3 P-1 Reactor

The NEPA studies had indicated that the successful technical development of nuclear power plants for aircraft propulsion was feasible and that there were useful applications for such power plants. However, there was still considerable uncertainty as to whether the utilization and maintenance of aeronautical nuclear propulsion systems was operationally practicable. The resolution of this question was considered to be essential before commitments could be made to use nuclear power in military aircraft weapons systems. Consequently, an Air Force objective was established for early ground and flight operation of a nuclear power plant in a modified conventional aircraft, the Convair X-6. If operational practicability were thus established, the data developed in the flight program would be applied to the design of high performance prototype military aircraft and propulsion systems.

Early availability rather than high performance was the dominant requirement for the nuclear propulsion system. It was considered desirable but not necessary that sufficient thrust be provided to sustain flight at low speeds and altitudes without chemical assistance. A power plant designated the P-1 was designed to meet this objective.

P-1 Materials and Design Selection

Because of the early flight requirement, a prime requisite in the design of the P-1 power plant was to make maximum use of previously developed materials. Using the NEPA reactor concept, in which the water moderator cooled the reactor structure, permitted the use of readily available aluminum as the structural material. The performance requirements were such that the temperature levels achievable with uranium-bearing, stainless-steel fuel elements seemed adequate. Because the early development and production of such a fuel element appeared more likely than the development of suitable ceramic elements, and since the question of uranium availability was not as critical as it had been, the stainless steel fuel element approach was adopted.

The reactor design selected for the P-1 consisted of large-diameter, concentric annular rings, in which the water moderator was alternated with air passages containing the fuel elements. This was a symmetrical configuration with a uniform composition at any specific radius. Gross radial power could be flattened by radial variation of the moderator-to-fuel ratio. The selection of this configuration was based largely on nuclear considerations since theoretical methods for the nuclear analysis of highly heterogeneous, compact, gas-cooled reactors were in the early

stages of development, and there was little experimental data on which to base an empirical design. Mechanical considerations were not as critical because the reactor structure would operate at low temperatures.

Description of P-1 Reactor

The P-1 power plant used four turbojet engines powered by a single nuclear reactor. This arrangement was chosen because a single, large, reactor-shield assembly would weigh less than four smaller assemblies of the same total power and airflow. The reactor was to be mounted within the X-6 aircraft, with the engines extending below the fuselage. General Electric J47 turbojet engines, modified by replacing the combustion section with a compressor outlet scroll and a turbine inlet scroll, were to be used in the ground test.

The reactor, is illustrated in Figure 1.1.3. There were nine annular air passages containing fuel elements, nine 1-inch rings of moderator water, and a 1-1/2-inch central water tube. The diameter of the air passages was increased with distance from the center of the core. For radial power flattening, the fuel elements varied in width from approximately 3 inches near the core center to about 1 inch at the outermost ring. The fuel element was fabricated in a honeycomb structure (Figure 1.1.4). The fuel, uranium oxide dispersed in 310 stainless steel, was "sandwiched" between a cladding of unfueled stainless steel (similar to the aluminum fuel stock used in the fuel elements for the Materials Testing Reactor). The reflector consisted of two concentric stainless steel cylinders.

The shield was primarily water, supplemented by a lead and steel gamma shield at the forward end. The shield configuration is shown in Figure 1.1.5.

Final Status of P-1 Development

The early flight objective was withdrawn in May 1953, and the P-1 power plant development was discontinued. The basis for this decision was that early flight demonstration with a system not fitting a specific military requirement was not warranted. When the P-1 program was discontinued, the reactor and propulsion-system components were under final design and development. The fuel element development was proceeding satisfactorily. Significant advances had been made in the techniques of analyzing heterogeneous, hydrogen-moderated, gas-cooled reactors. Exploratory critical experiments had been performed. A full-scale shield mockup had been built and was later tested in the Oak Ridge Tower

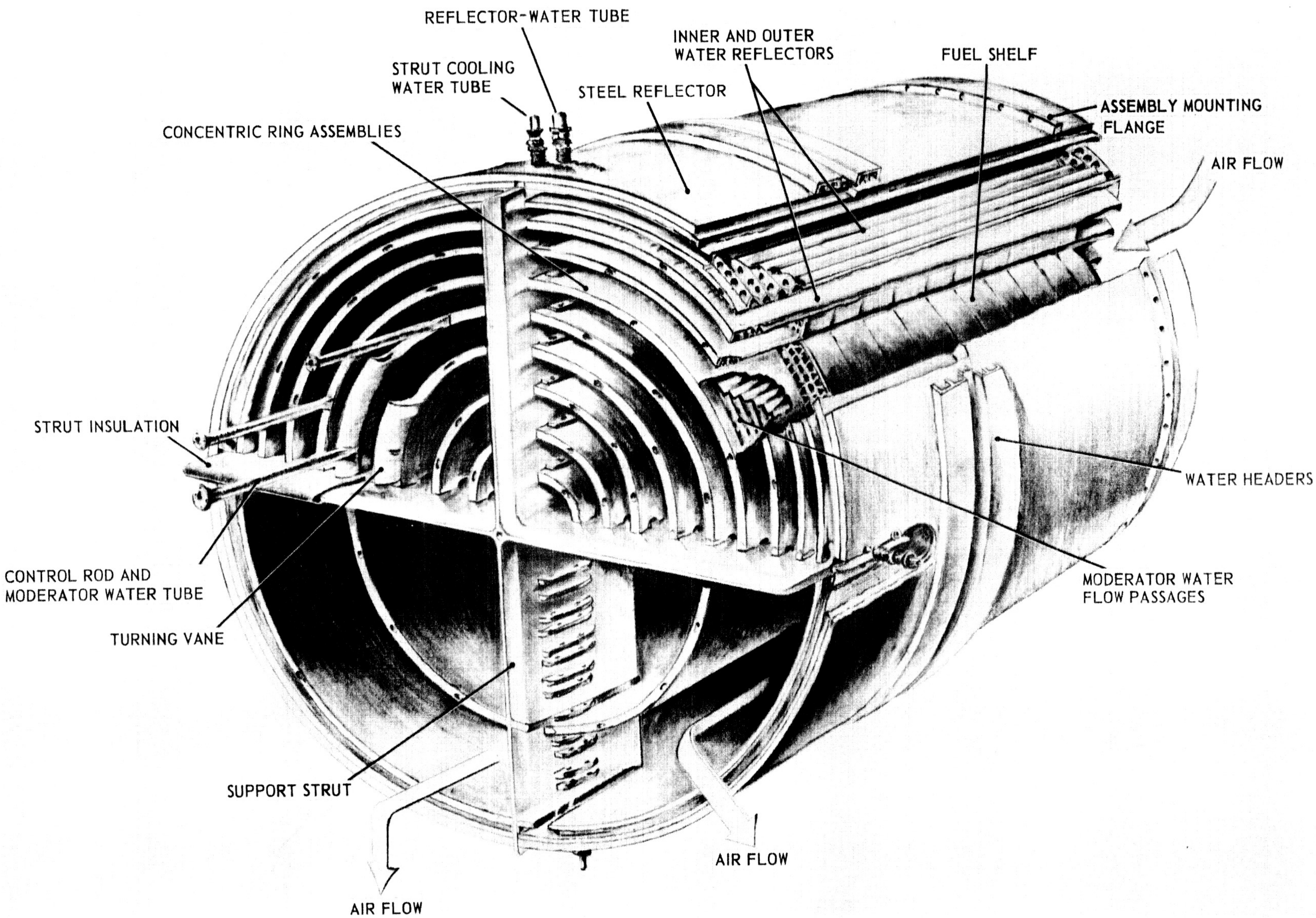

Fig. 1.1.3 - P-1 reactor structural arrangement

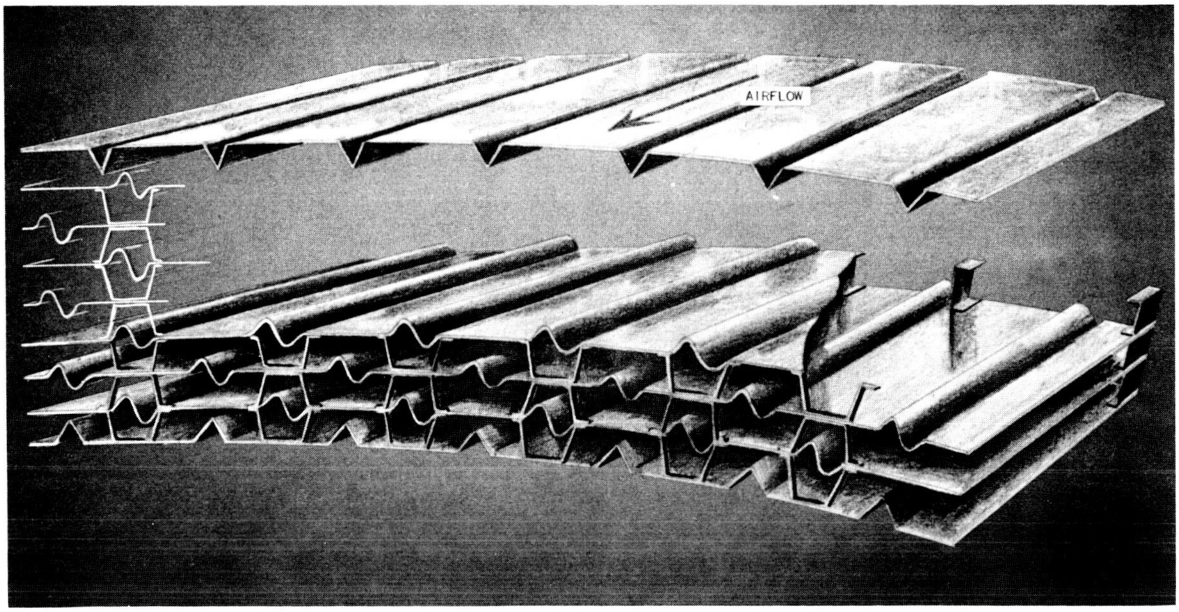

Fig. 1.1.4 -P-1 type of fuel element

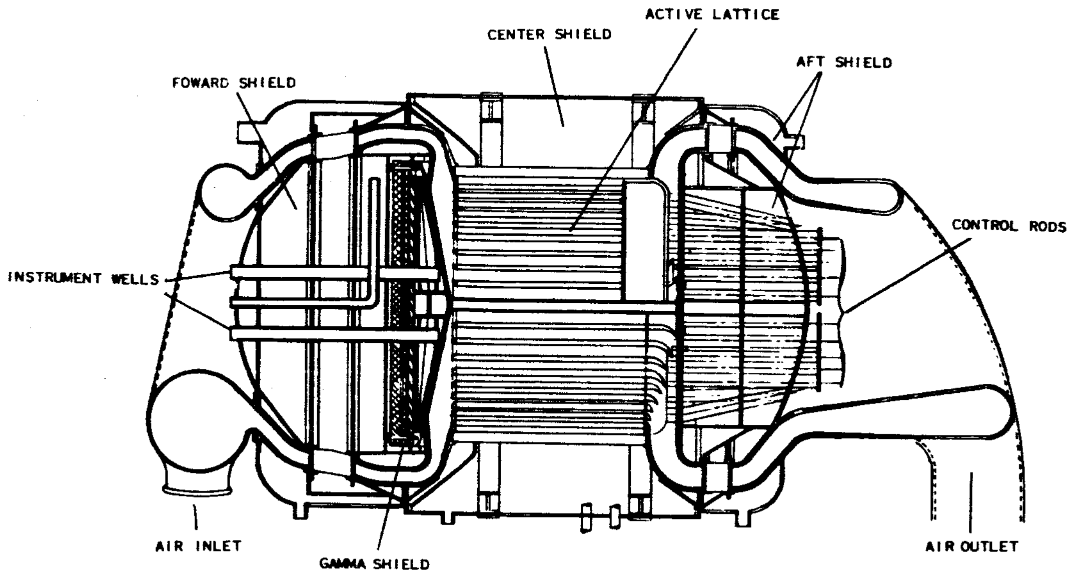

Fig. 1.1.5 -P-1 reactor-shield assembly

Shielding facility. Control rod actuators and other controls components had been developed and were later used in HTRE-1.

A power plant, designated the Propulsion Unit Test (PUT), had been constructed in the P-1 configuration, using a single chemical combustion chamber to simulate the nuclear reactor. The PUT demonstrated that several turbojet engines could be operated stably from a common heat source. The modified turbomachinery required for the P-1 ground test had been completed and tested successfully in the PUT operation; these engines were also used in HTRE-1.

Concurrent with the development of the stainless steel fuel elements, development work had been performed on ceramic fuel elements and on metallic fuel elements of higher operating temperature capabilities.

1.1.4 HTRE 1 Reactor

After withdrawal of the P-1 power plant objective, General Electric's efforts were redirected toward applied research and development applicable to a broad spectrum of potentially useful nuclear propulsion systems. The objective of the applied research phase was the development of improved materials and methods of engineering physics. It was decided that the most effective way of providing direction to component and design development, in the absence of a specific power plant objective, would be to perform one or more preliminary nuclear reactor experiments using reactor types with potential application to aircraft propulsion systems. These operations, known as the Heat Transfer Reactor Experiments, were used as development tools from 1953 through the ANP termination in 1961. The first operation of a turbojet engine exclusively on nuclear power occurred in January 1956, in Heat Transfer Reactor Experiment No. 1 (HTRE-1). This was followed by HTRE-2 and HTRE-3 using more advanced reactor components. A fourth Heat Transfer Reactor Experiment (HTRE-4) was studied but set aside, in favor or proceeding directly to a prototype propulsion system.

Although no new objective had been specified to succeed the P-1 power plant, several potential applications had been identified for nuclear propulsion systems. A number of configurations were considered. Single- or dual-engine systems were favored over the four-engine P-1 configuration, primarily because of the easier handling of smaller power packages and the added versatility for application to aircraft with different power requirements. The dual-engine configuration shown in Figure 1.1.6 was representative of the designs studied. This system demonstrates the trend, that continued throughout the ANP program, toward increasingly closer integration of the reactor and turbomachinery. The design studies indicated a potential use for reactors incorporating materials similar to those used in the P-1 reactor but with higher performance capabilities. It was decided to develop such a reactor for test operation. The general objectives of HTRE-1 were to:

1. Demonstrate the feasibility of operating a turbojet engine on nuclear power
2. Evaluate and further develop the materials and design technology of the reactor and other system components for application to the design of prototype propulsion systems
3. Develop operating and maintenance procedures and establish the practicability of ground operation and maintenance of nuclear turbojet systems.

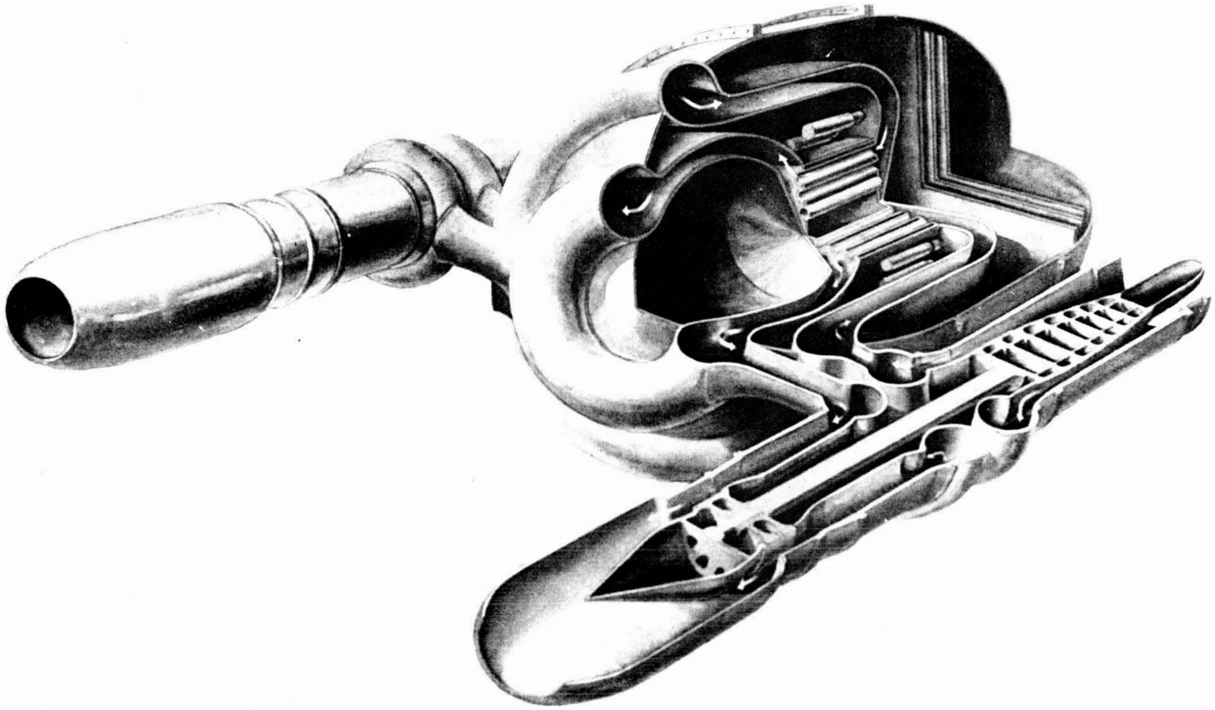

Fig. 1.1.6 - Early ANP power plant configuration

Core Test Facility

In order to provide a test vehicle for the first Heat Transfer Reactor Experiment, a Core Test Facility (CTF) was built in which various experimental reactor types could also be tested. The requirements for the CTF were established in 1953. It was completed in 1955, first used with HTRE-1 in 1956, and continued in use as the HTRE-2 test vehicle through 1961.

The CTF is shown schematically in Figure 1.1.7. The assembly consisted of two turbojet engines, a large shield tank (which was not designed to aircraft standards), and accessory equipment, all mounted on a mobile platform. The experimental reactors and shield plug were inserted as an integral unit into the shield. The entire test assembly was then delivered to the test stand by a shielded traction vehicle. After operation, it was returned to the hot shop for inspection, disassembly, maintenance, or reactor replacement.

HTRE-1 Reactor Materials and Design Selection

Although designed for higher performance than the P-1, the HTRE-1 reactor incorporated many of the P-1 reactor design features. Specifically, water was again used both as moderator and structural coolant. In an aircraft installation, it was planned to use a liquid hydrocarbon of high boiling point rather than water to facilitate waste heat rejection at high speeds.

A tubular reactor configuration was selected because (1) it appeared to have better structural characteristics than the P-1 annular ring configuration and (2) nuclear analysis methods had been developed sufficiently to take into account the greater heterogeniety of the tubular geometry.

Clad metallic fuel stock of the same type used in the P-1 was selected for the HTRE-1 fuel elements. A nickel-chromium alloy was selected in preference to stainless steel, however, because of its longer life potential at the required operating temperatures. A reactor operating life of 100 hours at full power was established as the design and development objective, with reactor exit-air temperatures in the range from 1200° to 1400°F.

Description of HTRE-1 Reactor and Test Assembly

An artist's concept of the HTRE-1 reactor is shown in Figure 1.1.8. The reactor is shown during construction in Figure 1.1.9.

