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COMPREHENSIVE TECHNICAL REPORT

DIRECT AIR CYCLE | AIRCRAFT NUCLEAR PROPULSION PROGRAM

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PROGRAM SUMMARY AND REFERENCES

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FLIGHT PROPULSION LABORATORY DEPARTMENT

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GENERAL ELECTRIC DIRECT-AIR-CYCLE AIRCRAFT NUCLEAR PROPULSION PROGRAM

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Authors: G. THORNTON
A. J. ROTHSTEIN
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(Formerly Aircraft Nuclear Propulsion Department)
FLIGHT PROPULSION LABORATORY DEPARTMENT
Cincinnati 15, Ohio

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ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume discusses the background to the General Electric program, and summarizes the various direct-air-cycle nuclear test assemblies and power plants that were developed. Because of the requirements of high performance, low weight, and small size, vast improvements in existing technology were required to meet the flight objectives. The technological progress achieved during the program is also summarized.

The last appendix contains a compilation of the abstracts, tables of contents, and reference lists of the other twenty volumes.

ACKNOWLEDGMENT

Much of the material in this volume was originally prepared for other volumes of the Comprehensive Technical Report. Additional contributions were made by J. I. Trussell and G. F. Hamby.

Special acknowledgement is given to D. R. Shoults, M. C. Leverett, W. H. Long, J. W. Morfitt, E. B. Delson, H. C. Brassfield, and W. J. Condon for helpful comments.

PREFACE

In mid-1951, the General Electric Company, under contract to the United States Atomic Energy Commission and the United States Air Force, undertook the early development of a militarily useful nuclear propulsion system for aircraft of unlimited range. This research and development challenge to meet the stringent requirements of aircraft applications was unique. New reactor and power-plant designs, new materials, and new fabrication and testing techniques were required in fields of technology that were, and still are, advancing very rapidly. The scope of the program encompassed simultaneous advancement in reactor, shield, controls, turbomachinery, remote handling, and related nuclear and high-temperature technologies.

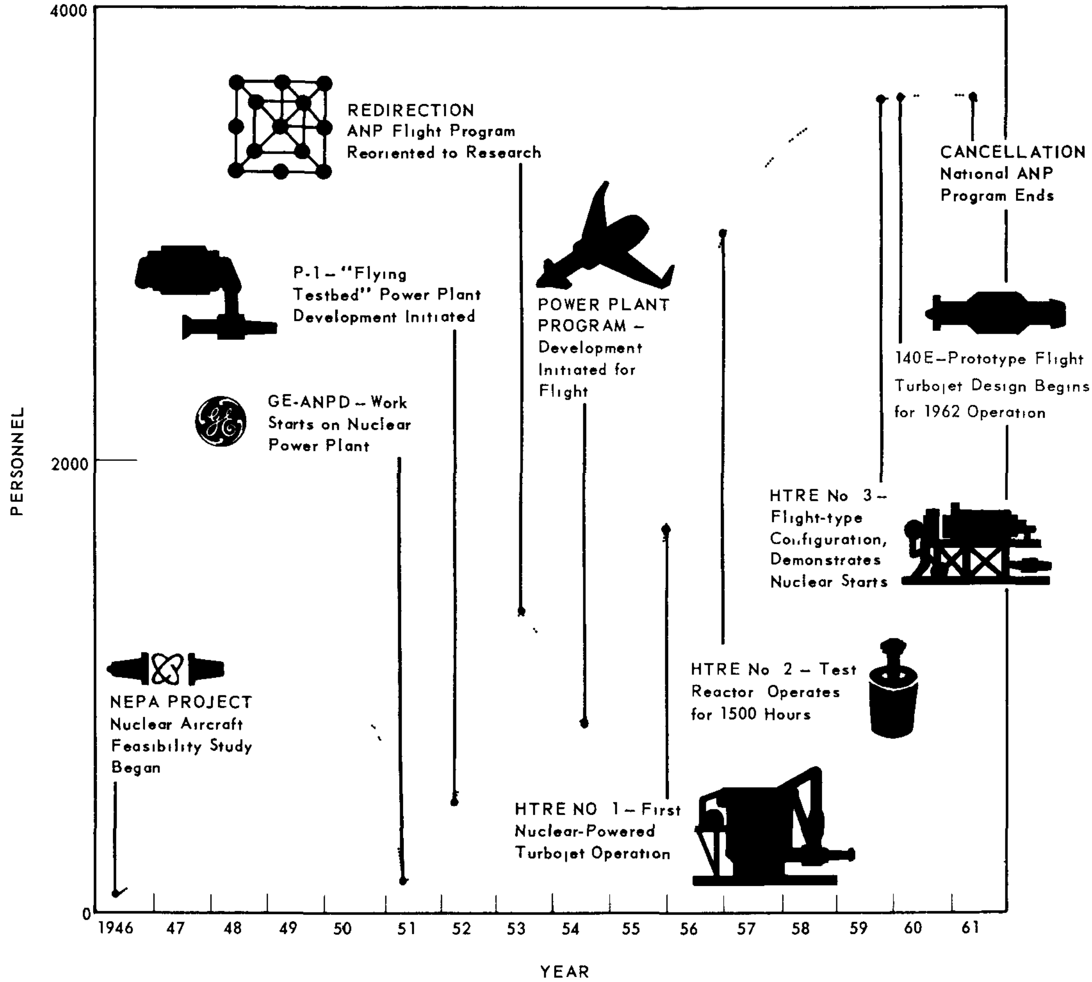
The power-plant design concept selected for development by the General Electric Company was the direct air cycle turbojet. Air is the only working fluid in this type of system. The reactor receives air from the jet engine compressor, heats it directly, and delivers it to the turbine. The high-temperature air then generates the forward thrust as it exhausts through the engine nozzle. The direct air cycle concept was selected on the basis of studies indicating that it would provide a relatively simple, dependable, and serviceable power plant with high-performance potential.

The decision to proceed with the nuclear-powered-flight program was based on the 1951 recommendations of the NEPA (Nuclear Energy for the Propulsion of Aircraft) project. Conducted by the Fairchild Engine and Airplane Corporation under contract to the USAF, the five-year NEPA project was a study and research effort culminating in the proposal for active development of nuclear propulsion for manned aircraft.

In the ensuing ten years, General Electric's Aircraft Nuclear Propulsion Department carried on the direct air cycle development until notification by the USAF and USAEC, early in 1961, of the cancellation of the national ANP program. The principal results of the ten-year effort are described in this and other volumes listed inside the front cover of the Comprehensive Technical Report of the General Electric Direct Air Cycle Aircraft Nuclear Propulsion Program.

Although the GE-ANPD effort was devoted primarily to achieving nuclear aircraft power-plant objectives (described mainly in APEX-902 through APEX-909), substantial contributions were made to all aspects of gas-cooled reactor technology and other promising nuclear propulsion systems (described mainly in APEX-910 through APEX-921). The Program Summary (APEX-901) presents a detailed description of the historical, programmatic, and technical background of the ten years covered by the program. A graphic summary of these events is shown on the next page.

Each portion of the Comprehensive Report, through extensive annotation and referencing of a large body of technical information, now makes accessible significant technical data, analyses, and descriptions generated by GE-ANPD. The references are grouped by subject and the complete reference list is contained in the Program Summary, APEX-901. This listing should facilitate rapid access by a researcher to specific interest areas or



Summary of events – General Electric Aircraft Nuclear Propulsion Program*

*Detailed history and chronology is provided in Program Summary APEX-901 Chronology information extracted from Aircraft Nuclear Propulsion Program hearing before the Subcommittee on Research and Development of the Joint Committee on Atomic Energy 86th Congress of The United States First Session July 23 1959 United States Government Printing Office Washington 1959 sources of data. Each portion of the Comprehensive Report discusses an aspect of the Program not covered in other portions. Therefore, details of power plants can be found in the power-plant volumes and details of the technologies used in the power plants can be found in the other volumes. The referenced documents and reports, as well as other GE-ANPD technical information not covered by the Comprehensive Report, are available through the United States Atomic Energy Commission, Division of Technical Information Extension, Oak Ridge, Tennessee.

The Report is directed to Engineering Management and assumes that the reader is generally familiar with basic reactor and turbojet engine principles; has a technical understanding of the related disciplines and technologies necessary for their development and design; and, particularly in APEX-910 through APEX-921, has an understanding of the related computer and computative techniques.

The achievements of General Electric's Aircraft Nuclear Propulsion Program were the result of the efforts of many officers, managers, scientists, technicians, and administrative personnel in both government and industry. Most of them must remain anonymous, but particular mention should be made of Generals Donald J. Keirn and Irving L. Branch of the Joint USAF-USAEC Aircraft Nuclear Propulsion Office (ANPO) and their staffs; Messrs. Edmund M. Velten, Harry H. Gorman, and John L. Wilson of the USAF-USAEC Operations Office and their staffs; and Messrs. D. Roy Shoults, Samuel J. Levine, and David F. Shaw, GE-ANPD Managers and their staffs.

This Comprehensive Technical Report represents the efforts of the USAEC, USAF, and GE-ANPD managers, writers, authors, reviewers, and editors working within the Nuclear Materials and Propulsion Operation (formerly the Aircraft Nuclear Propulsion Department). The local representatives of the AEC-USAF team, the Lockland Aircraft Reactors Operations Office (LAROO), gave valuable guidance during manuscript preparation, and special appreciation is accorded J. L. Wilson, Manager, LAROO, and members of his staff. In addition to the authors listed in each volume, some of those in the General Electric Company who made significant contributions were: W. H. Long, Manager, Nuclear Materials and Propulsion Operation; V. P. Calkins, E. B. Delson, J. P. Kearns, M. C. Leverett, L. Lomen, H. F. Matthiesen, J. D. Selby, and G. Thornton, managers and reviewers; and C. L. Chase, D. W. Patrick, and J. W. Stephenson and their editorial, art, and production staffs. Their time and energy are gratefully acknowledged.

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1. INTRODUCTION AND PROGRAM BACKGROUND

1.1 INTRODUCTION

The General Electric Aircraft Nuclear Propulsion Program started in 1951 following feasibility studies performed primarily by the NEPA Project of the Fairchild Engine and Airplane Company. The work was performed under simultaneous contracts with the United States Air Force and the United States Atomic Energy Commission.

Air Force Contract

Air Force Contract No. AF33(038)-21102 required that the General Electric Company should accomplish "a development program and the manufacture and ground testing of a nuclear power plant suitable for testing at the earliest feasible date." In this connection, it was expected that there would be "a series of nuclear power plants" each of which was to "employ turbojet engines operated in conjunction with the selected nuclear reactor." It was also stated that "successive power plants in the series shall have improving performance to the end that the final one fabricated under this contract shall be suitable for testing in a military prototype nuclear aircraft."

In addition, the General Electric Company was to "carry out secondary work including theoretical, analytical, design and experimental studies on alternate types of propulsion systems and components in order to determine their relative merits for future development."

AEC Contract

The Atomic Energy Contract No. AT(11-1)-171 stated that "It is the objective of the Commission to develop, within the shortest practicable time, a nuclear reactor which, in conjunction with propulsion equipment, will fulfill the Air Force's requirements for the propulsion of aircraft." It was expected that "Attainment of this objective will require the erection and operation of several preliminary nuclear reactors."

More explicitly, General Electric would undertake "such research and development work on reactors as may be necessary or desirable to establish suitable design and specifications" and would also undertake "fabrication, assembly and testing of reactors and their component parts."

Furthermore, "secondary work" was to be performed on "alternate types of nuclear reactors" consisting of "theoretical, analytical, design, and experimental studies to determine the merits of the alternate types for future development."

The program proceeded in accordance with these contractual statements until its termination in March of 1961.

Scope of the General Electric Program

As required by the AEC contract, a series of preliminary reactors were developed and tested. 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 continually improved reactor materials and components. Several power plants suitable for test and operation in military aircraft were designed to meet specific military objectives. Each of the objectives was withdrawn, in accordance with the constantly evolving requirements of the national defense establishment, prior to final fabrication and operation of a prototype unit.

The power plant under development at program termination, the XNJ140E nuclear turbojet (Figure 1.1) was designed in accordance with Department of Defense guidance for a nuclear propulsion system capable of propelling a Convair NX2 (Figure 1.2) or equivalent aircraft at high subsonic speeds for 1000 hours before refueling. This was equivalent to a total range of approximately 500,000 miles, exceeding the total range of equivalent fully loaded chemical aircraft by a factor of approximately 100. 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 in reconnaissance-counterstrike or similar missions at speeds of Mach 2.5 or greater. Advanced design and development of nuclear turbojets, ramjets, and rockets to meet other potential military objectives had been completed or were in process for both subsonic and supersonic applications.

Program Termination

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.

This summary volume of the Comprehensive Technical Report describes the program background, gives a brief physical description of the major power plant designs and test assemblies, and traces the development progression in basic technology to its final status. It also serves as a guide to the more detailed technical volumes of the Report (APEX-902 to 921 on the inside front cover), and to documents containing original source material.

Program background prior to initiation of the General Electric program is provided in the final report of the NEPA project (reference 1).

Matters of national policy concerning the program are discussed in transcripts of the 1959 Hearing of the Joint Congressional Committee on Atomic Energy and the report of the committee (references 2 and 3). The Congressional documents also enumerate various program evaluations which were performed by government committees.

The related activities of other contractors and government agencies participating in the aircraft nuclear propulsion program are described in reference 4 in the Congressional

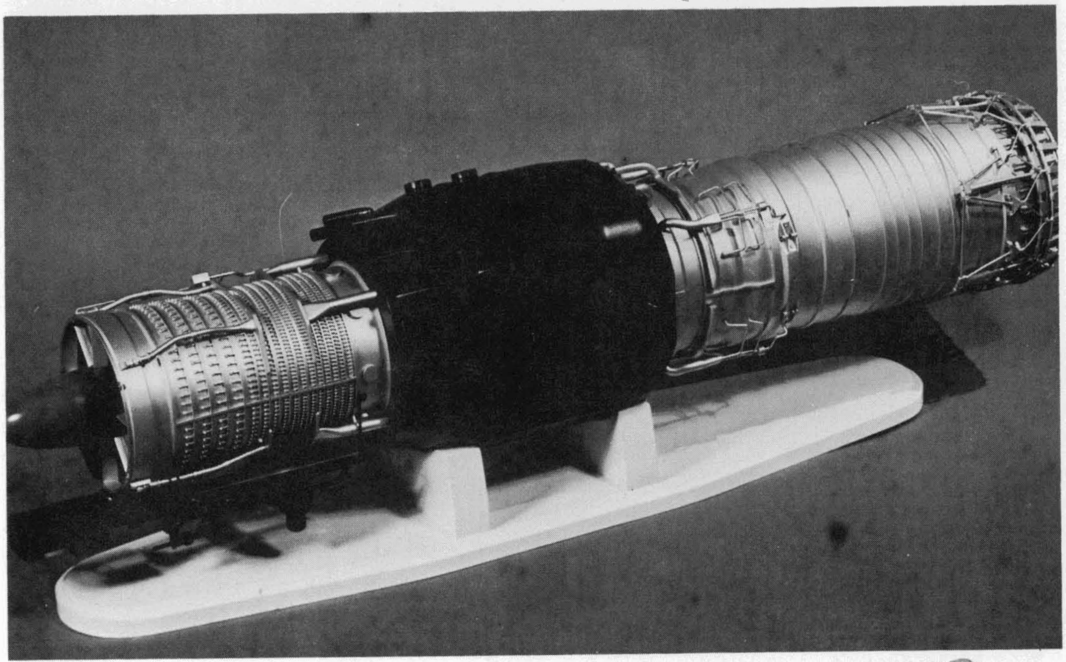

Fig. 1.1 – Model of the XNJ140E nuclear turbojet engine

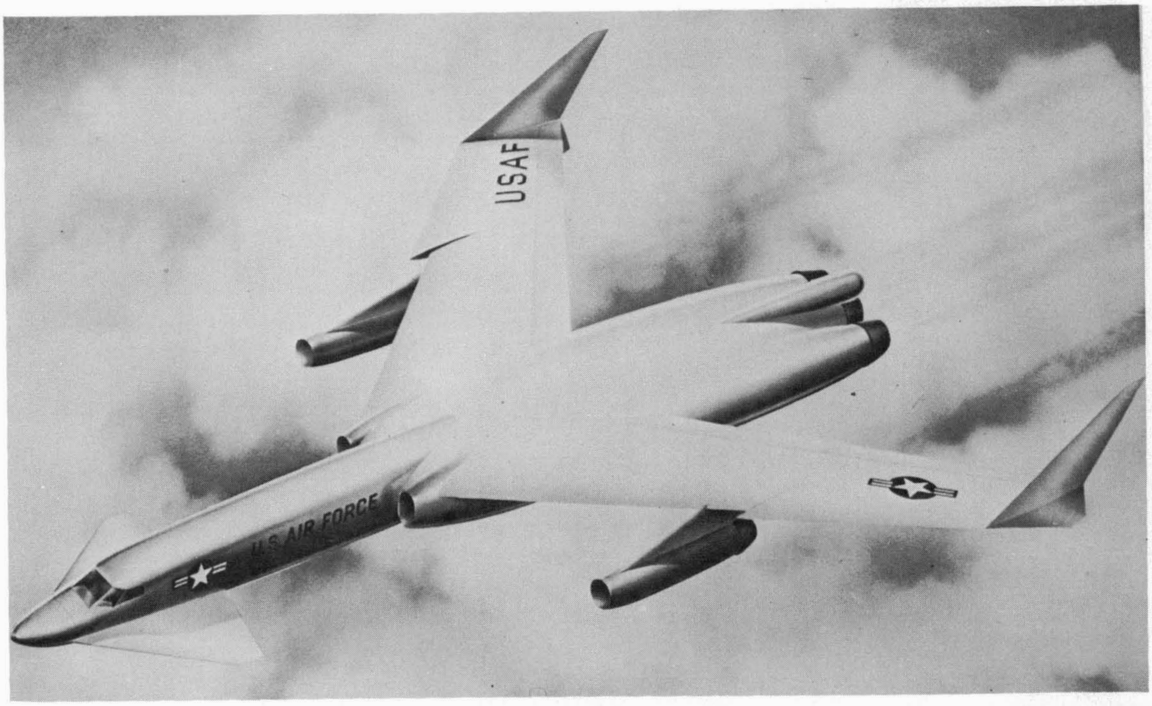

Fig. 1.2 – Convair NX2

committee reports, and in the technical reports of the participating contractors. The presidential budget message which prompted termination of the program is given in the Congressional Record (reference 5). Further information concerning the program termination and disposition of its activities are given in the annual report of the Atomic Energy Commission for the year 1961 (reference 6).

The detailed progression of the General Electric program is provided in the Quarterly Progress Reports (reference 7) and in the Annual Program Reports (reference 8).

1.2 PROGRAM BACKGROUND PRIOR TO 1951

The Aircraft Nuclear Propulsion Program resulted from actions during and shortly after World War II by individuals who were interested in applying the developments of the Manhattan District Program to aircraft propulsion. In 1946, the Air Force established the Fairchild NEPA (Nuclear Energy for the Propulsion of Aircraft) Project to study the feasibility of nuclear powered flight. In 1948, the Atomic Energy Commission contracted with the Massachusetts Institute of Technology for a group of prominent scientists, assembled as the "Lexington Project," to review the work being conducted in the field of aircraft nuclear propulsion. The Lexington Project recommended an expanded evaluation program and predicted that nuclear flight could be achieved in a subsonic system in "approximately 15 years at a cost well in excess of one billion dollars" (reference 9). Subsequently, in 1949, the Atomic Energy Commission entered the program on an active basis in a joint effort with the Air Force, the Navy, and the National Advisory Committee for Aeronautics. The feasibility studies continued at NEPA and at the Oak Ridge National Laboratories. The NEPA Project continued under joint auspices until May 1951 when the study was closed out on the basis that general feasibility had been established. The NEPA Final Status Report (reference 1) describes the early program background and the results of the studies that were performed.

1.2.1 MOTIVATION FOR NUCLEAR FLIGHT

The following statements, extracted from the NEPA report, define the motivation for establishing a program for the development of nuclear flight.

"The present air strategy of this country is defined by the speed, altitude, and range at which its aircraft can fly. As long as it is necessary to use conventional chemical fuels such as gasoline or kerosene, aircraft will be subject to limitations as to their characteristics of range, speed, and altitude. With chemical fuels, the factors of speed and range are incompatible...

"It is in the combination of high speed with long range that nuclear power promises to achieve results unobtainable by any other means... the only practical limitations on the range of a nuclear-powered aircraft would be the endurance of the crew against both radiation and ordinary fatigue, and the freedom of the power plant and aircraft from mechanical breakdown or battle damage.

"... For example, a nuclear-powered aircraft could:

- Not only fly to any point on the globe, but do so at very high speed, and by any route whatever, no matter how circuitous, and could return at high speed by an entirely different route. This is impossible for conventional aircraft.
- Fly at very low altitudes where radar detection is difficult and at high speeds for essentially unlimited distances, a performance impossible for chemically fueled aircraft.

- Fly completely around the earth at a speed equal to the easterly speed on the earth's surface at latitudes of military interest, thus conducting its mission entirely in darkness and hence with much reduced danger of enemy interference.
- Perform long-range reconnaissance missions or patrol for enemy submarines or aircraft for several days without stopping.
- Serve as an airborne communications center, particularly useful in polar areas, or as a mobile extension of the radar screen, or as a mobile airborne command post for large-scale military operations."

1.2.2 CONCLUSIONS OF NEPA PROJECT

It was the conclusion of the NEPA Project that nuclear powered flight was feasible. The following more specific conclusions (reference 1) of the NEPA Project provided the basis for the original orientation of the GE-ANP program.

- "A nuclear power plant can be developed which will be capable of propelling an inhabited bombardment aircraft at Mach 1.5 at 45,000 feet. The aircraft would be about the size of the largest present-day bombers. Its range to all intents and purposes would be unlimited."
- "Nuclear power plants to propel aircraft at subsonic speed and altitudes up to 35,000 feet can be built significantly sooner than those for Mach 1.5 and 45,000 feet... It may therefore be desirable to select as the first tactically useful power plant for development one designed for subsonic speed and altitude 35,000 feet or less..."
- "The most suitable type of propulsion machinery is the turbojet. Ducted fans and propellers are less desirable, but possibly of limited value... Nuclear ramjets and nuclear rockets may possibly become of interest in the distant future."
- "The air-cooled hydrogenous-moderated reactor is particularly attractive because of its relative simplicity and low vulnerability, but suffers from major uncertainties regarding the retention of fission products within the fuel elements and leakage of neutrons through the air ducts in the shield... The major uncertainties of the liquid-metal cycles are those of reliability and the degree to which the effects of a system leak can be controlled... Choice of any cycle for exclusive or even major development at this time would be arbitrary."
- "Further progress in the development of a nuclear power plant for an aircraft will require that the program contain a rapidly increasing proportion of the designing, building, and testing of full-scale power-plant components and assemblies. Basic investigation of the problems encountered, and development of fundamental supporting data, must also continue. New explorations, particularly in materials and metallurgy required for Phase II* performance, should be carried forward.
- "An accurate evaluation of the true worth of nuclear-powered flight will not be possible until a considerable amount of ground operation and flight experience is gained with this new propulsive system. In spite of the difficult nature of the development program and in spite of the inherent hazards of radiation, the potentialities of nuclear-powered flight are so great that its continued development, at least to the point where adequate ground and flight experience can be gained in one or more experimental articles, is mandatory in the interest of national defense."

*Defined in the NEPA report as "Tactical aircraft with the following design points...: 60,000 feet at Mach 0.9; sea level at Mach 0.9; 35,000 to 45,000 feet at Mach 1.5; and 35,000 feet at Mach 2.0 ..."

1.3 HISTORICAL SUMMARY OF THE GENERAL ELECTRIC PROGRAM

The Air Force and Atomic Energy Commission agreed, in early 1951, that feasibility had been sufficiently well established to warrant initiation of active research and development leading to militarily useful aircraft nuclear propulsion systems. General Electric was selected as the propulsion system contractor on the basis of its experience in the development of both nuclear reactors and aircraft turbomachinery. The Air Force contract with the Aircraft Gas Turbine Division of the General Electric Company, to undertake active development of a nuclear turbojet engine, became effective on March 21, 1951. The Atomic Energy Commission contract for reactor development became effective on June 29, 1951.

The General Electric Program encompassed:

1. An applied research program in materials, engineering physics, and component development to provide a basic technology applicable to a broad spectrum of potentially useful aircraft nuclear propulsion systems
2. The design and test of experimental nuclear reactors operating aircraft turbomachinery
3. The design and test of advanced turbomachinery
4. The design and development of prototype propulsion systems to meet specific military objectives
5. The advanced design and development of propulsion systems to meet anticipated military requirements

1.3.1 EARLY FLIGHT PROGRAM

The first several months of the General Electric program were devoted to the selection of a system for initial development. This phase of the program was completed on August 28, 1951 with a recommendation to the Air Force and Atomic Energy Commission to proceed with the development of a direct-air-cycle system using an air-cooled reactor with metallic fuel elements and a hydrogenous liquid as the moderator and structural coolant. This recommendation was made on the basis that it represented the best way to obtain early nuclear flight experience. It was anticipated that higher performance materials would be required for ultimate application to operational military aircraft.

An Air Force objective was established for nuclear ground and flight operation in a modified Convair aircraft at the earliest feasible date in order to evaluate the operational practicability of nuclear systems prior to commitment to a prototype military nuclear aircraft. Approval was granted to proceed with the development of a power plant, designated the P-1, to meet the early flight objective. The initial ground test was scheduled for 1954 and flight test for 1957. The early flight objective was withdrawn in March 1953 on the basis that early flight demonstration with a system not fitting a specific military requirement was no longer considered warranted.

1.3.2 APPLIED RESEARCH AND DEVELOPMENT

After discontinuation of work on the P-1 power plant, the program was redirected to an applied research and development program applicable to a broad spectrum of potentially useful military propulsion systems. The applied research activity continued until program termination, developing the basic materials and engineering analysis methods used in a series of subsequent reactor operations and power plant designs. The materials program encompassed the development of metallic and ceramic fuel elements, hydrided metallic moderators, and shield, controls, and structural materials for use in both subsonic and supersonic aircraft. This was supported by extensive in-pile test programs. Engineering analysis techniques were developed in reactor and shield nuclear physics, aerothermodynamics, controls, mechanics, and nuclear safety. These activities were supported by a strong experimental program including the performance of reactor critical experiments, shielding experiments, aerothermodynamic tests, and mechanical testing in simulated environments and in nuclear test reactors. The applied research activity is discussed in section 7 of this volume.

1.3.3 HEAT TRANSFER REACTOR EXPERIMENTS

A series of experimental reactors were built and operated using materials and methods developed in the applied research activity.

Heat Transfer Reactor Experiment No. 1

The first reactor operated in the General Electric program was in Heat Transfer Reactor Experiment No. 1. This was a direct-air-cycle reactor using nickel-chromium, uranium-oxide-dispersion fuel elements, with water serving the combined function of moderator and structural coolant. The HTRE-1 reactor first operated a modified General Electric J47 turbojet engine exclusively on nuclear power in January 1956. Operation of the HTRE-1 continued throughout the Calendar Year 1956, accumulating a total of 150.8 hours of operation at high nuclear power levels, exceeding the design requirement of 100 hours.

Heat Transfer Reactor Experiment No. 2

The HTRE-2 reactor was a modification of HTRE-1, providing a hexagonal center hole, 11 inches across flats with an active length of 30 inches, for use in testing insert sections for advanced reactors. HTRE-2 operation started in July 1957 and continued during the remainder of the program, accumulating 1299 hours of high power nuclear operation. Insert test sections consisted of metallic fuel elements combined with air-cooled hydrided zirconium moderators and beryllium oxide fuel elements for use in ceramic reactors. Inserts were operated at materials temperatures up to 2800°F for extended periods and for short periods at higher temperatures.

Heat Transfer Reactor Experiment No. 3

The HTRE-3 reactor was built in a full-scale aircraft reactor configuration using Ni-Cr fuel elements of the HTRE-1 type and an air-cooled hydrided zirconium moderator. Two modified J47 turbojets were operated by the reactor with full nuclear power being achieved in 1959. The system operated for a total of 126 hours; the design objective was 100 hours operation.

Proposed Heat Transfer Reactor Experiment No. 4

A ceramic reactor using beryllium oxide fuel tubes was designed and received extensive component development for a proposed fourth Heat Transfer Reactor Experiment in the HTRE-1 or HTRE-3 test assemblies. Consideration of HTRE-4 was dropped in favor of proceeding directly to a prototype propulsion system incorporating the ceramic reactor design features and components which had been developed.

1.3.4 TURBOMACHINERY DEVELOPMENT

A propulsion system is characterized primarily by the propulsion machinery; the heat source, chemical or nuclear, plays a secondary role. By 1955, sufficient progress had been made to define the characteristics of a basic turbojet propulsion unit into which nuclear reactor heat sources of successively higher performance capabilities could be incorporated with minimum modification to the turbomachinery. Development of this unit, the X211 turbojet engine, began in 1955. The X211 was a single-rotor, variable-stator, high-pressure-ratio engine with an airflow of approximately 400 pounds per second at sea level static; it had growth potential to turbine inlet temperatures above 2000°F for supersonic operation.

1.3.5 PROTOTYPE PROPULSION SYSTEMS

Prototype propulsion system designs all used X211 turbomachinery with reactors that had been tested or were planned for development in the reactor program. The power-plant configurations, performance requirements, and materials selection were based on specific military objectives or anticipated future requirements. Two prototype power plants, the XMA-1 and the XNJ140E, were designed to meet specific military objectives.

XMA-1 Power Plant

A specific military objective requiring a nuclear propulsion system was issued by the Air Force on March 22, 1955, as Systems Operational Requirement No. 81. This required a "piloted nuclear powered intercontinental strategic bombardment weapon system" capable of extended cruise without in-flight refueling, penetrating enemy defenses at high altitudes and supersonic speeds, and low-level attack at subsonic speeds. The design and development of a power plant designated the XMA-1 was undertaken to meet these requirements, assuming a nuclear cruise and chemically augmented sprint. The XMA-1 combined two sets of X211 turbomachinery with a single reactor. Initial ground test was scheduled for 1959 and the first flight test for 1960. A decision was made, late in 1956, to de-emphasize aircraft development but to continue developing the propulsion system at a reduced level without reference to a specific military objective.

A new objective was provided by the Air Force on October 28, 1958, as Systems Operation Requirement No. 172 for "a Continuous airborne missile launcher and low level weapons system" (CAMAL). The CAMAL mission retained the extended cruise and low-level penetration of SOR 81 but substituted the use of long-range, air-to-ground ballistic missiles for the high altitude supersonic portion of the flight regime. The XMA-1 development was redirected toward the CAMAL objective. First flight in 1963 was assumed as a target date using an early model of the power plant, the XMA-1A with a reactor of the type tested in HTRE-3. An improved model, the XMA-1C, using an advanced metallic or ceramic reactor was placed in preliminary design. The proposed Convair Model 54 aircraft was considered to be the flight vehicle.

XNJ140E Power Plant

In 1959, the early flight objective for the CAMAL aircraft was eliminated. In place of a specific weapons system objective, general guidance was provided to direct the applied research and development program toward a propulsion system capable of propelling an aircraft at a speed of Mach 0.8 to 0.9 at an altitude of approximately 35,000 feet. The reactor was to be capable of 1000 operating hours at the specified performance level and was to have development potential for even higher performance. In view of the revised objective, work on the XMA-1A reactor was discontinued.

After an evaluation of the relative development status of high temperature metallic and ceramic materials that had been under development for the XMA-1C reactor, a BeO ceramic reactor was selected to meet the Department of Defense guidance. Simultaneously, it was determined that a single-rather than dual-engine configuration was better suited to meet the growth potential requirement.

The development of such a power plant, designated tha XNJ140E, was proposed in March 1960 and was subsequently approved as a development objective. The XNJ140E used the basic components of the X211 turbomachinery and the reactor and shield materials that had been under development for the XMA-1C. A target date for ground test of a prototype unit, the XNJ140E-1, was set for December 1962. The ground test was referred to as the "Advanced Core Test," turbomachinery having been previously tested under chemical power.

On November 9, 1960, the Air Force issued Advanced Development Objective No. 20 defining the objectives of the Air Force Nuclear Aircraft Development program. The immediate objective was to achieve nuclear flight in a military prototype aircraft at subsonic performance levels matching the Department of Defense guidance. Supersonic nuclear flight was the ultimate objective.

The initial system would be used to evaluate the practicality of subsonic nuclear missions of long endurance, such as air alert, missile launching, low-level penetration, logistics, reconnaissance, air early warning, antisubmarine warfare, and airborne command posts. The subsonic system would also serve to develop a basic equipment and operational technology leading toward supersonic nuclear aircraft capability.

In accordance with the ADO No. 20 objective a target date was established to furnish XNJ140E power plants for initial flight operation in the Convair NX2 aircraft in 1965 after completion of ground testing of the XNJ140E-1.

1.3.6 ADVANCED PROPULSION SYSTEMS

In addition to the design of power plants to meet specific military objectives, propulsion systems were being designed to meet anticipated future military requirements. Primary emphasis was placed on a power plant capable of propelling a B-70 type aircraft at a speed of Mach 2.5 and an altitude of 45,000 feet. The basic XNJ140 power plant configuration was used in these studies, with advanced versions of the X211 turbomachinery designed to higher temperature capability. Smaller reactors of higher temperature capability were under development for the supersonic application. Additional design studies were being made on power plants of even higher performance capability and on propulsion systems for subsonic, load-carrying aircraft, closed-gas-cycle systems, nuclear ramjets, and nuclear rockets.

1.3.7 STATUS AT PROGRAM TERMINATION

The XNJ140E program was on schedule when the Aircraft Nuclear Propulsion Program was terminated. The basic propulsion machinery had been operated for a total of 758 hours with a chemical heat source, using several engine buildups. Structural modifications were in process to adapt the turbomachinery to the final configuration of the XNJ140E reactor-shield assembly. Design of the reactor-shield assembly and subassembly had been completed. Manufacturing drawings of individual parts and components were completed or in process. Prototypes of critical components had been proof-tested. Beryllium oxide fuel tube assemblies for use in the reactor had been tested for a total of 10,683 hours in the MTR, ORR, and as inserts in an experimental aircraft reactor at temperatures approximating or exceeding design requirements. Reactor critical experiments had been performed to verify fuel element loading specifications. Permission had been requested to proceed with manufacture and assembly of the reactor into the propulsion machinery. Ground test operation was scheduled for late 1962. Nuclear flight test was predicted in a test bed aircraft for 1963, and nuclear flight operation of the NX2 prototype military aircraft in 1965.

Additional information on the status of the turbomachinery, on the methods used in design, and on the proposed ground test and flight test programs, are incorporated in references 10 through 13.

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2. CYCLE SELECTION AND FIRST POWER PLANT

2.1 TECHNICAL BACKGROUND

2. 1. 1 ENERGY CONVERSION AND HEAT TRANSFER CONCEPTS

The primary difference between chemical and nuclear propulsion systems is 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 and to a much higher temperature because the kinetic energy of the fission fragments is much higher than that of combustion products. However, 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, the NEPA and General Electric studies were confined primarily to nuclear systems in which the fissionable material was 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.

The propellant can be heated directly by passing it across the reactor fuel elements, or indirectly by circulating an intermediate fluid through the reactor to a heat exchanger where the heat is transferred to the propellant. In a variation of the latter system, the fuel can be contained within the fluid which circulates first through a critical region and then to the radiator. The propellant temperature that can be achieved is limited by the temperature capability of the fuel-bearing material. In general, this means that the propellant temperatures and performance of nuclear systems can approximate but not exceed those of their chemical counterparts. As a corollary, conventional propulsion machinery such as gas turbines can be used.

2. 1. 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.

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 as with a chemical rocket for the same weight of propellant. In an air-breathing nuclear system on the other hand, a given thrust level can be sustained virtually indefinitely because the supply of propellant (air) is unlimited. Nevertheless, in rocket applications the advantage of nuclear power may be of critical significance.

2. 1. 3 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.

2.1.4 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.

Reference 1 contains additional details on the technical status of the ANP program at the end of the NEPA project.

2.2 CYCLE SELECTION

The first major problem confronting the General Electric Company, upon assuming responsiblity for the Aircraft Nuclear Propulsion Project, was the selection of thermodynamic cycle for the first development system. The first 4 months were devoted to a study of this question. The choice quickly narrowed to direct air cycle and indirect liquid-metal cycle reactors powering turbojet engines. An intensive program of experimental and analytical studies was undertaken to resolve the obvious uncertainties in the two approaches.[2] On the basis of these studies a program for development of a direct air cycle power plant was presented to the Air Force and the Atomic Energy Commission on August 28, 1951, and was subsequently approved.

The decision to proceed with the development of the direct-air-cycle concept was based on the available evidence which indicated that this represented the best way to achieve the immediate objective of nuclear flight at the earliest feasible date. The decision was re-served as to the technical course best suited for the ultimate development of nuclear power for aircraft propulsion.

A number of factors were considered in arriving at the recommendation to proceed with the direct air cycle. NEPA had identified the retention of fission products and the leakage of neutrons through the air ducts as the major uncertainties in the direct air cycle. Fission product release could be reduced or eliminated by using a clad metallic fuel element. This approach allowed more time to develop the ceramic fuel element technology for use in later power plants. Upon closer examination, neutron leakage through the air ducts appeared to be less of a problem than originally anticipated. This was due to recognition of the fact that duct-scattered neutrons were so reduced in energy that penetration of the crew shield was improbable.

Reliability and the degree to which the effects of a system leak could be controlled were the major uncertainties identified by NEPA in the liquid-metal cycle. GE-ANPD studies indicated that an early solution of this problem was unlikely.

Several factors concerning the ultimate desirability of the two cycles were also considered before the selection was made. Factors that appeared to favor the direct air cycle were development potential to extremely high temperatures, dependability, vulnerability, operational problems, and fuel reprocessing. The indirect liquid-metal cycle appeared to have a weight advantage because of the small size of its reactor. However, the weight of liquid-metal pumps, plumbing, heat exchangers, and other components of sufficient reliability for aircraft utilization were expected to nullify a major portion of the weight advantage of the reactor-shield assembly.*

2.3 P-1 POWER PLANT

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

*The question of the choice between the direct and indirect cycle was periodically reviewed throughout the GE-ANP program in the light of the continually improving technology of both systems. The relative development status at any point in time during the program was always such that earliest flight could be achieved with the direct air cycle.

It continued to be the belief of the General Electric Company that the direct air cycle, in addition to its early flight potential, also offered the greatest promise for ultimate successful and practicable utilization in operational nuclear aircraft. This belief was based on the Company's experience as a major supplier of military aircraft engines, on its work in the Aircraft Nuclear Propulsion program, and on its experience in developing the liquid-metal system for the Naval Reactors Program and the mercury vapor turbine for electric power generation.

The direct-air-cycle nuclear turbojet engine followed directly in the tradition of air-cooled aircraft engines. A direct-air-cycle propulsion system is relatively invulnerable to damage by foreign objects; the coolant (air) does not freeze or burn; and coolant leaks are of little importance since air is available in unlimited quantity. All of these were major considerations in arriving at the ultimate conversion of aircraft to the use of engines in which the materials of construction were cooled by air rather than by a liquid (references 3 through 7). Furthermore, as with a conventional aircraft engine, the direct air cycle is always in a ready condition: it can remain unattended for prolonged periods but be restarted on short notice with minimum warmup and checkout. This was graphically demonstrated in the HTRE operations, where the systems were shut down or started up with little preliminary preparation after either prolonged operation or prolonged shutdown.

In view of these considerations, the GE-ANP development progressed steadily in the direction of using air not only as the primary working fluid, but also to replace all liquids required for power plant operation. Although the NEPA concept for a liquid-cooled reactor structure was adopted to achieve early reactor operation, air-cooled reactor structure and moderator materials were substituted later. Early shields using hydrogenous liquids were replaced with air-cooled solid shields operating at high temperatures. Nuclear startup without chemical assistance was demonstrated in actual turbojet operation, thus suggesting the ultimate elimination of flammable aircraft fuel. Pneumatic rather than hydraulic control systems were under consideration for supersonic operation. Even the engine lubricating oil could be eliminated if air bearings, which were being studied, could be developed. Thus, the danger of mission failure due to the loss or combustion of aircraft fluids could be reduced or possibly eliminated.

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. Active development of the power plant components was authorized early in 1952. The initial ground test was scheduled for 1954, with flight test in 1957.

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.

2. 3. 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.

2. 3. 2 DESCRIPTION OF P-1 POWER PLANT

The P-1 power plant arrangement is shown in Figure 2. 1. Four turbojet engines were 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, designated the R-1, is illustrated in Figure 2. 2. 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 2. 3). 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.

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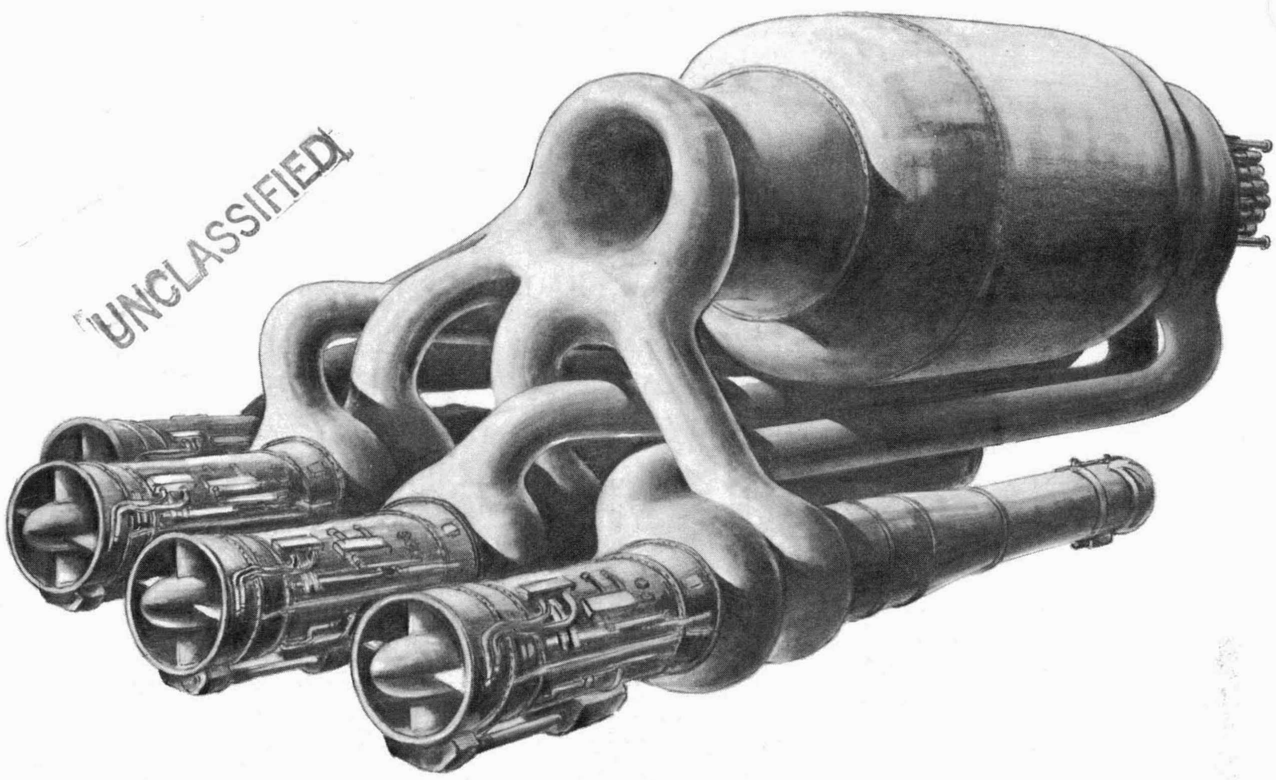

Fig. 2.1 – Artist's concept of a 4-engine version of the P-1 power plant

The shield was primarily water, supplemented by a lead and steel gamma shield at the forward end. The shield configuration is shown in Figure 2. 4.

2. 3. 3 FINAL STATUS OF P-1 DEVELOPMENT

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 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, work was undertaken on ceramic fuel elements and on metallic fuel elements of higher operating temperature capabilities.

Further details of the P-1 power plant are given in APEX-902, "P-1 Nuclear Turbojet," of this Report, and in references 8 through 14.

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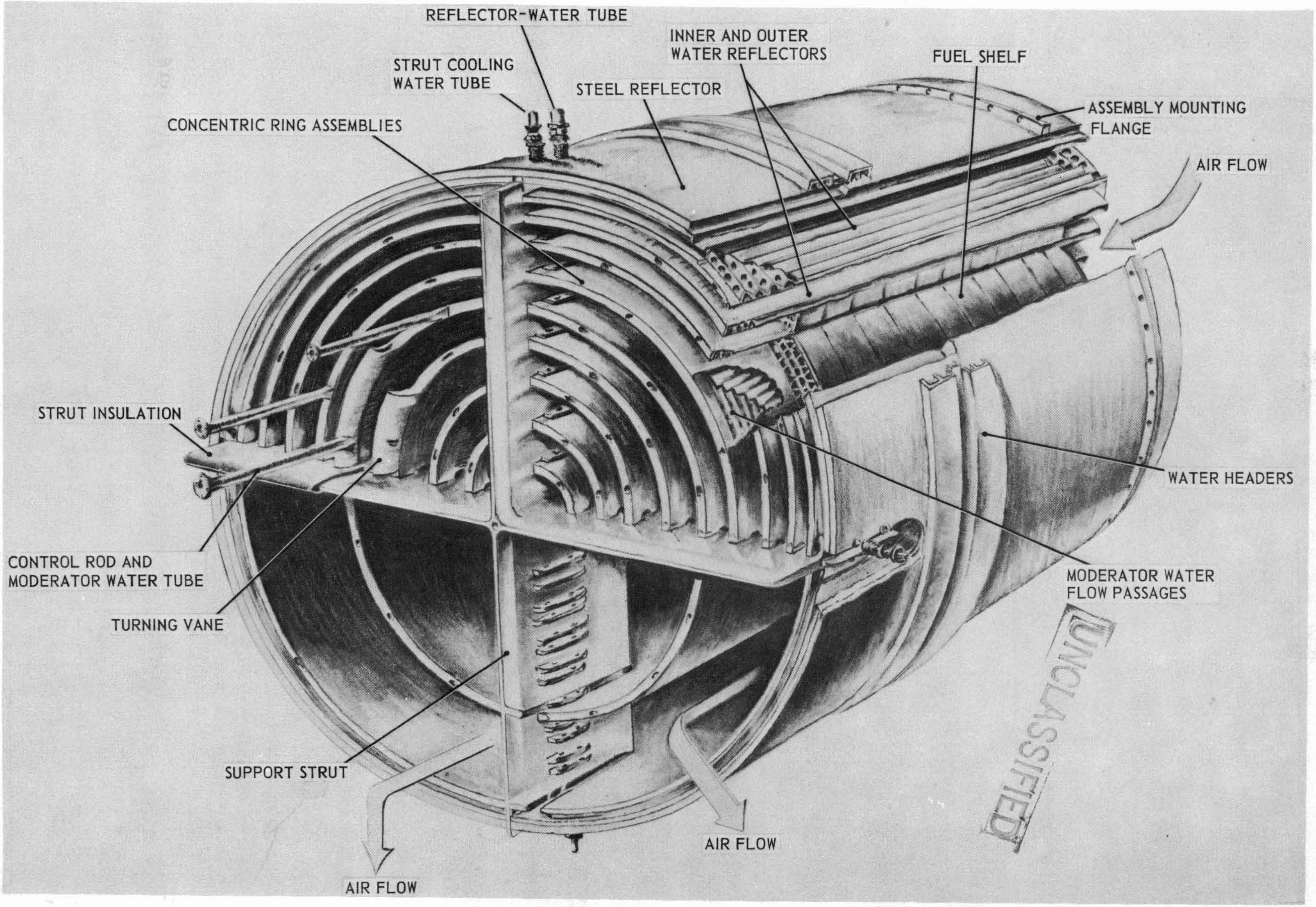

Fig. 2.2 – P-1 reactor structural arrangement

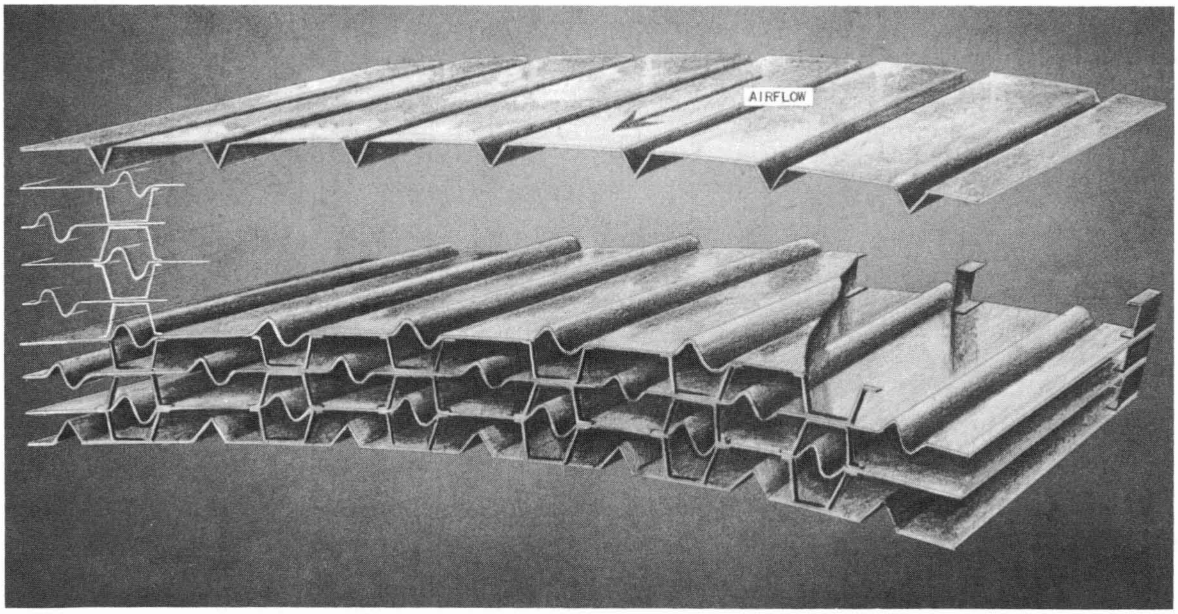

Fig. 2.3 – P-1 type of fuel element

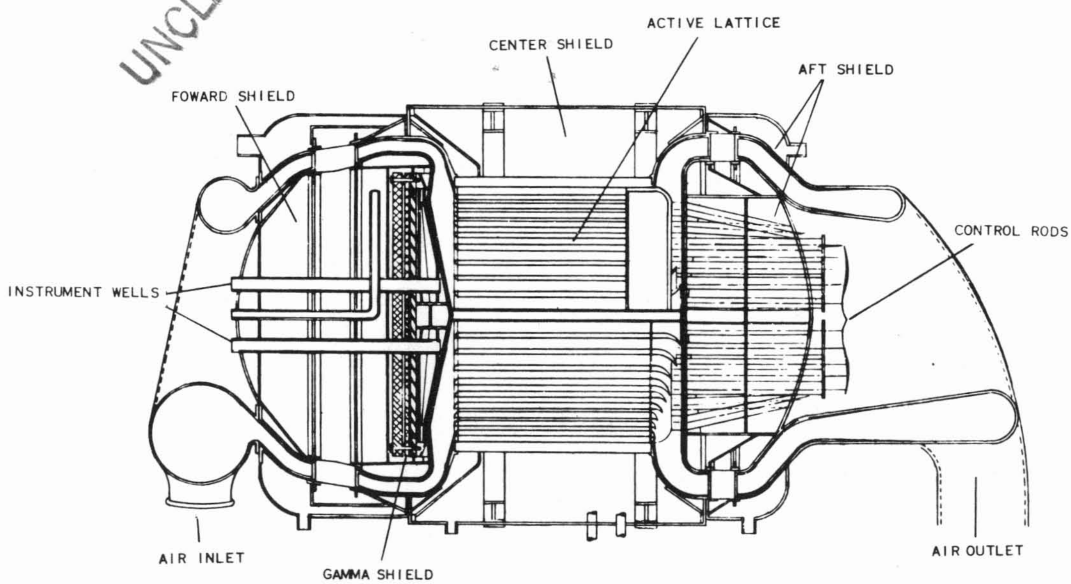

Fig. 2.4 – P-1 reactor-shield assembly

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3. HEAT TRANSFER REACTOR EXPERIMENTS (HTRE)

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 of proceeding directly to a prototype propulsion system.

3.1 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 in Figure 3.1. 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 as shown in Figure 3.2. 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.

The turbojet engines which had been modified for the P-1 ground test were used as the air supply for the CTF. All the principal elements of a nuclear propulsion system, reactor, engine, and controls were thus incorporated in the test assembly. The system components could be treated with relative independence from a mechanical standpoint, but were very closely coupled thermodynamically and aerodynamically. The resultant dynamic interaction of the reactor, controls, and turbomachinery required that the reactor be designed with a discipline that would not have been necessary if a fixed air supply, such as an electrically driven compressor, had been used. The reactor experiments, therefore, served not only to develop reactor technology, but also the technology of the other system components and the integration of these components.

The CTF is described in APEX-903, "Core Test Facility" of this Report.

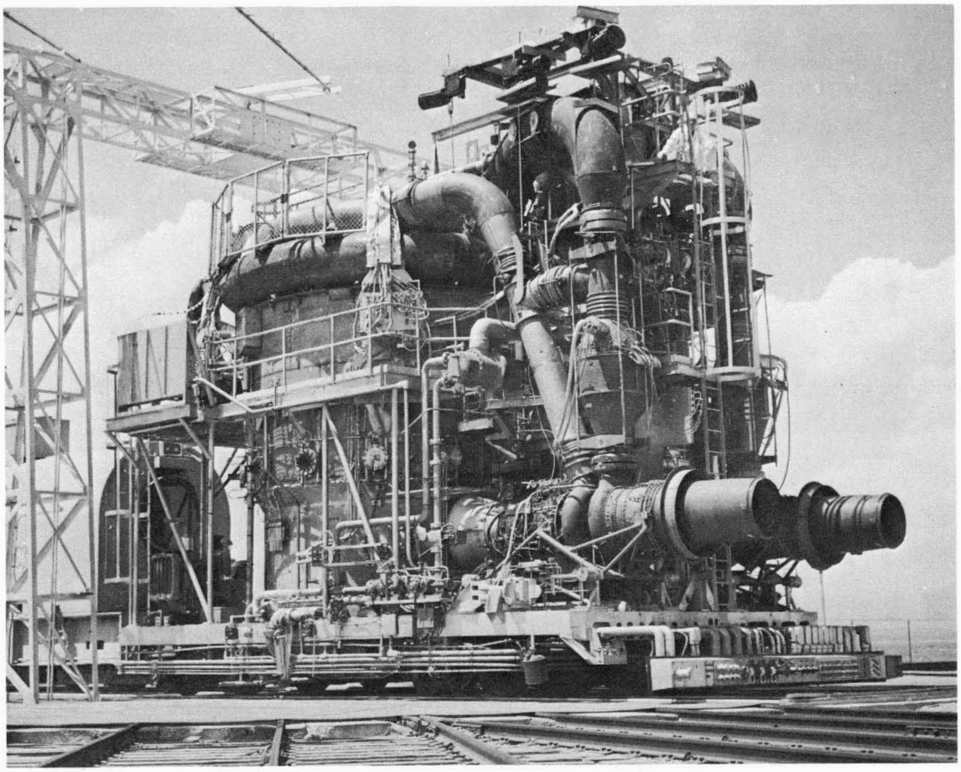

Fig. 3.1 – Core Test Facility

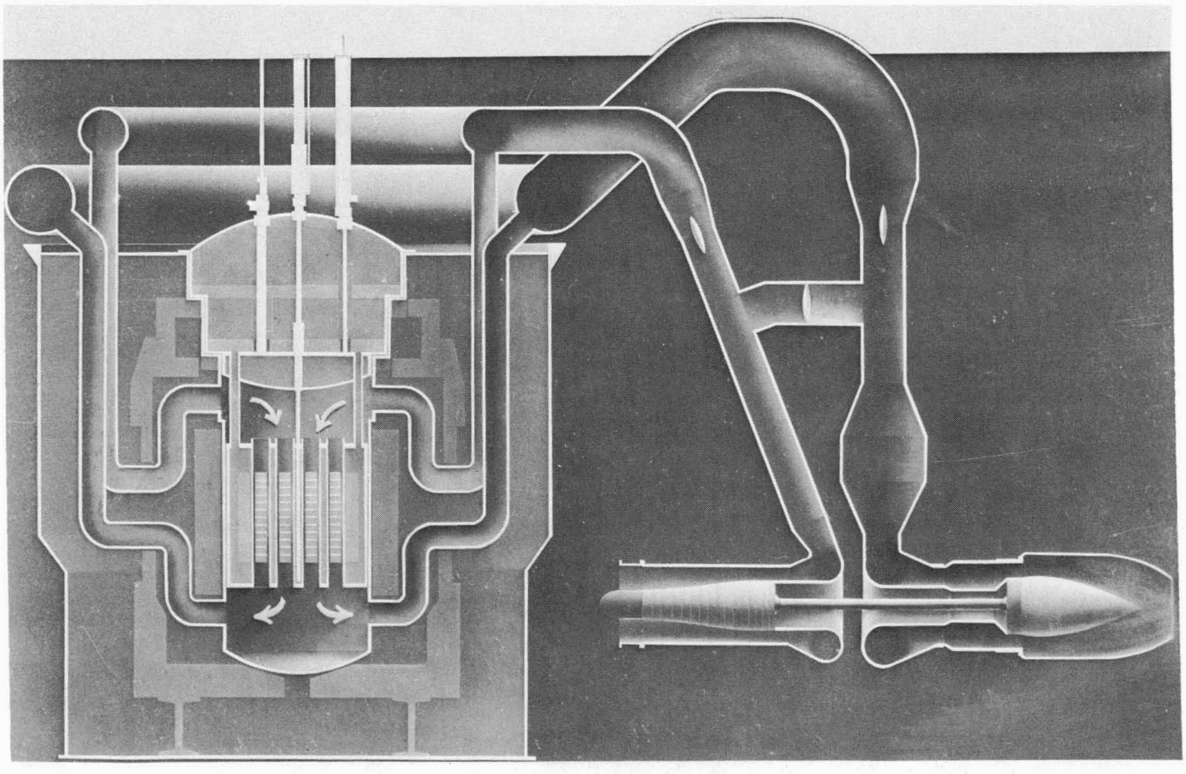

Fig. 3.2 – Schematic illustration of Core Test Facility with reactor

3.2 HTRE-1

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 3.3 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 in the CTF. 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

The preliminary design of the HTRE-1 reactor was completed in February 1954, and the final design and development of the reactor was then initiated. The first known operation of a turbojet engine on nuclear power was achieved in HTRE-1, on schedule, in January 1956. Test operations continued for the rest of that year. All of the program objectives were realized.