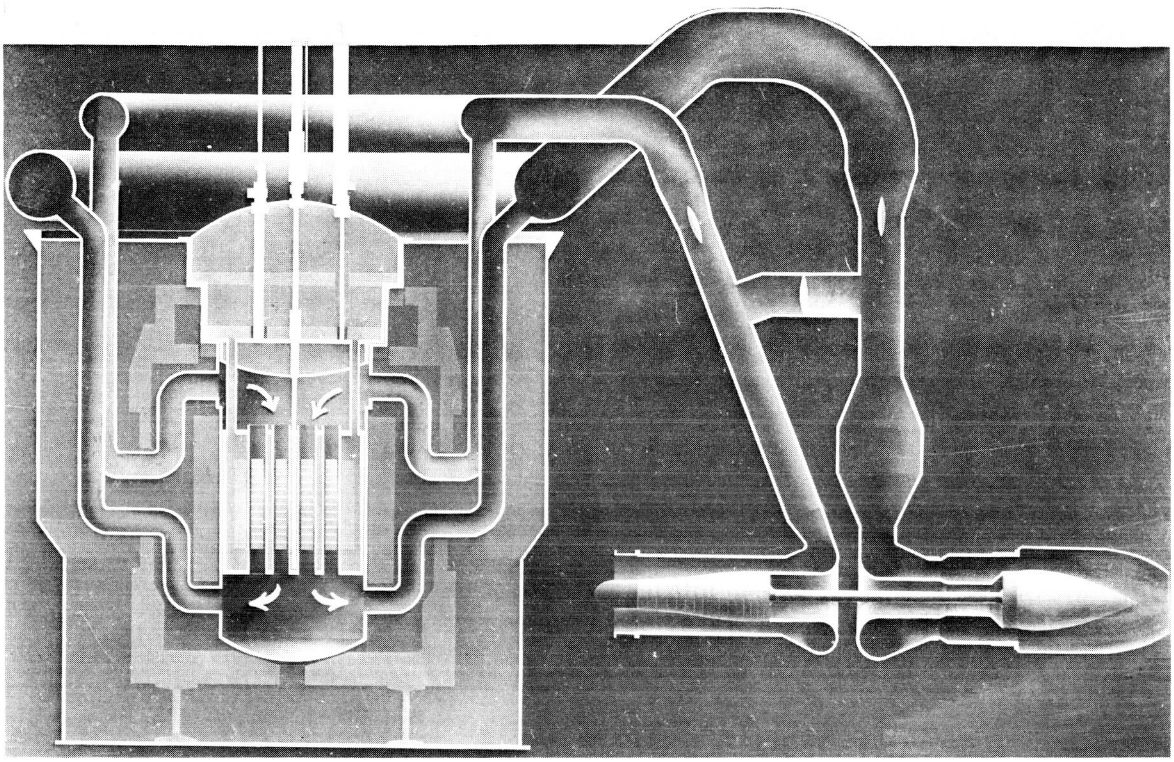

Fig. 1.1.7 - Schematic illustration of Core Test Facility with reactor

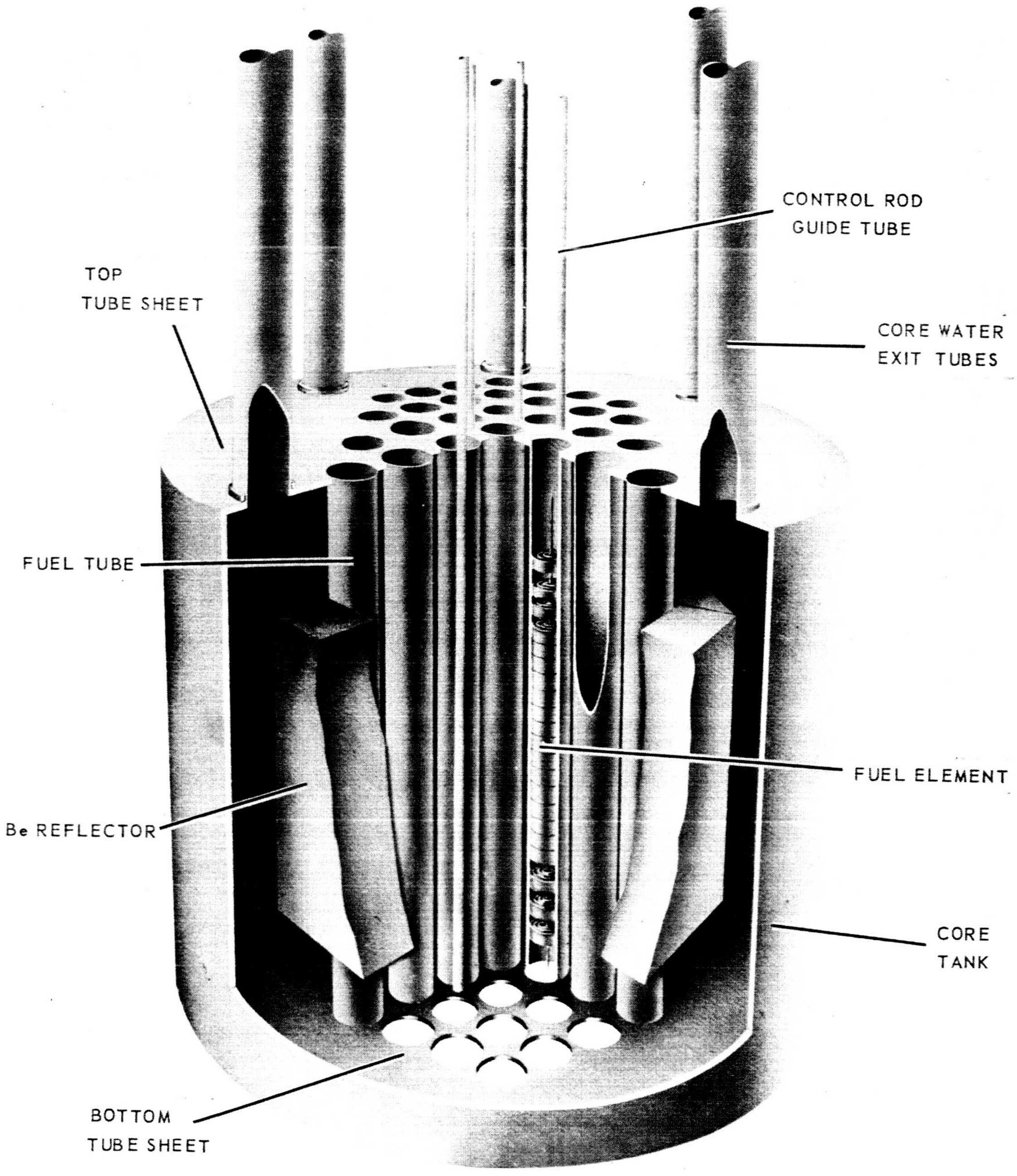

Fig. 1.1.8 HTRE-1 reactor

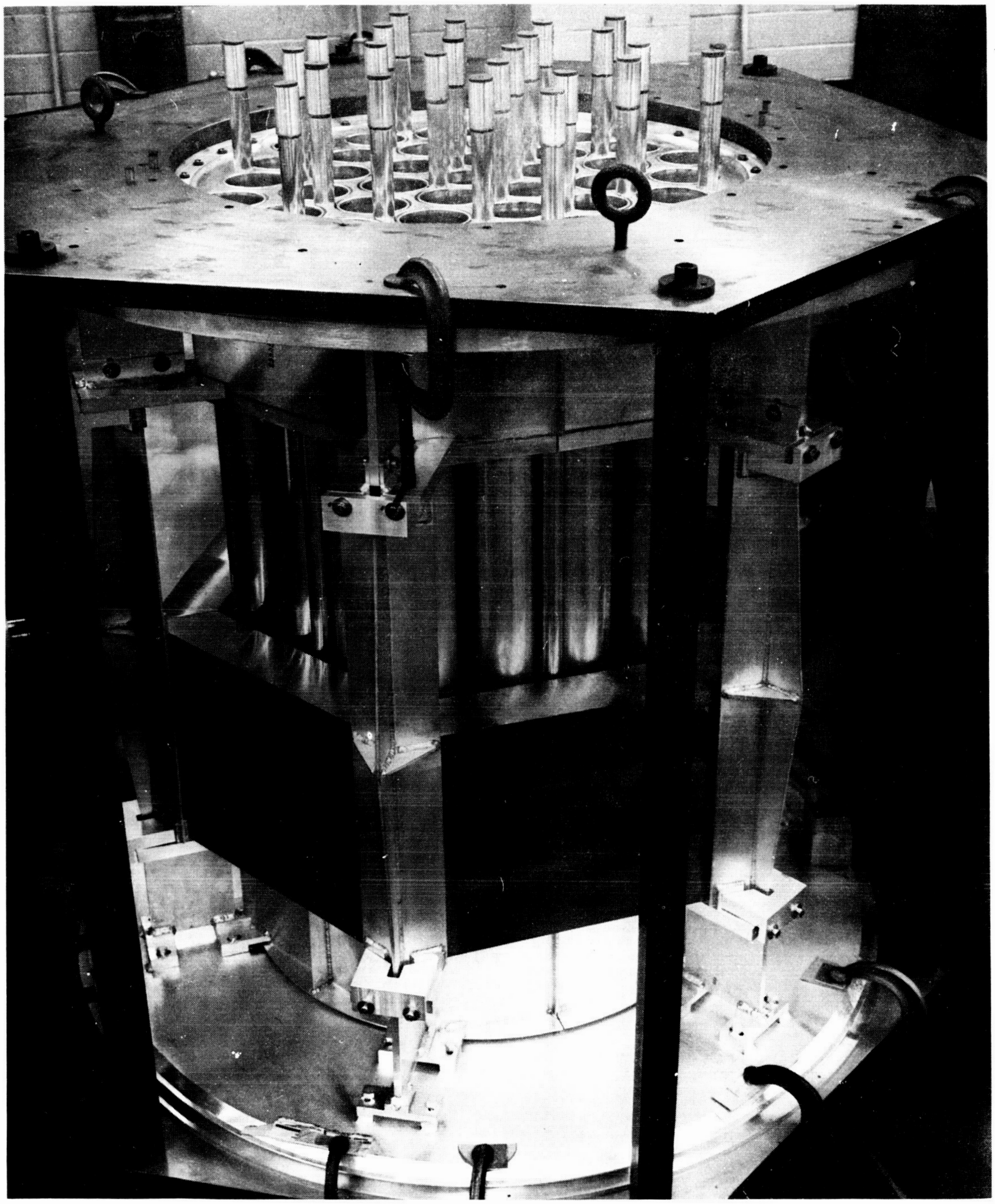

Fig. 1.1.9 -HTRE-1 reactor during construction

The reactor consisted of a cylindrical aluminum water vessel penetrated by 37 tubular air passages in a hexagonal pattern 30 inches across flats. Each of the air passages contained a concentric ring fuel cartridge (Figure 1.1.10) made of an 80 Ni - 20Cr alloy impregnated with UO_2. The active length of each cartridge was 29 inches. The air passage tubes were lined on the inner surface with a thin sleeve of stainless-steel-jacketed insulation to reduce the direct transmission of heat into the water moderator. The control rod guide tubes also served as inlet tubes for the moderator water which filled the entire reactor vessel except for the air passages and cooled the beryllium reflector and aluminum structure. The water pressure was only that required for pumping, and was maintained at a temperature of $160^{\circ}F$ by circulation to an external radiator, while the fuel elements operated at a temperature of approximately $1700^{\circ}F$, heating the air to about $1350^{\circ}F$.

Each fuel element within the reactor generated the same power. This was accomplished by varying the tube spacing, with the maximum spacing occurring near the outside of the reactor, where the power would normally be low. Thus, more moderator was associated with each tube and the thermal flux from tube to tube was equalized. The beryllium reflector also helped to maintain a sufficiently high flus in the outer tubes. The fine radial power distribution was flattened within a fuel cartridge by radial variation of the fuel loading from ring to ring.

The reactor vessel was attached to the top shield plug and both were inserted as an integral assembly into the cavity in the Core Test Facility shield. The control rod actuators were mounted on the top plate of the shield plug, as were the nuclear sensor supports, the neutron source actuators, the water inlet and outlet pipes, and the instrumentation leads for the reactor assembly.

A schematic diagram of the HTRE-1 aerothermal and control systems is shown in Figure 1.1.11. The air entered the turbojet engine, was compressed to approximately five times atmospheric pressure, and was ducted to the reactor. After being heated in the reactor (or the chemical burner downstream from the reactor), it was returned to the turbine and was then exhausted to a stack.

Summary of HTRE-1 Operation

The HTRE-1 engine was initially started and operated on chemical fuel with compressor air passing through the cold reactor. To transfer to nuclear power, the reactor control rods were gradually withdrawn and the reactor was brought to power by demanding an increase in neutron

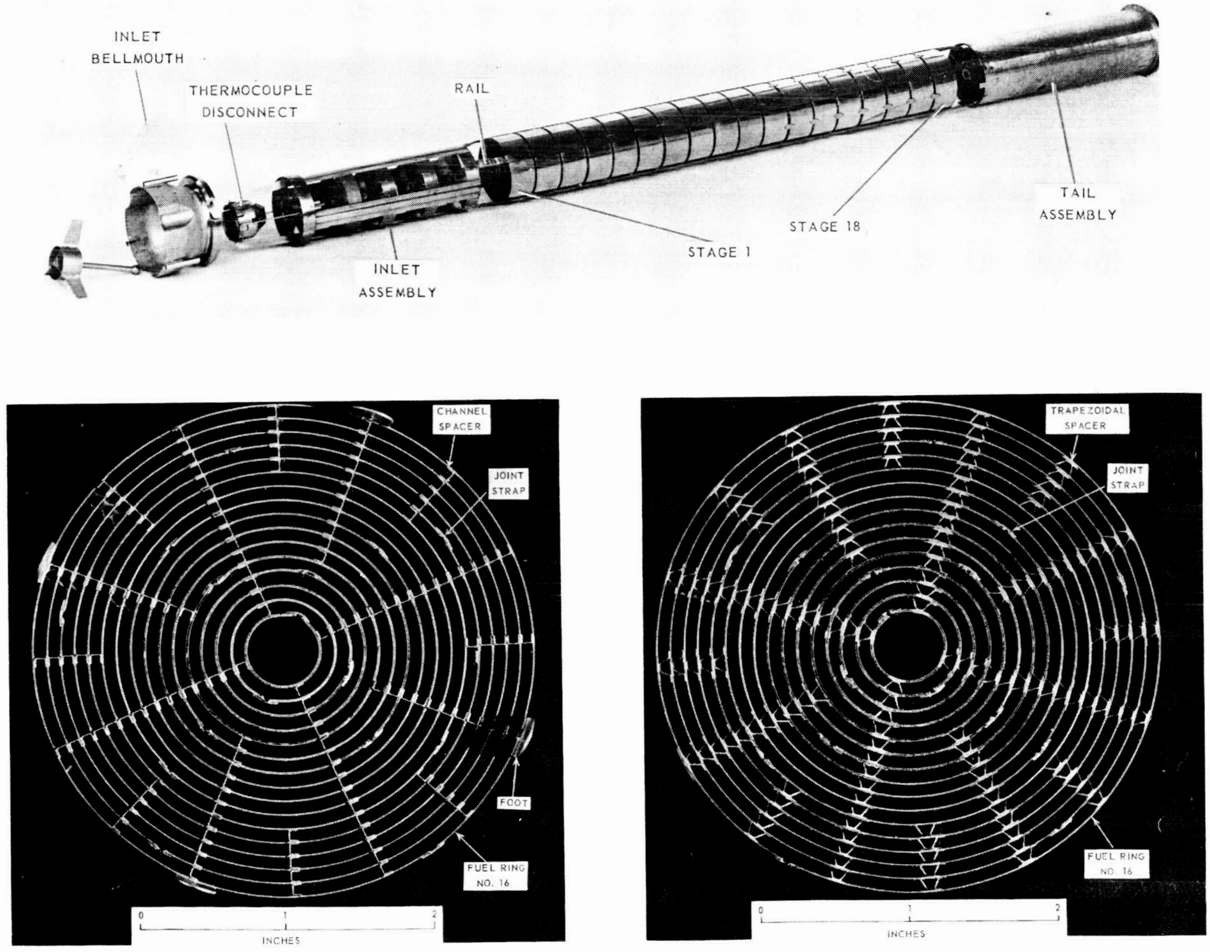

Fig. 1.1.10 -HTRE-1 fuel element and cartridge assembly

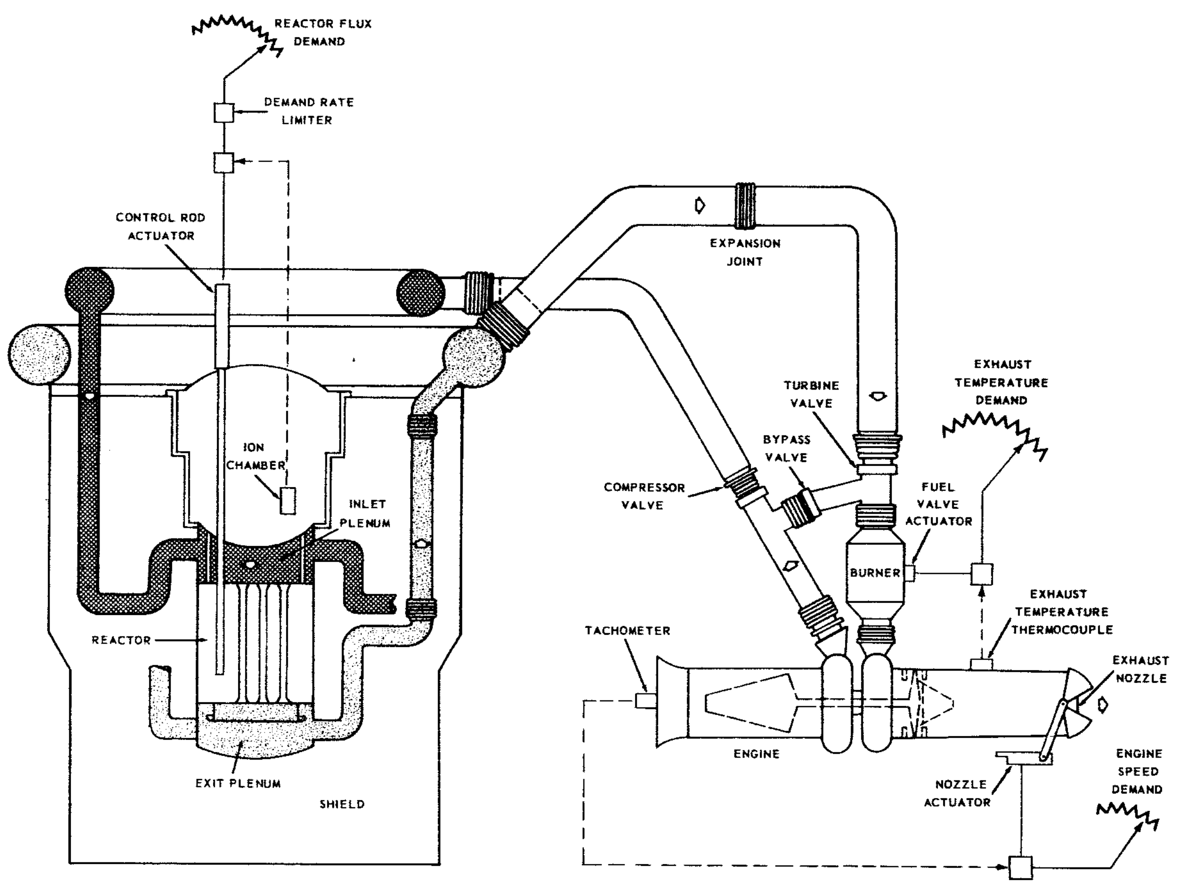

Fig. 1.1.11 - Schematic diagram of HTRE-1 aerothermal and control systems

flux. As more heat was supplied to the airstream by the reactor, the chemical fuel valve, sensing the temperature rise, gradually closed until the system was operating exclusively on nuclear power. The engine speed was held constant by controlling the area of the exhaust nozzle. To shut the system down, a reverse procedure could be followed, or shutdown could be achieved simply by scramming the reactor and allowing the engine to coast to a stop. The air supplied by the engine during coastdown provided sufficient aftercooling for the initial decay of afterheat. Auxiliary blowers provided aftercooling subsequent to engine coastdown.

The first full power test of the HTRE-1 system on nuclear power only took place in January 1956. A total of 5004 megawatt-hours of operation was completed during the test program, at power levels up to 20.2 megawatts. HTRE-1 operated above 200 kilowatts for 485.6 hours and for 150.8 hours at full nuclear power without chemical assistance. During the first 6 hours of full power operation, fuel element damage occurred in three cartridges caused by a defect in the insulation liners. After the damaged elements were replaced, power operation was resumed. An endurance test of 100 hours was run at a reactor-discharge air temperature of $1280^{o}F$, followed by 44 hours at $1380^{o}F$, thus exceeding the original test objective of 100 hours operation.