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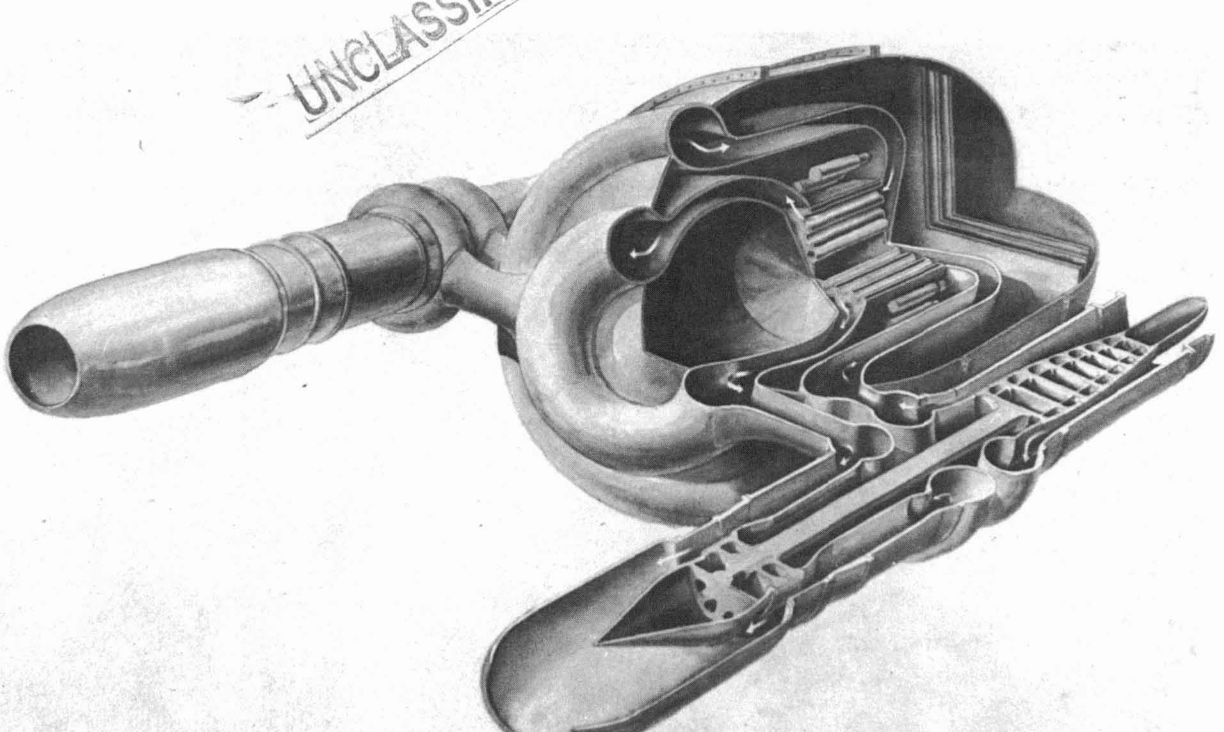

Fig. 3.3 – AC101B type power plant

3.2.1 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.

3.2.2 DESCRIPTION OF HTRE-1 REACTOR AND TEST ASSEMBLY

An artist's concept of the HTRE-1 reactor is shown in Figure 3.4. The reactor is shown during construction in Figure 3.5.

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 3.6) made of an 80Ni - 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°F by circulation to an external radiator, while the fuel elements operated at a temperature of approximately 1700°F, heating the air to about 1350°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 flux 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 3.7. 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.

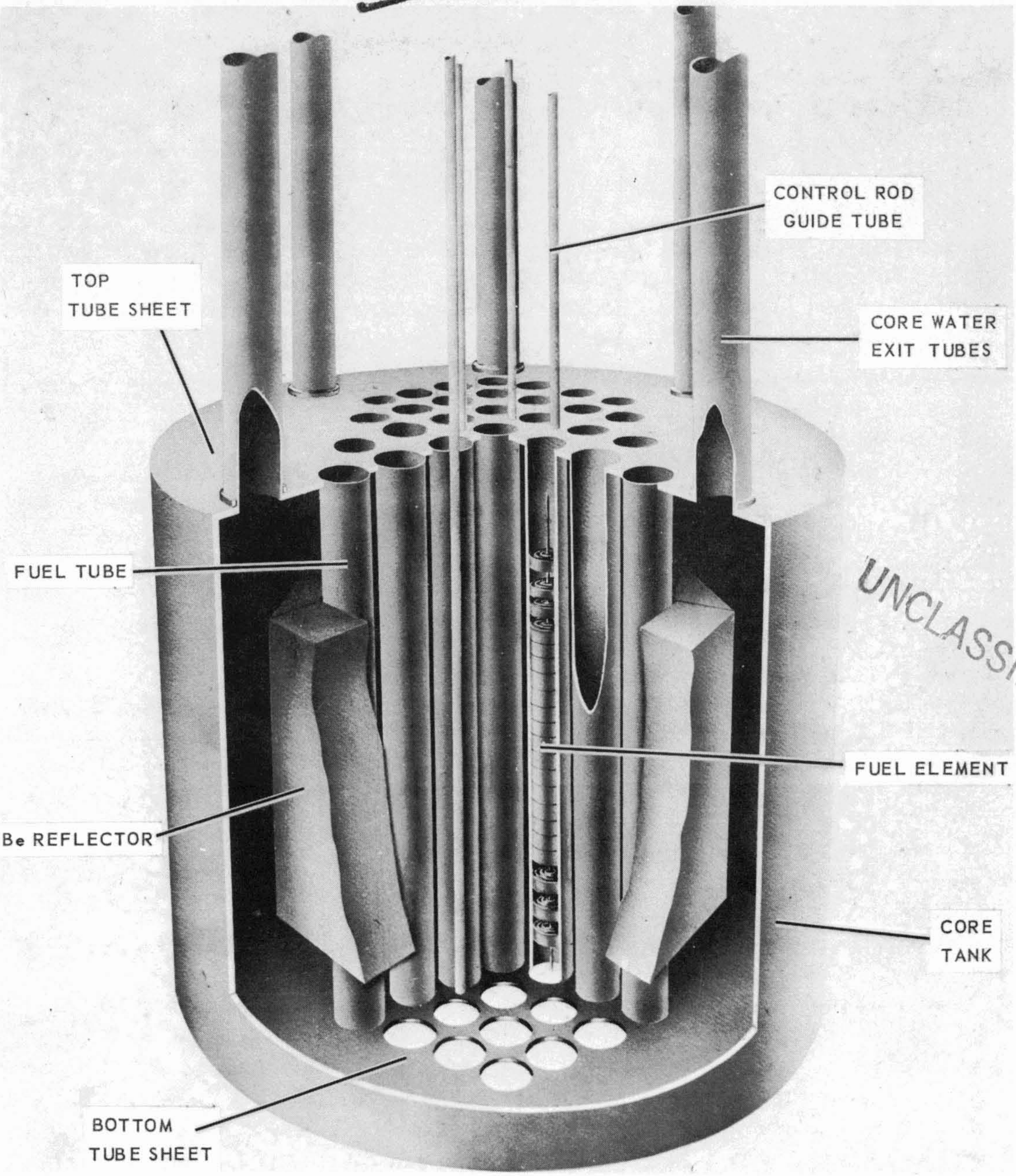

Fig. 3.4 – HTRE-1 reactor

The 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 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.

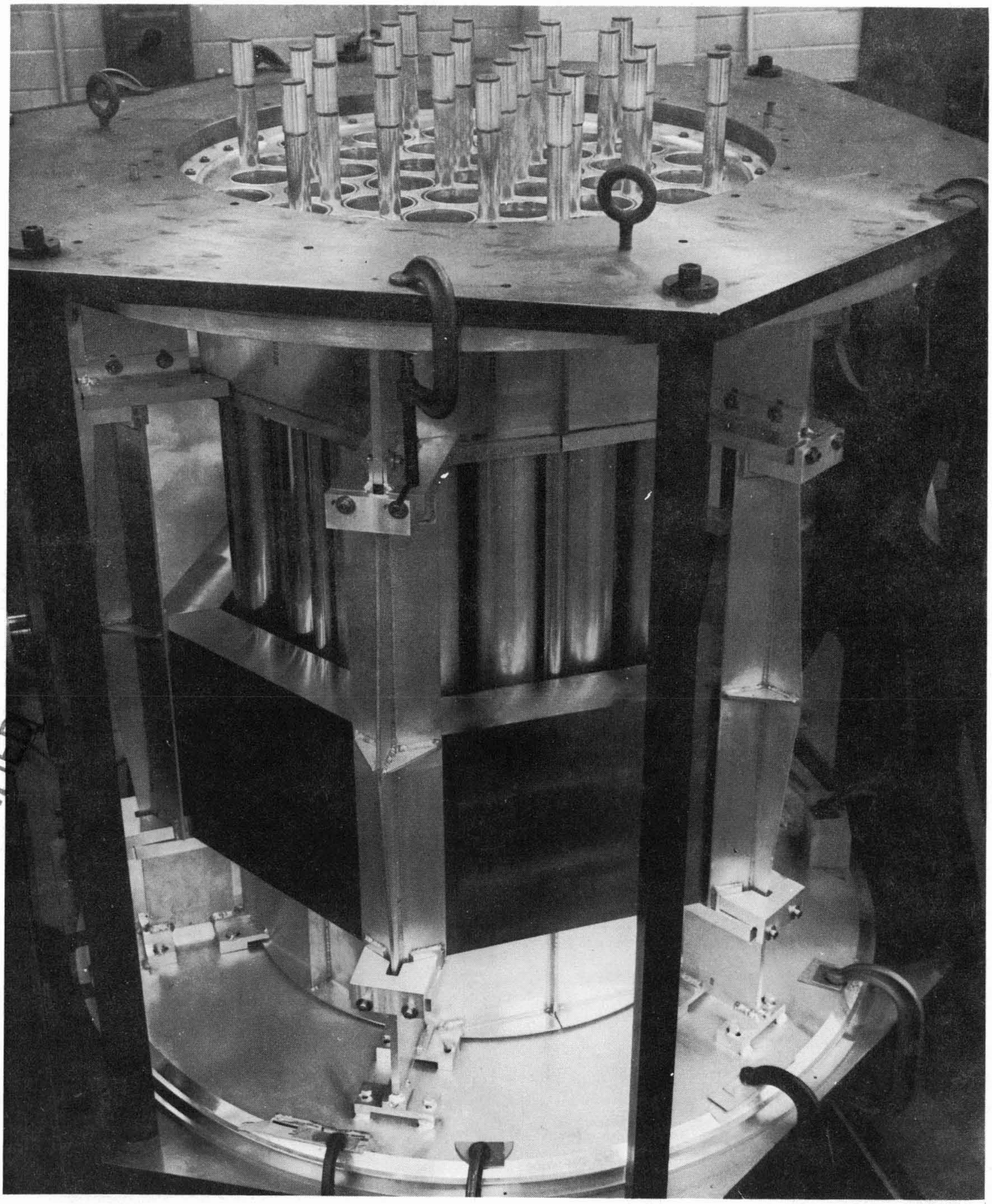

Fig. 3.5 – HTRE-1 reactor during construction

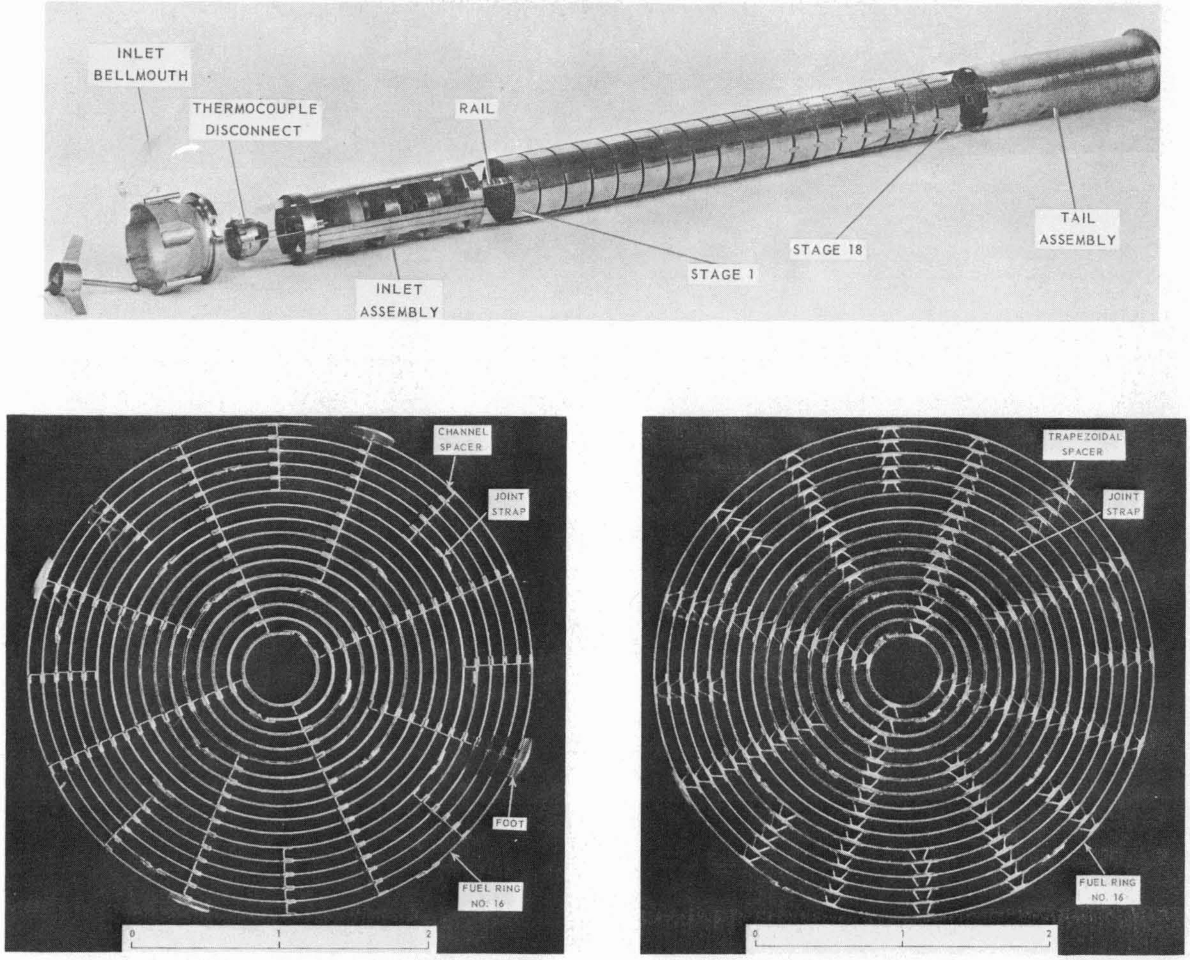

Fig. 3.6—HTRE-1 reactor fuel element and cartridge assembly

3.2.3 SUMMARY OF HTRE-1 OPERATION

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°F, followed by 44 hours at 1380°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 3.1.

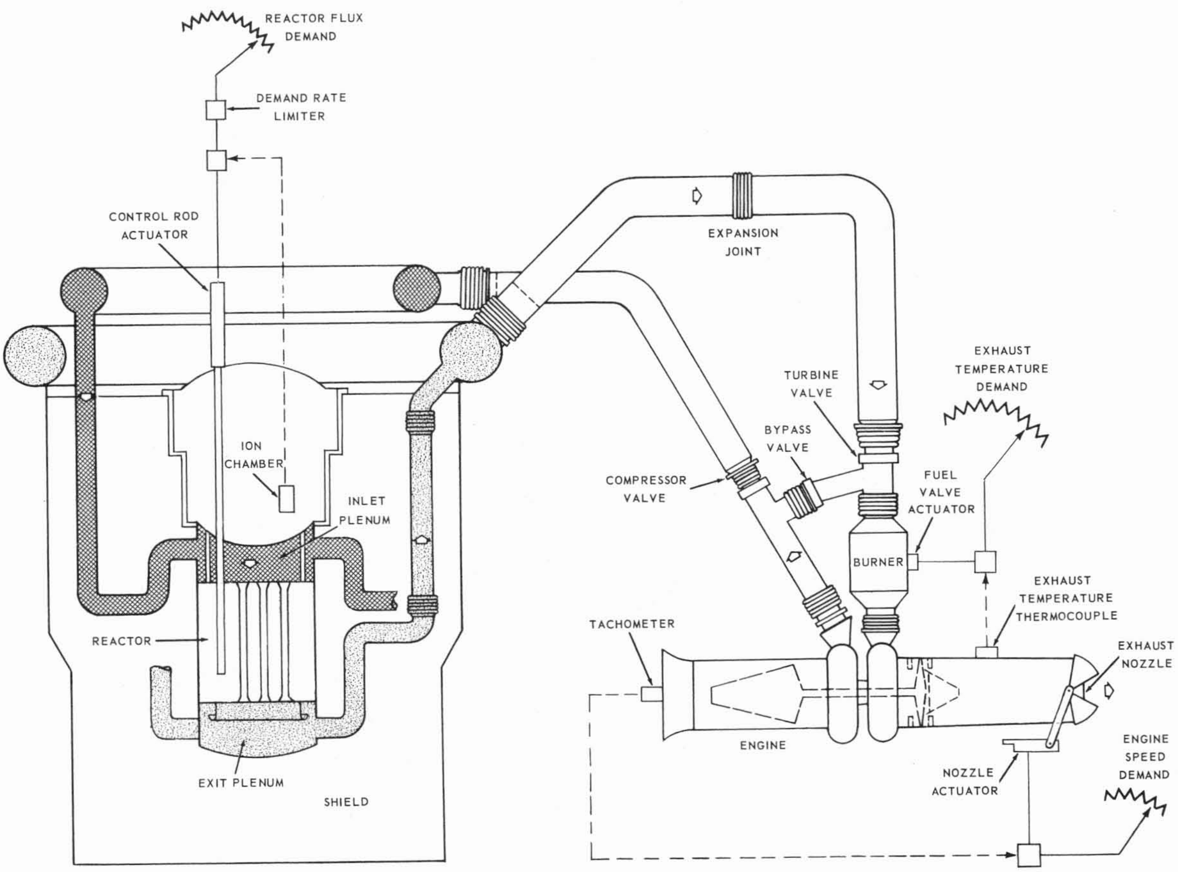

Fig. 3.7 – Schematic diagram of HTRE-1 aerothermal and control systems

TABLE 3.1

HTRE-1 THERMODYNAMIC 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, °F	393
Turbine inlet temperature, °F	1295
Reactor	
Reactor inlet air temperature, °F	359
Reactor inlet pressure, psia	54.95
Reactor exit-air temperature, mean, °F	1335
Maximum average fuel element operating temperature, °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°F for 100 hours and 1380°F for 44 hours.

3.2.4 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 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 more fully in APEX-904, "Heat Transfer Reactor Experiment No. 1," of this Report.

3.3 HTRE-2

The status of reactor development achieved in HTRE-1, if applied to a prototype aircraft propulsion system, would have made possible the flight of a load-carrying aircraft a distance of approximately 50,000 miles at intermediate subsonic speeds without refueling or touching down. This exceeded the range of equivalent chemically powered aircraft by a large factor. However, military application studies indicated that both higher performance levels and longer endurance were desirable.

Endurance and performance are interchangeable to a large extent because of the tradeoff between material operating temperatures and operating life. Therefore, 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.

3.3.1 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 3.8.) A corresponding opening was made in the top shield plug so that sec-

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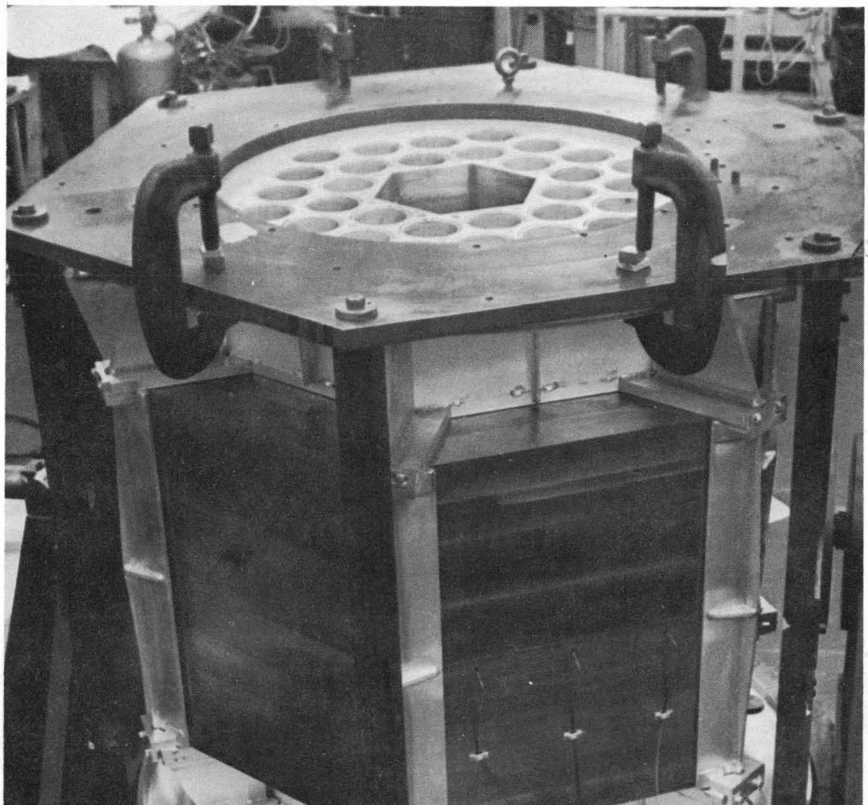

Fig. 3.8 – HTRE-2 reactor during construction, showing the hexagonal cavity used to test advanced reactor components (C-04013)

tions 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 3.9.) No special cooling air circuit was provided for the insert. The air was drawn from the common plenum chamber above the reactor.

Since it was expected that some of the inserts would contribute less to reactivity than the seven fuel cartridges which had been removed, an additional 4 inches of beryllium side reflector was added to the parent core to maintain an acceptable reactivity balance.

3.3.2 SUMMARY OF HTRE-2 OPERATION

HTRE-2 was used principally for the testing of BeO ceramic fuel cartridges of the type planned for the XNJ140E-1, although some tests were performed using the metallic cartridges and hydrided zirconium moderator of the type used in HTRE-3.

Metallic Reactor Tests in HTRE-2

Four metallic fuel element and moderator tests were run in the HTRE-2 test assembly.

1. Insert 1B

   Insert 1B consisted of seven hydrided zirconium moderator elements of hexagonal cross section, each containing a metallic fuel element cartridge similar to HTRE-1 cartridges. The moderator tubes were clad with stainless steel. A primary purpose of the test was to evaluate clad hydrided zirconium as a moderator material at temperatures of 1600°F. The insert was operated at this temperature for 38 hours. The 1B test was also used to establish the operating characteristics of the HTRE-2 reactor.

2. Insert 1C

   Insert 1C was similar to 1B, but no cladding was used on the moderator elements. A photograph of the insert is shown in Figure 3.10. The insert was operated for 100 hours with a maximum moderator temperature of 1200°F. The test verified that the prediction that the hydrided zirconium moderator could be used in an unclad condition at about 1200°F without excessive oxidation or loss of hydrogen.

3. Insert 1D

   Insert 1D was similar to 1C except that remotely operable pneumatic valves were mounted at the inlet end of two of the air passages to reduce airflow to the fuel elements during operation. The purpose of the test was to evaluate the behavior of a fuel element meltdown caused by eliminating airflow to the fuel cartridge. Post-operation examination showed that melting had occurred and that a substantial fraction of the fuel cartridge had broken away. A considerable amount of the portion that broke away was held up in the tailcone assembly and ducting; a smaller amount escaped through the exhaust. The moderator element within which the cartridge was contained was essentially unaffected by the cartridge meltdown.

4. L2C1 Insert Cartridge

   The L2C1 cartridge was a test of a single metallic fuel cartridge with a central hydrided zirconium moderator rod. The test specimen was surrounded by a water-cooled beryllium adapter in order to peak the neutron flux. The fuel cartridge was fabricated from fuel sheet consisting of UO_2 dispersed in chromium-titanium and clad with an iron-chromium-yttrium alloy. This was a fuel element material proposed for operation in the XMA-1 reactor for operation at temperatures exceeding

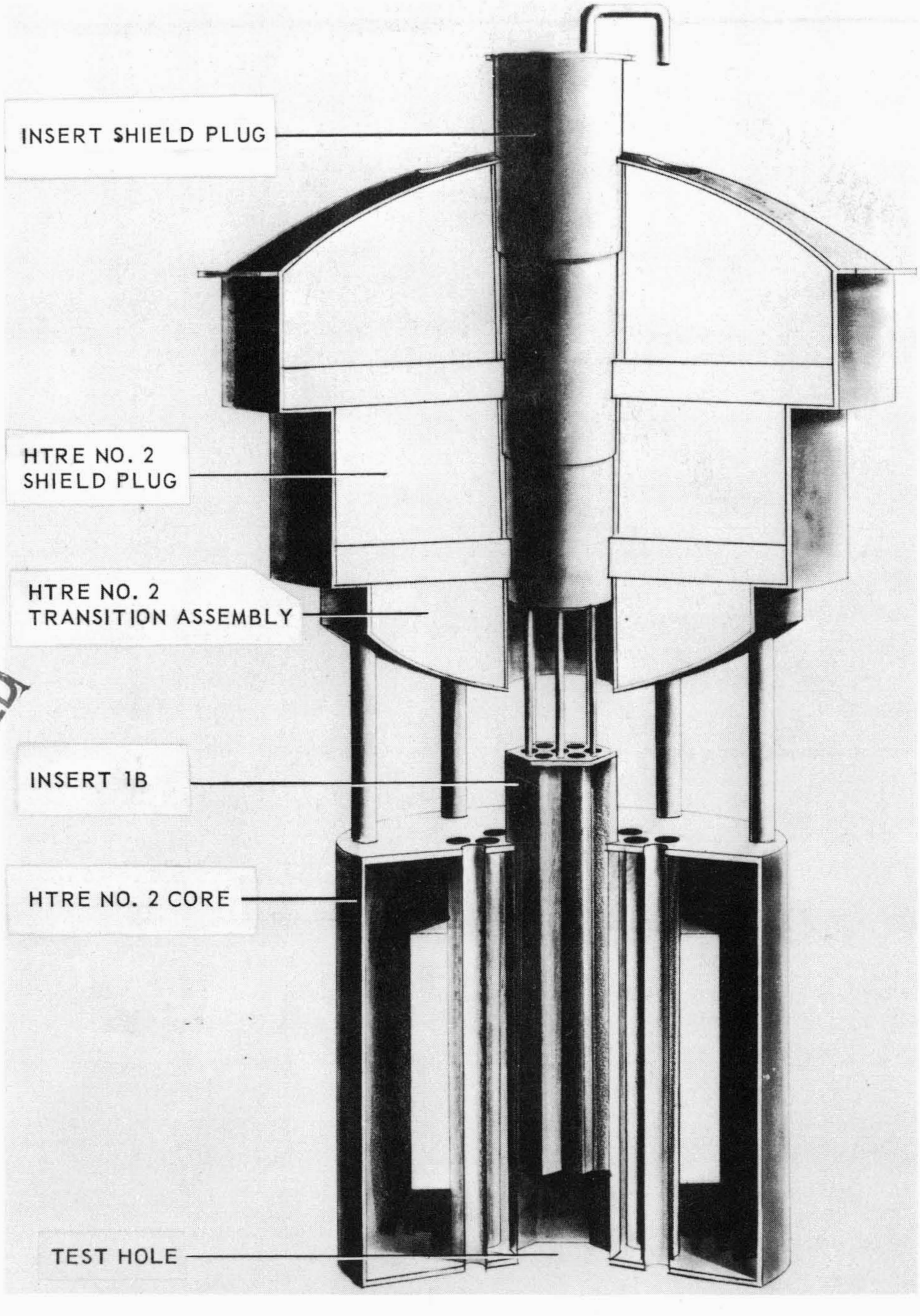

Fig. 3.9 – Artist's conception of HTRE-2 parent reactor, shield plug, and test insert (040-513)

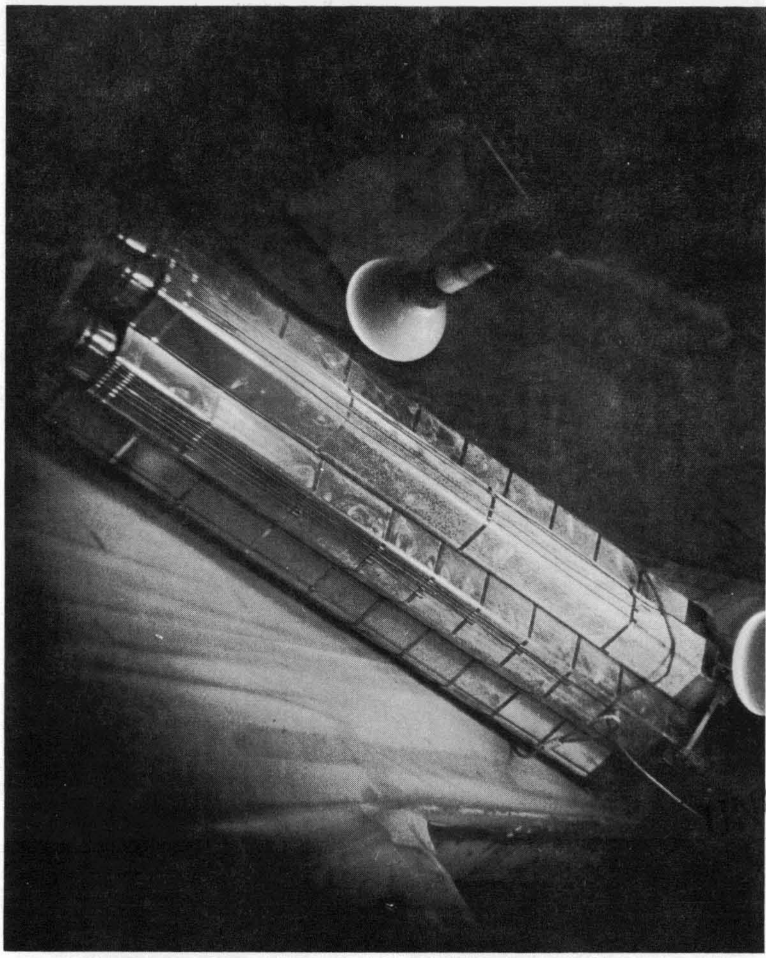

Fig. 3.10 – Insert 1C after test (Neg. C-13630)

the capability of nickel-chromium. The cartridge was tested for a total of 80 hours with a maximum indicated fuel plate temperature of 2090°F. The maximum moderator temperature was 1790°F. Post-operation examination of the cartridge showed that the general mechanical condition was good. However, there were a number of fuel sheet blisters which in some cases had ruptured. This accounted for an increase in effluent activity which had been observed during operation.

Ceramic Test Inserts

HTRE-2 was operated for more than 1100 hours while testing a variety of ceramic inserts. The first insert, 2B (Figure 3.11) occupied the entire reactor test cavity. Subsequent insert "cartridges" were smaller in cross section and were surrounded by a water-cooled beryllium adapter to peak the neutron flux in the test specimen. An end view of a typical cartridge after test is shown in Figure 3.12. A summary of ceramic insert tests is given in Table 3.2.

1. Insert 2B Cartridge

Insert 2B 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. This was an

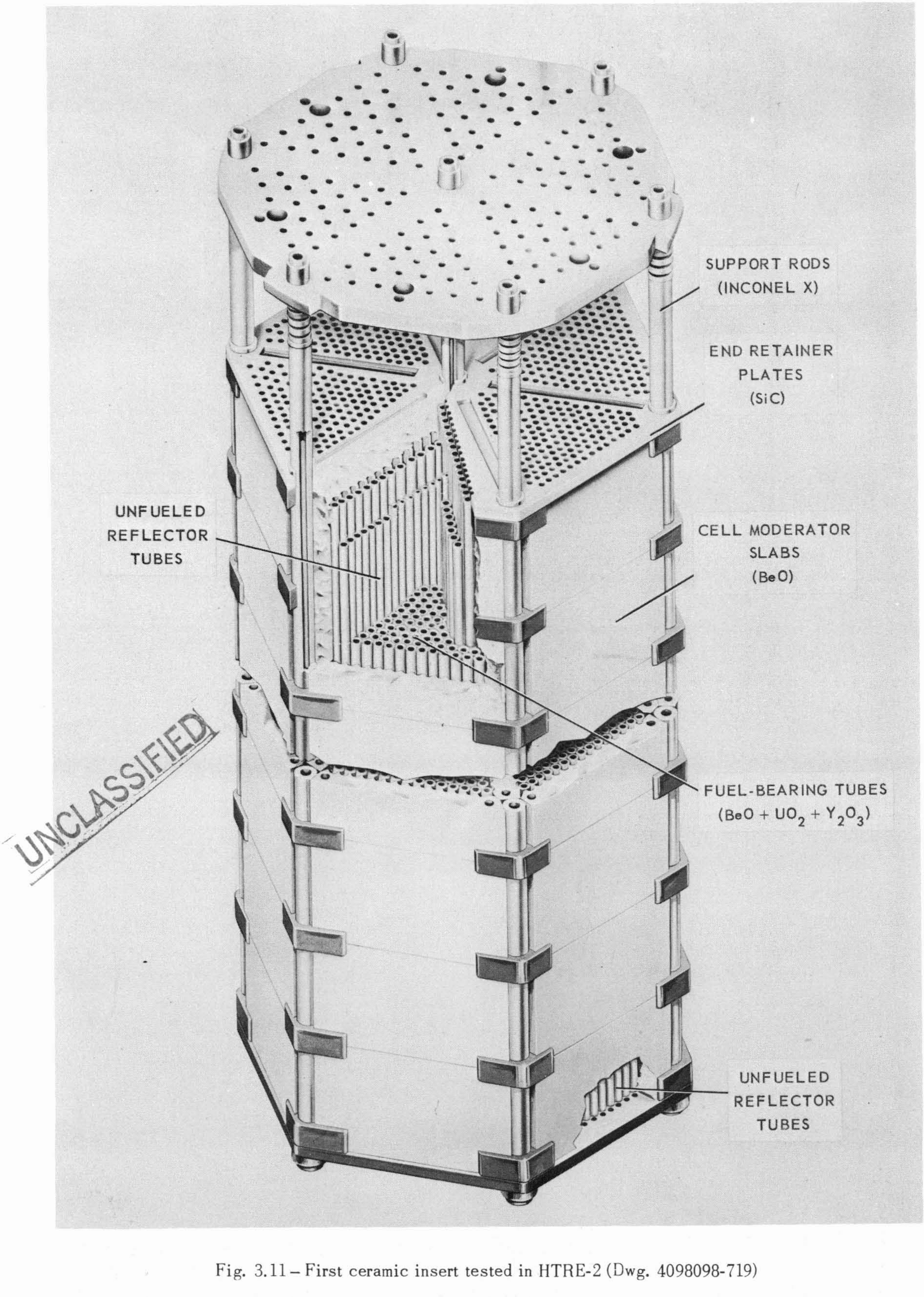

Fig. 3.11 – First ceramic insert tested in HTRE-2 (Dwg. 4098098-719)

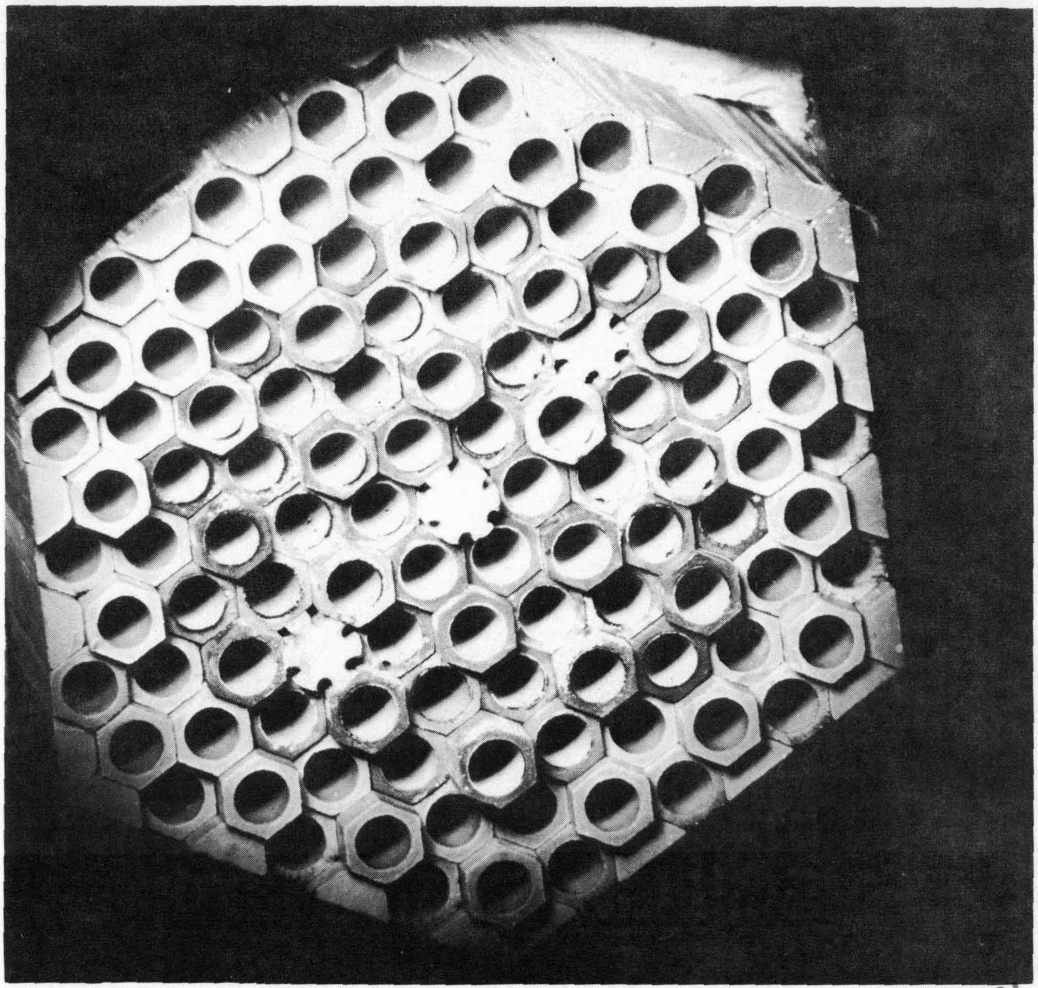

Fig. 3.12 – BeO insert tested in HTRE-2 representative of XNJ140E-1 fuel element design

accurate mockup of a proposed beryllium oxide reactor design. The use of separate moderator slabs was eliminated in later reactor designs.

The insert survived the test operation at an average temperature of 2830°F, in good mechanical condition without tube failure. Several slabs were cracked in both the moderator and aft retainer plate, however, this did not affect the over-all mechanical integrity of the insert. A considerable amount of BeO transport had occurred from one portion of the insert to later stages because of the hydrolysis of the BeO by atmospheric water vapor. The predicted need for a coating to protect against this effect had been one of the problems which had delayed early adoption of BeO reactors. The hydrolysis was considerably more severe than would have been the case in an operational reactor under high altitude conditions because of the relatively large amount of water in the air at the altitude of the National Reactor Testing Station.

2. L2A1, L2A2 Insert Cartridges

The L2A1 and L2A2 inserts used round, uncoated beryllium oxide tubes similar to those used in the 2B. The inserts were operated at temperatures of 2500°F and 2700°F respectively, whereas 2B operated at 2830°F. Again the tubes survived the test in good mechanical condition. Water vapor corrosion was less than in the 2B test, suggesting that this effect was a sensitive function of operating temperature.

TABLE 3.2

SUMMARY OF HTRE-2 BERYLLIUM OXIDE FUEL ELEMENT TESTS

Insert No.	Fuel Tube, I.D. Clad	Nominal Maximum Insert Temperature, °F	Maximum Power Density, Btu/in^3-sec[a]	Test Time, hr
2B	bare	2550	1.86	25
		2830		100
L2A1	bare	2500	2.57	100
L2A2	bare	2700	2.51	94
L2E1	Al_2O_3	2500	2.69	106
L2E2	Al_2O_3	2500	2.75	46
		2600		99
L2E3	ZrO_2	2500	3.37	102
		2600		102
L2E4	b	4400	3.15	(10 min.)
L2E5	bare	3700	3.22	2
L2E6	ZrO_2	2500		50
		2600		54
		2650	2.82	56
		2700		193
		2750		3

[a]Based on nuclear analysis.
[b]Bare, ZrO_2 and Al_2O_3 clad tubes were used.

3. L2E1, L2E2, and L2E3 Insert Cartridges

The L2E1 and L2E2 tubes were coated on the inside surface with a 0.0015-inch-thick layer of aluminum oxide (Al_2O_3) to protect against water vapor corrosion. The L2E3 tubes used a 0.005-inch thickness of zirconia (ZrO_2) coating on the inside surface. The L2E2 and L2E3 fuel tubes were hexagonal rather than round in cross section.

Again all tubes survived in mechanically good condition with no breakage. The coatings effectively eliminated the problem of water vapor corrosion and verified that protection was not required on the outer surface of the tube.

4. L2E4 Insert Cartridges

Both coated and uncoated hexagonal fuel tubes were used in the L2E4 insert cartridge. The test was a deliberate attempt to evaluate the nature and propagation of fuel element damage that could result from a complete loss of airflow in the center 18 air passages with normal airflow in adjacent tubes. Post-test examination indicated that melting temperatures had been reached in the center of the cartridge but that the damage was restricted to the uncooled region. The immediately adjacent tubes had fused together, thus bridging around the damaged section while maintaining completely unobstructed air passages.

5. L2E5 Insert Cartridge

Uncoated hexagonal tubes were used in the L2E5 insert. The insert was used similarly to the L2E4 except that approximately 10 percent of normal airflow was allowed to pass through the center tubes. The L2E5 test verified the prediction that tubes deprived of 90 percent of the airflow would overheat but would not melt. Tube temperatures of 3800° to 3900°F were reached in the central region with no obstruction of the air passages except for a deposit of beryllia crystals due to hydrolysis of the uncoated tubes.

6. L2E6 Insert Cartridge

The L2E6 fuel tubes were hexagonal and coated in the inner surface with 0.003 inch of ZrO_2. The test objective was to verify the mechanical integrity of the fuel tube and cladding material at temperatures higher than those planned for the proposed XNJ140E-1 power plant. The fuel elements were operated at various temperatures from 2500° to 2750°F for a total of about 350 hours. Post-operation examination revealed that the cartridge survived the test in good mechanical condition. Although a number of individual tubes were cracked, the over-all mechanical integrity of the cartridge was not affected.

HTRE-2 Parent Core

Although intended primarily for insert testing, the HTRE-2 parent core supplied additional useful information about the life potential of the metallic fuel elements used in the basic HTRE-1 type design. After the first 552 accumulated hours at test power levels, 25 of the 30 fuel cartridges were replaced in the parent core. The reason for this was that fission product poisoning and fuel depletion had reduced the excess reactivity margin sufficiently to warrant a recharge of the reactor. The fuel elements themselves appeared to be metallurgically and mechanically in good condition and would be capable of operation for an indefinitely longer period if reinstalled into a reactor of greater excess reactivity margin. Additional replacements were made after 997 hours operation.

In summary, the HTRE-2 parent core operated at power for an accumulated total of 1299 hours with fuel element temperatures between 1200° and 1750°F and reactor exit-air temperature between 875° and 1125°F and was still in operation at the termination of the ANP program. Five of the fuel cartridges were in use for 997 hours with no external manifestation of metallurgical or mechanical difficulties. A photograph of a parent core fuel element after 997 hours operation is shown in Figure 3.13.

3.3.3 FINAL STATUS OF HTRE-2 DEVELOPMENT AND APPLICATION

HTRE-2 operation verified the use of hydrided zirconium as a reactor material, providing a firm basis for the HTRE-3 and XMA-1A reactor designs. 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. The continued integrity of the fuel tubes surrounding a locally overheated region was demonstrated.

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

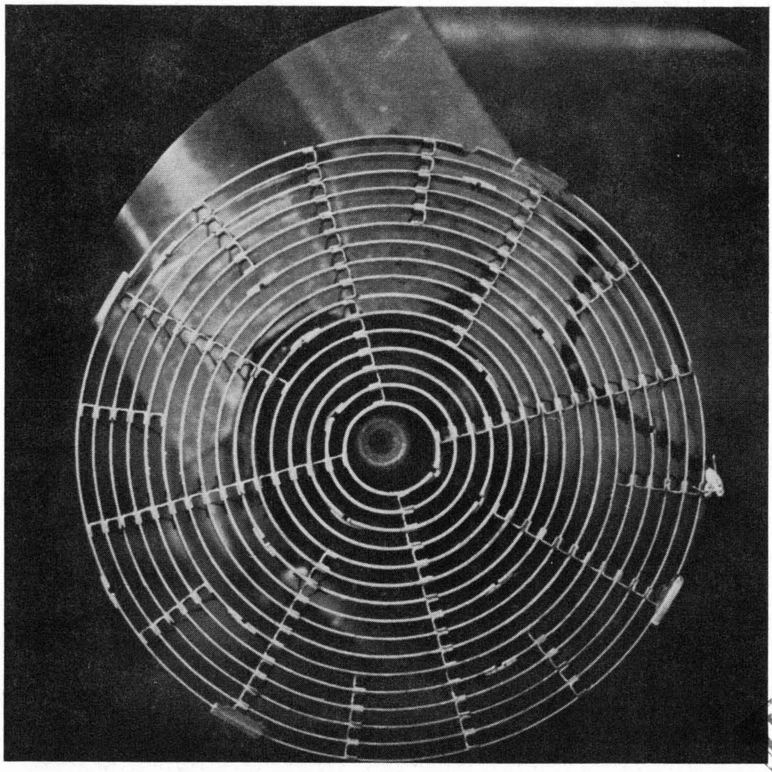

Fig. 3.13 – Upstream face of stage 18 cartridge 452 after 997 hours on test (Neg. U-4198-44)

fragment release rate of nuclear propulsion systems using clad ceramic fuel element materials was within tolerable limits.

HTRE-2 is presented in greater detail in APEX-905, "Heat Transfer Reactor Experiment No. 2," of this Report.

3.4 HTRE-3

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

The design of the HTRE-3 reactor and other components started in early 1956. The first operation on nuclear power occurred in 1958; further operation continued through 1960.

3.4.1 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 compatability 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.

3.4.2 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 3.14. These components and the required test support equipment were mounted on a mobile dolly, similar to the CTF dolly, as shown in Figure 3.15. 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 3.16. 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 3.17. A drawing of the reactor is shown in Figure 3.18.

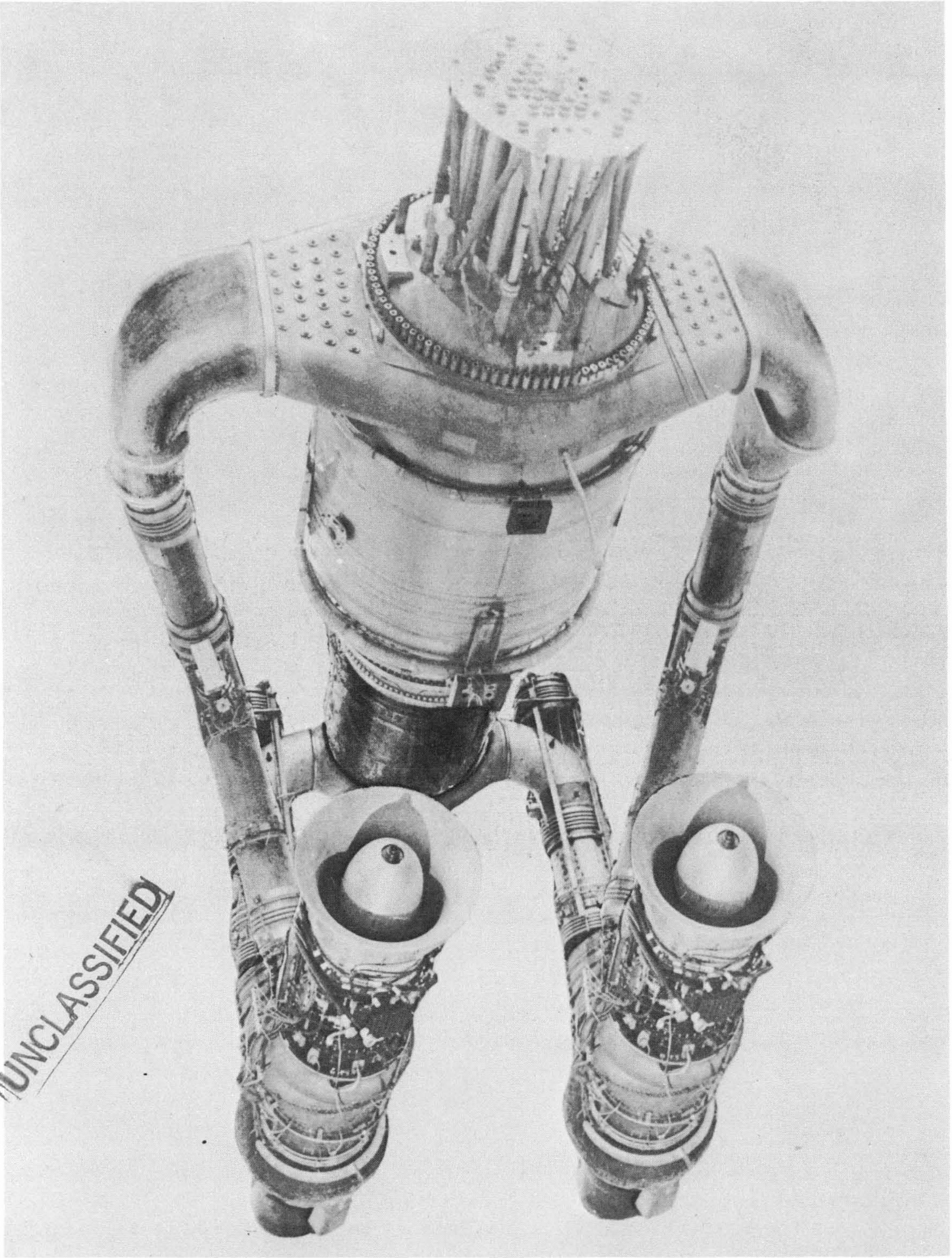

Fig. 3.14 – HTRE-3 basic components (U-2189)

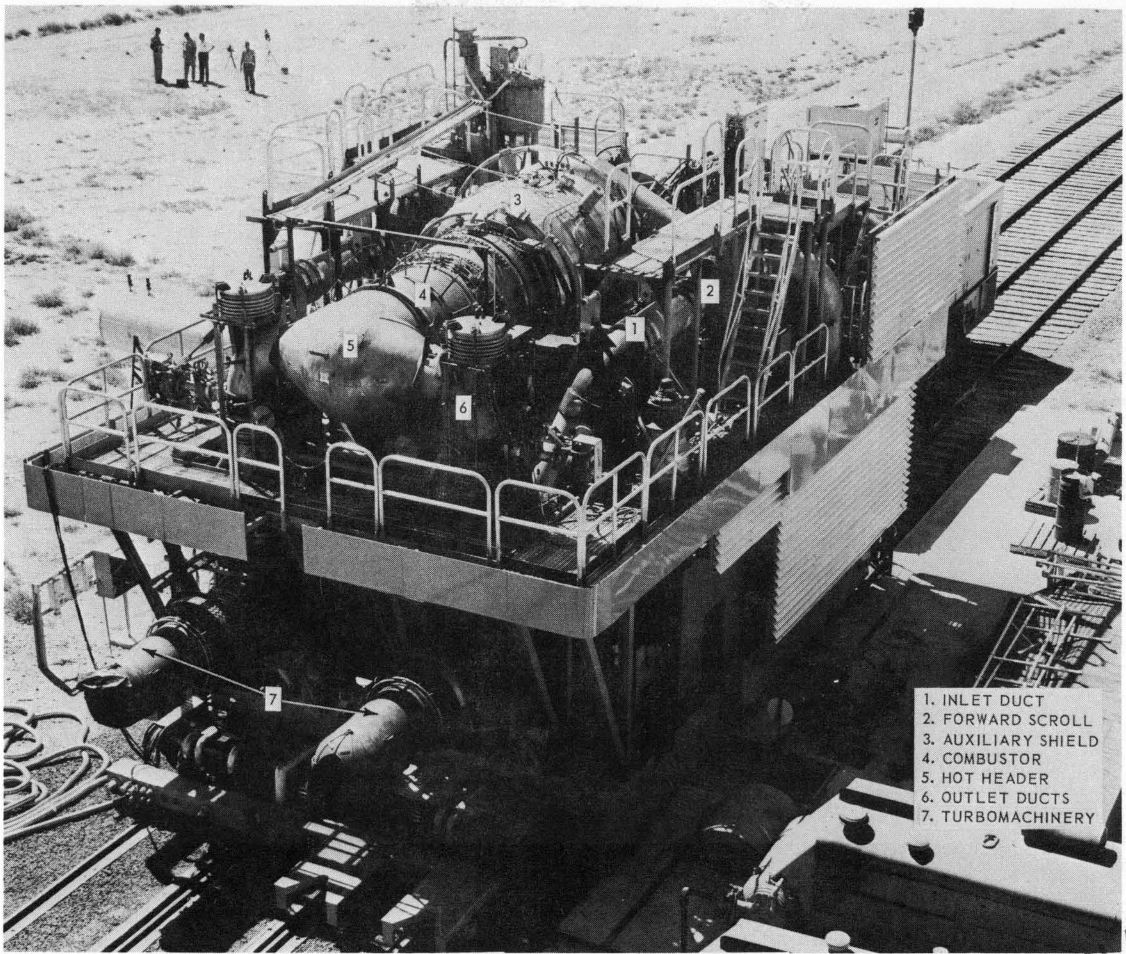

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Fig. 3.15 – HTRE-3 test assembly (U-2700-11)

An assembled fuel cartridge is shown in Figure 3. 19. 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 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.

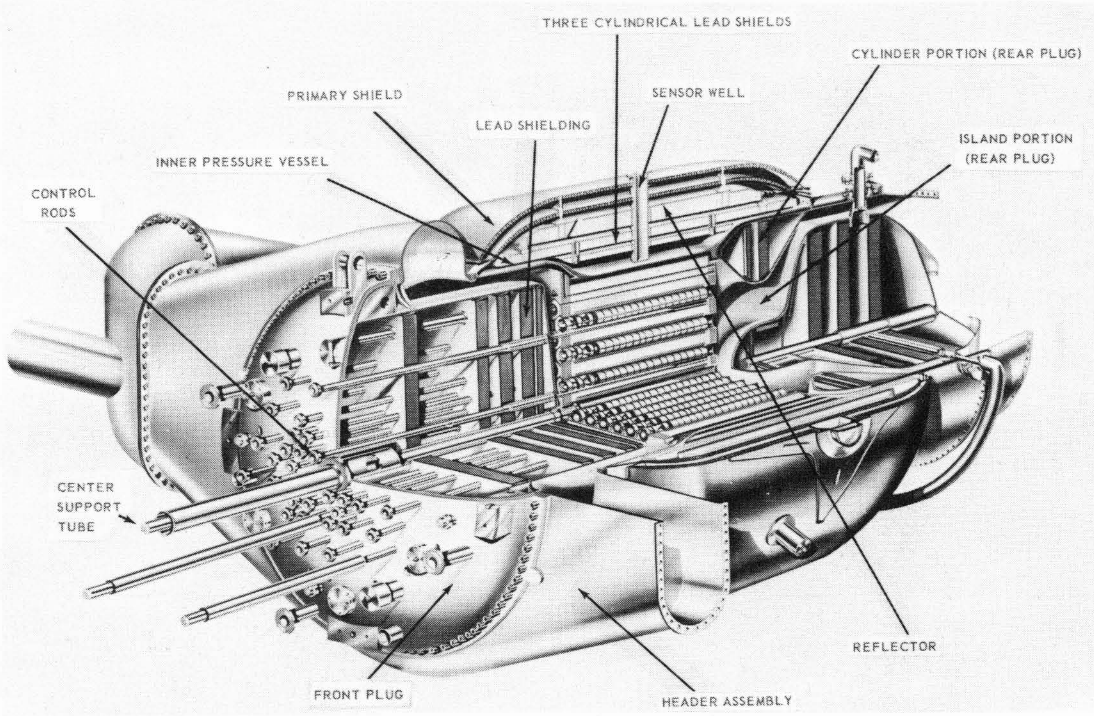

Fig. 3.16 – HTRE-3 reactor-shield assembly

3.4.3 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

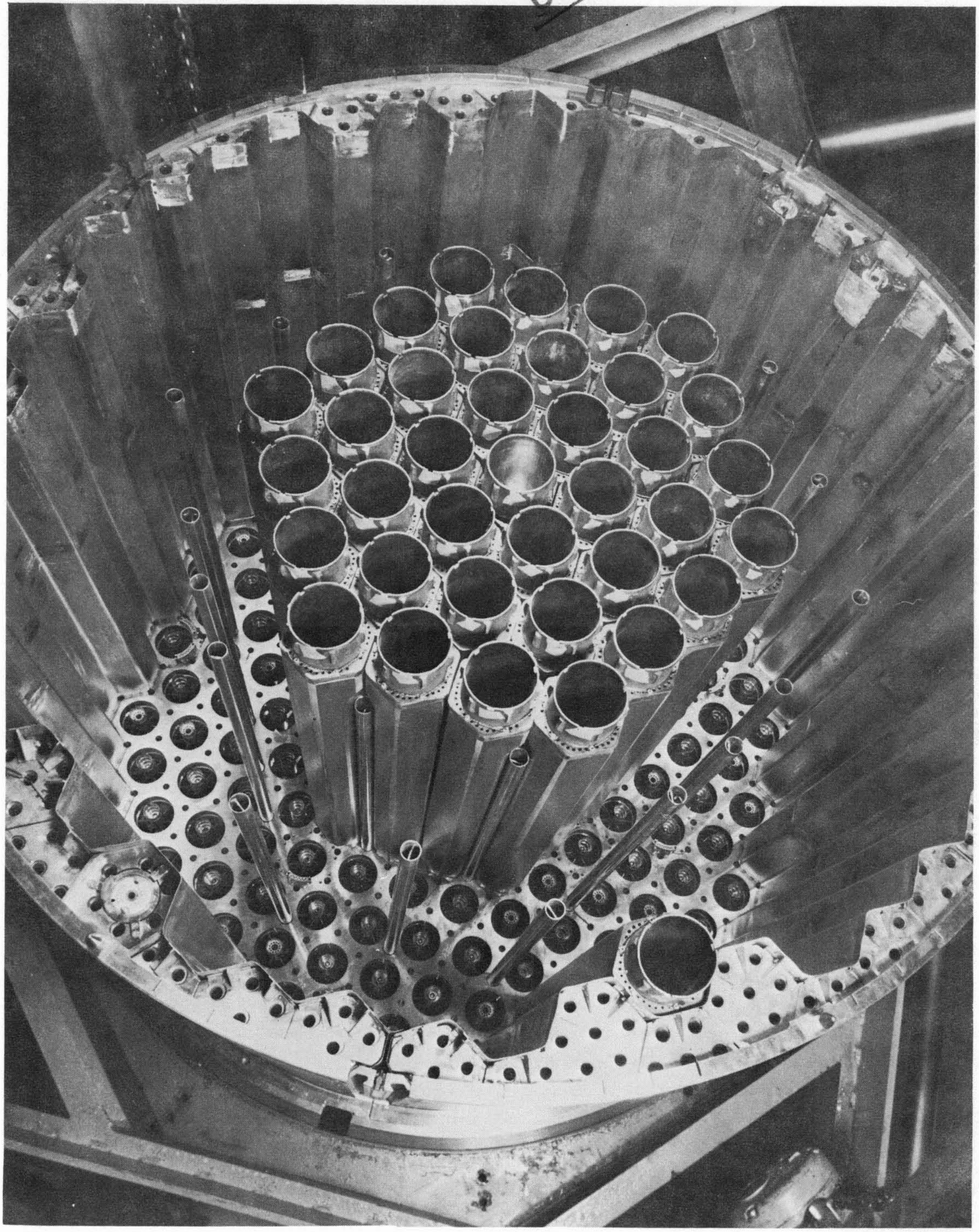

Fig. 3.17 – HTRE-3 reactor during assembly (C-13822)