Post-operation examination revealed that the fuel elements used in the endurance run incurred no gross oxidation or mechanical damage. A number of small blisters was observed in the fuel stock; these were caused during fabrication by weld spatter which had damaged the clad material. Upon exposure to air, the UO_2 fuel was oxidized to U_3O_8, and in expanding had produced the blistering. This defect was eliminated in subsequent fuel element fabrication.

The aerothermal design data for HTRE-1, under typical conditions, is summarized in Table 1.1.1.

Final Status and Application of HTRE-1 Development

All objectives of the HTRE-1 program were met or exceeded. The reactor was tested beyond its life requirements and was capable of continued operation at completion of the test program. The feasibility of nuclear turbojet engine operation with a direct air cycle reactor had been demonstrated. This was the first known operation of a high-temperature, gas turbine engine on nuclear power.

High-temperature, oxide-dispersion, metallic fuel elements demonstrated a life capability in excess of design requirements. Further improved fuel elements of the same type were used in the subsequent HTRE-3

TABLE 1.1.1

HTRE-1 THERMOCYNAMIC DATA[a]

Engine	
Engine	One modified GE J47
Compressor pressure ratio	4.95
Altitude, ft (NRTS)	5,000
Air weight flow, lb/sec	59.5
Compressor discharge temperature, ^{o}F	393
Turbine inlet temperature, ^{o}F	1295
Reactor	
Reactor inlet air temperature, ^{o}F	359
Reactor inlet pressure, psia	54.95
Reactor exit-air temperature, mean, ^{o}F	1335
Maximum average fuel element operating temperature, ^{o}F	1700
Total heat transfer area, ft^2	1194
Core air pressure drop, psi	7.11
Reactor power-to-air, Btu/sec	15,100
Reactor power-to-water, Btu/sec	1,500
Total reactor power, Btu/sec	16,600

[a]The cycle conditions varied from these conditions during operation depending on the ambient air conditions and the value at which the operator set the control parameters. For example, the reactor was operated at an exit-air temperature of 1280^{o}F for 100 hours and 1380^{o}F for 44 hours.

reactor operation and in the XMA-1 power plant design.

The predictions of neutron flux distributions and the methods used to achieve uniform radial power were verified both in critical experiments and during power operation. Predictions of fuel element and air temperatures to reflect gross radial, longitudinal, and fine radial power distributions as well as perturbations produced by control rods, airflow maldistributions, and manufacturing tolerances, were in close agreement with test results. The nuclear and aerothermal analytical techniques were further developed and used in subsequent metallic reactor designs.

Test experience verified the analytical predictions that the reactor was stable in operation with transient temperature variations well within the

capability of the response characteristics of the control system. Test results indicated that the extremely fast response that had been provided in the control system was unnecessary as were other control refinements such as continuous indication of the position of the control rods. Later control system designs were simplified accordingly.

Safe operational and maintenance procedures were developed and the practicability of ground operation and maintenance of nuclear turbojet systems was proved. A realistic basis was established for determining the extent to which prototype propulsion systems could be maintained manually rather than remotely, e.g., manual decontamination and maintenance of the turbomachinery proved to be feasible. After remote removal of the fuel elements, the other system components could be maintained manually after relatively short decay times.

HTRE-1 is described in the GE report APEX-904, "Heat Transfer Reactor Experiment No. 1".

1.1.5 HTRE 2 Reactor

After committing HTRE-1 to hardware, the materials and component development effort was directed toward a number of moderator and fuel element materials of potentially greater temperature and/or life capability. Liquid hydrocarbons, hydrided metals, and ceramics were under development as moderator materials. Improved nickel-chromium and other, even higher temperature metals, as well as ceramics, were being developed as fuel element materials. Active in-pile test programs were in process or planned for these materials. However, the size of test specimens that could be accommodated and the type of experiment that could be performed were limited in the available in-pile test facilities, such as the Materials Testing Reactor. Therefore, a decision was made to modify the HTRE-1 reactor to accommodate large test specimens of more advanced reactors with which the capabilities and interactions of moderator, fuel elements, and structural materials could be evaluated. This modified reactor, designated HTRE-2, was used to test a variety of metallic and ceramic reactor components as well as to perform other special purpose tests.

The modification of the HTRE-1 reactor was started in early 1956. Modification was completed and the first specimen brought to test in July 1957. HTRE-2 continued to test further improved or alternative reactor materials and components until the ANP program was terminated in 1961.

Description of HTRE-2 Reactor and Test Assembly

The HTRE-2 "parent core" was similar to the HTRE-1 core, except that the central seven air tubes were removed and replaced by a hexagonal void 11 inches across flats. (See Figure 1.1.12.) A corresponding opening was made in the top shield plug so that sections of advanced reactors could be inserted into the HTRE-2 parent core without requiring removal of the core from the shield. The inserts were suspended from a small diameter shield plug, which filled the opening in the main shield plug. (See Figure 1.1.13.) No special cooling air circuit was provided for the insert. The air was drawn from the common plenum chamber above the reactor.

Final Status and Application of HTRE-2 Development

HTRE-2 was used principally for the testing of BeO ceramic fuel cartridges although some tests were performed using the metallic cartridges and hydrided zirconium moderator of the type used in HTRE-3.

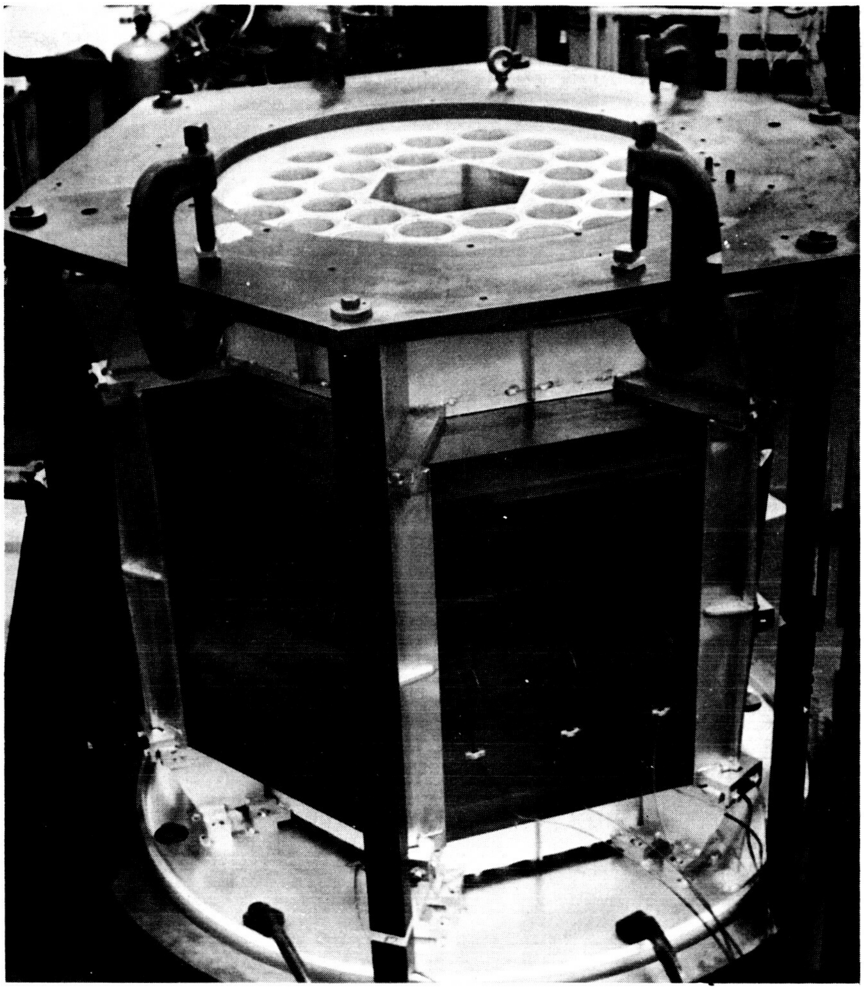

Fig. 1.1.12 -HTRE-2 reactor during construction, showing the hexagonal cavity used to test advanced reactor components (C-04013)

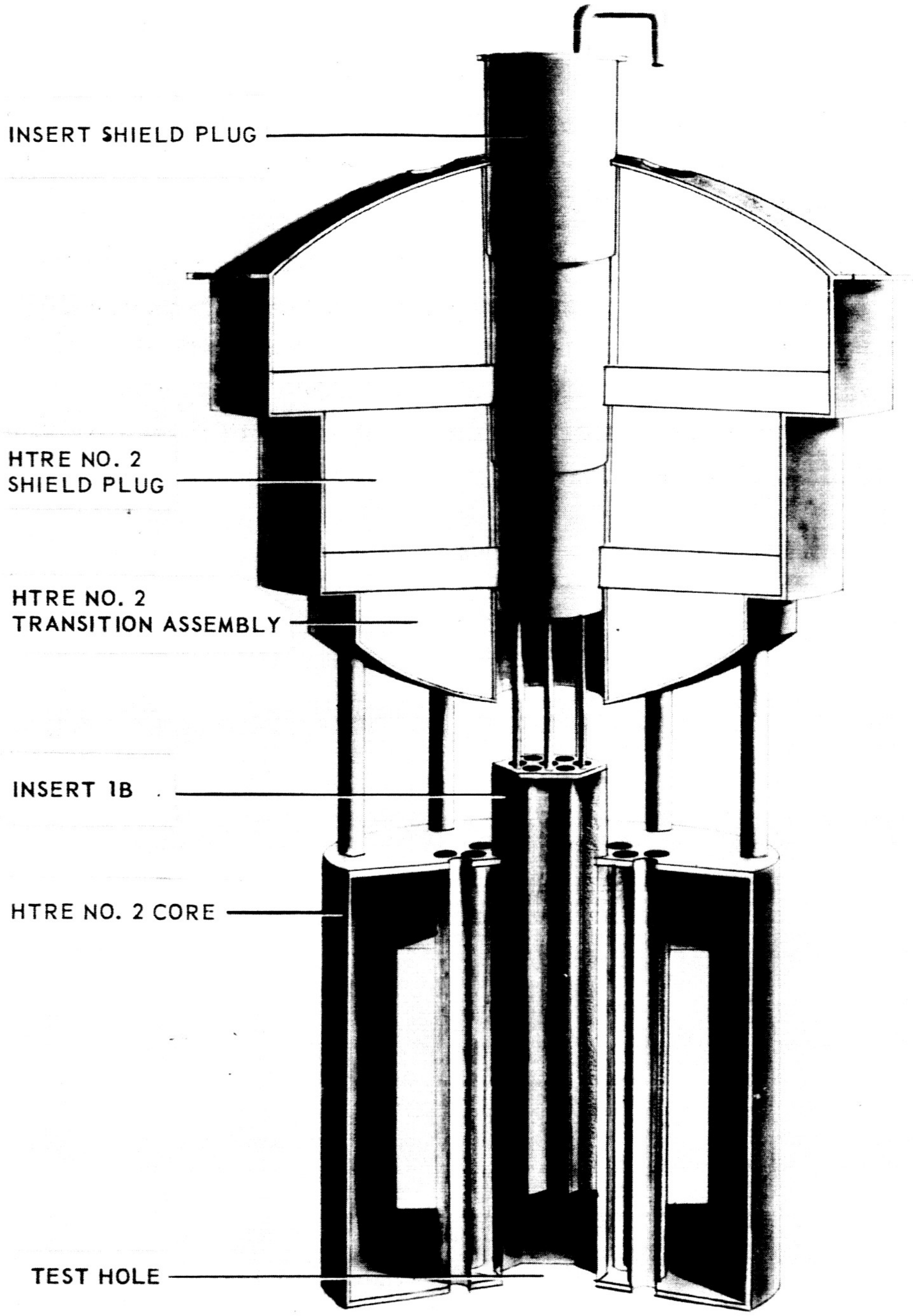

Fig. 1.1.13 - Artist's conception of HTRE-2 parent reactor, shield plug, and test insert (040-513)

HTRE-2 operation verified the use of hydrided zirconium as a reactor material, providing a firm basis for the HTRE-3 reactor design. Operation of the HTRE-2 parent core provided further data applicable to the design of subsequent reactors using metallic fuel elements.

Over 1100 hours of power testing were completed on test specimens of beryllium oxide reactors. These tests verified the integrity of clad materials to prevent water vapor corrosion. The mechanical integrity of the fuel tubes was established at temperatures in excess of those proposed in the subsequent XNJ140E design.

Valuable data was collected on fission fragment release, deposition on ducting and other components, filtration, and atmospheric diffusion. This data was applied to subsequent operational analyses of nuclear propulsion systems. The data verified that the fission fragment release rate of nuclear propulsion systems using clad ceramic fuel element materials was within tolerable limits.

1.1.6 HTRE-3 Reactor

At the same time the HTRE-2 program was initiated, designs were started for a full-scale reactor test, designated Heat Transfer Reactor Experiment No. 3 (HTRE-3). Although similar to HTRE-1 and HTRE-2, HTRE-3 was dissimilar in three basic ways: (1) the reactor was mounted horizontally and was equipped with flight-type shield; (2) a high temperature, solid moderator was used; and (3) the power plant was designed for simultaneous operation of two turbojet engines from a single heat source.

The development of HTRE-2 was scheduled sufficiently ahead of HTRE-3 that the materials and components selected for HTRE-3 could be evaluated in HTRE-2 before HTRE-3 was fully committed to hardware.

The objectives of the HTRE-3 were to:

1. Evaluate and further develop the materials and design technology of a direct air cycle reactor in which all components were air cooled and operated at high temperatures
2. Develop and evaluate other propulsion system components more closely resembling those required in an aircraft power plant configuration
3. Gain operating and maintenance experience with a nuclear system whose external radiation levels were similar to those anticipated in aircraft installations

HTRE-3 Materials and Design Selection

Following the successful operation of HTRE-1, the reactor development progression could logically have included further full-scale tests using an intermediate-temperature hydrogenous liquid in place of the water moderator used in HTRE-1. This was not warranted, however, since in-pile tests and design studies had indicated the feasibility of such materials for the duty cycle required in an aircraft. The use of a solid hydrogenous moderator, on the other hand, introduced complex new mechanical and aerothermal problems. The severity of many of these problems could be reduced by providing excessive airflow and thus overcooling the moderator and structure, but this would have resulted in both a performance penalty and a larger, heavier reactor. To avoid these penalties required a precise balance of the airflow distribution between moderator, fuel elements, control rods, and structure so that each operated closely enough to its life-temperature capability to provide maximum performance while still retaining an adequate margin for reliability. Hydrided zirconium had been developed

to a sufficiently advanced state to be used to evaluate the aerothermal and mechanical design technology in a full-scale, solid moderated reactor test. Thus, hydrided zirconium for the moderator and nickel-chromium for the fuel elements were the major materials selected for the HTRE-3 reactor. Europium oxide was used as the control rod poison primarily because of its high-temperature compatibility with the containment materials.

Fully developed materials, specifically lead, steel, and water, were selected for the shield because high temperature shield evaluation was not a test objective. Nevertheless, shield design objectives, particularly in the vicinity of the ducts, more closely approximated aircraft requirements.

Structural load requirements simulating landing, maneuvering, etc., were imposed on both the shield and the reactor in accordance with aircraft power plant standards.

Description of HTRE-3 Reactor and Test Assembly

The major HTRE-3 components, reactor, shield, single chemical combustor mounted behind the reactor-shield assembly, two modified J47 turbojet engines and interconnecting ducting, are shown in Figure 1.1.14. These components and the required test support equipment were mounted on a mobile dolly, similar to the CTF dolly. In HTRE-3, the flow of air and the method of operation were much the same as in HTRE-1.

The HTRE-3 reactor shield assembly is shown in Figure 1.1.15. The radial and end shields consisted of alternate layers of lead and water. The active core, 30 inches long and 51 inches in diameter, contained 150 cells inside a 3-inch-thick beryllium reflector. Each cell consisted of a fuel cartridge inside a hydrided zirconium moderator element; the moderators were hexagonal on the outside and circular on the inside. All the reactor components were cooled by primary air from the turbojet compressor. A view of the partially assembled reactor is shown in Figure 1.1.16. A drawing of the reactor is shown in Figure 1.1.17.

An assembled fuel cartridge is shown in Figure 1.1.18. Each cartridge had 19 stages made up of 12 concentric metallic rings. The UO_2 fuel, dispersed in a matrix of 80Ni - 20Cr, was clad with 80Ni - 20Cr stabilized with niobium.

Power flattening was achieved by (1) varying the hydrogen content of the moderator for gross radial control, (2) shimming the fuel elements with boron steel for circumferential power control, (3) extending the

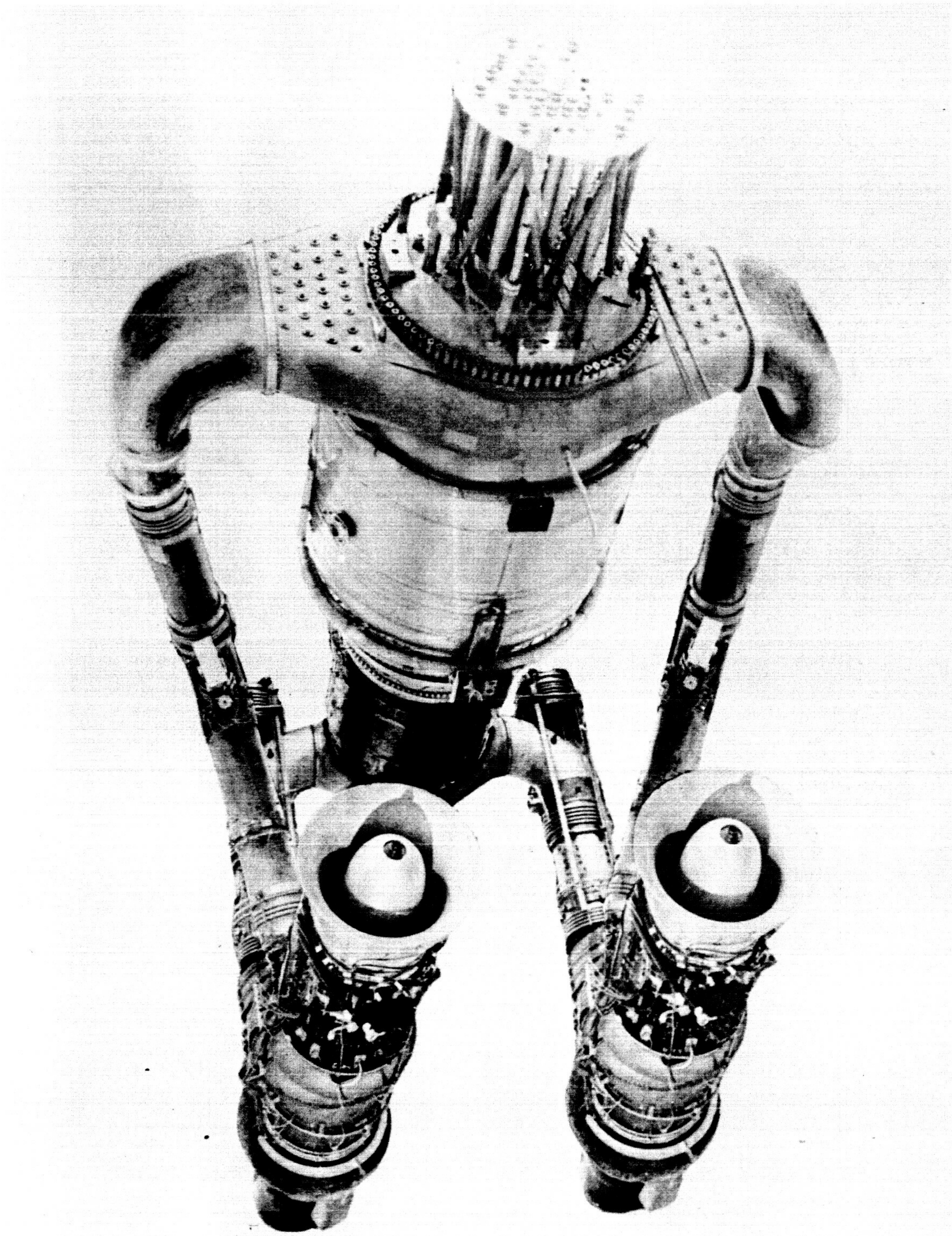

Fig. 1.1.14 - HTRE-3 basic components (U-2189)

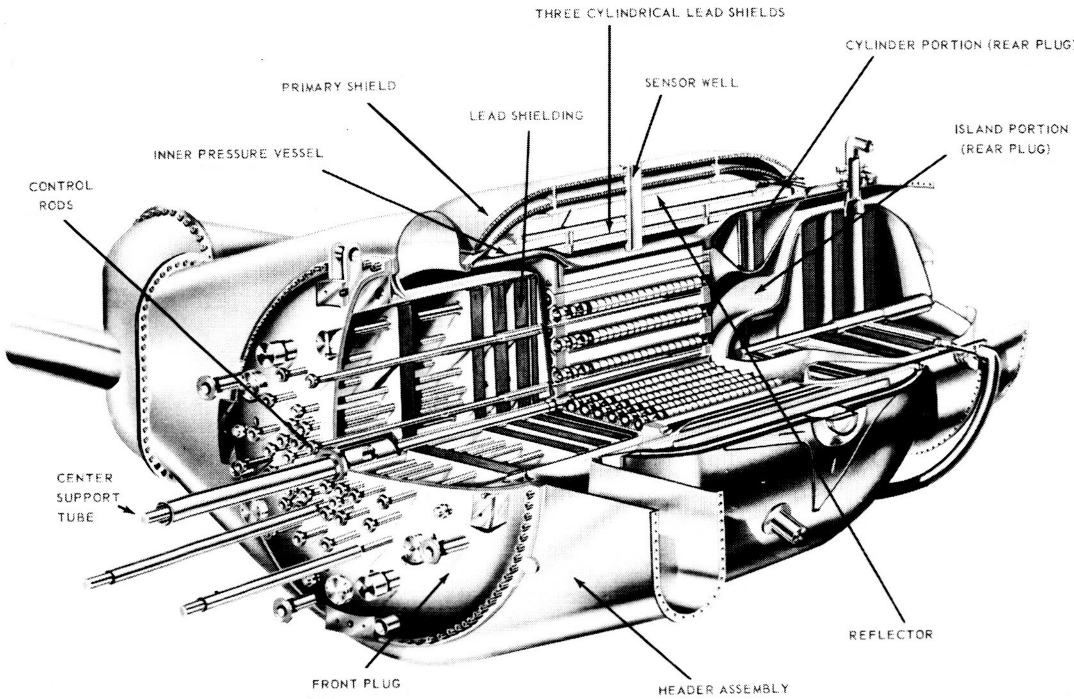

Fig. 1.1.15 -HTRE-3 reactor-shield assembly

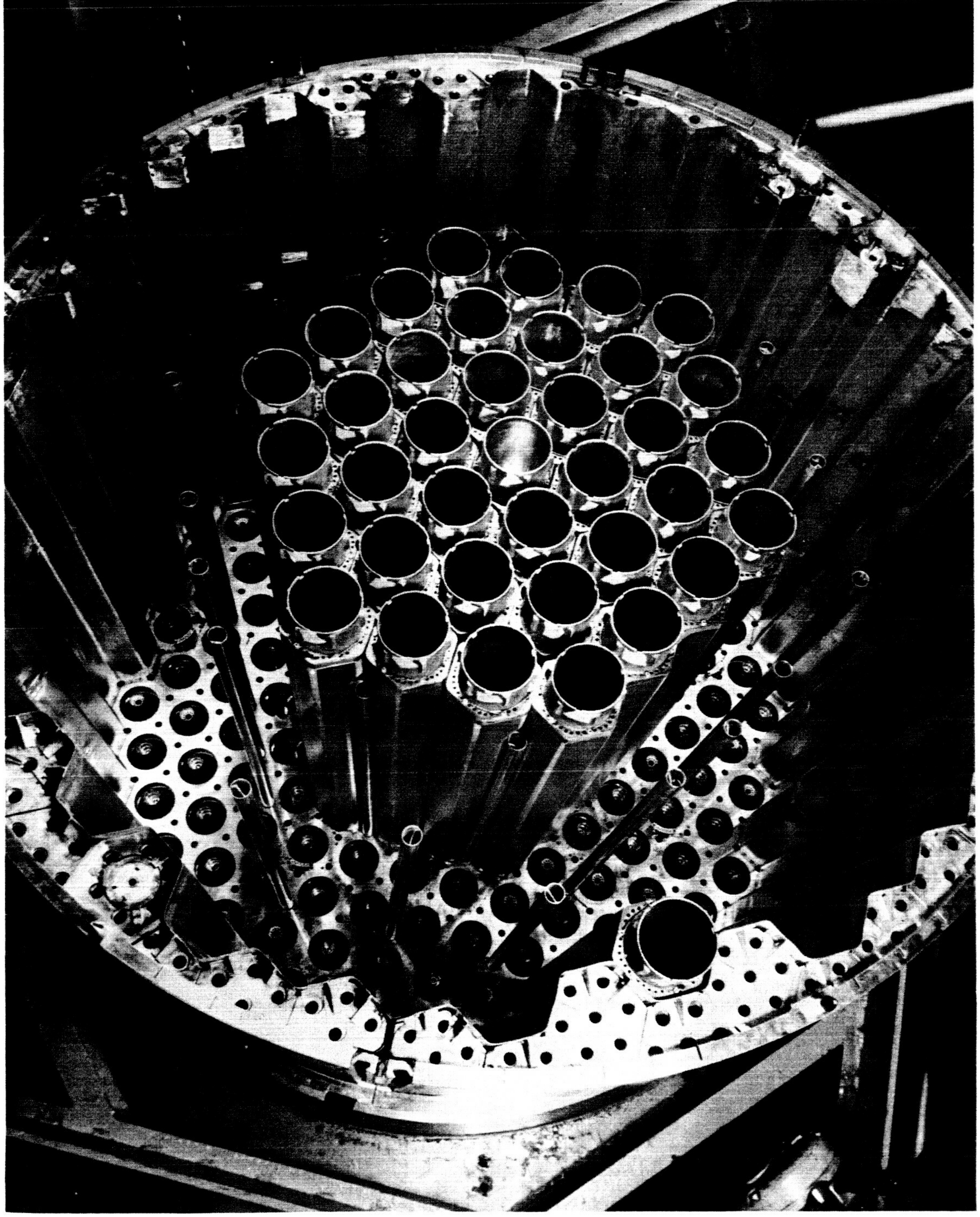

Fig. 1.1.16 - HTRE-3 reactor during assembly (C-13822)

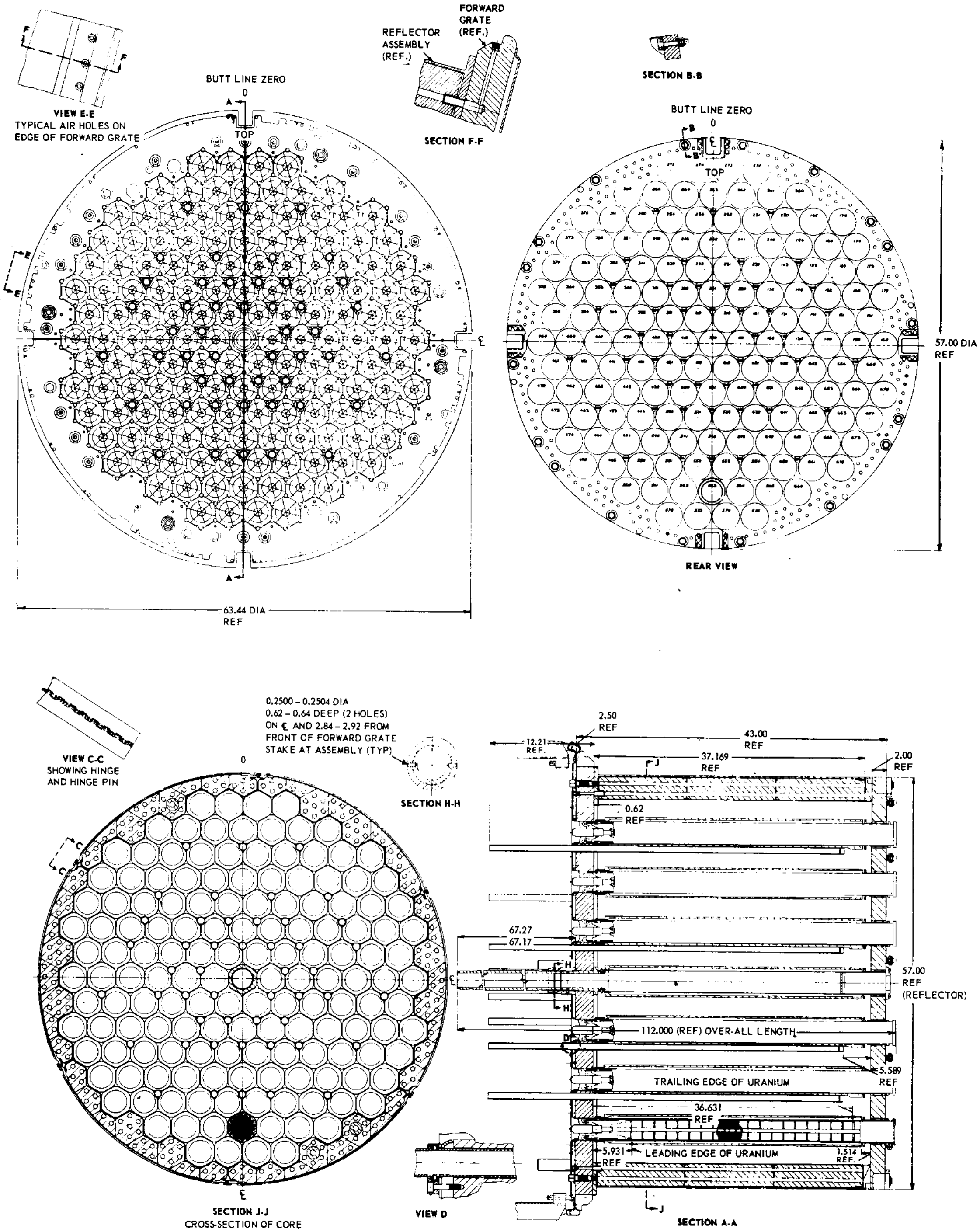

Fig. 1.1.17 – HTRE-3 core assembly (Dwg. 7018R51)

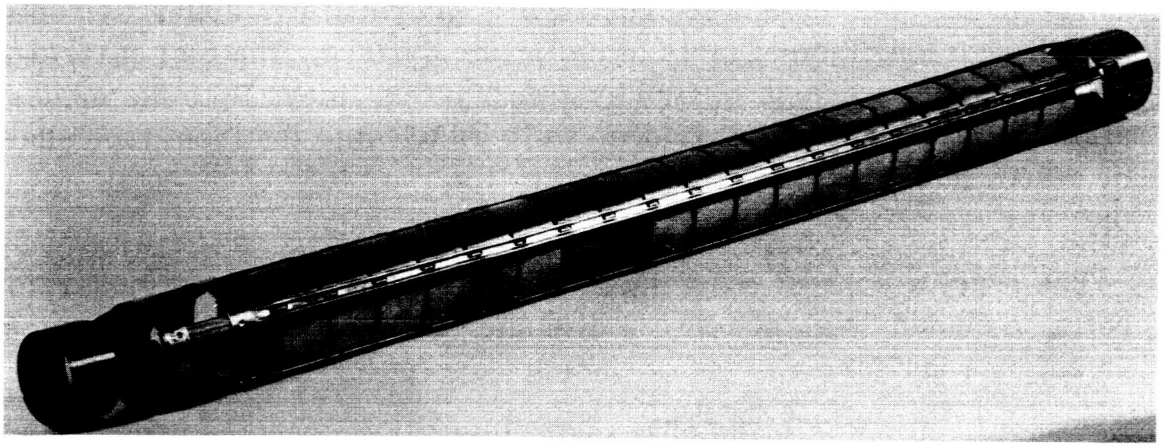

Fig. 1.1.18 - HTRE-3 fuel cartridge

moderator beyond the active core for longitudinal power control, and (4) varying fuel loading in the individual fuel rings for fine radial power control.

The moderator tubes were cooled to approximately 1200°F by routing air through longitudinal slots in the inside surface. The fuel cartridge and moderator were separated by an insulation liner. Both the moderator tubes and the fuel cartridges were attached to the front tube sheet by disconnects and freely supported by the rear tube sheet to allow for thermal expansion.

The reactor control rods, located at the junction of three moderator cells, included 30 shim rods, 3 dynamic rods for power changes, and 15 safety rods normally out of the core except for shutdown.

The reflector was made of hexagonal beryllium sectors provided with longitudinal holes for cooling.

HTRE-3 Test Operation

To evaluate nuclear characteristics and to provide a shakedown of control and other components prior to power operation, low power testing of HTRE-3 was started in 1958. Power operation was delayed by damage to the reactor fuel elements in a power excursion. This resulted from control rod withdrawal under the influence of an erroneous reactor power indication caused by a fault in an electronic component. The reactor airflow, which was being supplied by low capacity blowers rather than the turbojet engines, was insufficient to prevent fuel element overtemperature.

Power operation, using the turbojet engines, was started after replacement of damaged fuel elements. The first operation was a check of the chemical engine performance to establish temperature, pressure, and flow rates over the range of engine speed and nozzle position. Preliminary runs were made to determine the part-chemical, part-nuclear characteristics of the system prior to transfer to full nuclear power. Subsequently, six transfers to full nuclear power were made. System variables were examined over a range of engine speeds and reactor powers, including the lowest possible engine speed, to examine some of the system characteristics associated with a full nuclear start.

The reactor and engines were operated for 126 hours on full nuclear power in successive runs of 1.4, 29.0, 5.5, 25.4, and 64.9 hours of continuous operation. Since this exceeded the initial objective of 100 hours operation, the test assembly was returned to the hot shop for inspection in February 1960. Visual inspection revealed that the fuel

elements were in excellent condition. Detailed radiochemical analysis verified that power generation was within the predicted range.

HTRE-3 testing was resumed in late 1960 to demonstrate the capabilities of the fuel elements above design temperatures and to confirm that a nuclear turbojet power plant could achieve a full nuclear start without the use of chemical fuel. Previously, nuclear operation had been achieved in three steps; (1) using the engine starter to turn the engine rotor and obtain a low airflow, (2) igniting the chemical fuel and bringing the engine up to speed and full airflow, and (3) bringing the reactor up to full power while closing the chemical fuel valve. In the nuclear start, the engine starter was used as before to obtain initial airflow, but the intermediate chemical operation was omitted, and the engine was brought up to speed and full airflow by a gradual increase of reactor power. The first full nuclear start was made in December 1960; subsequently, two more nuclear starts were made. Reactor materials temperatures stayed within design limits throughout these nuclear startups.

Following the second nuclear start, in order to evaluate nuclear shutdown, the reactor was maintained at a power of approximately 29 megawatts for 1 hour and then was manually scrammed and the engine allowed to coast down. An aftercooling blower supplied 8.6 pounds of cooling air per second to the reactor after scram. Transient recordings were made of selected system parameters. All temperatures started to decline after the scram and continued to fall for the remainder of the 1-hour recording period.

An additional 20.3 hours of full nuclear operation was accumulated after the evaluations of nuclear start were completed. This operation was performed at a maximum fuel element temperature of approximately 2050°F, to demonstrate temperature capability in excess of design requirements. At the termination of this operation, the reactor appeared to be fully capable of continued operation.

A summary of the HTRE-3 performance data during these tests is given in Table 1.1.2.

Final Status and Application of HTRE-3 Development

The HTRE-3 operation demonstrated the feasibility of an air-cooled reactor using nickel-chromium fuel elements and a hydrided zirconium moderator. The fuel elements were operated at temperatures and for time periods in excess of design requirements. Verification was achieved of the analytical design methods for balancing airflows, uranium distribution, and hydrogen distribution to flatten material- and air-

TABLE 1.1.2

HTRE-3 PERFORMANCE DATA

	Endurance Run		Elevated Performance
Reactor power to air, mw	32.4		34.2
Reactor airflow, lb/sec	123		125.6
Mixed core discharge air temperature, °F	1330		1370
Compressor discharge temperature, °F	385		376
Compressor discharge pressure, psia	53.5		
	Predicted	Measured	Measured
Maximum temperature, °F			
Fuel element	1880	1900	1986 2050 (extrapolated)
Moderator	1175	1120	1160
Reflector	1100	1030	1010[a]
Discharge air from fuel	1640	1640	1720
Average temperature, °F			
Discharge air from moderator	968	880	900[a]
Discharge air from reflector	955	940	845[a]
Discharge air from control	805	480	485[a]

[a]Thermal equilibrium not reached

temperature distributions. Mechanical design features were proved to be adequate. A reactor of this type, with further design refinements, was incorporated in the XMA-1A prototype propulsion system design.

Nuclear starts were demonstrated. This improved the prospect of ultimately eliminating auxiliary chemical burners from nuclear propulsion systems. This would reduce the length and weight of the power plant as well as reduce the system air pressure drops, and thus improve the over-all performance.

Measurements of radiation levels obtained in the shield, especially in the vicinity of the ducts, were applicable to the design of prototype systems, particularly the XNJ140E, in which a similar annular duct configuration was used. Of necessity, previous shielding measurements had used reactor radiation sources which differed in configuration and radiation leakage from full-scale aircraft-type reactors.

The practicability of ground operation and maintenance of turbojet engines with a nuclear heat source had been further verified.

At the termination of the ANP program, the HTRE-3 reactor and engine assembly were in a standby condition, capable of resuming nuclear operation at any time.

Details of HTRE-3 are provided in the GE report, APEX-906, "Heat Transfer Reactor Experiment No. 3.

1.1.7 Ceramic Reactors

Because of the steadily improving status of ceramic materials relative to metals in applications above 2000°F, two ceramic reactor design studies and concurred development were carried into considerable detail for a proposed test of an experimental ceramic reactor. The testing of one of these, the D101E reactor, was to be performed in the CTF which had been used for HTRE-1 and HTRE-2. This proposed test was designated HTRE-4. The other, the D141A, was to be tested in the HTRE-3 test assembly, modified to accommodate the high performance X211 engine planned for use in prototype propulsion systems. The proposed test of the D141A reactor was referred to as the Ceramic Core Test (CCT). These approaches were dropped in favor of proceeding directly to operation in a prototype power plant configuration.

The D101E reactor concept was based on a geometry of triangular cells in which ceramic moderator slabs, arranged to form triangular cells, contained and supported bundles of round, fueled, ceramic tubes. The triangular cells, in turn, were supported and contained by metallic external structure and internal longitudinal support tubes. Circular bores in the fueled tubes were provided for the passage of cooling air. The reactor is shown in Figure 1.1.19.

A section of the D101E type reactor was tested as an insert in HTRE-2. An illustration of this insert is shown in Figure 1.1.20. The insert incorporated round, uncoated beryllium oxide fuel tubes, and additional beryllium oxide moderator in the form of slabs which also served as structure by bearing the radial compressive loads. The longitudinal loads were borne by air-cooled metallic tubes penetrating to silicon carbide aft retainer plates.

The D141A-1 reactor used hexagonal ceramic tubes as unit building blocks. The portion of the reactor constituting the active core contained fueled tubes in which the ceramic matrix acted as both moderator and fuel carrier. The unit building block concept was used also in an outer annular region comprising the outer reflector. The tube bundle was contained in, and supported by external metallic radial and longitudinal support systems. As in the D101E reactor, cooling air was channeled through circular bores in the tubes. The D141A-1 is shown in Figure 1.1.21.

A number of test sections of the D141A-1 type of reactor were operated as inserts in HTRE-2. The hexagonal tubes were coated on the inner surface to prevent hydrolysis of the BeO by atmospheric water vapor.