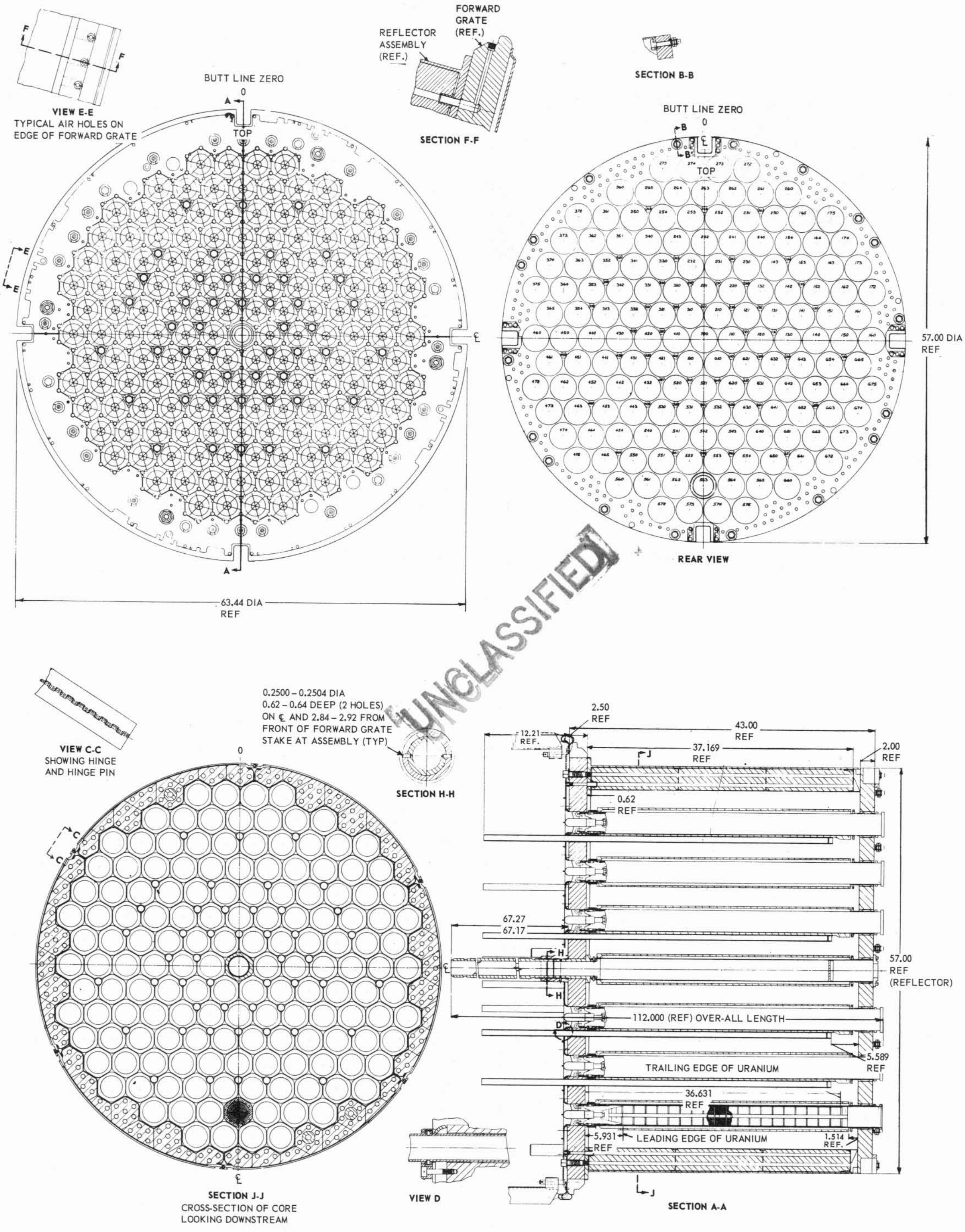

Fig. 3.18 – HTRE-3 core assembly (Dwg. 7018R51)

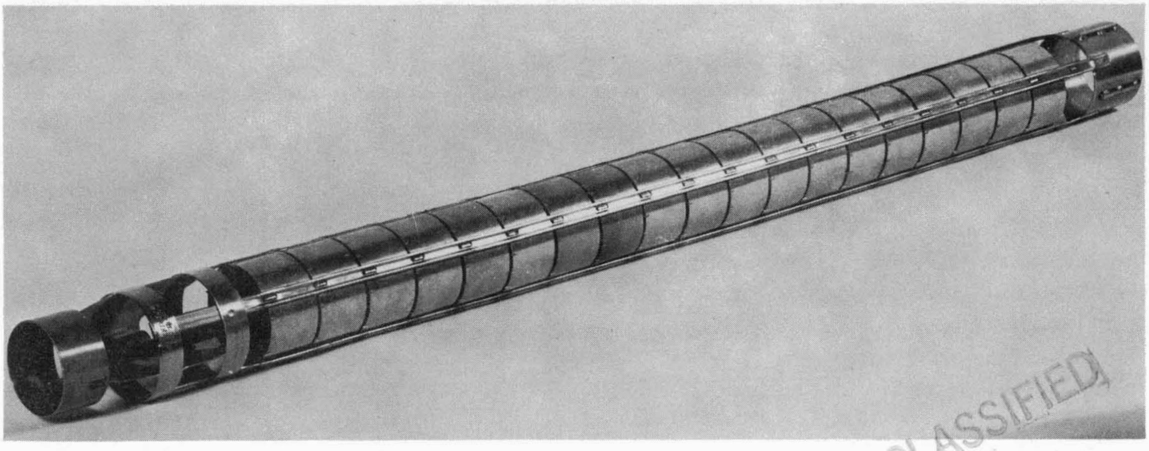

Fig. 3.19 – HTRE-3 fuel cartridge

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 3.3.

3.4.4 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-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.

TABLE 3.3

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, ^{o}F	1330		1370
Compressor discharge temperature, ^{o}F	385		376
Compressor discharge pressure, psia	53.5		
	Predicted	Measured	Measured
Maximum temperature, ^{o}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, ^{o}F			
Discharge air from moderator	968	880	900[a]
Discharge air from reflector	955	940	845[a]
Discharge air from control rods	805	480	485[a]

[a]Thermal equilibrium not reached.

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 APEX-906, "Heat Transfer Reactor Experiment No. 3," of this Report.

3.5 PROPOSED HTRE-4 AND CERAMIC CORE TEST

Because of the steadily improving status of ceramic materials relative to metals in applications above 2000^{o}F, two ceramic reactor design studies and concurrent 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 the Advanced Core Test in a prototype power plant configuration, the XNJ140E (discussed in section 4).

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 3.20.

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 channelled through circular bores in the tubes. The D141E-1 is shown in Figure 3.21.

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 study were used as the basis of subsequent XNJ140E design.

Both HTRE-4 and the CCT are described in APEX-908, "XNJ140E Nuclear Turbojet," of this Report.

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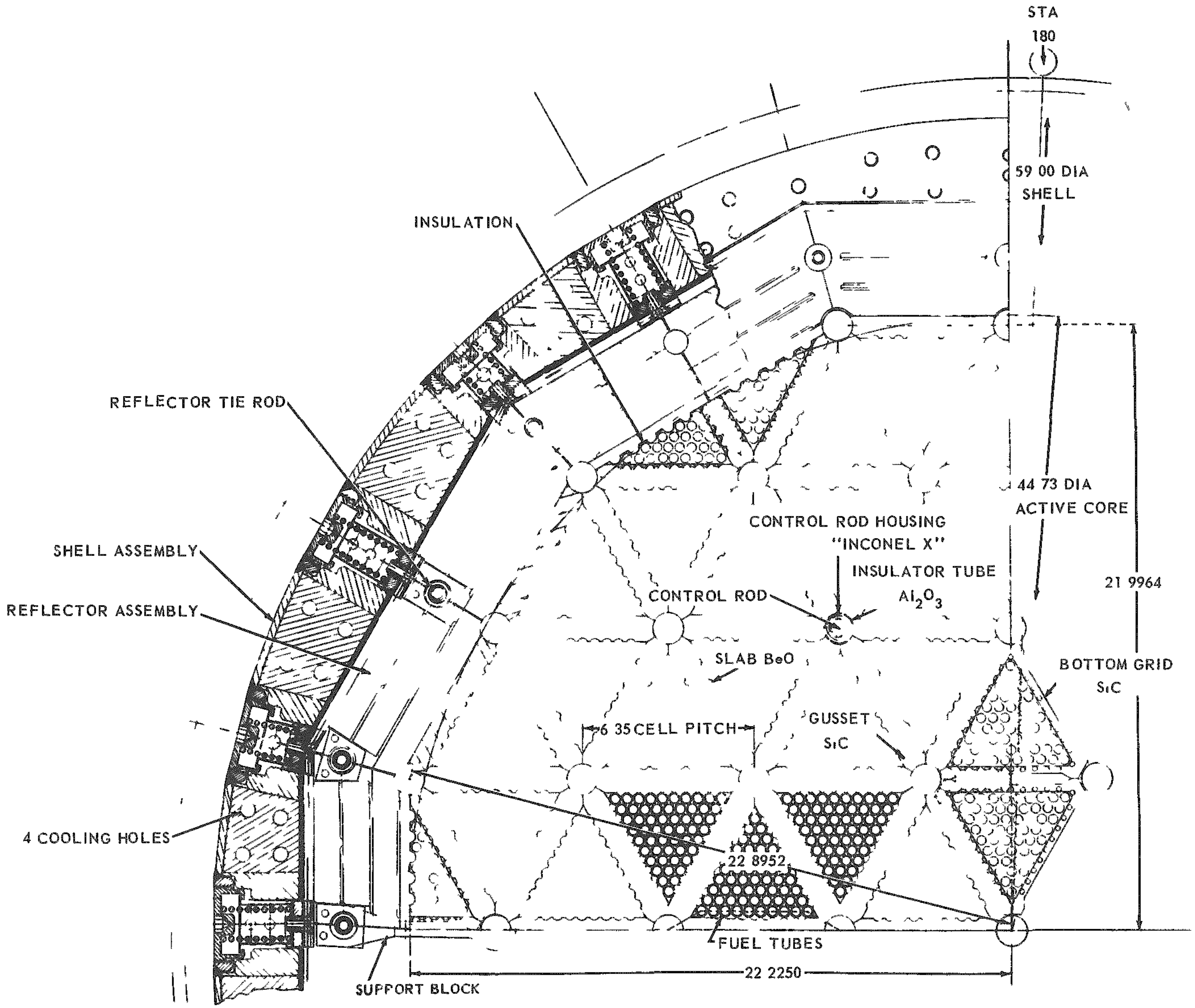

Fig. 3.20 – D101L ceramic reactor for the proposed HTRL-4

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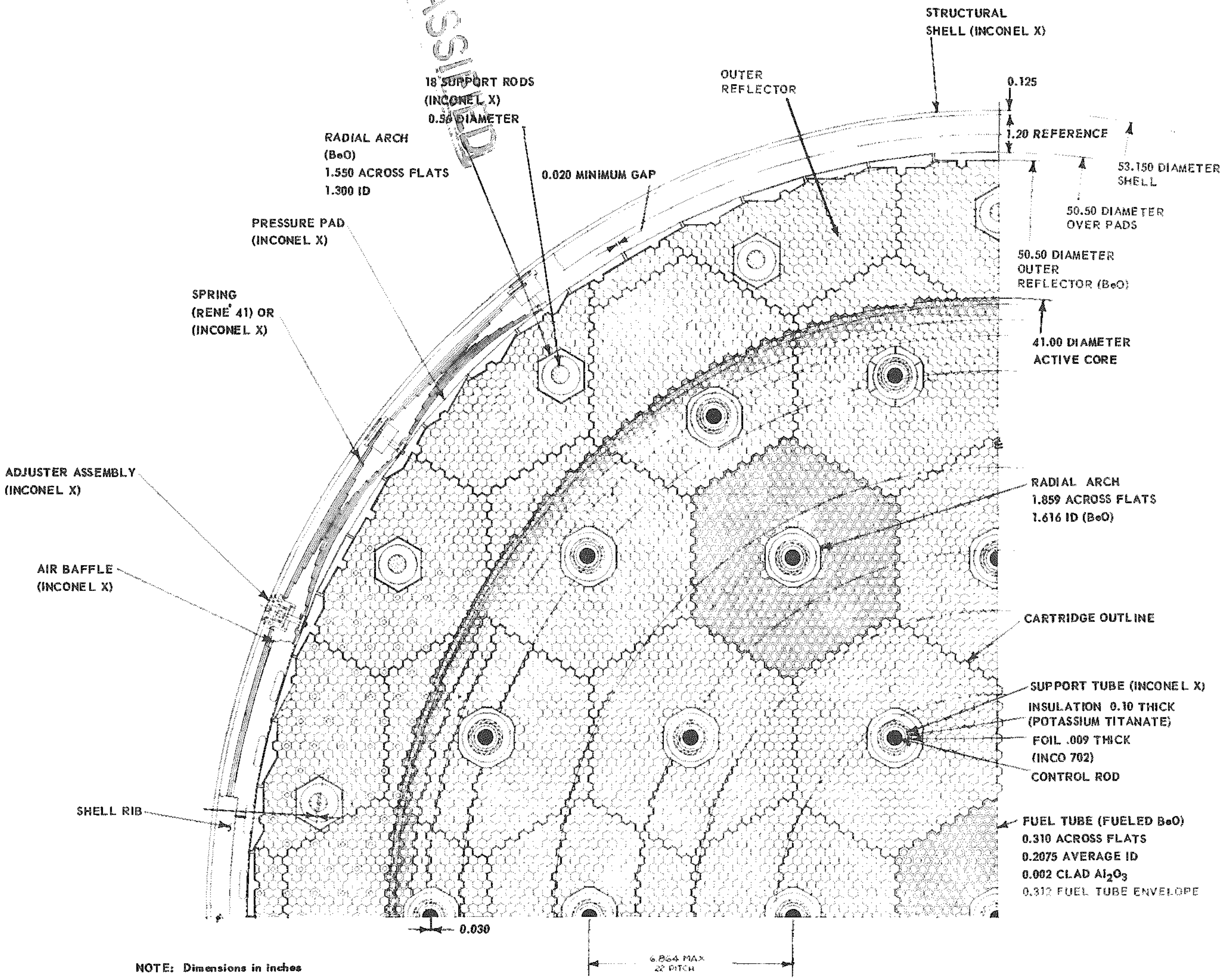

Fig. 3.21 – D141A-1 reactor for the proposed CCT

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4. PROTOTYPE POWER PLANT DESIGN AND DEVELOPMENT

4.1 XMA-1 POWER PLANT

With the issuance, in 1955, of Air Force Systems Operational Requirement No. 81 for the proposed 125A Weapons System, a specific military objective was established for the ANP program. SOR No. 81 required "A Piloted Nuclear-Powered Intercontinental Strategic Bombardment Weapons System" capable of extended cruise without in-flight refueling, penetrating enemy defenses at very high altitudes and supersonic speeds, and low-level attack at subsonic speeds. The following specific requirements were established:

1. Continuous cruise for a minimum of 40 hours at Mach 0.9 and 20,000 feet
2. Sprint over the target area for a distance of 2000 nautical miles at Mach 2.5 and 55,000 to 60,000 feet
3. Low-level penetration at Mach 0.9 and 500 feet

The design and development of a power plant, designated the XMA-1, was undertaken to meet these requirements. The cruise portion of the mission was entirely nuclear with chemical afterburning during the supersonic sprint. Initial ground test was scheduled for 1959, with the initial flight test in 1960. A model of the XMA-1 power plant is shown in Figure 4.1; an artist's concept of the principal components is shown in Figure 4.2. The Convair Division of General Dynamics, Ft. Worth, Texas, undertook a study of the aircraft to meet the 125A Weapons System requirements.

Late in 1956, a decision was made to de-emphasize aircraft development but to continue developing the propulsion system at a reduced level. The 125A Weapons System objective was withdrawn. XMA-1 development continued at the reduced level.

A new objective was provided by the Air Force in 1958 as Systems Operational Requirement No. 172 for "A Continuous Airborne Missile Launcher and Low Level Weapons System" (CAMAL). The CAMAL mission retained the extended cruise and low-level penetration of the 125A mission but substituted the use of long-range air-to-ground ballistic missiles for the high altitude supersonic portion of the flight regime. The following specific requirements were established:

1. Continuous cruise for a minimum of 120 hours at Mach 0.85 and 30,000 feet
2. Air launch of ballistic missiles from outside the target area
3. Low-level penetration at Mach 0.9 and 500 feet

The XMA-1 development was redirected toward the new objective. A first flight in 1963 was assumed as a target date, using an early development model of the reactor. The proposed Convair Model 54 aircraft was considered to be the flight vehicle.

The CAMAL objective continued as the basis for the XMA-1 development until 1959, when new guidance was furnished by the Department of Defense. The early flight objective was withdrawn in favor of concentrating on the development of materials and designs

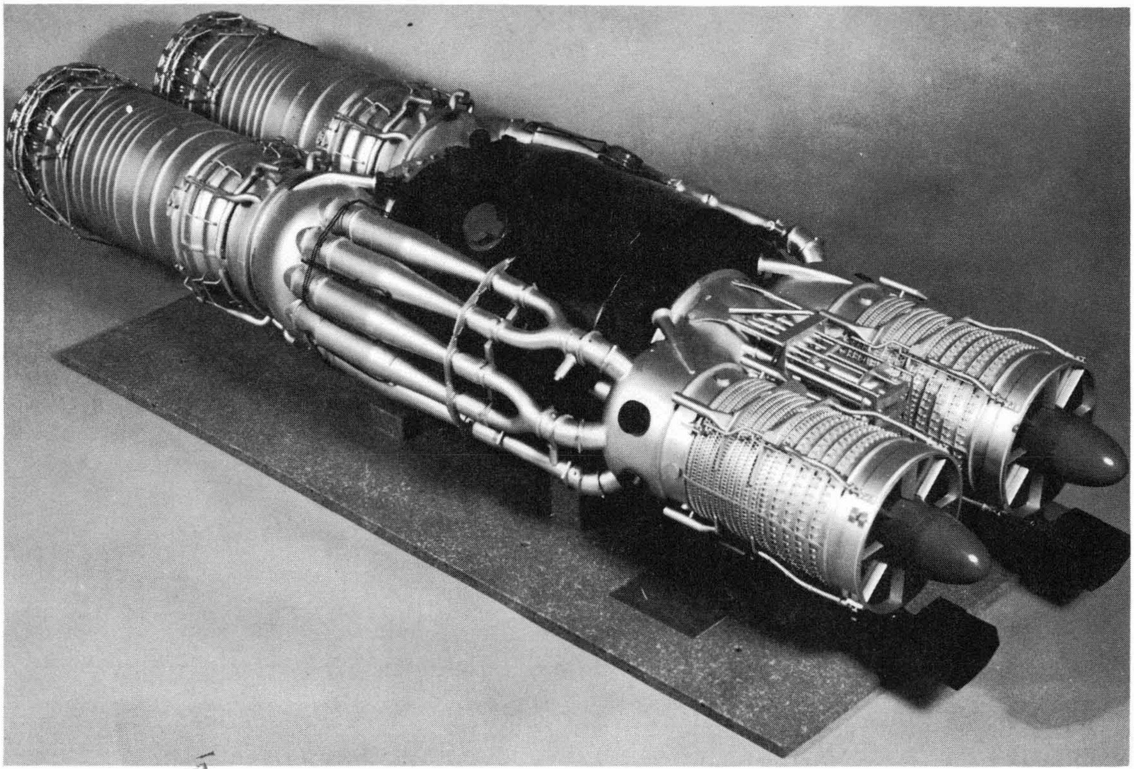

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Fig. 4.1 – Model of XMA-1 power plant (U-39336B)

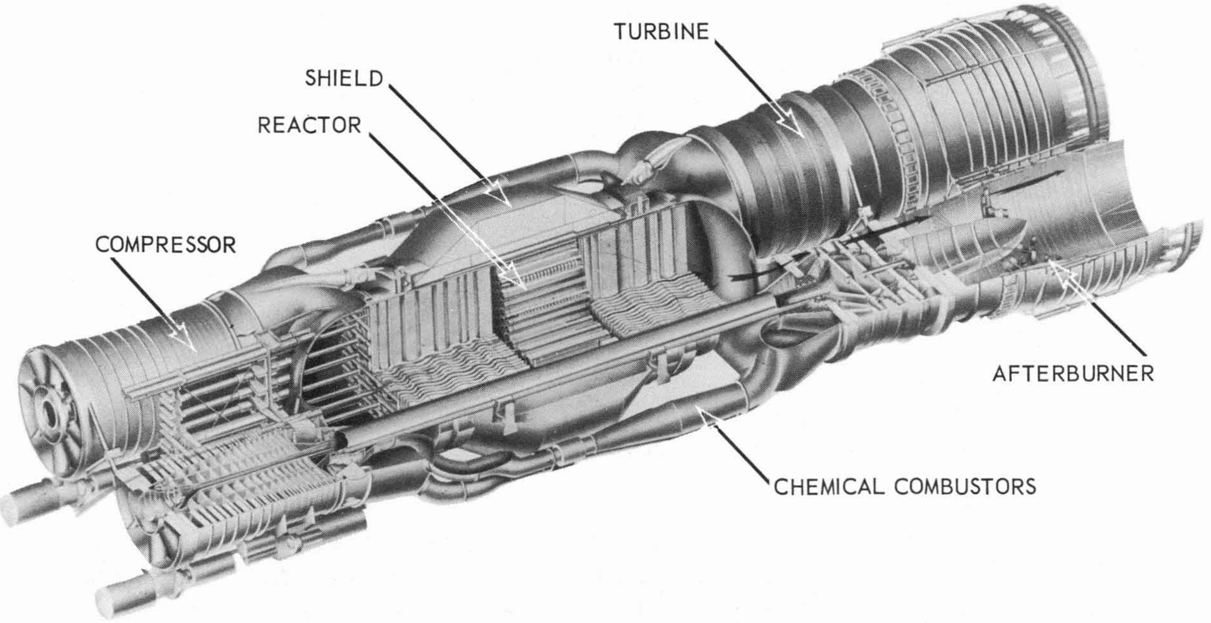


Fig. 4.2 – Cutaway of XMA-1 power plant

capable of higher levels of performance. The XNJ140E power plant, using a more advanced reactor, was adopted to meet the growth requirement.

4.1.1 MATERIALS AND DESIGN SELECTION

Selection of Turbomachinery and Configuration

Turbojet engines were best suited for the flight regimes required in both the 125A and CAMAL Weapons Systems. Anticipating that similar flight system requirements would be issued, optimization studies had been performed in 1954 and 1955 to select the compressor pressure ratio, airflow, turbine inlet temperatures, and midsection pressure drops required for an engine of this type (reference 1). Several factors entered into these studies. In a conventional turbojet engine the specific weight of the engine per unit of airflow increases with increased airflow. The trend in the nuclear system is just the opposite. The specific weight of the reactor-shield assembly per unit of airflow and power decreases. On this basis, a single large reactor as the heat source for several small engines would appear to be preferable. However, such a system is more complex than one using fewer engines.

High compressor discharge pressures are desirable because a smaller flow area can be achieved per unit of airflow. This results in a smaller power plant and less frontal area. A high pressure, dense working fluid is particularly desirable in minimizing reactor size. On the other hand, high pressures impose higher stress loads on a number of components. Furthermore, high compression increases the air temperatures. Because the maximum allowable temperature in a nuclear system is imposed by materials, the amount of additional heat which can be added by the reactor is limited if the reactor inlet temperature is already high. Still another factor to be considered is the high ram-air compression at higher flight speeds, which reduces the necessity for further mechanical compression. In a supersonic ramjet, for example, mechanical compression is not used at all.

The selection of the design for the XMA-1 turbomachinery was based on a consideration of these and other factors. The following performance objectives were established: a compressor pressure ratio at sea level static of approximately 14:1, an airflow of approximately 400 pounds per second, an allowable pressure drop of 20 to 30 percent between compressor discharge, and a growth potential to turbine inlet temperatures of about 2000°F and higher. The engine, designated the X211, was placed under development in the Large Jet Engine Department of the General Electric Company.

An even lower specific weight could have been achieved using an even higher rate of airflow, perhaps in excess of 1000 pounds per second. However, the unit size of such an engine would have reduced the flexibility of application, and would have required a size extrapolation well beyond existing turbomachinery practice.

In the power plant configuration selected for the XMA-1, two sets of X211 turbomachinery were coupled to a single reactor-shield assembly. This was similar in concept to the AC-101B configuration (described in section 2) except that the reactor and engines were much more closely coupled. This followed the trend toward closer integration of the power plant components.

Reactor Selection

Both ceramic and metallic fuel elements were considered for the XMA-1 reactor. With further materials development, the metallic fuel elements of the configuration used in the HTRE tests appeared to be capable of meeting the XMA-1 temperature and endurance requirements. Although the potential of the ceramics was even higher, they were in a less advanced state of development than the metallics. Therefore, the principal design and component development was directed toward higher temperature metallic fuel elements, with a lesser effort continuing on ceramics.

A solid hydrogenous moderator was selected for the XMA-1. This choice was made because the 125A supersonic sprint and the low-level, high subsonic speed requirements for both the 125A and CAMAL Weapons Systems were severe environmental conditions in which to use liquid moderators, for two reasons. First, since available hydrogenous liquids with acceptable chemical properties must remain at a relatively low temperature, waste heat rejection becomes difficult at high speeds because of the high ram-air temperature. Secondly, a liquid system is particularly subject to leakage and damage by foreign objects, especially at low altitudes. Historically, these same two factors had been instrumental in the adoption of air-cooled rather than liquid-cooled reciprocating engines for aircraft.

With the advent of the CAMAL objective, and in view of successful experience with HTRE-3, the decision was made to use HTRE-3 materials in the design of an initial model of the XMA-1 reactor in order to achieve early ground and flight test operation of the power plant. Thus, the basic power plant could be checked out at reduced performance levels, using fully developed reactor materials. Conversely, the reliability of the basic power plant would have been developed to a relatively advanced state prior to substitution of a higher temperature reactor. The early model incorporating the HTRE-3 materials was designated the XMA-1A.

The objective power plant was designated the XMA-1C. Both a ceramic reactor and a metallic reactor, using a hydrided yttrium moderator and refractory metal fuel elements, were under consideration for the XMA-1C.

4.1.2 DESCRIPTION OF XMA-1 POWER PLANT

The XMA-1 power plant was a nuclear turbojet system designed to operate either on nuclear or chemical power. It consisted of a single reactor-shield assembly coupled with two sets of X211F turbomachinery, as shown in Figures 4.1 and 4.2. The engines were mounted from the forward and aft flanges of the reactor-shield assembly and arranged so that the compressor-turbine coupling shafts passed through the outer portion of the reactor side shield with the chemical interburner combustion ducts on both sides of and parallel to the reactor side shield. Although the XMA-1A and XMA-1C arrangements were identical, the basic power plant was designed to XMA-1C requirements. The XMA-1A reactor was designed to fit into the same reactor cavity as the XMA-1C.

The design conditions imposed on the XMA-1 power plant are summarized in Table 4.1.

Although the XMA-1A reactor was similar to the HTRE-3 reactor, a number of design improvements were incorporated. A sketch of the reactor is shown in Figure 4.3. The reactor core consisted of 151 cylindrical, metallic fuel cartridges in a matrix of unclad, hydrided zirconium moderator elements. Designed with a triflute cross section, the reactor moderator elements were located at the interstices between the fuel cartridges. To provide additional moderation, round zirconium hydride bars were placed at the center of the fuel cartridges inside the smallest fuel ring. Tubular members passed through the full length of the core at 129 positions. The tubes accommodated the control rods and, at some locations, provided the structural support between the forward and rear tube sheets.

The core was designed to operate in a horizontal position with the moderator bars and fuel cartridges supported by tube sheets at each end. The reflector assembly formed the support between the tube sheets and completed the cylindrical configuration. The complete assembly was cantilevered from the forward shield plug.