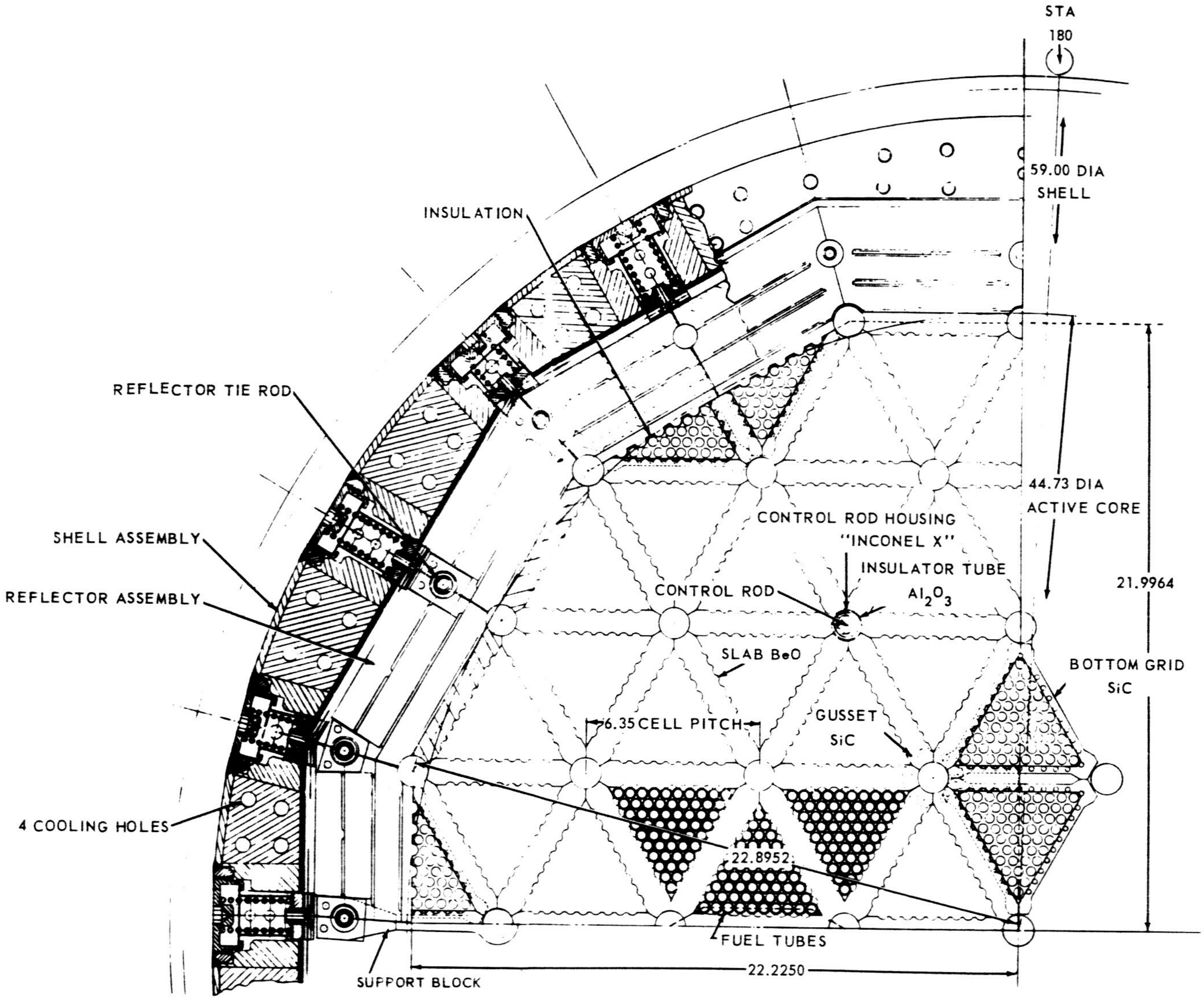

Fig. 1.1.19 - D101E ceramic reactor

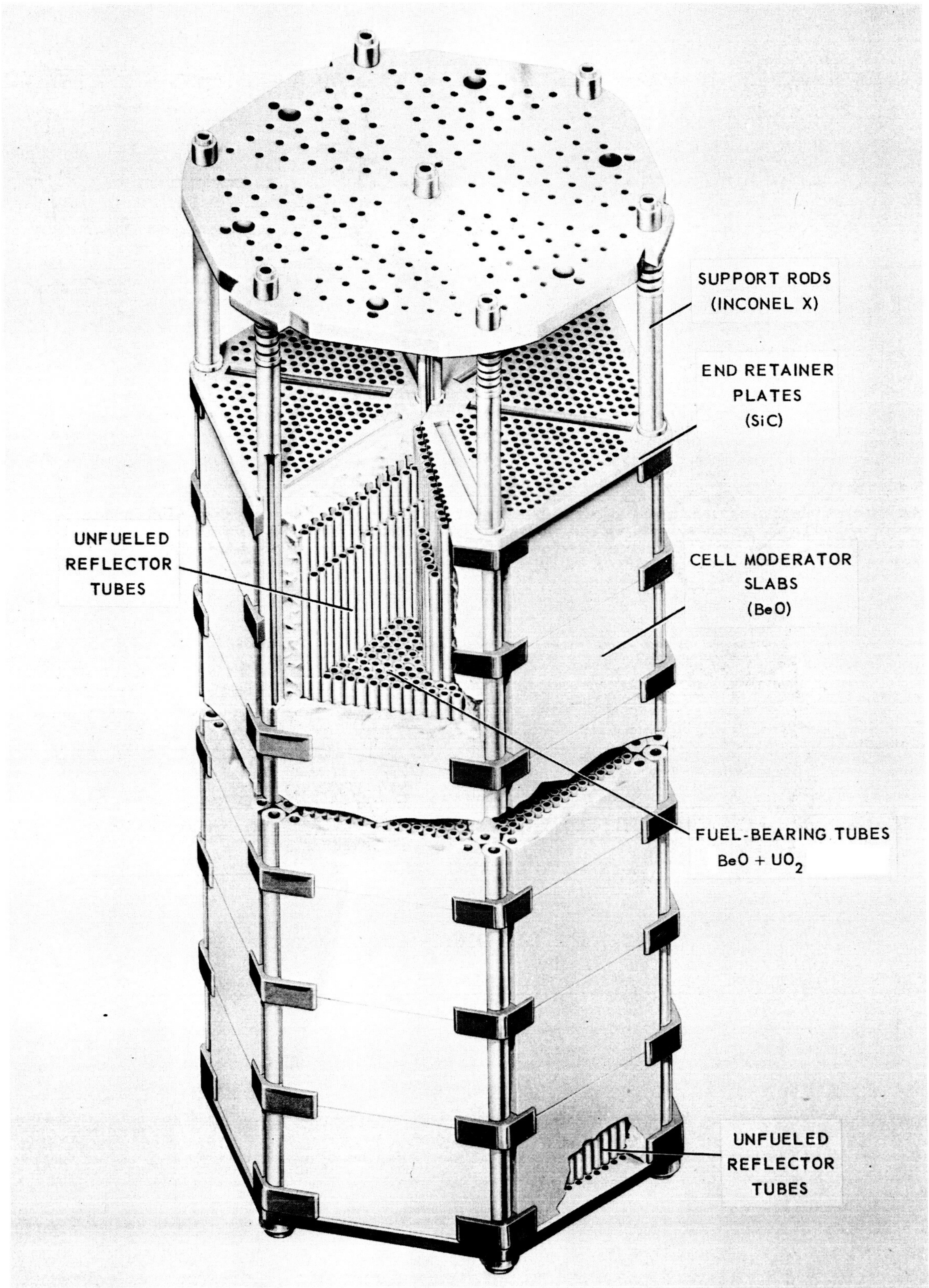

Fig. 1.1.20 - First ceramic insert tested in HTRE-2 (Dwg. 4098098-719)

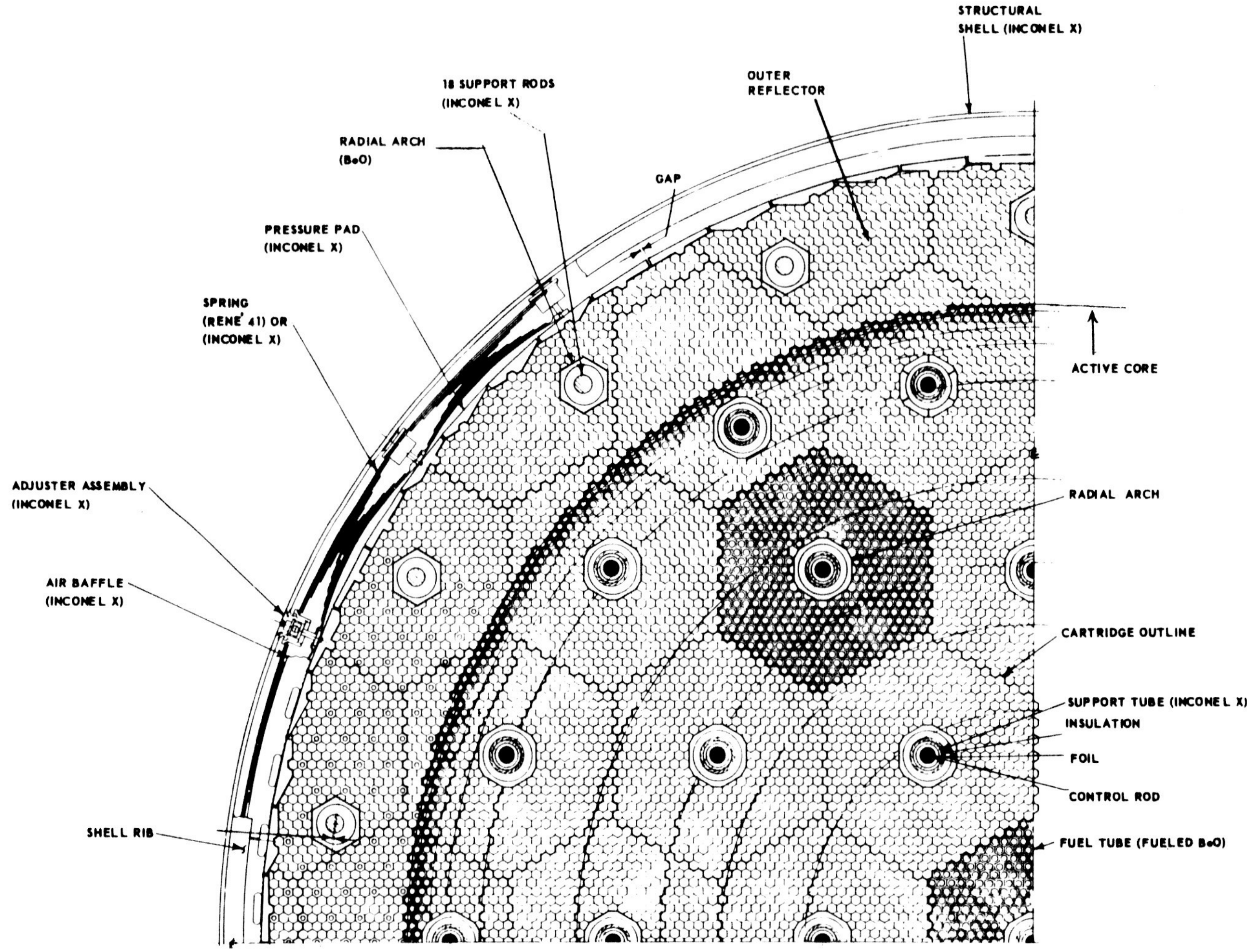

Fig. 1. 1. 21 - D141A-1 reactor

Comparing the results of these design studies indicated that the D141A-1 geometry was the preferred design. The XNJ140E reactor design concept evolved directly from the D141A-1 reactor design. Several significant engineering developmental tests supporting the D141A-1 design were used as the basis of subsequent XNJ140E design.

1. 1. 8 Prototype Propulsion System Reactors

Both metallic and ceramic reactors were carried into final design for use in prototype military power plants. The metallic reactor in its most highly developed configuration is shown in Figure 1. 1. 22 with a fuel cartridge shown in Figure 1. 1. 23 and individual fuel stages in Figure 1. 1. 24. The most highly developed ceramic reactor design was in a configuration similar in general to that of the D141A-1 shown in Figure 1. 1. 21 but differing in a number of details.

Two major power plant configurations were carried into final design. One of these, the XMA-1, Figure 1. 1. 25, was a dual engine system with a single reactor heating air for both turbines. The other, the XNJ140E, Figure 1. 1. 1, was a single engine single reactor system in which the engine shaft penetrated through the center of the reactor. The single engine XNJ140E with the ceramic reactor was the system under development to meet the requirements which existed at the time of program termination.

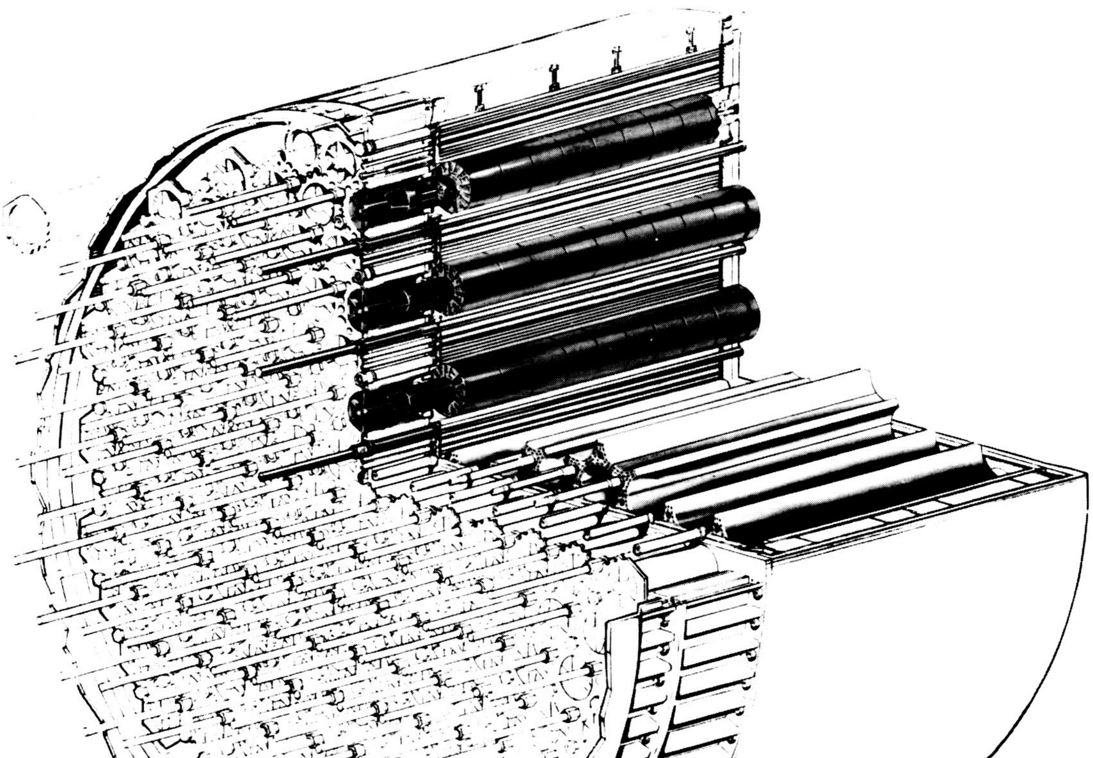

Fig. 1.1.22 - XMA-1A reactor core

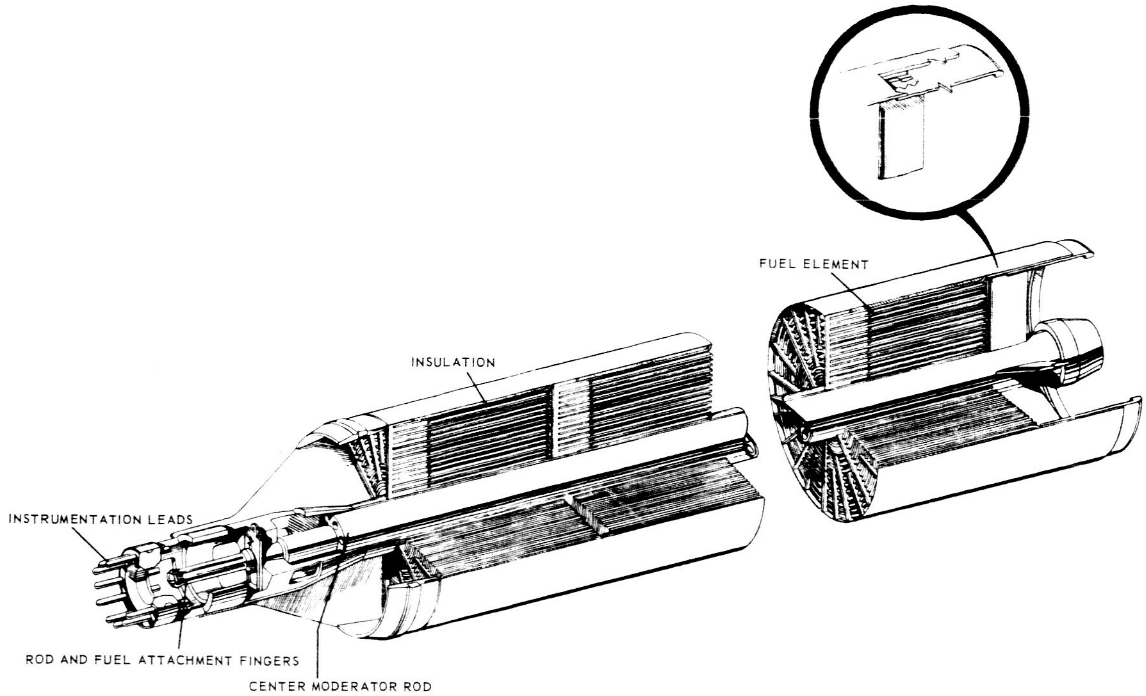

Fig. 1.1.23 – XMA-1A fuel cartridge (DI-37)

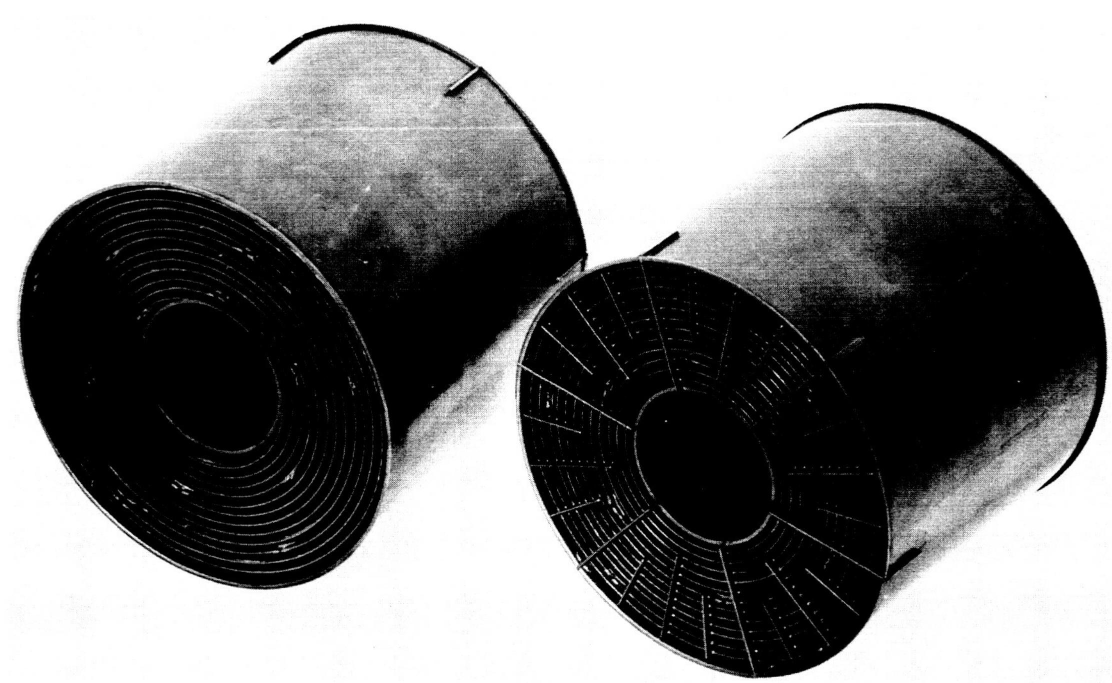

Fig. 1.1.24 - XMA-1A fuel stage (C22689)

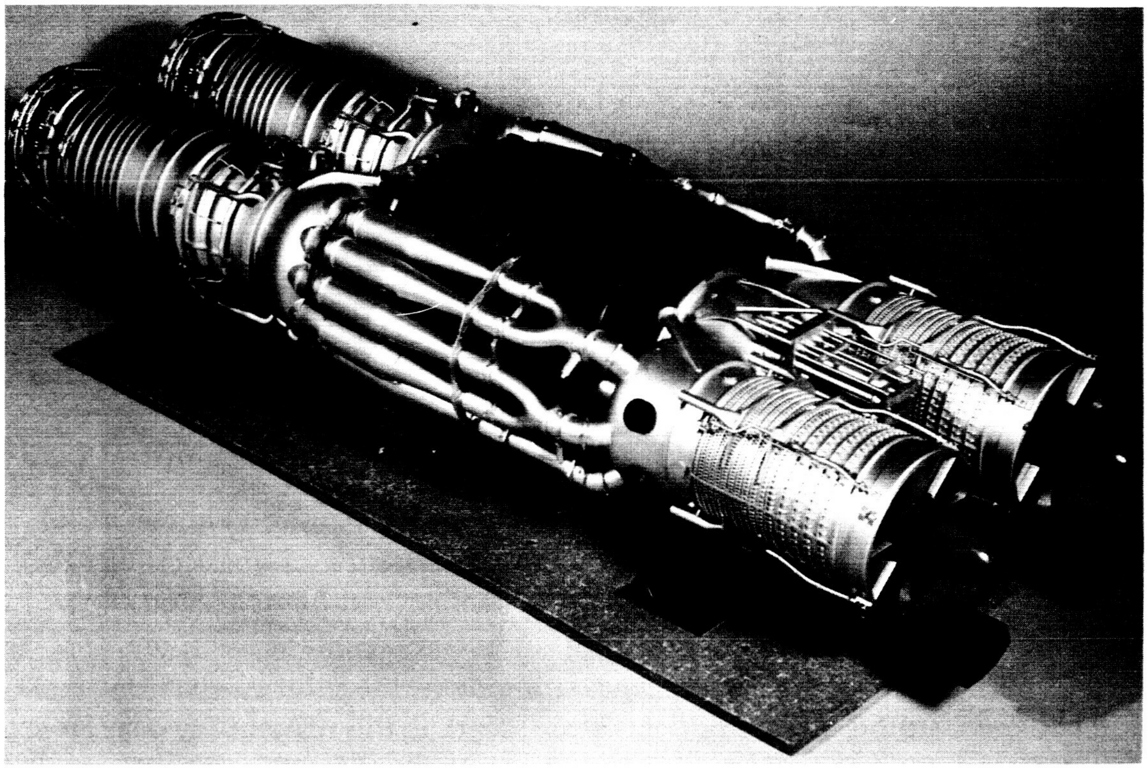

Fig. 1.1.25 - Model of XMA-1 power plant (U-39336B)

1.2 NUCLEAR RAMJET REACTORS*

The Pluto Reactor Program was established at the University of California Lawrence Radiation Laboratory to demonstrate the feasibility of a nuclear reactor which would be able to propel a supersonic ramjet missile. Feasibility has been defined as the successful ground operation of a reactor, with the desired characteristics, for short periods of time together with suitable laboratory experiments to allow extrapolation to the desired lifetime.

The nuclear ramjet consists of an inlet diffuser followed by a single-pass, straight-through heat exchanger (the reactor) and an exhaust nozzle, Figure 1.2.1. The purpose of the diffuser is to reduce the velocity of the intake air and recover as much of the ram or stagnation pressure as possible. In passing through the reactor, heat is added to the air. The random molecular motion due to heating is changed into directed motion in the nozzle, resulting in an increase in air momentum and a net thrust for the missile.

1.2.1 TORY II-A Reactor

For a Mach-3 sea-level ramjet, air is supplied to the reactor at a pressure of approximately 350 psi and a temperature of about 1000°F. Flow rates greater than 2000 pounds per second can be achieved. It is apparent that the ground test facility, in particular the air supply, for testing such a reactor is large. It was decided, therefore, that the first ground test would be of a small reactor, so that the air supply and BeO reactor materials requirements would be minimized. The small, high-temperature, fueled core of this reactor, designated the TORY-II-A, would be tested in order to obtain data on the materials, physics, and engineering required for the design of a full-scale reactor. In order to enable the reactor to reach criticality, the core would be surrounded with a thick carbon reflector operating at room temperature. In addition, the reactor controls would be placed in the reflector so that the problem of operating control rods in a reactor at high temperature

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*UCRL-6923 The Pluto Program, by H. L. Reynolds, University of California, Lawrence Radiation Laboratory, Livermore, California, May 17, 1962.

UCRL-6305 TORY II-A Mechanical and Aero-Thermodynamic Design, by J. W. Cox and P. M. Uthe, University of California, Lawrence Radiation Laboratory, Livermore, California, February 2, 1962.

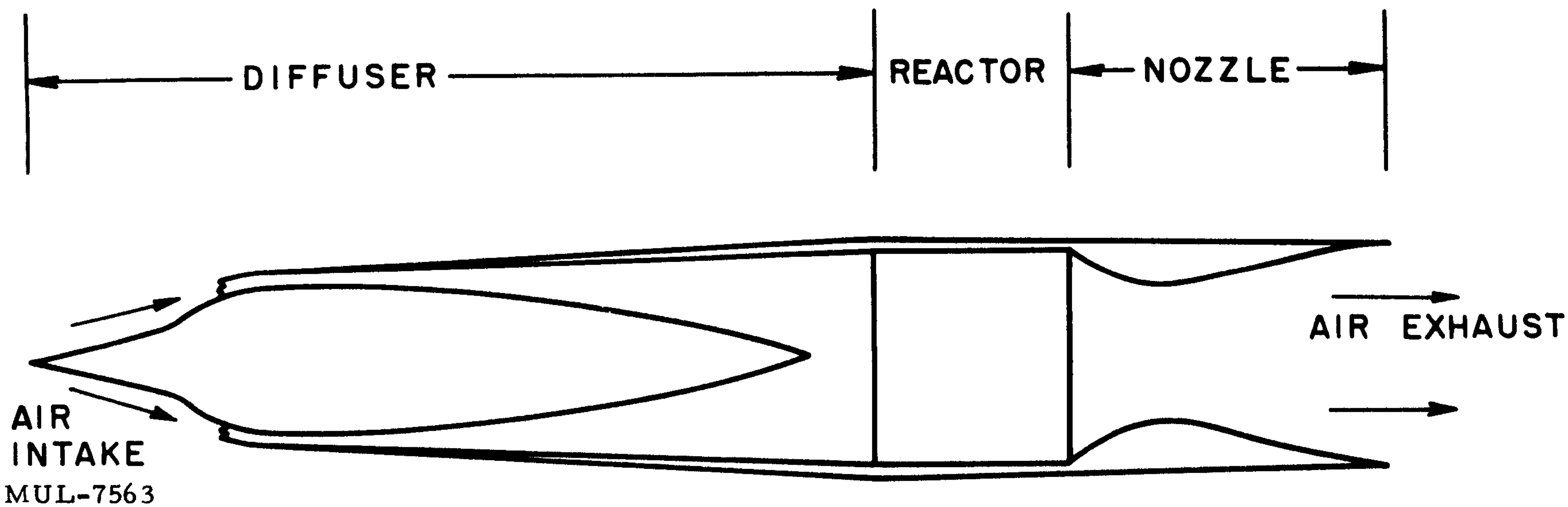

Fig. 1.2.1 - Conceptual arrangement of a nuclear ramjet

could be avoided in the first tests.

The main parameters of the TORY II-A reactor are listed below:

Power	155 megawatts
Flow rate	708 pounds/sec
Maximum fuel-element wall temperature	2250°F
Exit gas temperature (tube)	1975°F
Core diameter	32 inches
Core length	48 inches
Side reflector thickness (graphite)	24 inches

The reactor core contains approximately 100,000 hexagonal fuel elements 4 inches long, 297 mils across flats with 200-mil-diameter holes for the passage of air. These tubes are arranged in hexagonal bundles about 5 inches across. The bundles are contained in unfueled BeO structural elements as shown in Figure 1.2.2. A close up of a reactor section is shown in Figure 1.2.3. Typical ceramic components are shown in Figure 1.2.4. An air-cooled Hastelloy tube is placed in each corner of the hexagonal bundles. The Hastelloy tubes are attached to a massive front support structure. At the exit end of the reactor the Hastelloy tubes are attached to coated molybdenum base plates. The pressure drop through the fueled BeO tubes appears as a load on these base plates and is transmitted through the air cooled Hastelloy tubes to the front support structure. The core is cantilevered from the front support structure by an air-cooled Hastelloy shroud which is separated from the reflector by a water-cooled aluminum pressure shell. The graphite reflector is in two sections which can be separated horizontally so that the aluminum pressure shell can be removed from the reflector. The reflector contains eight graphite cylinders with boron steel at the outer edge of one quadrant of each. These cylinders rotate and control the reactor by increasing or decreasing the effective size of the reflector. Additional fast control is obtained from four linear rods placed near the inner wall of the reflector. All control elements are moved by hydraulic actuators and can be operated singly or in unison. The reflector is water-cooled.

The core during assembly is shown in Figure 1.2.5. The unfueled BeO structural elements and Hastelloy tubes can be seen. The dark tubes are fueled BeO. The painted dots are a code system to indicate the percent uranium content. The front (upstream end) of the core is shown in Figure 1.2.6.