TABLE 4.1

XMA-1 DESIGN CONDITIONS

	XMA-1A	XMA-1C
Turbine Inlet Temperature, ^{o}F		
Normal	1450	1600
Military	1500	1700
Emergency	-	1750
Life, hr		
Nuclear operation	150	900
Chemical operation	100	100
Total Life, hr	250	1,000
Weight, lb	100,000	115,740
Cruise Speed	M 0.6	M 0.85
Cruise Altitude, ft	10,000	30,000
Maximum Speed	-	M 1.0
Maximum Altitude, ft	-	45,000
Length of Single Mission, hr	-	120

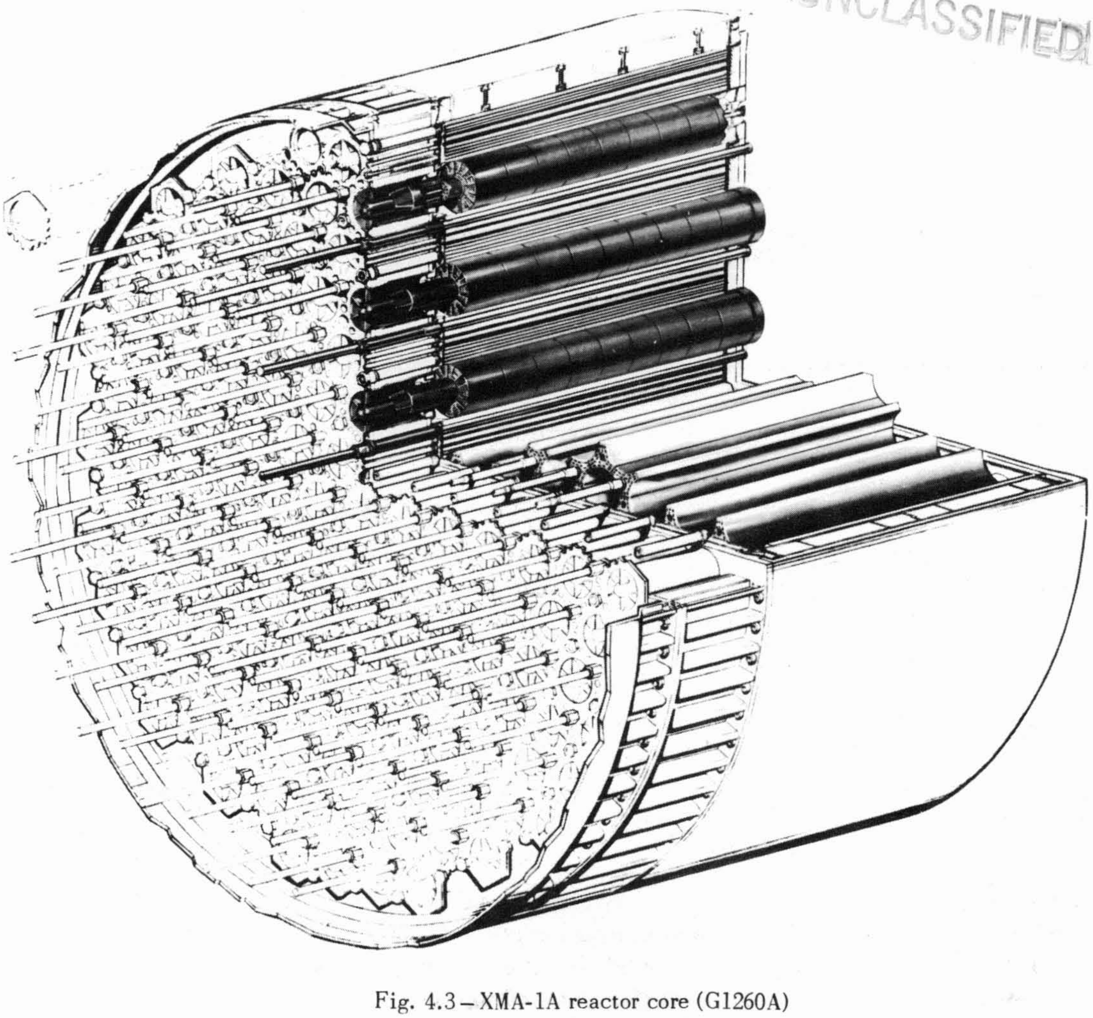


Fig. 4.3 – XMA-1A reactor core (G1260A)

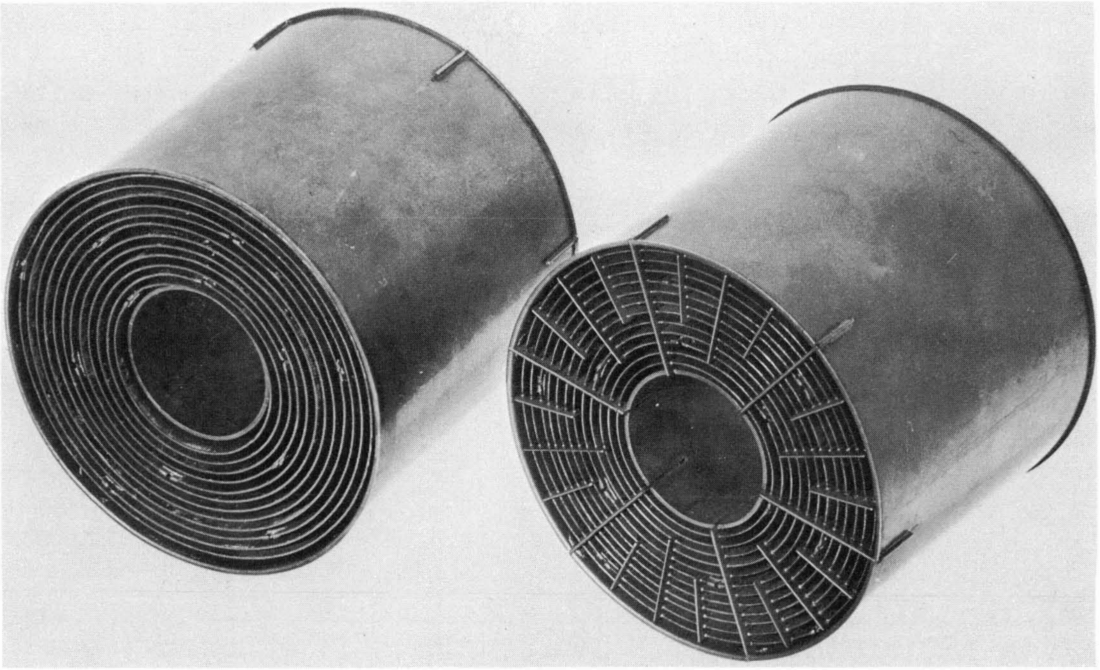


Fig. 4.4 – XMA-1A fuel stage (C22689)

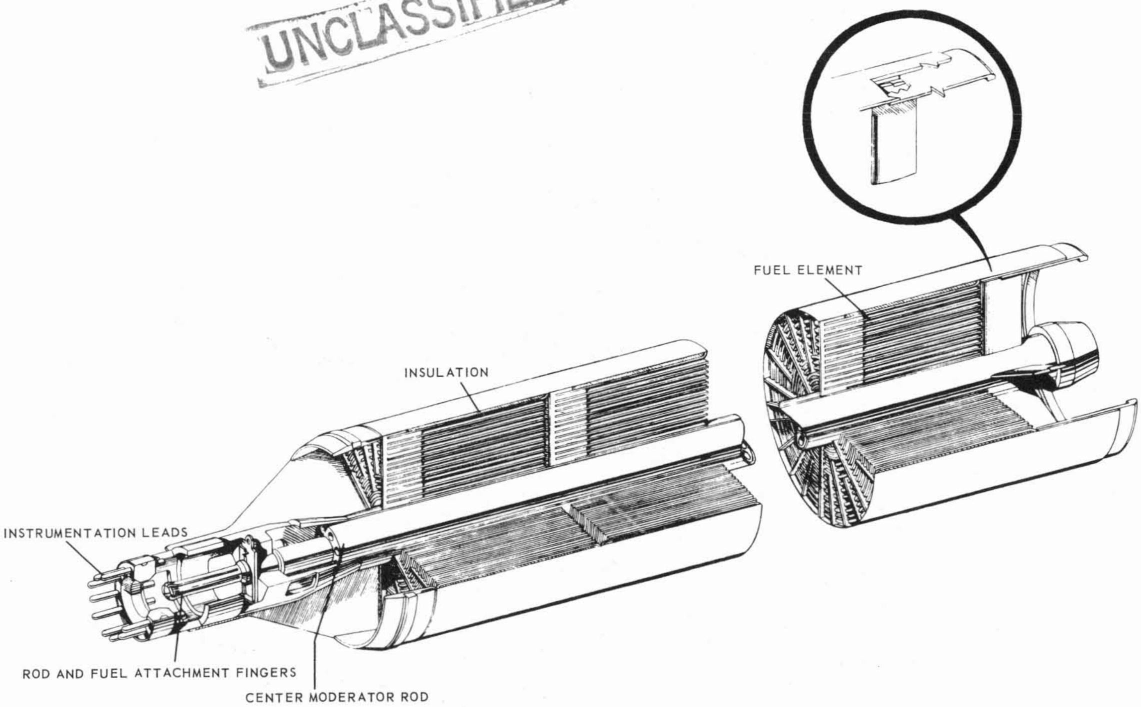


Fig. 4.5 – XMA-1A fuel cartridge (DI-37)

The fuel stock used in the XMA-1A reactor was essentially the same as that used in HTRE-3, but of further improved product integrity. The fuel element stages, shown in Figure 4. 4, were 3 inches long rather than 1. 5 inches to reduce the effective friction factor and the volume of poison material associated with support hardware. The fuel element cartridges included a nonfueled ring containing a thin layer of insulation to form the outer wall of the outermost annulus. Making this ring an integral part of the cartridge provided closer dimensional control of the annulus size than in the HTRE-3 design. A fuel cartridge is shown in Figure 4. 5. As a result of introducing several sizes of center moderator rods for gross radial power flattening, the number of fuel rings per stage was varied for different radial regions of the core. Uniform temperature distribution within a stage was achieved by varying the fuel loading from ring to ring. More rings, spaced closer together, were used in the six rear stages to increase heat transfer surface area and partially flatten the longitudinal temperature.

The displacement of moderator volume from the area surrounding the fuel elements by use of the center moderator rod, plus a lower moderator volume fraction than that associated with HTRE-3, rendered a hexagonal moderator element impractical because the walls would have been too thin. The triflute moderator element was used to overcome this difficulty.

4.1.3 FINAL STATUS OF XMA-1A POWER PLANT DEVELOPMENT

The XMA-1A power plant was in final design when the objective shifted to the XNJ140E. Numerous prototype components had been built and tested. In general, the turbomachinery, shield materials, controls accessory components, and analytical design techniques were directly applicable to the XNJ140E.

The X211 turbomachinery had logged many hours of operation under XMA-1A operating conditions. It was applied directly to the XNJ140E power plant; although the shaft, bearing frames, and chemical burners required modification.

Control system performance had been determined using a large analog simulator. The dynamic behavior of the control system was similar to that of the XNJ140E and could be applied with relatively little change. Directly applicable prototype control system components had been built and tested.

Nuclear and mechanical tests had been performed on XMA-1 shield components; the same materials were used in the XNJ140E shield, although in a different configuration.

The XMA-1A reactor design had reached the point of issuance of manufacturing drawings. The critical experiment had been performed and detailed uranium loading specifications established. Metallic fuel elements had been tested under simulated heat conditions in the Engineering Test Reactor (ETR) at the National Reactor Testing Station in Idaho. The test results indicated that the fuel elements were capable of meeting or exceeding the XMA-1A life and temperature requirements.

The XMA-1C reactor materials development and preliminary design had progressed sufficiently that a selection could be made between the ceramic and metallic reactors with confidence. This design and materials development was also directly applicable to the XNJ140E reactor.

Additional details on the XMA-1 power plant are contained in APEX-907, "XMA-1 Nuclear Turbojet," of this Report.

4.2 XNJ140E POWER PLANT

A Department of Defense decision was made in mid-1959 to eliminate consideration of an early flight objective for the CAMAL aircraft. In place of a specific Weapons System objective, general guidance was provided to direct the applied research and development program toward a propulsion system capable of propelling an aircraft at a speed of Mach 0.8 to 0.9 at an altitude of approximately 35,000 feet. The reactor was to use materials capable of providing a reactor life of 1000 operating hours at the specified performance level, and was to have development potential for even higher performance.

In view of the revised objective, work on the XMA-1A reactor was discontinued and an evaluation was made of the relative development status of the high temperature metallic and ceramic materials which had been under development for the XMA-1C reactor. Simultaneously, a study was undertaken to compare the XMA-1 with alternative configurations. One of the primary objectives of the study was to determine whether single-engine or dual-engine power plants were best suited for growth to higher performance levels. The results indicated that a propulsion system consisting of a single X211 turbojet engine with a beryllium oxide reactor would meet the Department of Defense guidance and offered the greatest potential for further growth.

The development of such a power plant, designated the XNJ140E and illustrated in Figure 4.6, was proposed in March of 1960 and was subsequently approved as a development objective. A target date for ground test of a prototype propulsion system, the XNJ140E-1, was set for December 1962. The ground test was referred to as the Advanced Core Test, the turbomachinery having been previously tested under chemical power. A target date for flight test in the proposed Convair NX-2 prototype aircraft (see Figure 1.2) was set for 1965.

The XNJ140E program was on schedule when the Aircraft Nuclear Propulsion program was terminated in April 1961. Manufacturing drawings for the XNJ140E-1 were being released; the last release was planned for mid-year. Long lead time materials had been ordered, developmental models of critical components had been proof-tested under simulated service conditions, and reactor critical experiments had been performed. Approval to proceed with fabrication and assembly had been requested.

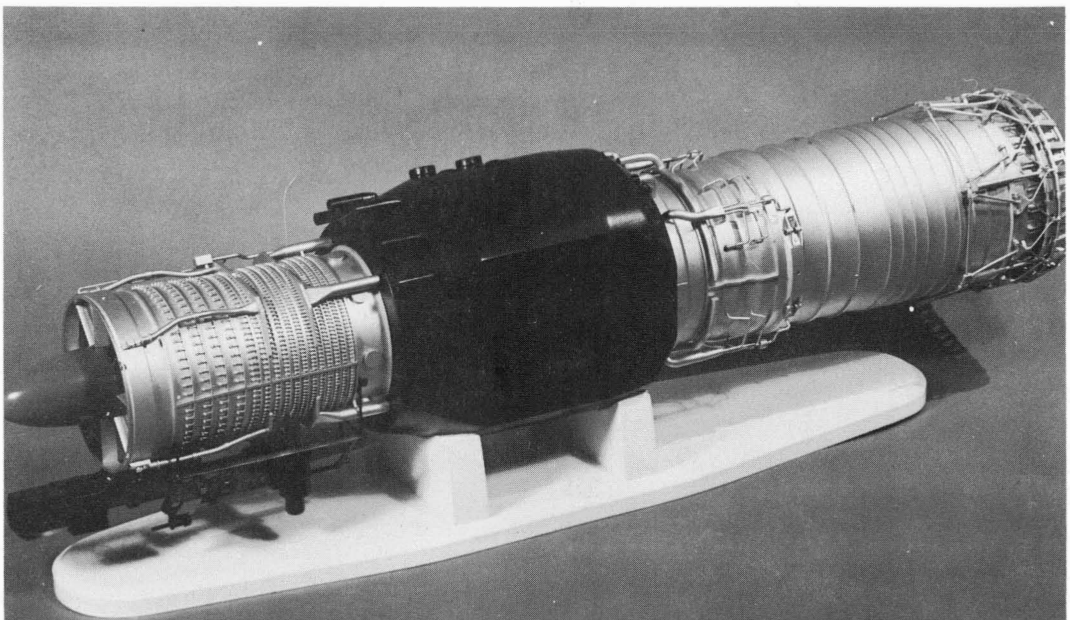


Fig. 4.6—Model of XNJ140E nuclear turbojet engine

4.2.1 XNJ140E Design Requirements

Design requirements, consistent with the Department of Defense guidance, were established. Based on work with Convair, the profile for a typical 5-day mission was derived (Figure 4.7). The cycle conditions of airflow, temperature, pressure, thrust, etc., at the various design points in the typical mission and the apportionment of the required 1000-hour reactor life over these points is shown in Table 4.2. These conditions were used to establish the reactor design and temperatures of materials.

TABLE 4.2

CYCLE CONDITIONS FOR THE XNJ140E POWER PLANT[a]

Flight Conditions	Ground Checkout	Chemical Takeoff	Climb To Station, NC	Cruise On Station	Maneuver On Station	Two-Engine Operation	Nuclear Takeoff[c]
Ambient temperature, ^{o}F	59	59	-13	-66	-66	23	59
Speed, Mach No.	0	0	0.7	0.8	0.8	0.6	0
Altitude, ft	0	0	20,000	35,000	35,000	10,000	0
Power setting[b]	NC	Max	NC	NC	Mil	NC	Max
% rpm	98	100	98	98	100	98	100
Total airflow, lb/sec	412	426	280	173	176	361	426
Reactor airflow, lb/sec	369	382	252	155	157	323	381
Compressor discharge pressure, psia	171	162	117	72.8	74.9	150	179
Compressor discharge temperature, ^{o}F	665	663	638	583	597	665	685
Turbine inlet temperature, ^{o}F	1740	1640	1740	1740	1800	1740	1800
Reactor power, mw	112	-	78.1	50.5	53.6	98.5	121
Thrust, lb	21,600	34,160	12,050	8,120	8,570	14,830	35,310
Life, hr	20	5	20	885	20	50	-
Torque, lb-ft	92,680	93,420	62,830	38,240	38,970	81,330	97,000

[a]Standard day conditions.
[b]NC - Normal continuous
Mil - Military (full power without afterburner)
Max - Maximum (full power with afterburner).
[c]Alternative to normal chemical takeoff.

The power plant was also to be capable of operating at any speed or altitude up to Mach 1.0 at 45,000 feet within the envelopes shown in the flight map in Figure 4.8, which is a composite of aricraft and propulsion system limits. Extended operation at other points in the flight map could affect the life of the power plant. In general, extended operation at low speeds and high altitudes would increase the life beyond the 1000-hour objective, whereas extended high speed operation at low altitude would decrease life expectancy.

The reactor was to be replaced during normal engine overhaul which was assumed to be after approximately 1000 hours of power operation.

The objective radiation levels at a distance of 50 feet from the reactor are shown in Figure 4.9. Additional shielding provided at the crew compartment reduced the crew dose rate to 0.02 r per hour during operation at the cruise condition. These radiation levels were within radiation damage limitations on aircraft components. Induced activities were sufficiently low to permit manual maintenance of the aircraft after removing the power plant.

4.2.2 MATERIALS AND DESIGN SELECTION

Reactor Selection

Both metallic and ceramic reactor materials had been under development for the XMA-1C power plant. A satisfactory, high-strength, oxidation-resistant clad had not yet been fully developed for use with refractory metal materials such as niobium. The beryllium oxide reactor, on the other hand, had by then been evaluated through in-pile testing, component testing, and design, as a result of the HTRE-2 tests and the work done on the proposed

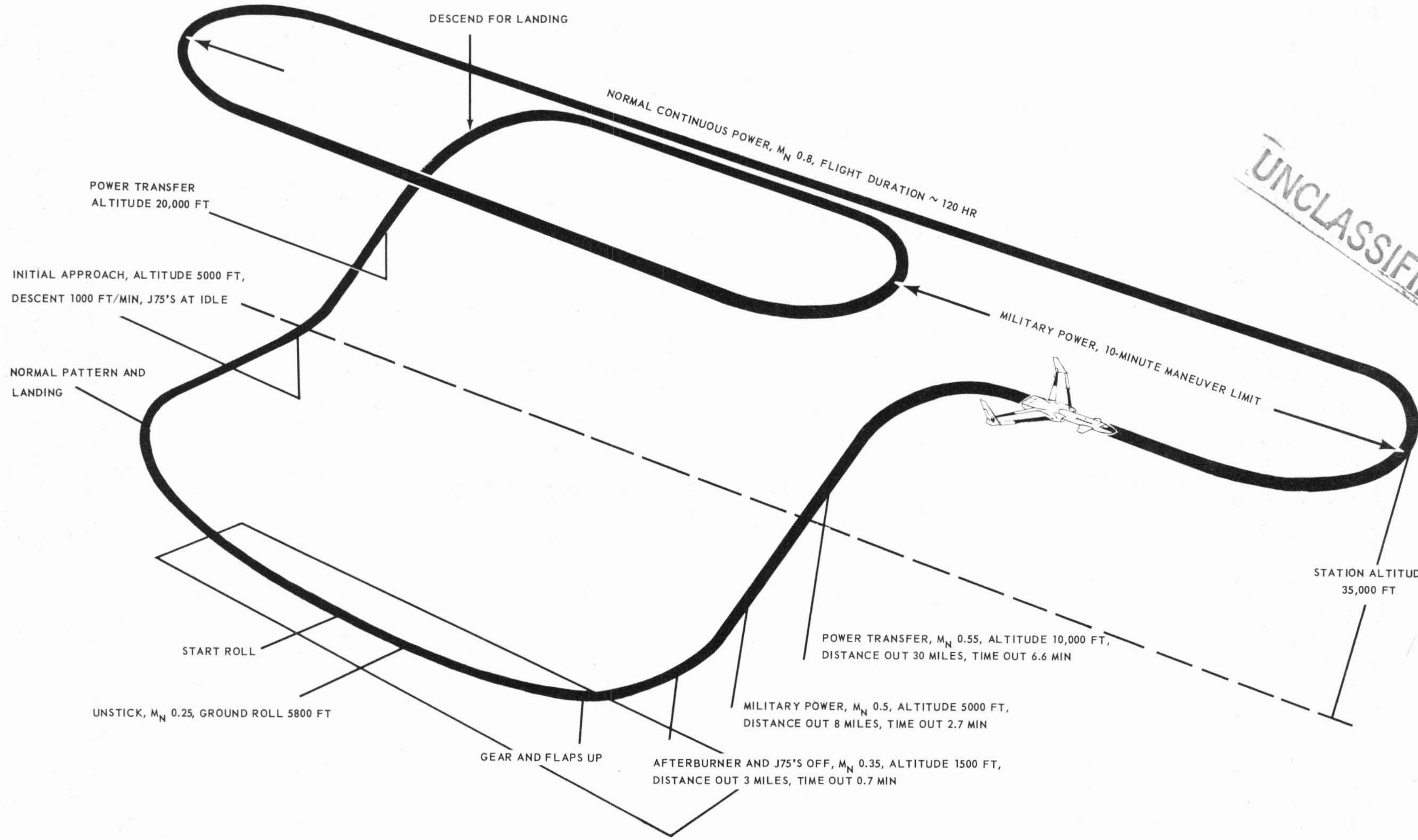


Fig. 4.7 – Typical mission profile of XNJ140E objective power plant

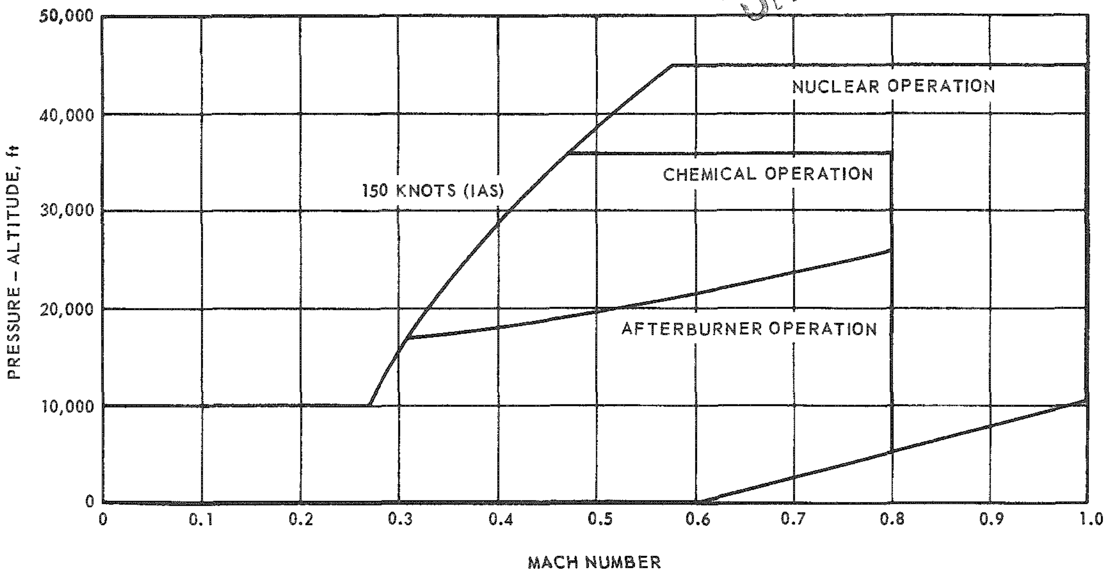


Fig. 4.8 – XNJ140E flight map

HTRE-4 and Ceramic Core Test reactors. Sufficient progress had been demonstrated in the development of the ceramic materials that the beryllium oxide reactor was adopted for the XNJ140E.

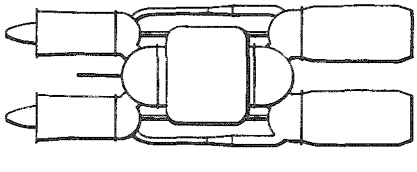
Selection of Propulsion System Configuration

Because of the revised power plant objective, the advent of the ceramic reactor, and the extensive development which had been performed on other system components since the dual-engine XMA-1 configuration had first been adopted, an "Advanced Configuration Study" (ACS) was performed in late 1959 and early 1960.

Configurations using a single engine coupled with each reactor had been under design for advanced propulsion systems. The purpose of the ACS was to compare two of the single engine configurations with the dual-engine XMA-1 configuration. The study was performed in depth, with detailed preliminary designs being prepared for each configuration. Great care was used to maintain consistency between the designs. Corresponding components of each configuration were designed by the same engineers. The three configurations were compared on the basis of performance, weight, maintainability, growth potential, and other factors.

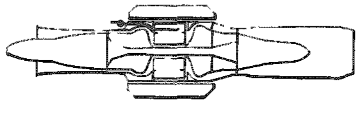
The configurations studied were as follows:

Dual-Engine Configuration



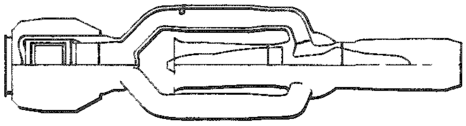
The dual-engine configuration was an advanced version of the XMA-1 power plant with two sets of turbomachinery mated to a single reactor.

Integral Single-Engine Configuration



In this configuration, the nuclear reactor and engine were coaxial, with the engine shaft passing through the center of the reactor.

Separable Single-Engine Configuration



In this configuration, the reactor and turbomachinery were coaxial but were separate entities with interconnecting ducting.

The following were the most significant findings of the study.

Performance - The power plant performance (thrust as a function of speed) during normal nuclear operation had been held as an independent variable and hence was identical for the three configurations. However, the dual-engine configuration had a potential performance disadvantage during flight because a reactor shutdown would cause the loss of thrust from two engines.

Weight - Both single-engine systems suffered a weight penalty because more shielding was required for two small reactors than for a single large reactor of the same total power. This disadvantage was partially eliminated by reducing the thickness of adjacent side shields when two or more units were installed side by side. The penalty was further reduced in the integral single because the symmetrically disposed metal in the turbomachinery, particularly the compressor, was effective as shielding material. The integral single also had a compensating weight advantage because heavy external ducting was not used. The corresponding absence of duct pressure drop provided a further advantage: a greater pressure drop could be tolerated in the reactor, which resulted in a smaller reactor and less shield weight. The net result was that the combined weight of two integral single engines was approximately equal to the weight of the dual-engine configuration. A pair of separable single engines, however, suffered from both the shield-weight and duct-weight disadvantages and was substantially heavier than the other two configurations.

Turbomachinery - The engine shaft for the integral single, and to a lesser extent for the dual engine, required cooling because of the internal heat generated by the absorption of radiation in the shaft material. Calculations indicated that the shafts in both configurations could be maintained at temperatures approximating those used in conventional turbojet engines. The shaft of the integral single would be machined to provide internal helical fins, thus providing added strength as well as added heat transfer surface. The dual engine configuration required the longest shafts and the separable single required the shortest. Bearings could be spaced to avoid critical speed problems in all three configurations. The integral single was preferred from the standpoint of turbomachinery development because of its symmetrical design and absence of external ducting.

Maintainability - In both the separable-single and the dual-engine configuration, the turbomachinery could be separated as a unit from the reactor-shield assembly. In the integral single, on the other hand, the turbomachinery was separated from the reactor-shield assembly by rotating an internal screw thread to uncouple the shaft connecting the turbine and compressor before removal of the reactor. This was not an unfamiliar disassembly feature because a similar disconnect procedure is used during overhaul of the General Electric J79 turbojet, a production engine.

Among the important maintenance advantages of the integral single were that the assembled engine was compact and easy to handle, externally mounted components and disconnects were unobstructed by ducting, and vertical assembly and disassembly methods were readily achievable. Consequently, despite the requirement that the compressor and turbine be uncoupled prior to reactor removal, the integral single configuration was considered to be the most suitable from the standpoint of maintenance.