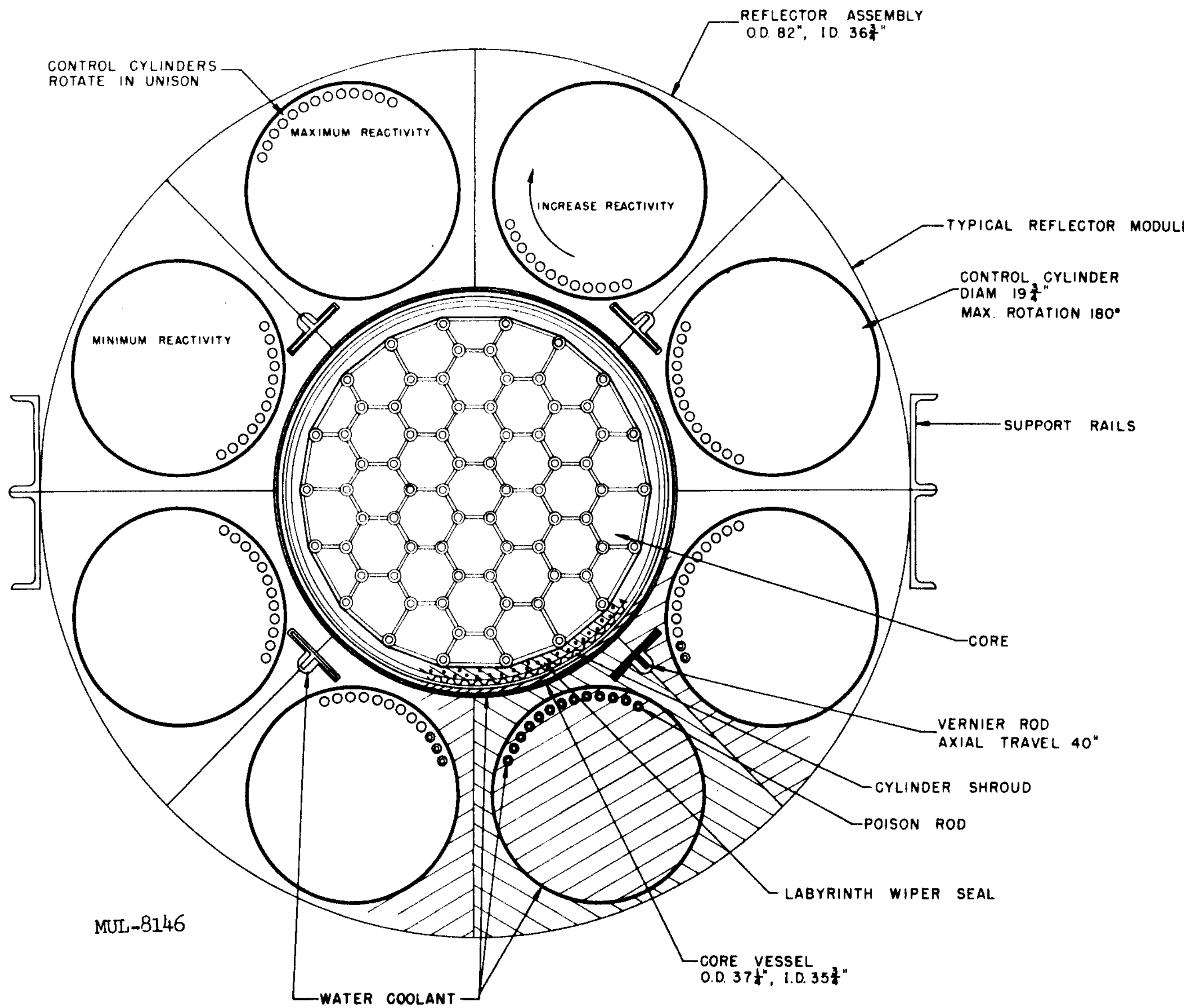

Fig. 1.2.2 - Cross section of TORY II-A core and reflectors

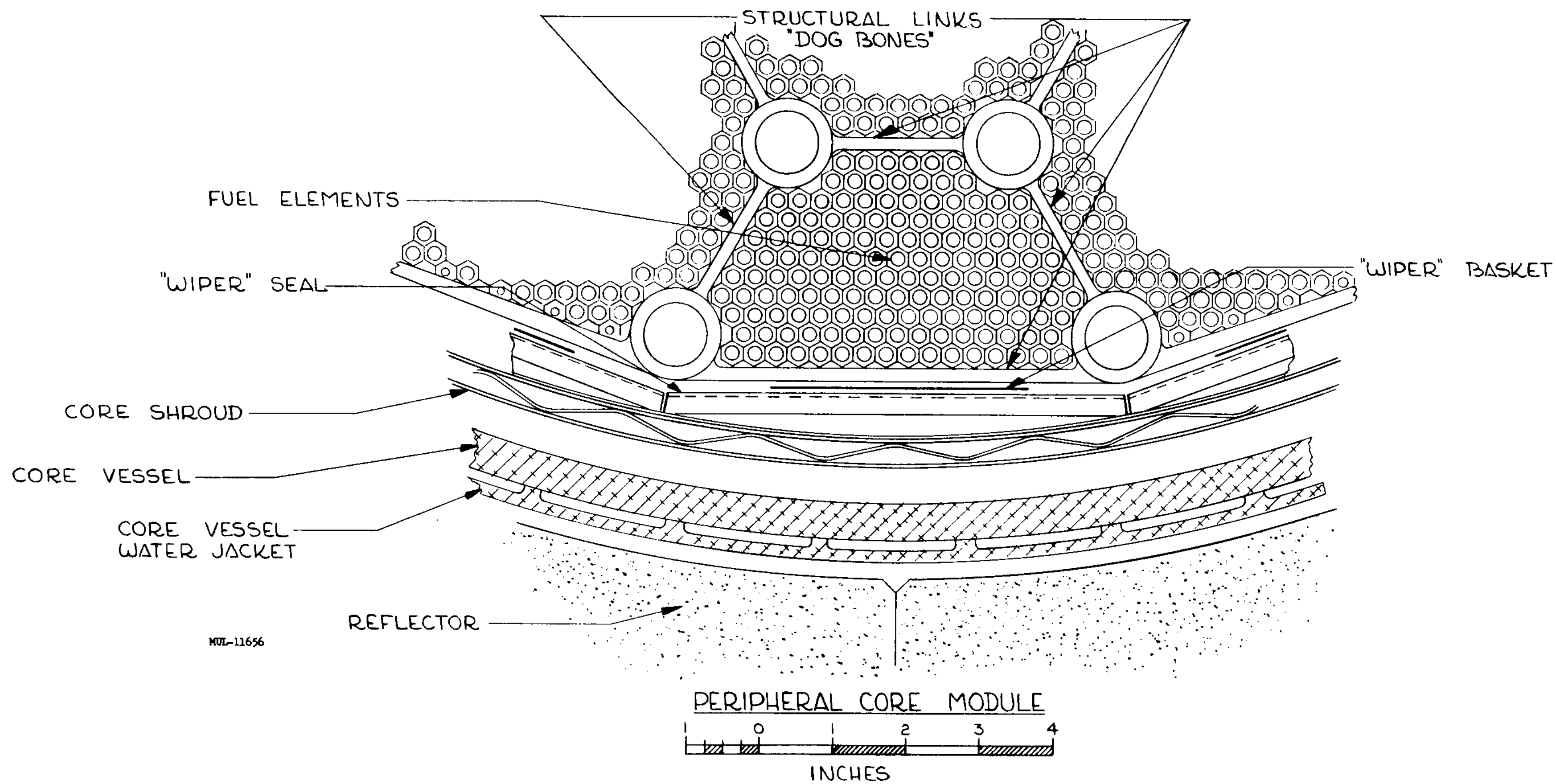

Fig. 1.2.3 - TORY II-A reactor core module

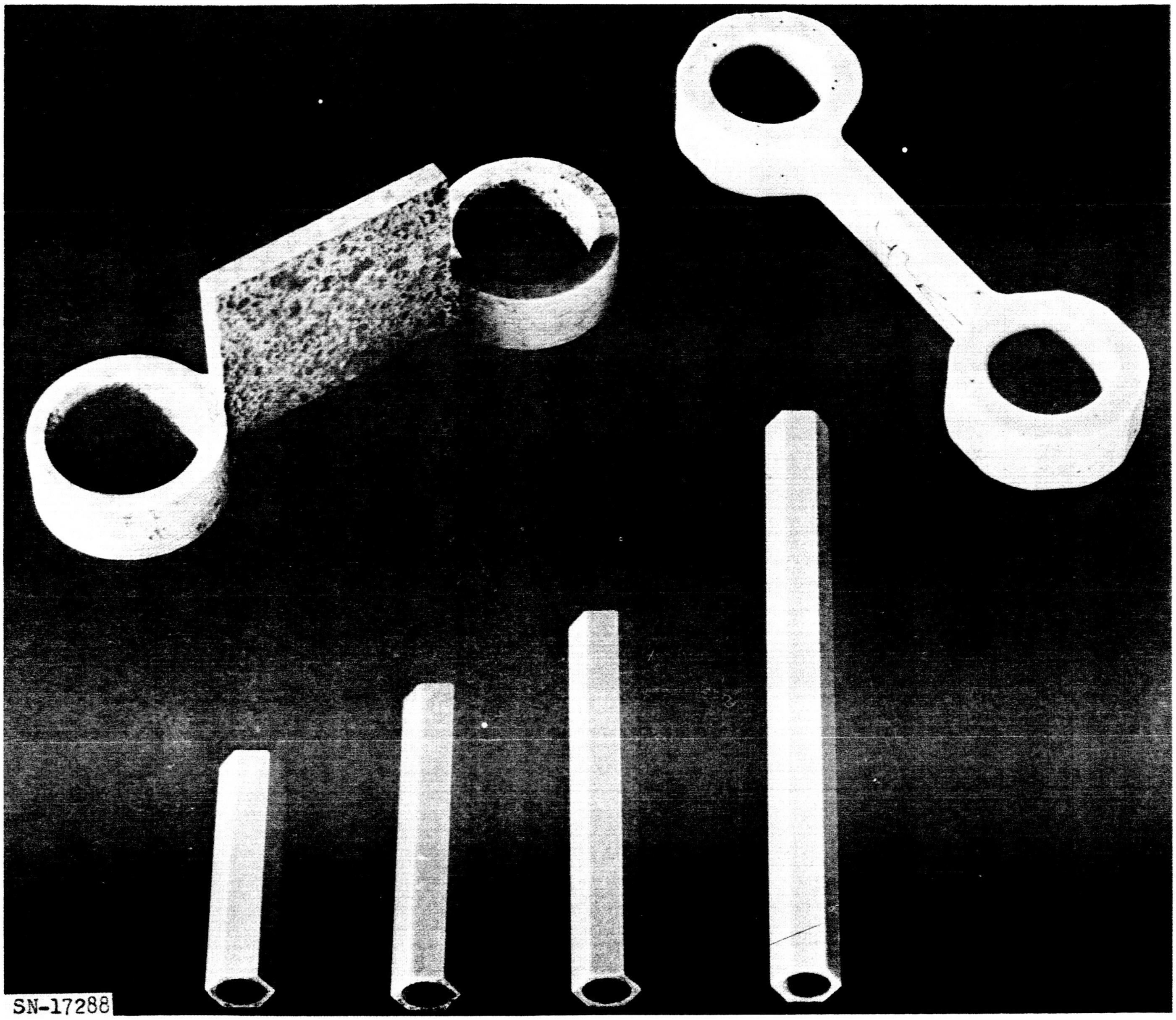

Fig. 1.2.4 - Typical ceramic components

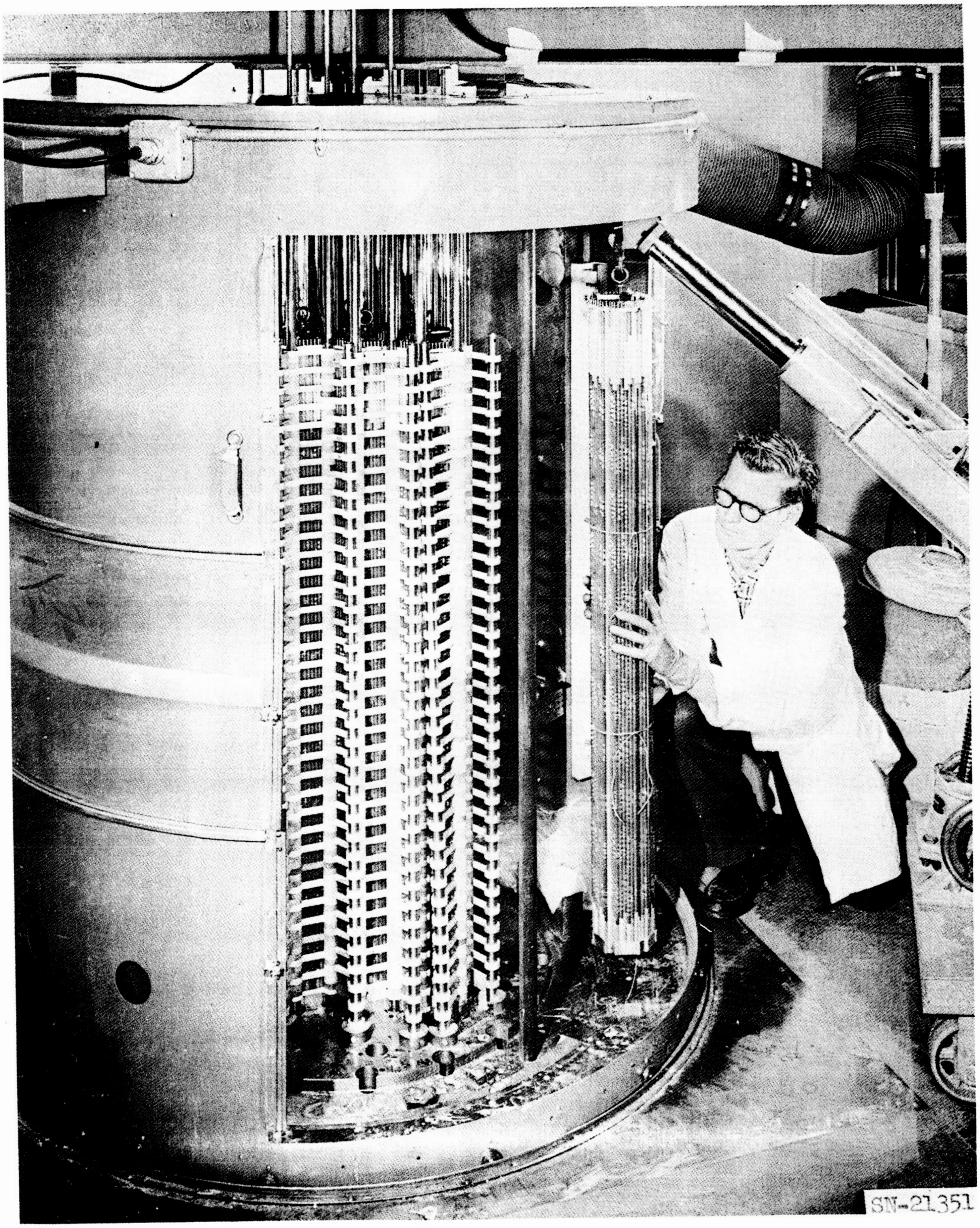

Fig. 1.2.5 - TORY II-A core during assembly

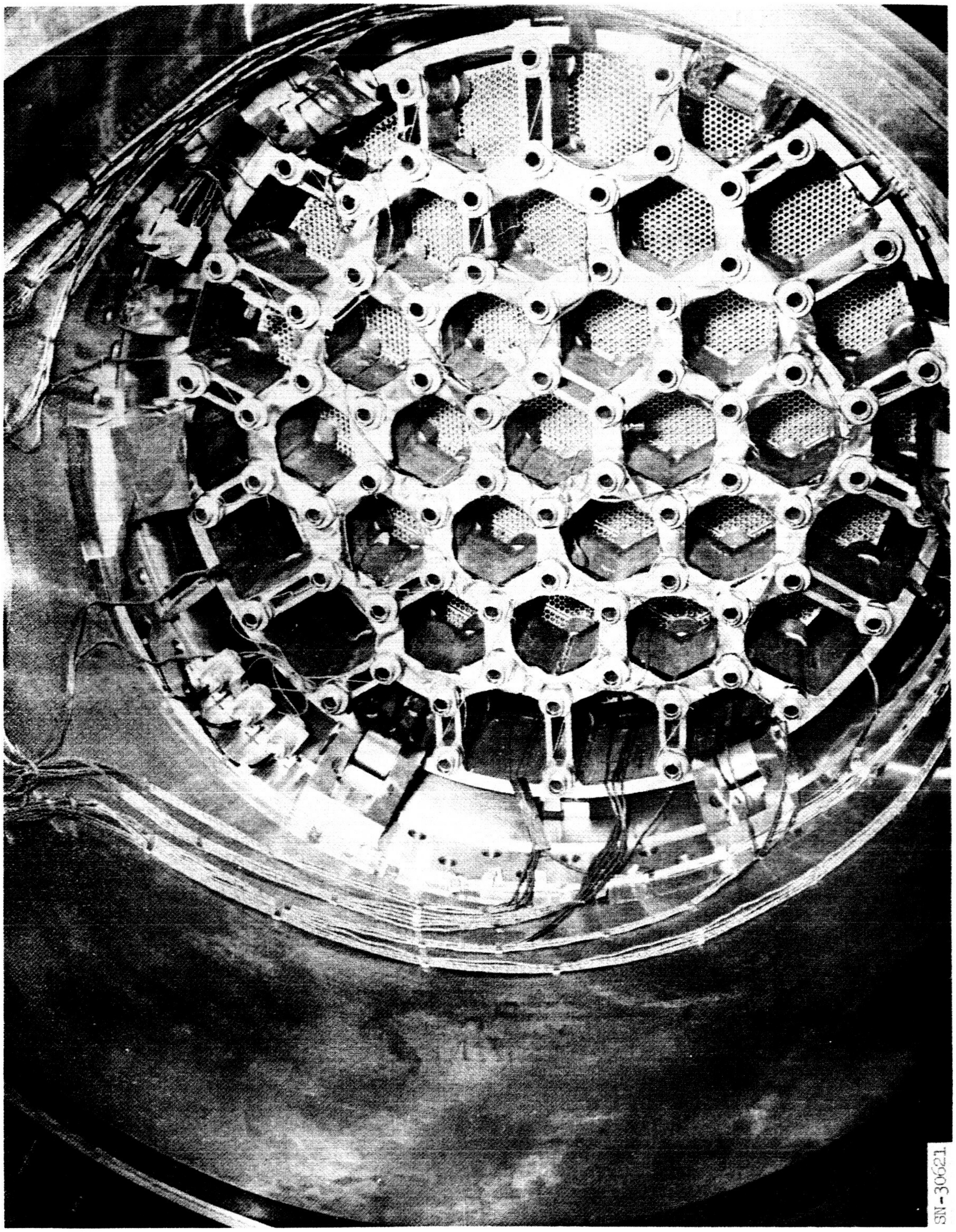

Fig. 1.2.6 - Front of TORY II-A core

The front support structure and Hastelloy tubes can be clearly seen. The aft end of the core is shown in Figure 1.2.7.

The test facility is located at the Nevada Test Site. The reactors are tested on unshielded railroad flat cars which can be remotely moved between a test bunker and a disassembly facility. Electrical, air, and water-cooling connections and disconnections to the test bunker can be made remotely. A schematic of the TORY II-A test vehicle assembly is shown in Figure 1.2.8. Air is supplied by blowing down previously pressurized air tanks.

The first high-power operation of TORY II-A took place in May 1961, a power level of 50 megawatts was achieved with temperatures in excess of 2000^{o}F. Necessary modifications to the bunker piping took place in the period from May to September. Additional power runs took place on September 28 and October 5 and 6.

1.2.2 TORY II-C Reactor

The TORY II-C reactor which will be tested in 1963 is a full-scale, missile-like reactor and is intended to demonstrate the feasibility of the Pluto ramjet reactor. A drawing of the TORY II-C test vehicle and ducting is shown in Figure 1.2.9. There will be no water-cooling on the test vehicle. The nozzle and reactor duct will be air-cooled. The reactor will be controlled by linear rods in the reactor core rather than by the rotary reflector controls used in TORY II-A.

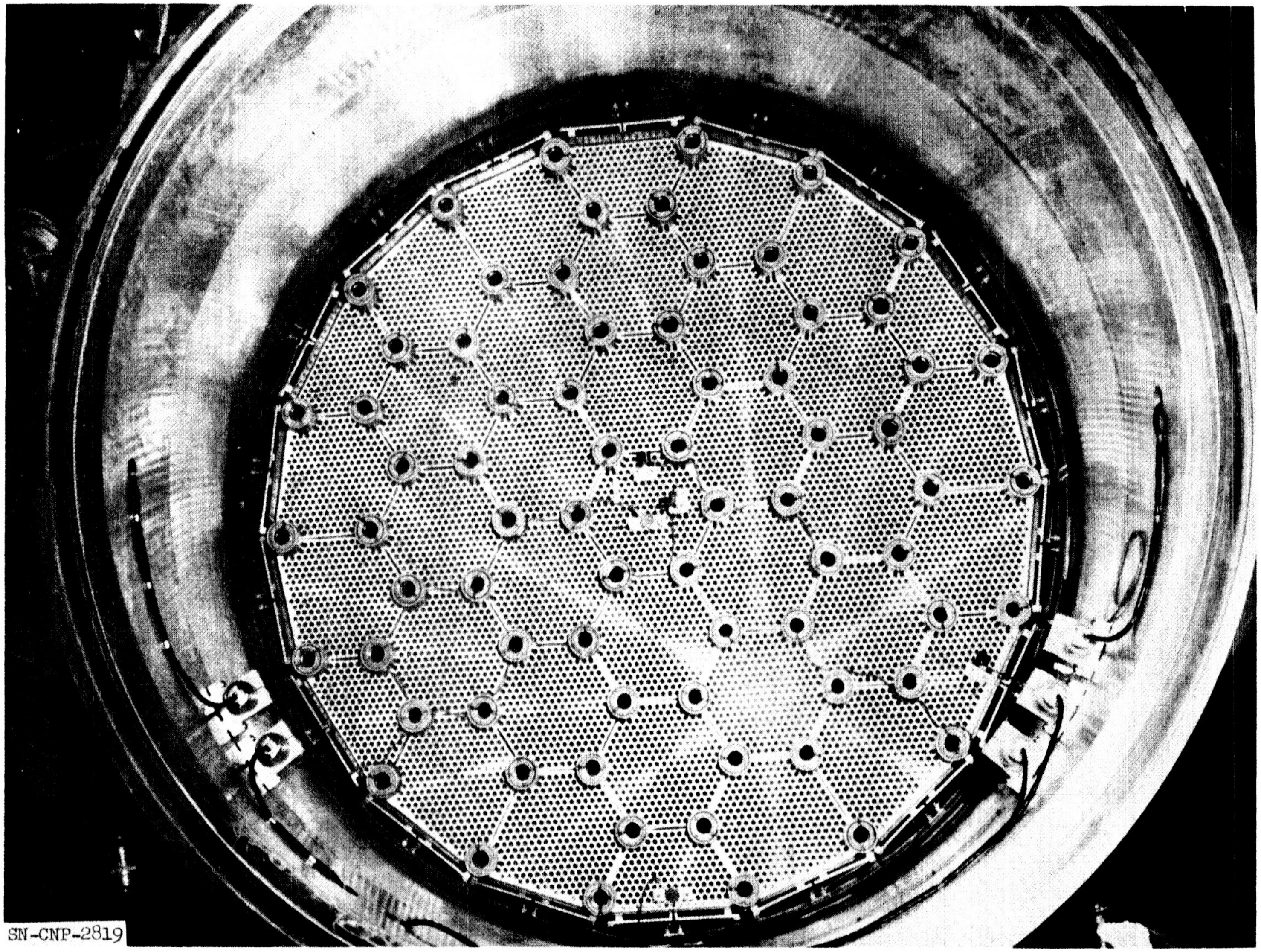

Fig. 1.2.7 - Exhaust face of TORY II-A reactor

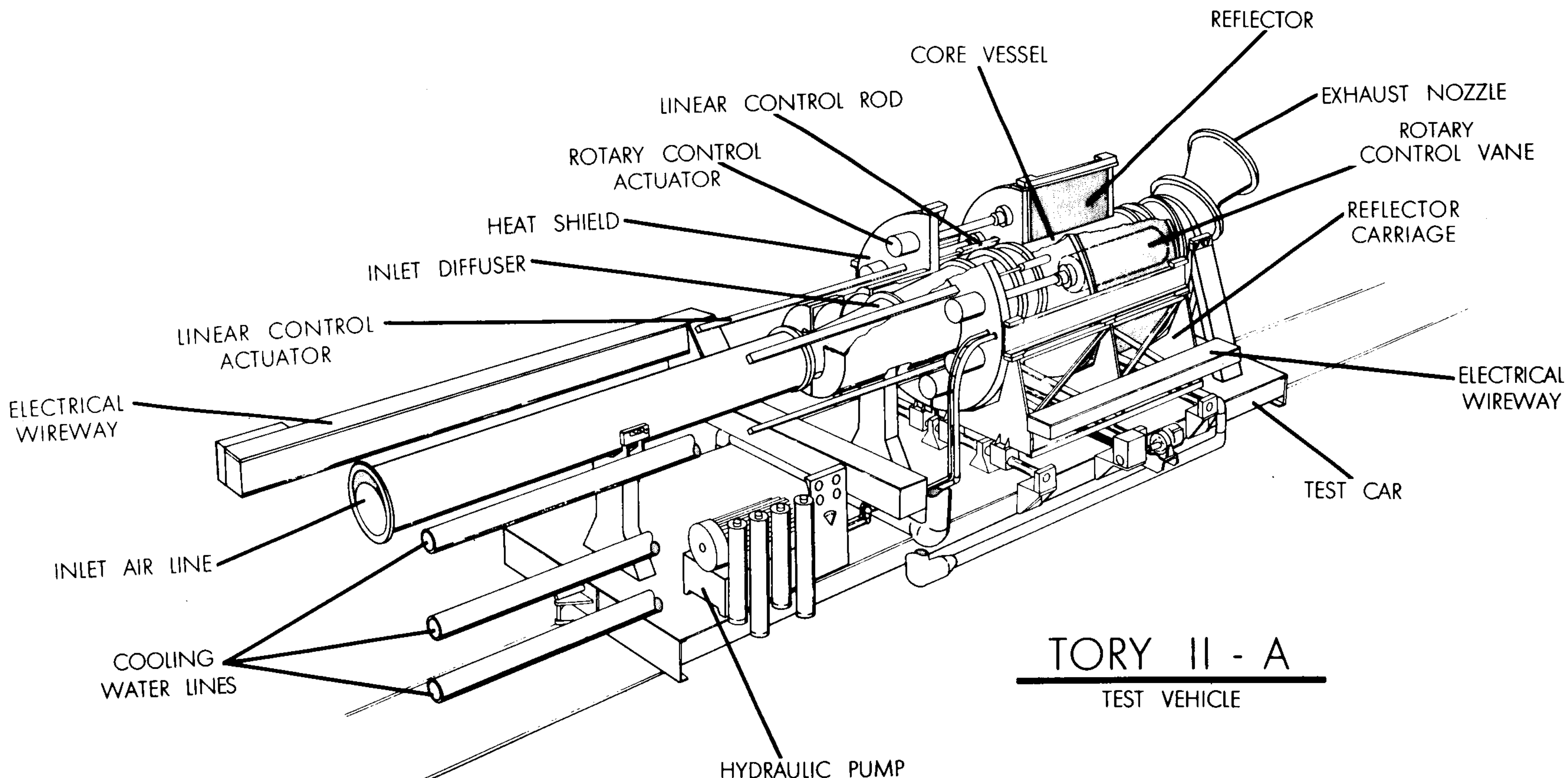

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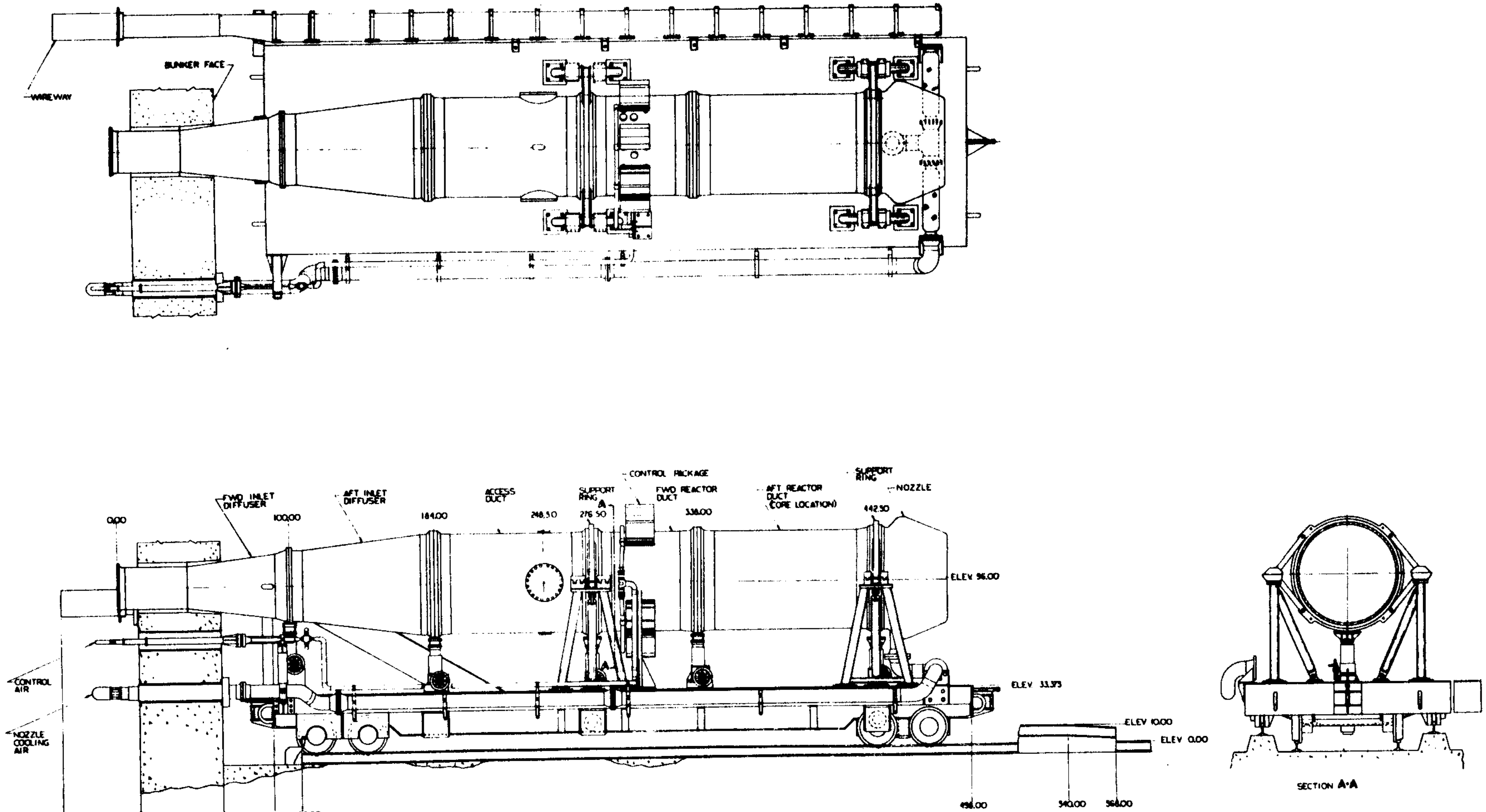
Fig. 1.2.8 - Test vehicle schematic



TORY IIC TEST VEHICLE

Fig. 1.2.9 - Schematic of the TORY II-C test vehicle

1.3 NUCLEAR ROCKET REACTORS*

The principal emphasis in the ROVER nuclear rocket program has been on the use of graphite reactors. A cross sectional view of what a typical graphite reactor might look like is shown in Figure 1.3.1. Since the details of the graphite rocket reactors are classified we will not include detailed illustrations in the lecture notes at this time.