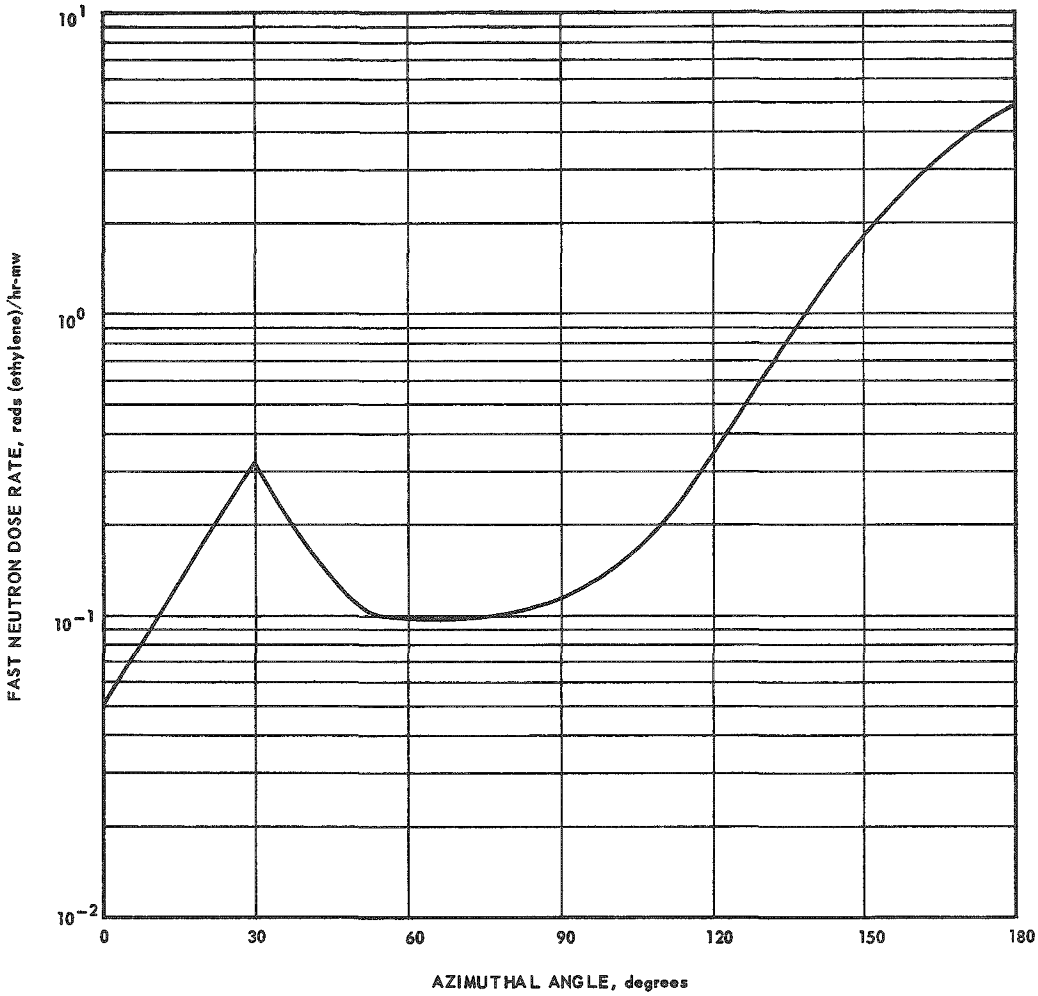


Fig. 4.9a—Objective fast neutron dose rates per megawatt of reactor power 50 feet from reactor midpoint

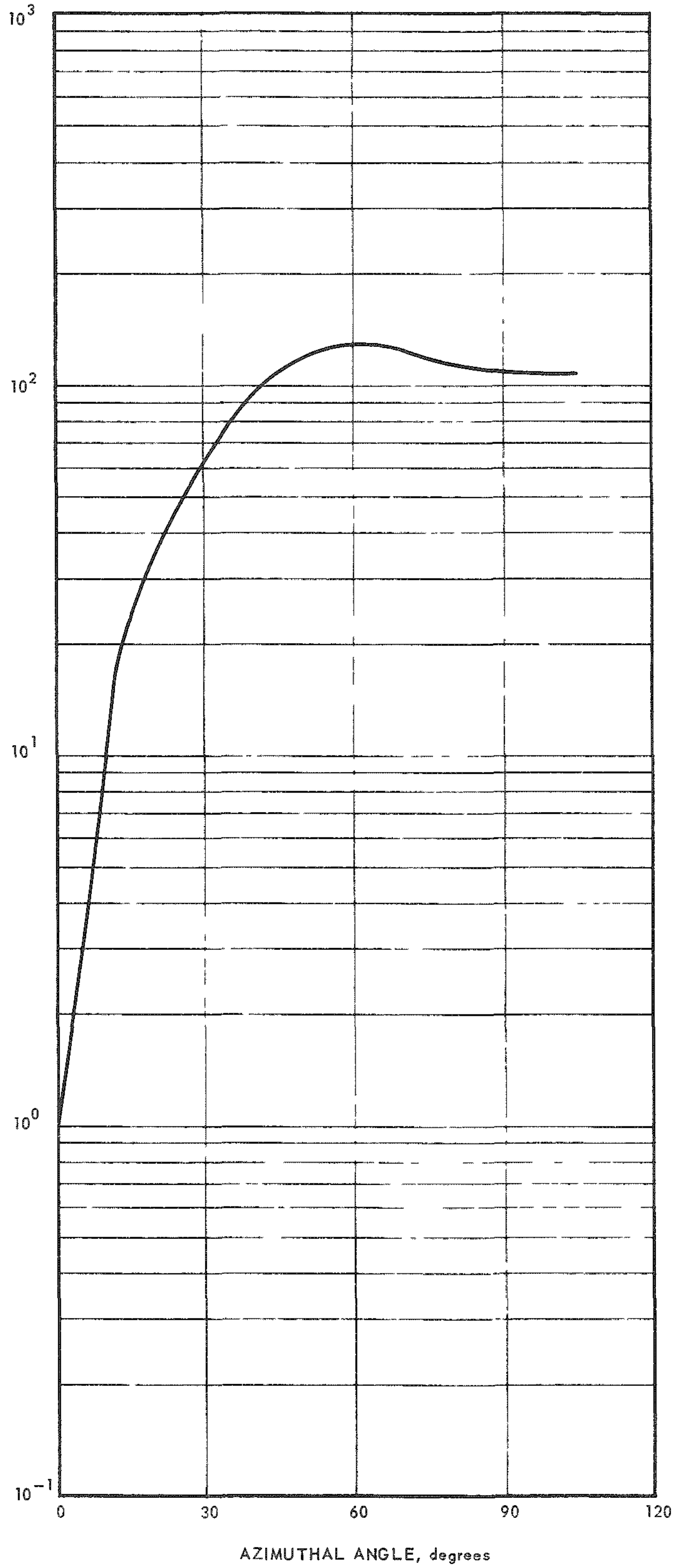


Fig. 4.9b – Objective gamma dose rates per megawatt of reactor power 50 feet from reactor midpoint

Growth Potential - The integral single-engine configuration also appeared to have the most favorable growth potential, based on the belief that growth would be in the direction of more compact reactors with higher exit-air temperatures and, possibly, higher pressures. Design and weight studies indicated that, at higher temperatures and pressures, the external ducting used in the other configurations became disproportionately heavy and less reliable.

Other Factors - Reliability, development costs, fabricability, and adaptability to aircraft with different installation and power requirements, also appeared to favor the single-engine configurations.

Recommended Configuration

As a result of the Advanced Configuration Study, the integral single engine configuration using a BeO ceramic reactor was recommended (and subsequently approved) for development to meet the Department of Defense guidance. The primary consideration in this selection was the requirement to direct the program toward higher temperature systems where the use of external high temperature ducting becomes impractical. Other factors, such as weight and performance, had been shown to be equivalent or to favor the chosen configuration.

The details of the Advanced Configuration Study are contained in references 2 to 7.

4.2.3 DESCRIPTION OF XNJ140E NUCLEAR TURBOJET

The XNJ140E was an integral single-engine nuclear turbojet using X211 turbomachinery and a beryllium oxide ceramic reactor. An artist's illustration of the power plant is shown in Figure 4.10. Major components are identified in Figure 4.11. A model of the power plant was previously shown in Figure 4.6.

Turbomachinery

The X211 turbomachinery was an axial-flow, high-pressure-ratio, single-rotor engine with a 16-stage variable stator compressor and a 3-stage turbine. The hollow engine shaft was cooled by ninth-stage compressor air to approximately the same temperatures used in conventional turbojet engines. To increase heat dissipation and provide added strength for torque loads, helical fins were machined in the internal face of the shaft.

The primary power plant structure consisted of the compressor and turbine casings connected by an Inconel X pressure shell completely encasing all nuclear components except the side shield. The rotor was supported by four bearings, of which the second served as the thrust bearing. The bearing loads were carried to the outer casing by support frames. The engine pickup points were connected to the second and third bearing support frames.

The reactor was located between the compressor and the turbine in the region normally occupied by the combustion chambers of a chemical turbojet engine. The engine shaft passed through a tunnel in the reactor. The compressor air penetrated the front shield in an annular duct, was heated in the reactor, passed through a rear shield annular duct, drove the turbine, and was exhausted as a jet. A combustion zone for burning JP fuel was provided in series with the reactor and located in the rear annular duct directly ahead of the turbine. The engine could be operated on nuclear heat, or chemical heat, or any combination of the two. A chemical afterburner was also provided for use in takeoff.

Thrust was controlled by demanding a change in turbine air temperature while holding airflow approximately constant. Control rods were automatically repositioned until the desired air temperature level was reached. An ion chamber in the control actuator loop was used to limit the rate of change of reactor power. While operating on chemical fuel,

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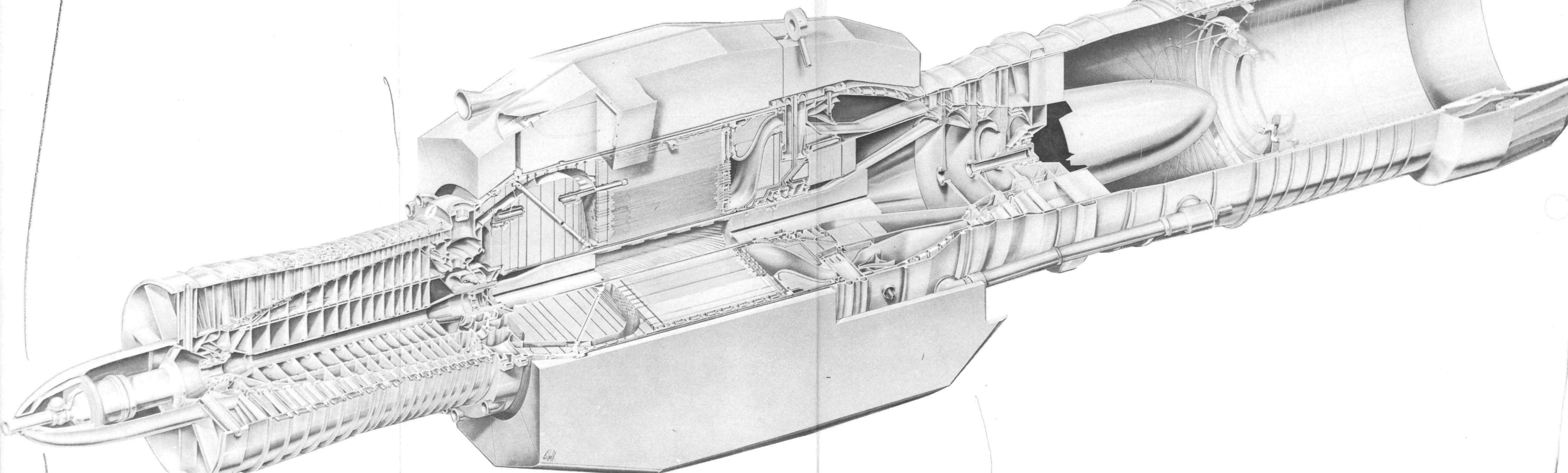

Fig. 4.10 – XNJ140E nuclear turbojet engine (DI-400)

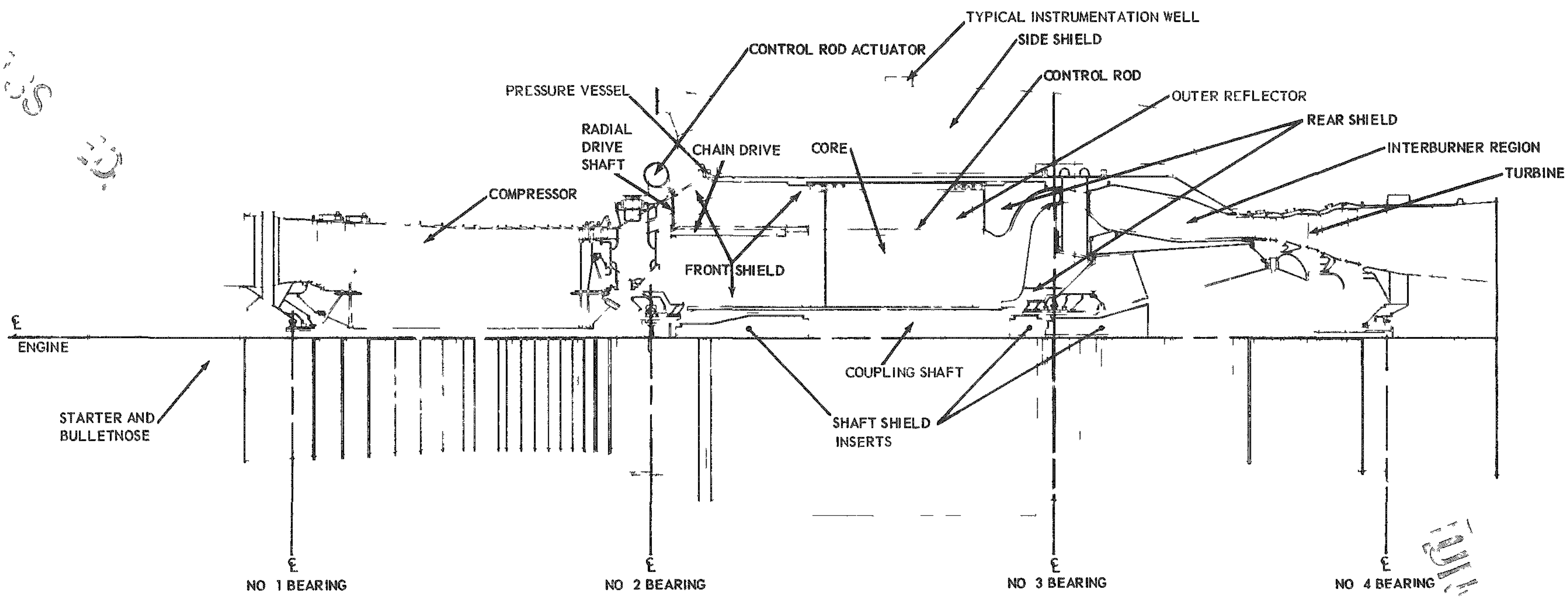

Fig. 4.11 – XNJ140E engine schematic, exhaust duct removed

the air temperature demand opened or closed the fuel valve as required. The engine speed, and thus the airflow, were adjusted by controlling the exhaust nozzle area. Air could be bypassed around the turbine for fast correction of engine overspeed. Engine speed was maintained at approximately 5000 rpm at full power and at 98 percent of this value during cruise conditions.

The power plant was designed for installation and removal from the aircraft using shielded transport vehicles. Externally mounted controls and accessories were combined into integral units capable of remote replacement. Tubing and wiring were combined into harnesses equipped with remote handling connections. All major sections of the power plant, such as compressor, front shield plug, reactor, etc., could be remotely assembled or disassembled and maintained as required. Most of this work could be performed manually after removal of the reactor.

Design dimensions of the XNJ140E-1 (developmental prototype for use in the Advanced Core Test) are shown in Figure 4.12.

The estimated maximum dry weight of the XNJ140E-1 power plant was 60,600 pounds with one side-shield cheek removed. This included 18,320 pounds for the turbomachinery. The estimated maximum weight of the reactor-shield assembly plus control system was 42,230 pounds. The estimated maximum weight of a single side-shield cheek was 2,300 pounds.

Reactor

The XNJ140E-1 reactor consisted of (1) an annular cylindrical assembly of ceramic tubes which formed the active core, the outer reflector, the inner reflector, and the end reflectors, (2) control rods, (3) the longitudinal support structure, (4) the radial support structure, (5) the shaft tunnel, (6) the core liner, and (7) the enclosing structural shell. Figure 4.13 is an isometric cutaway view of the reactor. Figure 4.14 is a radial cross section of the reactor and Figure 4.15 is a longitudinal cross section. Reactor materials and representative calculated operating temperatures are shown in Figure 4.16.

The use of ceramic materials was a logical method of providing the desired high temperature capability; however, thermal stress considerations inherent in the use of ceramics necessitated small, simple shapes, and a small hexagonal tube was chosen as the basic modular element. The tubes were fitted together to form an assembly that was 62 inches in diameter and 33 inches long. A central void, 13.23 inches in diameter, accommodated the coupling shaft that joined the compressor and turbine.

The fuel tubes (Figures 4.17 and 4.18) were made of yttria-stabilized beryllia containing dispersed enriched urania. Each fuel element was a small hexagonal tube 0.249 inch across flats and 4.28 inches long with an inside diameter of 0.167 inch. The inside surface was clad with a 0.003-inch-thick layer of yttria-stabilized zirconia. The coefficient of thermal expansion of the clad matched that of the fuel tube.

There were approximately 25,000 airflow passages through the reactor and approximately 170,000 separate fuel elements in the active core. During engine test operation simulating the extended cruise-flight condition of the operational engine, the reactor fuel elements operated at a calculated peak temperature of approximately 2530°F.

The annular cylindrical central island located inside the active core was composed of (1) alumina tubes with the same outer dimensions as the fuel elements, (2) a metallic core liner, and (3) a metallic shaft tunnel. The alumina region served as an inner reflector, provided thermal insulation for the metallic components in the central island, and acted as a gamma shield to reduce the secondary heating rate in the metallic components. The core liner served as a structural arch permitting the inner reflector tubes to bridge the central void. The shaft tunnel was a structural component of the longitudinal support sys-

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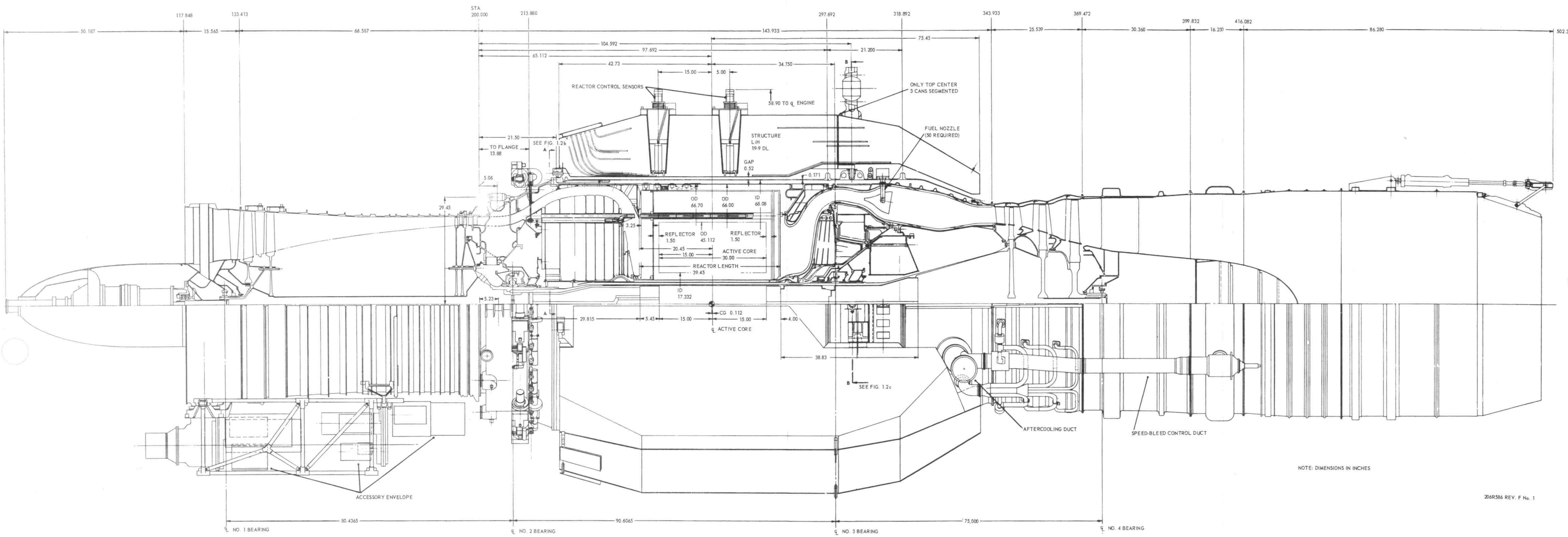

Fig. 4.12—XNJ140E-1 engine design layout

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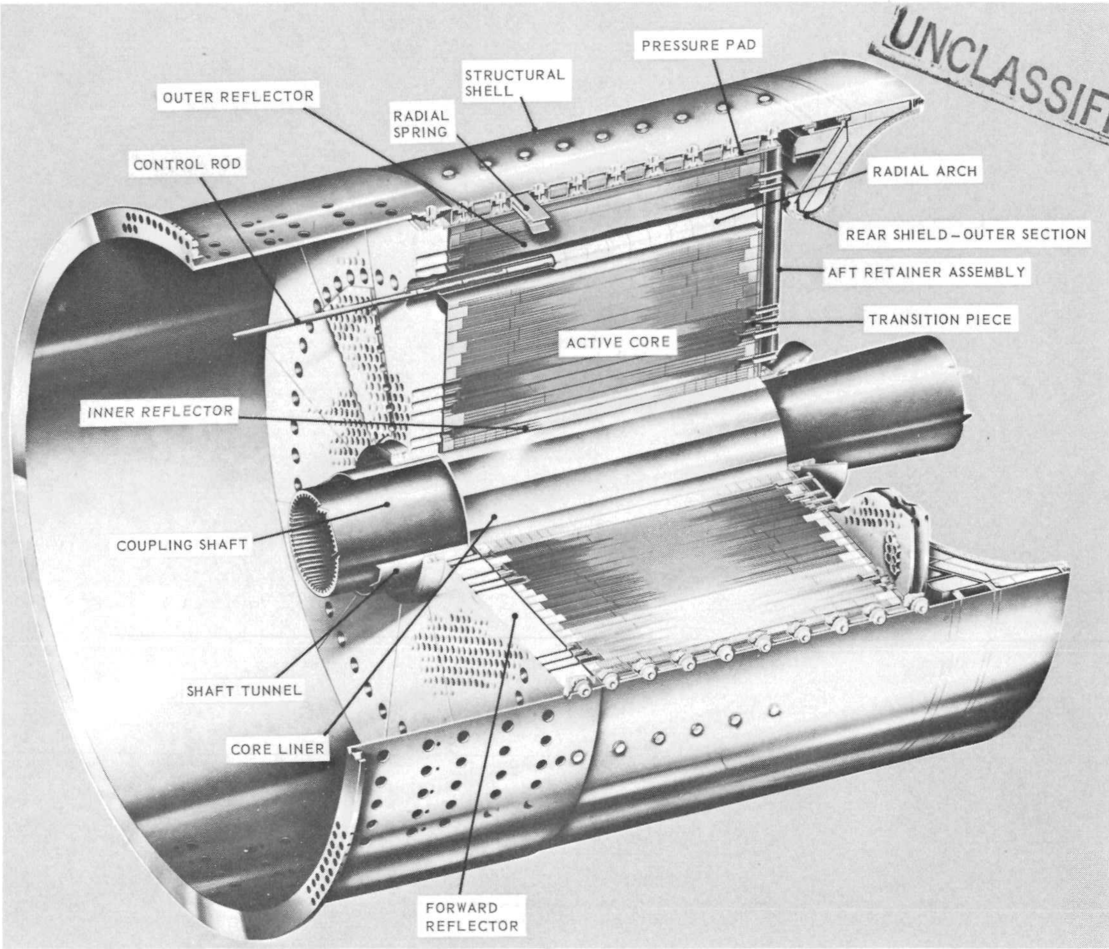
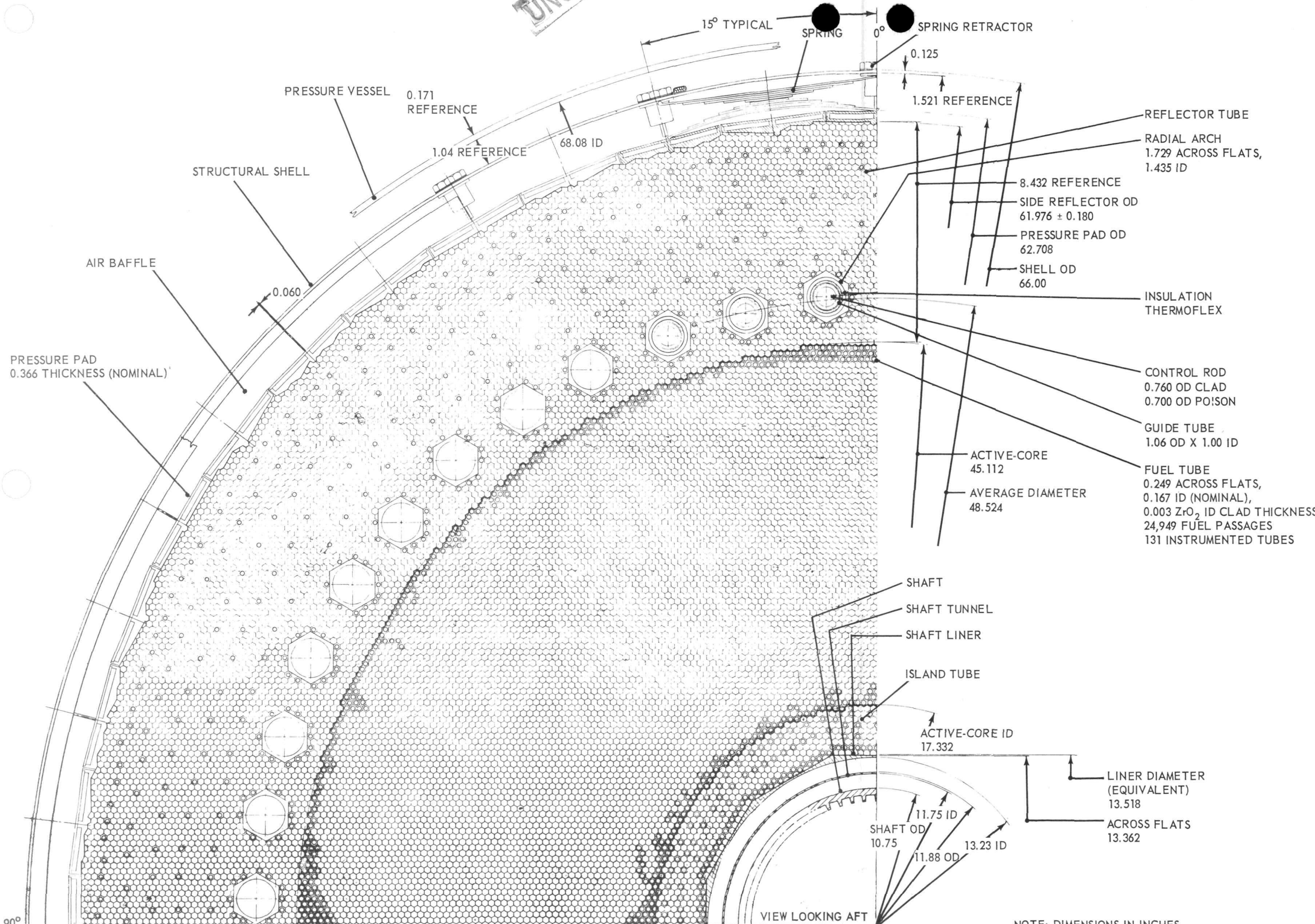


Fig. 4.13 – Isometric cutaway view of the XNJ140E-1 reactor

tem and carried part of the longitudinal loads on the reactor from the aft-retainer assembly to the front shield. The shaft tunnel and core liner formed an annular duct that channeled cooling air from the front shield to the rear shield. The shaft tunnel was supported in a manner that maintained concentricity with the core liner so that cooling air flowing through the annular passage was not affected by deflections of the reactor under flight loads.

The outer reflector was an 8.5-inch-thick annular region of unfueled beryllium oxide tubes surrounding the active core. Control rods were located at 48 equally spaced places within the outer reflector, 1.75 inches from the boundary of the active core. The control rods (Figure 4.19) contained Eu_2O_3 as a neutron absorber in a nickel matrix and were clad with 80Ni - 20Cr. Radial arches (Figure 4.20) provided tunnels through the outer reflector for the control rod guide tubes.

The rear reflector was 1.5 inches thick, and was formed by the multiple beryllia fuel-tube transition pieces. Each transition piece received air from 19 fuel element channels and collected it into a single large-diameter channel (see Figure 4.21). Transition pieces also were used at the forward end of the reactor between the front reflector sectors and the active core. These transition pieces permitted the use of large-diameter channels in the end structural components and facilitated the structural and aerodynamic design.