The present status of reactor development for the ROVER program was summarized in a recent talk by H. B. Finger.* The following is quoted directly from this talk.

"The program started in 1955 at the Los Alamos Scientific Laboratory of the Atomic Energy Commission and it has continued to receive its major effort and emphasis at that Laboratory. After several years of comprehensive analysis and laboratory tests on materials, physics, heat transfer, dynamics and other disciplines, the Los Alamos Scientific Laboratory initiated the testing of complete research reactors, the KIWI-A reactor series, in 1959. Those tests were designed primarily to determine the ability of uranium-loaded graphite fuel elements to heat hydrogen to a temperature of interest for nuclear rocket propulsion. The three KIWI-A reactors which were tested in 1959 and 1960 provided enough confidence in the design techniques and the materials to permit us to go ahead with the KIWI-B reactor series. The KIWI-B reactor series is aimed at providing a basic reactor design which can lead directly, with continued engineering development effort, to a flight reactor system. The engineering development is to be done by the NERVA contractors who were brought into the program in July of 1961. The NERVA developers are Aerojet-General as the prime contractor, with the Westinghouse Astronuclear Laboratory as the principal subcontractor responsible for engineering the reactor portion of the NERVA engine.

"In the KIWI-B series of reactors, the Los Alamos Scientific Laboratory established several designs which represented different approaches to the solution of problems associated with the use of a brittle material in the environment of a nuclear rocket. The first of these, the KIWI-B1A reactor, was tested with gas coolant flow in December of 1961.

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*Ref. Missions for Nuclear Rockets; Remarks by Harold B. Finger, Manager, Space Nuclear Propulsion Office, AEC-NASA and Director, Nuclear Systems, NASA at the 31st annual meeting of the Institute of Aerospace Sciences, Hotel Astor, New York City, January 23, 1963.

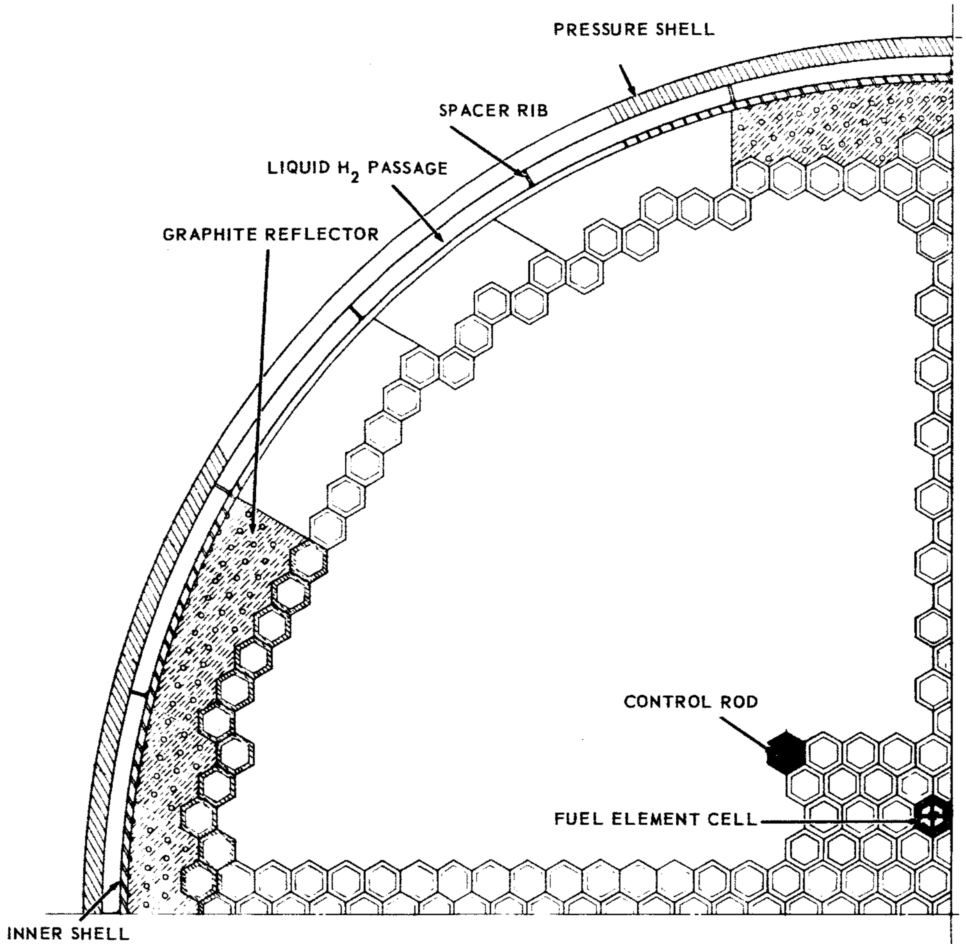

Fig. 1.3.1 - Typical graphite rocket reactor configuration

A similar reactor (KIWI-B1B) was then tested with liquid hydrogen inlet flow, as is required in a flight rocket engine, in September of 1962. A photograph of that reactor at the test cell is shown on the first slide.* This is the general configuration of the test setup of all reactors run to date. They have been fired with the exhaust jet pointing upward to simplify the facility installation. The nozzle in this test was regeneratively cooled with liquid hydrogen. The results of this test indicated that the reactor could be started stably with liquid hydrogen. However, in this KIWI-B1B design, damage occurred in the reactor core similar to damage that had occurred in certain of the KIWI-A tests. The fact that this damage has not been explained through extensive laboratory tests and analysis has made us discard the KIWI-B1 design, for the present, as a candidate for the NERVA engine. It is important to point out that prior to this test, the decision had been made to proceed with the KIWI-B4 type reactor design as a basis for the first NERVA reactor design. This decision was made on the basis of the best available analytical and laboratory experimental data and the fact that the KIWI-B1 type of design had failed in the KIWI-A tests and the failure was not explained. It was also based on the belief that the KIWI-B4 provided greater margin of performance for operation of the reactor.

"The first of the KIWI-B4 reactors, the KIWI-B4A, was tested by Los Alamos in November of 1962. A photograph of that reactor is shown on the next slide.* It is externally very similar to the KIWI-B1 reactor; however the core design is substantially different. Almost as soon as the test of the KIWI-B4A reactor was started, flashes of light were noted in the exhaust jet. These flashes of light were an indication that material from the core was being carried away in the jet. However, the test was continued until the frequency of these flashes became so rapid that it was apparent that more could be learned by shutting down and examining the reactor than by trying to barge ahead to design power conditions. Upon examination, it was found first, that a thermal insulation component around the reactor had broken and parts of those thermal insulation components had been ejected from the reactor. Upon continued disassembly, it was found that fuel elements had been cracked and, now that the disassembly is complete, it is apparent that there was extensive damage in the reactor. The data indicate that vibrations probably took place in the reactor and that the vibrations were probably generated within the reactor. Work is now actively underway by Los Alamos and

*This illustration is not reproduced in these notes but is available in the reference.

Westinghouse to modify the mechanical design so as to reduce to a minimum the possibility of such mechanical vibrations. Although there is very strong feeling among all participants of the program that this vibration is the explanation for the damage, we have determined that before further full-scale, hot tests are run, component, subassembly, and full-scale mechanical testing and cold-flow testing will be conducted to evaluate the failure mode hypothesis that has been made of the KIWI-B4A reactor damage. Such testing will also be conducted to check the suitability of redesigns of that reactor to overcome the mechanical difficulties experienced.

The current status of the reactor program is, therefore, that major accomplishments have been made by the Los Alamos Scientific Laboratory in the development of the materials technology required for nuclear rocket reactors including the development of techniques for fabricating fuel elements and protecting them in a hydrogen environment. In addition, Los Alamos has made major progress in evaluating and accomplishing the start-up of a nuclear reactor rapidly with liquid hydrogen. As part of this phase of the program, they have gone a long way toward establishing the control parameters and control methods for nuclear rocket reactors. The nuclear physics aspects of these reactors are well understood and designs can proceed with a fairly high level of confidence in this area. The greatest area of concern at the present time is in the area of mechanical engineering design of the reactor within known limitations of the materials, physics, and heat transfer processes. Through a thorough design and test effort, I am convinced our program will lead to a successful nuclear rocket reactor of the general type that we have been discussing for use in our NERVA engine and RIFT flight test programs.

"A full-scale mock-up of the NERVA engine is shown on the next slide* The engine stands 22 feet high. Shown in the slide are the reactor, the regeneratively-cooled nozzle, the control drum actuators, and the thrust structure at the top of the engine. The turbopump, shut-off valve, and gimbal bearing about which the entire engine may be swiveled to direct the thrust vector are mounted within the upper thrust structure section. The large spheres at the top of the engine are pressurizing gas bottles used as a drive source for the pneumatic actuators in the system. These bottles are refilled during operating cycles of the engine.

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*This illustration is not reproduced in these notes but is available in the reference.

"The reactor used in the NERVA engine will be a direct outgrowth of KIWI reactor work. These reactors are already very similar in design and are becoming more similar as Westinghouse and Los Alamos cooperate and collaborate in the design modifications, to be made as a result of the KIWI-B4A experiment. We are, therefore, conducting a single reactor development program in which all tests will be directed toward the goal of achieving a reliable NERVA engine.

"The objective of the RIFT program is to flight test this NERVA engine. In addition, the objective of the RIFT program is to try to flight test the stage which should, with continued development, lead to an operational stage on a Saturn V vehicle. The RIFT stage is being developed by the Lockheed Missiles and Space Company. A drawing of the RIFT stage is shown in the next slide*. The stage will be 33 feet in diameter, the same diameter as the Saturn V vehicle, and it will stand approximately 80 feet high from the base of the engine to the top of the stage itself. With a nose cone added, the total stage will be about 137 feet tall. The trajectories for the RIFT flight have not yet been established, but the plan is to boost the RIFT stage by the first stage of the Saturn V using a dummy second stage and fly the RIFT stage over a limited range trajectory impacting in deep Atlantic Ocean water."

In addition to the talk by Mr. Finger, recent papers on nuclear rockets are given in the following.

NASA SP-II Proceedings of the NASA-University Conference on the Science and Technology of Space Exploration, Volume 2, Chicago, Illinois, November 1 - 3, 1962.

Session N - Nuclear Propulsion; Chairman: David S. Gabriel

Introduction - Davis S. Gabriel

Advanced Concepts for Nuclear Rocket Propulsion - Frank E. Rom and Robert G. Ragsdale

Nuclear Physics of Solid-Core Gas-Cooled Rocket Propulsion Reactors - Donald Bogart and Edward Lantz

Fluid Flow and Heat Transfer Problems in Nuclear Rockets - Herman Ellerbrock, John N. B. Livingood, and David M. Straight

*This illustration is not reproduced in these notes but is available in the reference.

Problems in Dynamics and Control of Nuclear Rockets - John C. Sanders, Herbert J. Heppler, Jr., and Clint E. Hart.

This compilation also provides a great deal of useful and interesting information about the non-nuclear aspects of rocket propulsion.

1.4 TECHNICAL DIFFERENCES BETWEEN CHEMICAL AND NUCLEAR SYSTEMS

1.4.1 Energy Conversion and Heat Transfer Concepts

The primary difference between chemical and nuclear propulsion systems in the manner in which heat is added to the working fluid. In chemical systems, the combustion process takes place in the propellant, which is heated by absorbing the kinetic energy and thermal radiation of the combustion products. The combustion products then become part of the propellant and are discharged in the jet. Presumably, the propellant in a nuclear system could be heated in a similar manner. However, most concepts that have been devised to achieve a nuclear fission process within a moving propellant result in an excessive loss of unused uranium and the release of radioactive fission products to the atmosphere. Consequently, although some progress has been made on systems which have intimate contact between the propellant and the fission products and their high temperature thermal radiation, most designs have been confined to nuclear systems in which the fissionable material is retained by containment in reactor fuel elements. In this approach, the kinetic energy of the fission products is absorbed by the fuel element materials and the resultant heat is transferred to the working fluid.

1.4.2 Propulsion Machinery

The same factors, in general, determine the choice of propulsion machinery in both nuclear and chemical systems. Turbojet engines are well suited for a wide range of subsonic and supersonic speeds. Propeller or ducted-fan variations of the turbojet provide superior performance at lower subsonic speeds, especially at takeoff. Ramjets are useful at speeds of about Mach 3 or higher, where sufficient ram-air compression is provided without the use of a turbine-driven compressor. Rockets are needed outside the planetary atmosphere and for extremely high speeds.

1.4.3 Thrust

Thrust is highly dependent on propellant temperature. Temperatures in excess of $5000^{\circ}F$ are achievable in chemical combustion processes using stoichiometric mixtures of oxidizer and fuel. However, in an air breathing system the maximum propellant temperature which can be realized is somewhat in excess of $3000^{\circ}F$ because of the dilution of the combustion products with atmospheric nitrogen. In a chemical rocket, full stoichiometric temperatures are theoretically achievable. However, since the specific impulse (lbs. of thrust per lbs. of propellant flow per second) is a function both of the temperature and molecular weight of

the propellant, a more optimum chemical system is obtained by using a non-stoichiometric mix with an excess of low molecular weight fuel such as hydrogen which dilutes the combustion products and lowers the temperature.

The temperatures to which a propellant can be heated by fission products is theoretically far in excess of that which can be achieved in chemical systems. However, no practical method has been developed for achieving these potentially higher propellant temperatures. The only nuclear systems which have been developed to date are those in which the fission process takes place within a solid fuel element material. The propellant temperature is therefore limited by the temperature at which the reactor material may be allowed to operate. Current nuclear materials technology limits achievable propellant temperatures to somewhat lower values than those achievable with chemical systems. Future materials development may make it possible to achieve propellant temperatures approximating but not exceeding those achievable with chemical systems. Hence, unless methods can be developed which can transfer the high initial energy of fission products directly into the propellant, nuclear systems are competitive with but do not appear to have an advantage over chemical systems from the viewpoint of the propellant temperatures which can be achieved.

The thrust advantage of the nuclear system does not therefore come from temperature, but rather from the fact that since it can use a low molecular weight propellant, its thrust per unit of propellant weight flow, i.e., the specific impulse, is high.

Weight

A weight comparison of nuclear and chemical systems will generally be reduced to a comparison of chemical fuel and propellant weight and nuclear shield weight.

Both chemical and nuclear systems operating in the upper atmosphere or in space will require some type of shielding of personnel and sensitive equipment against atmospheric heating, natural radiation, and high altitude bomb bursts. Nuclear systems require additional shielding against radiation of reactor origin. In the lower atmosphere the shield weight favors the chemical system, whereas at high altitude or in space, the difference in shield weights will be reduced but will still favor the chemical system. The magnitude of the shield weight difference depends on the duration of the mission and the nature of the subject which must be shielded. Manned vehicles require more shielding than unmanned

vehicles for protection against both reactor radiation and external radiation. For unamnned systems of short duration the difference in shield weight of nuclear and chemical systems may be insignificant.

The primary weight advantage of the nuclear system for air breathing engines is the fact that the fuel weight is essentially non-existent as compared to the engine weight, whereas for chemical systems the fuel load is generally much greater than the engine weight. The fuel load for chemical propulsion depends directly on the mission duration and the rate of fuel consumption. However; mission duration and the rate of energy utilization have only a fractional effect on the weight of nuclear systems.

The propellant weight is of concern only in rockets. Since an oxidizer is not needed with a nuclear system the propellant can consist entirely of low molecular weight material such as hydrogen. Hence, a significantly better specific impulse is achievable. Consequently, unless this advantage is offset by other weight penalties, a given mission can be performed with a smaller propellant load with a nuclear rocket than with a chemical rocket.

Generally speaking, from the weight viewpoint, missions requiring long endurance with intermediate or high rates of energy utilization will probably favor nuclear over chemical propulsion. Missions of short duration or with low rates of energy utilization will favor the chemical system over the nuclear.

1.4.4 Operational Considerations

Flight operational procedures for nuclear systems will be essentially identical with chemical systems. During flight operation, the crew will detect little difference between chemical and nuclear propulsion from the viewpoint of system response and behavior. Most future aerospace missions will require much closer control of environmental conditions for personnel and equipment. Confined crew quarters will be common whether or not radiation shielding is provided. The operation of the entire vehicle will generally be controlled from these confined quarters with increased use of automatic devices for both chemical and nuclear systems.

The principal operational difference between the nuclear and chemical systems will be in ground operation and maintenance. Experience in the ANP program in the operation and maintenance of turbojet engines operating with nuclear heat sources indicates that operating and maintenance

personnel quickly adapt themselves to working in an environment where the additional hazard of radiation is present. Nevertheless, the fact that special equipment and procedures are needed for such operation and maintenance will undoubtedly result in the selection of chemical systems in some cases where nuclear systems would otherwise be more desirable.

1.4.5 Summary

Consideration of the performance, weight, operational and other factors will favor the choice of nuclear propulsion systems for some applications and chemical systems for others. In some cases combinations of chemical and nuclear propulsion will be desirable.

The application of a nuclear reactor to rocket propulsion is limited by the supply of rocket propellant that must be carried, even though the consumption of nuclear fuel is negligible. Nevertheless, since the propellant can consist entirely of a substance of low molecular weight, such as hydrogen, a heat transfer nuclear rocket has a potential advantage of a factor of approximately 2 in specific impulse. In other words, a given thrust level can be sustained twice as long by a nuclear rocket as compared with a chemical rocket for the same weight of propellant. Alternately, the given thrust level may be sustained for the same period of time but with a much lower mass of stored propellant and consequently lower overall weight. Since this weight must be accelerated, a much higher velocity can be attained with the nuclear system when the thrust is applied over a fixed period. This provides a decided performance advantage for the nuclear system for a number of potential space missions.

In air-breathing nuclear systems, such as turbojets and ramjets, a given thrust level can be sustained virtually indefinitely because the supply of propellant (air) is unlimited. However, since most air-breathing missions do not require acceleration to extremely high velocities, the fuel consumption during acceleration is usually not prohibitively large. Furthermore, most air-breathing missions operate within reach of additional fuel supplies. Consequently, the advantage of nuclear power for rockets appears to be a more critical one that is the case for air-breathing systems.

1.5 REFERENCES

Additional useful references in the field of nuclear propulsion are listed below. These references also provide data on the non-nuclear aspects of propulsion and aerospace missions.

1. Gantz, K. F., ed., "Nuclear Flight - The United States Air Force Programs for Atomic Jets, Missiles, and Rockets", Duell, Sloan & Pearce.
2. Bussard, R. W. & DeLauer, R. D., "Nuclear Rocket Propulsion", McGraw-Hill, 1958.
3. Siefert, H. S., ed., "Space Technology", Wiley, 1959.
4. Corliss, W. R., "Propulsion systems for space flight", McGraw-Hill, 1960.
5. Thring, M. W., ed., "Nuclear Propulsion", Butterworths, London, 1960.