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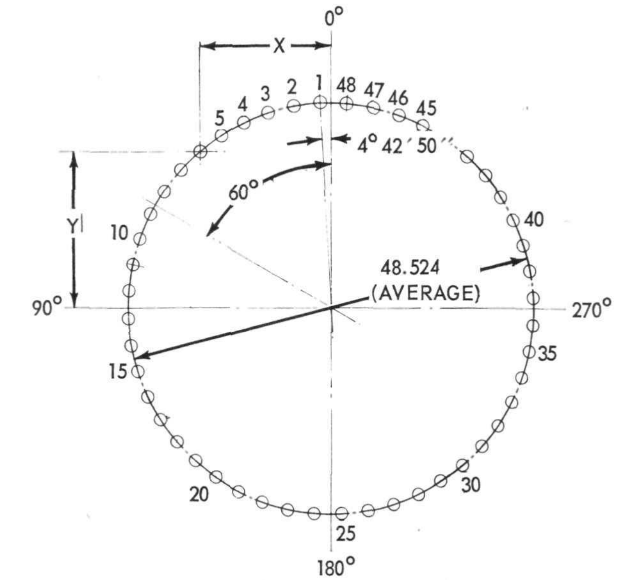

CONTROL ROD LOCATION

C.R.	C.R.	X	Y
1	25	2.003	24.291
2	26	5.134	23.640
3	27	8.139	22.773
4	28	11.144	21.471
5	29	13.774	19.954
6	30	16.403	18.001
7	31	18.532	15.615
8	32	20.410	13.230
9	33	22.038	10.410
10	34	23.040	7.374
11	35	23.791	4.338
12	36	24.167	1.084
13	37	24.167	1.951
14	38	23.791	5.205
15	39	22.789	8.241
16	40	21.662	11.061
17	41	20.035	13.881
18	42	17.906	16.266
19	43	15.652	18.435
20	44	13.023	20.387
21	45	10.393	21.905
22	46	7.388	23.206
23	47	4.257	23.857
24	48	1.252	24.291

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Fig. 4.14—Radial cross section of XNJ140E-1 reactor

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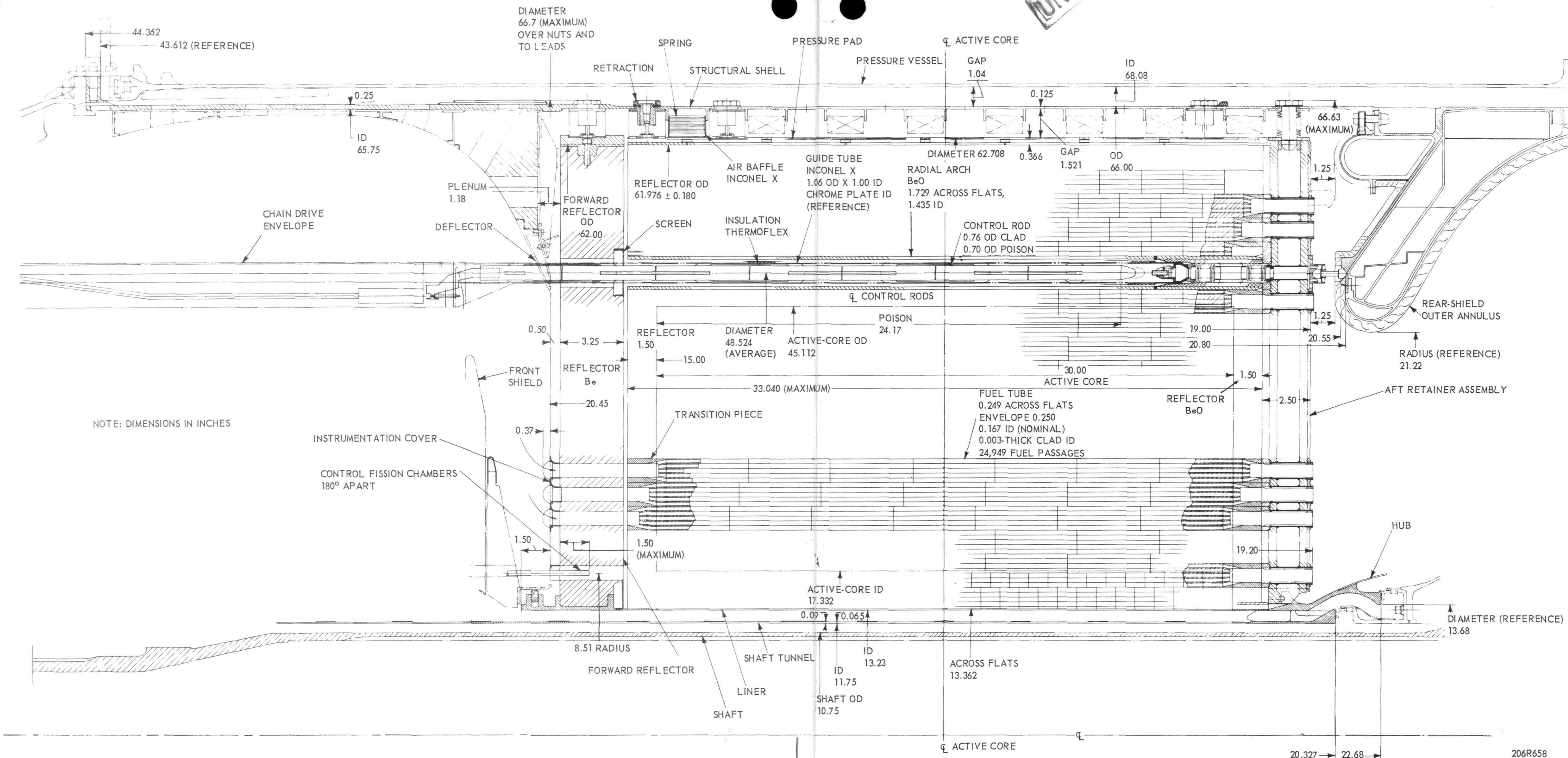

Fig. 4.15 – Longitudinal cross section of XNJ140E-1 reactor

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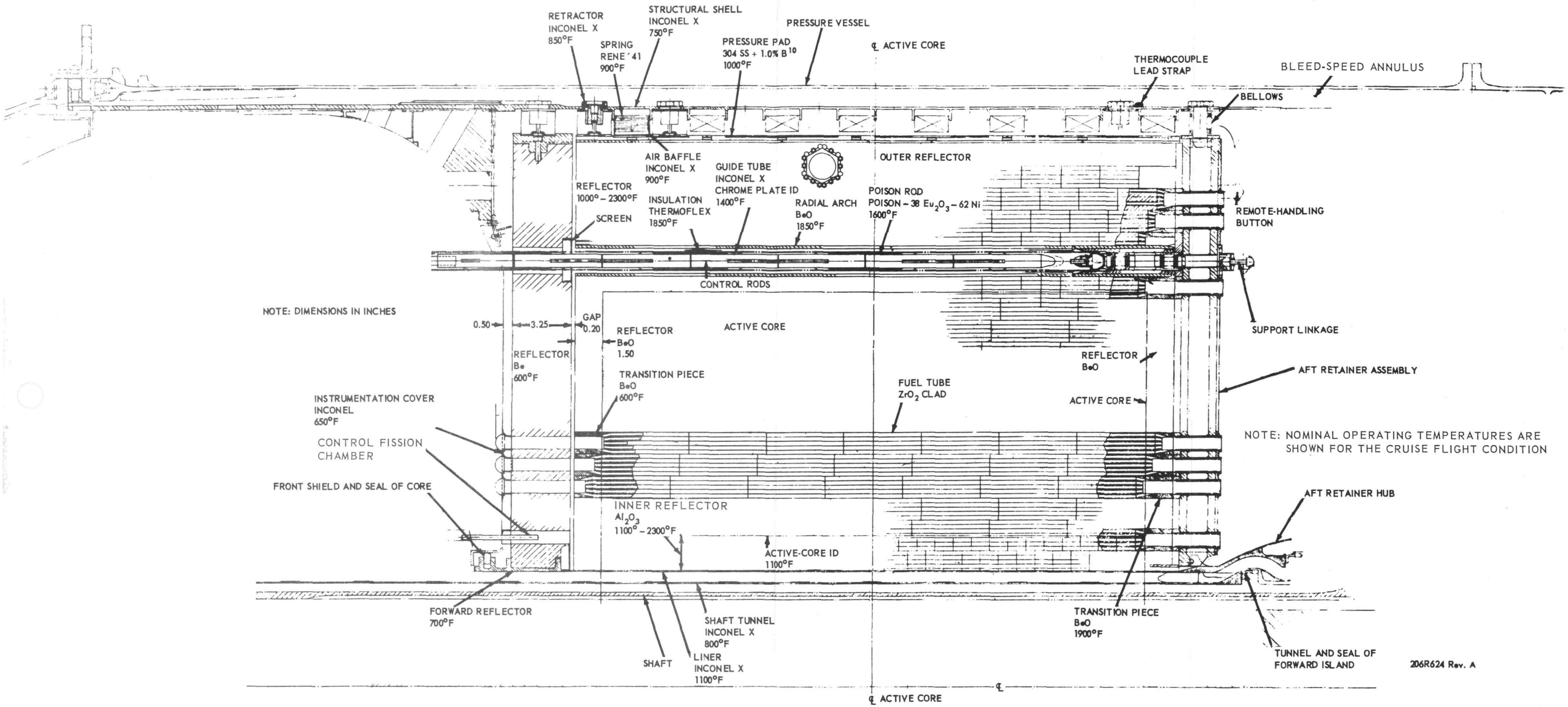

Fig. 4.16—Reactor materials and representative design operating temperatures (Dwg. 206R624 Rev A)

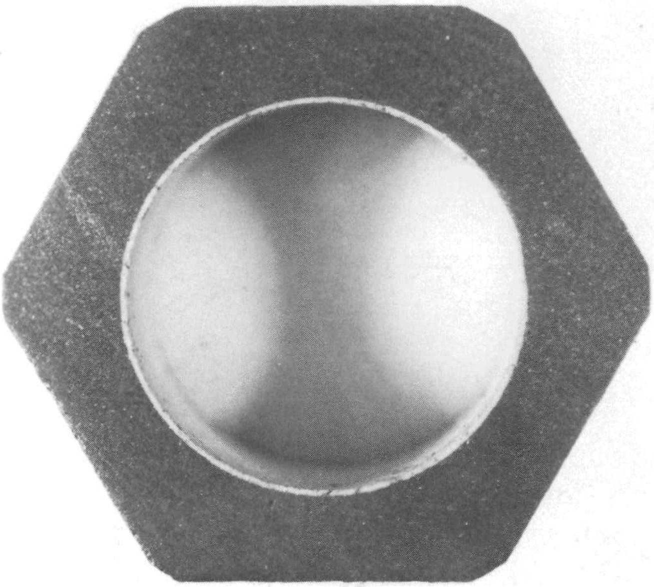

Fig. 4.17 – Coextruded fuel tube

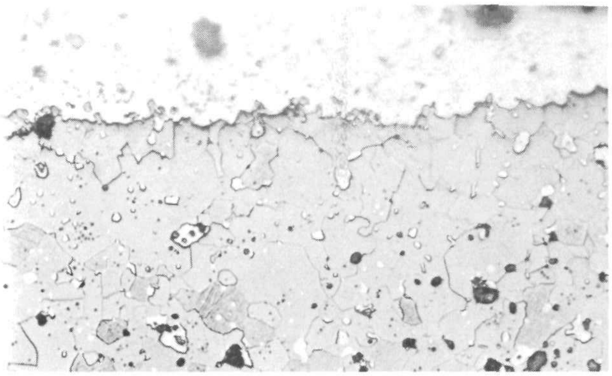

Fig. 4.18 – Clad-matrix interface of fuel tube

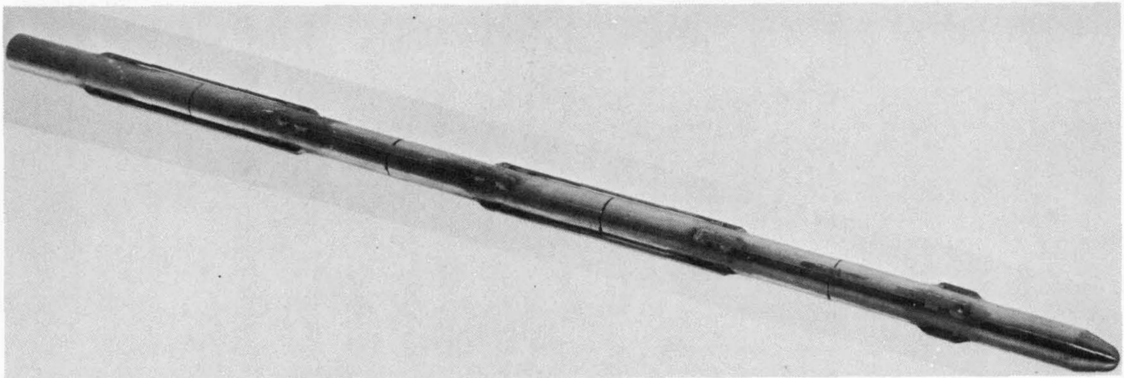

Fig. 4.19 – XNJ140E-1 control rod (Neg. U-38803B)

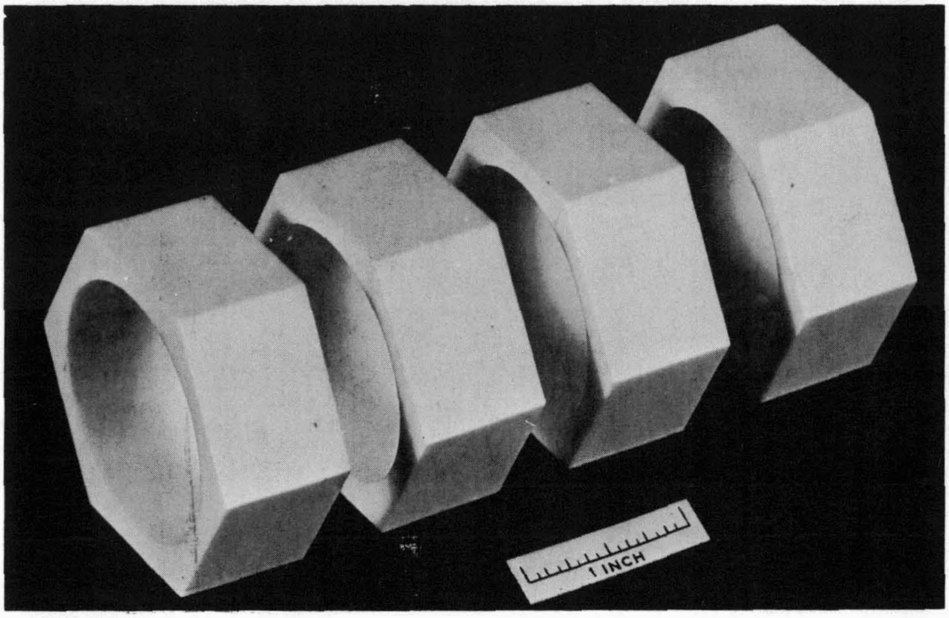

Fig. 4.20 – Radial arch pieces (Neg. C-23775)

The front reflector was composed of 12 beryllium sectors 3.25 inches thick. Perforations in the sectors served as passages for the primary airflow. In addition to acting as a neutron reflector, the front reflector also acted as a structural component that restrained the tube bundle against forward motion. The forward beryllia fuel-tube transition pieces also acted as an additional 1.5-inch-thick neutron reflector.

The external structure of the reactor was composed of a radial support system and a longitudinal support system. The radial structure restrained the ceramic tube bundle in a compressed unit assembly and resisted lateral loads. The longitudinal structure resisted aerodynamic drag on the reactor and axial inertial loads.

The radial support structure was composed of the structural shell, leaf springs (Figure 4.22), and pressure pads. The structural shell surrounded the reactor and was cantilevered at its forward end from the flanged connection to the front shield. The leaf springs

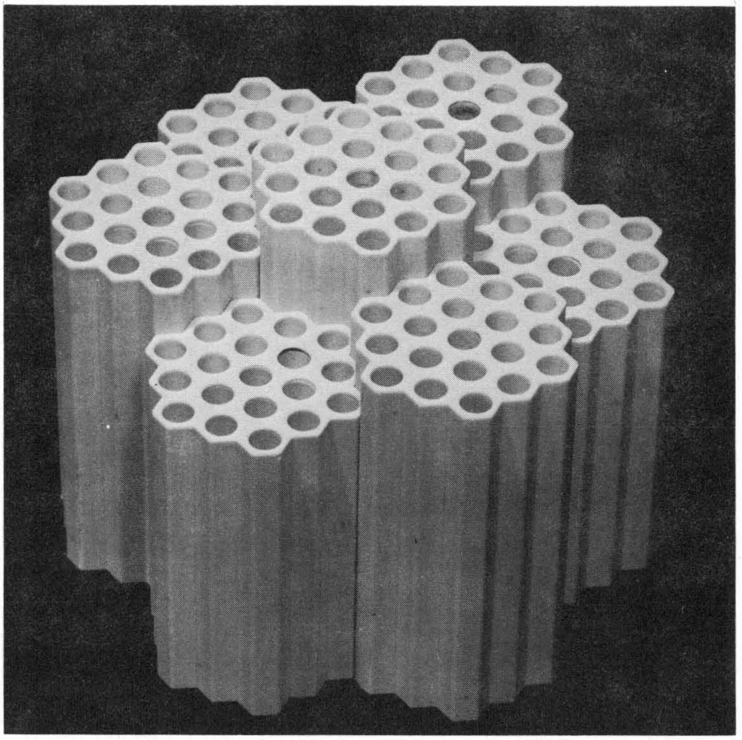

Fig. 4.21 – Beryllia fuel-tube transition pieces (Neg. C-23486)

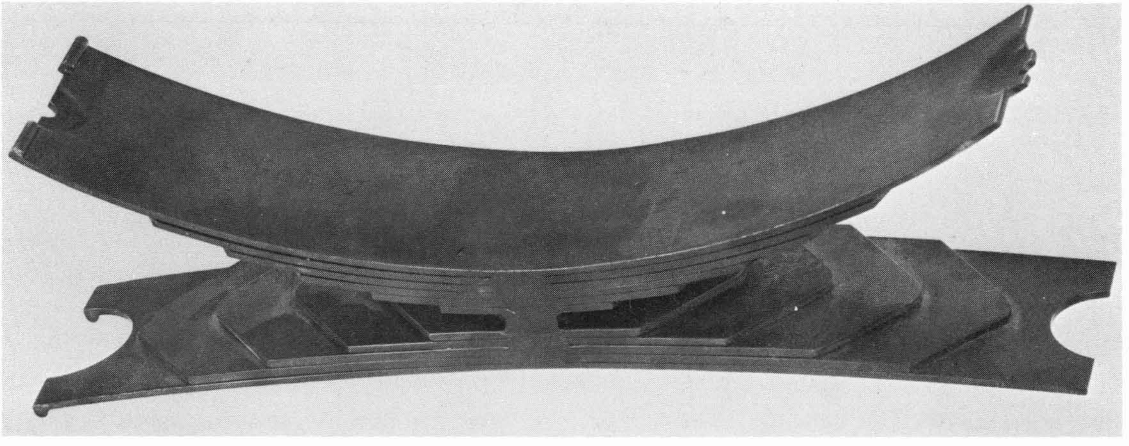

Fig. 4.22 – Leaf springs

were loaded outwardly against the structural shell and inwardly through the pressure pads into the tube bundle. The pressure pads served to distribute each spring load over several outer reflector tubes. Secondary heat due to neutron reactions in the side shield was reduced by neutron absorption in the pressure pads resulting from the addition of 1 weight percent of B^{10} added to the pressure pad material, which was cooled by compressor air.

The aft-retainer assembly, shown in Figure 4.23, was the main structural element of the longitudinal support system and resisted aft loads on the reactor. It was fabricated from twelve 30-degree sectors supported near the middle by the shaft tunnel, and near the outside by the rear shield outer section. Each sector consisted of parallel end-plates separated by tubes. The tubes acted as shear ties for the plates and also served as passages for primary-air discharged from the fuel elements. The assembly structure was internally air cooled.

The radial power distribution was flattened by varying the fuel concentration in annular regions of the active core, and resulted in radial power variations not exceeding 6 percent of the average over the core lifetime. The longitudinal power peak was shifted forward by the Be reflector at the forward end of the core to provide partial longitudinal temperature flattening.

The calculated fuel element average-channel maximum surface temperature was 2210°F at the design point. The corresponding fuel element maximum "hot spot" surface temperature due to temperature variations and perturbations was 2500°F. The temperature rise due to heat conduction through the fuel element was 30°F, and the maximum fuel element back-side surface temperature was 2530°F. These temperatures existed only toward the rear of the reactor. The variations and perturbations in fuel element temperature due to variations from flat power, control rod effects, manufacturing tolerances etc., are shown in Table 4.3. The relative distribution of fuel element temperatures is shown in Figure 4.24.

TABLE 4.3

MAXIMUM FUEL ELEMENT TEMPERATURE

	Temperature, °F
Average Maximum Surface Temperature (reference)	2210
Plus Built-in Temperature Deviations	120
Maximum Calculated Surface Temperature	2330
Plus Allowances	
Fabrication tolerance	100
Measurement uncertainty	70
Total	170
Maximum Estimated Surface Temperature	2500
Plus Internal Temperature Rise	30
Maximum Fuel Element Temperature	2530

Shield

The front shield consisted of an outer section and a central island supported by a structural member. Borated Type 304 stainless steel and borated beryllium were used in both regions. The slabs of shield material were slightly separated to allow passage of cooling air. Primary air was used to remove secondary heat in the front and rear shields.

The rear shield consisted of an outer section, which was attached to the reactor structural shell, and a central island, which was attached to the turbine front frame and to the structural wall of the combustor section. The shielding materials consisted of slabs of

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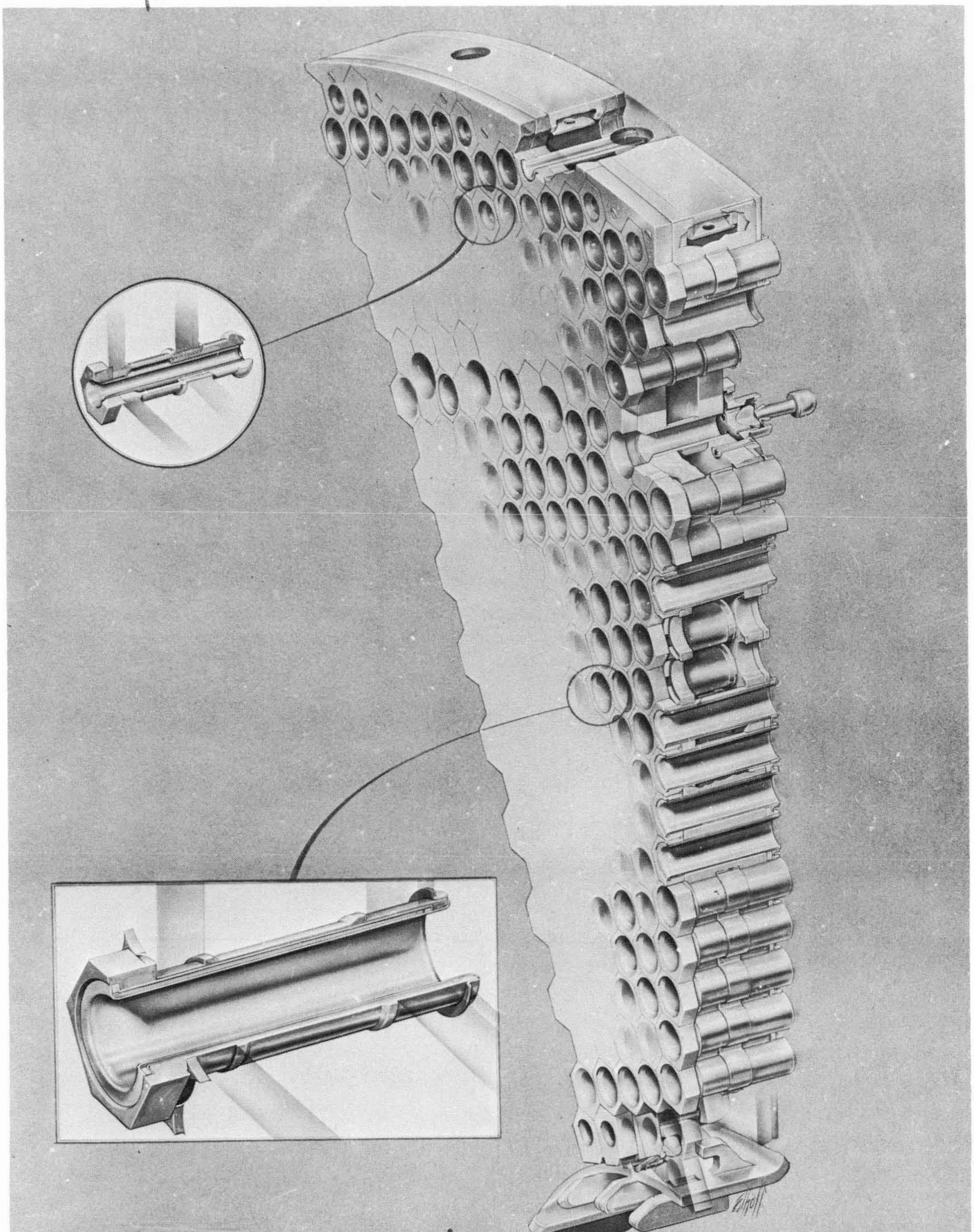

Fig. 4.23 – Aft-retainer sector, XNJ140E-1 (DI-533)

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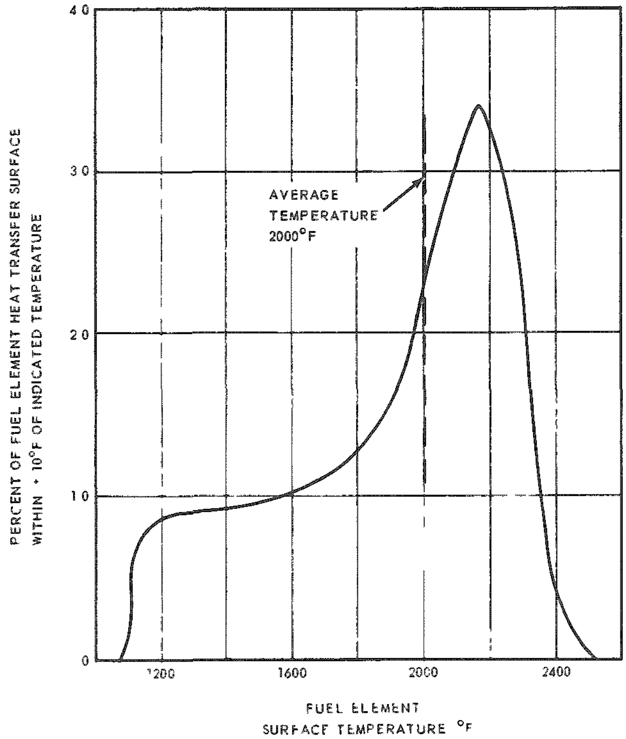

Fig. 4.24 – Temperature distribution of fuel elements in XNJ140E-1 reactor

borated beryllium, with the addition of borated stainless steel forward of the No. 3 bearing to act as gamma shielding for the bearing oil.

The side shield consisted of lithium hydride cast in stainless steel cans. Additional side shielding was provided by the beryllium oxide radial reflector. Cooling air flowed through stainless steel tubes dispersed throughout the side shield cross section. Air from an auxiliary blower was used to cool the side shield during ground testing; during flight, the side shield would be cooled with ram air.

Approximately 90 percent of the compressor inlet airflow was delivered to the nuclear midsection of the engine. Of this air, approximately 84 percent was used for cooling the active core and 16 percent was used for cooling the end shields and nonfueled components of the reactor. All air passing into the nuclear midsection was mixed and delivered to the turbine at various points upstream from the exhaust duct.

4.2.4 FINAL STATUS OF THE XNJ140E POWER PLANT

The XNJ140E program was on schedule when the Aircraft Nuclear Propulsion program was terminated in April 1961. Manufacturing drawings for the XNJ140E-1 were being released; the last release was planned for mid-year. Long-lead-time materials had been ordered, and developmental models of critical components had been proof-tested under simulated service conditions. Reactor critical experiments had been performed, and detailed planning of the test program was underway. Approval to proceed with fabrication and assembly had been requested.

Following is the development status of the major components at the termination of the program.

Turbomachinery

The X211 turbomachinery had accumulated 758 hours of operation at turbine inlet temperatures up to 1800°F and thrust levels up to 27, 370 pounds corrected to sea level static standard day without afterburner.

An illustration of the X211 engine equipped with a chemical burner to simulate the reactor is shown in Figure 4. 25. Four successive engines had been assembled in the single-engine configuration and tested using an engine shaft of the length required in the XMA-1. Redesign of the shaft, the bearing support frames, and the chemical burner had been completed for the XNJ140E configuration.

Later versions of the X211, capable of higher airflows and temperatures for use in more advanced models of the XNJ140 power plant, were being designed.

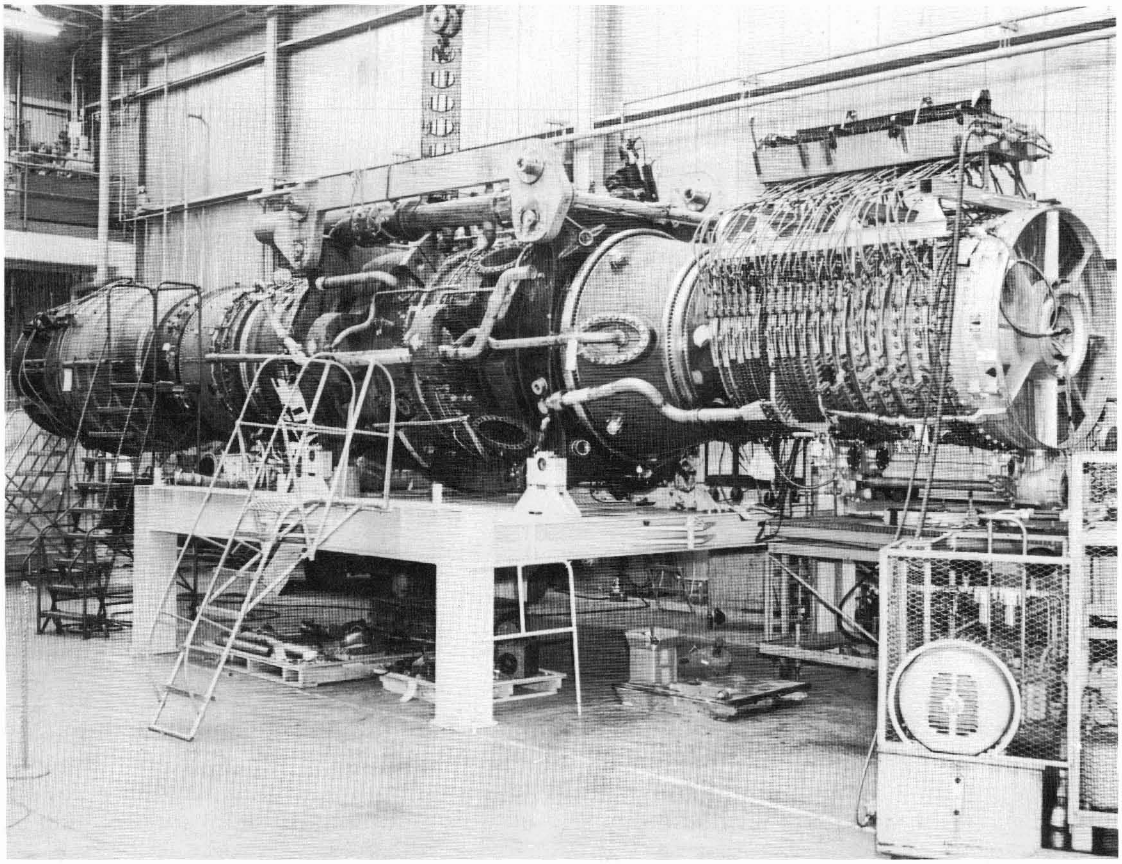

Fig. 4.25 – X211 turbomachinery with chemical combustion chamber replacing reactor

Shield

Shield materials had been fabricated in prototype component configurations for evaluation, which included tests of shield components under imposed radial temperature gradients to determine ability to sustain thermal stresses, thermal shock, and low cycle thermal fatigue. Tests had been performed to verify stresses and deflection of duct walls and structure under thermal and mechanical loading. Measurements had been performed to verify heat transfer predictions, particularly in the lithium hydride side shield. Full-scale and partial-scale airflow models had been operated to verify duct shape and cooling channel design.

Extensive nuclear measurements had been performed on the shield and on annular duct mockups. Evaluation of the duct leakage in HTRE No. 3, which had included annular ducts similar to those in the XNJ140E, indicated that duct radiation was within acceptable limits. This verified the prediction made in 1951 that duct leakage was not a critical problem in the direct air cycle.

Controls

XNJ140E reactor and turbomachinery control system concepts had been previously used and verified in the HTRE tests and X211 turbomachinery operation. The stability of the over-all control system had been verified on the analog simulator. Construction and proof testing of prototype components had been completed or were in process.

Reactor

Prototypes of most reactor components, as shown in Figures 4.17, 4.20, 4.21, and 4.22, had been built and tested under simulated operating conditions. Full diameter mechanical vibration and impact tests had been performed on large arrays of ceramic tubes. A typical test is shown in Figure 4.26. The purpose of this test had been to determine the effect of mechanical damage to the fuel tubes on the over-all mechanical integrity of the assembly. The test assemblies were subjected to vibration and impact loads simulating those expected in military aircraft due to gusts, evasive actions, and landing. It was found that mechanical integrity was retained with broken tubes and even if large voids were created. Frictional forces between the tubes, under the external constraint of expansion springs and pressure pads, were sufficient to bridge around the voids.

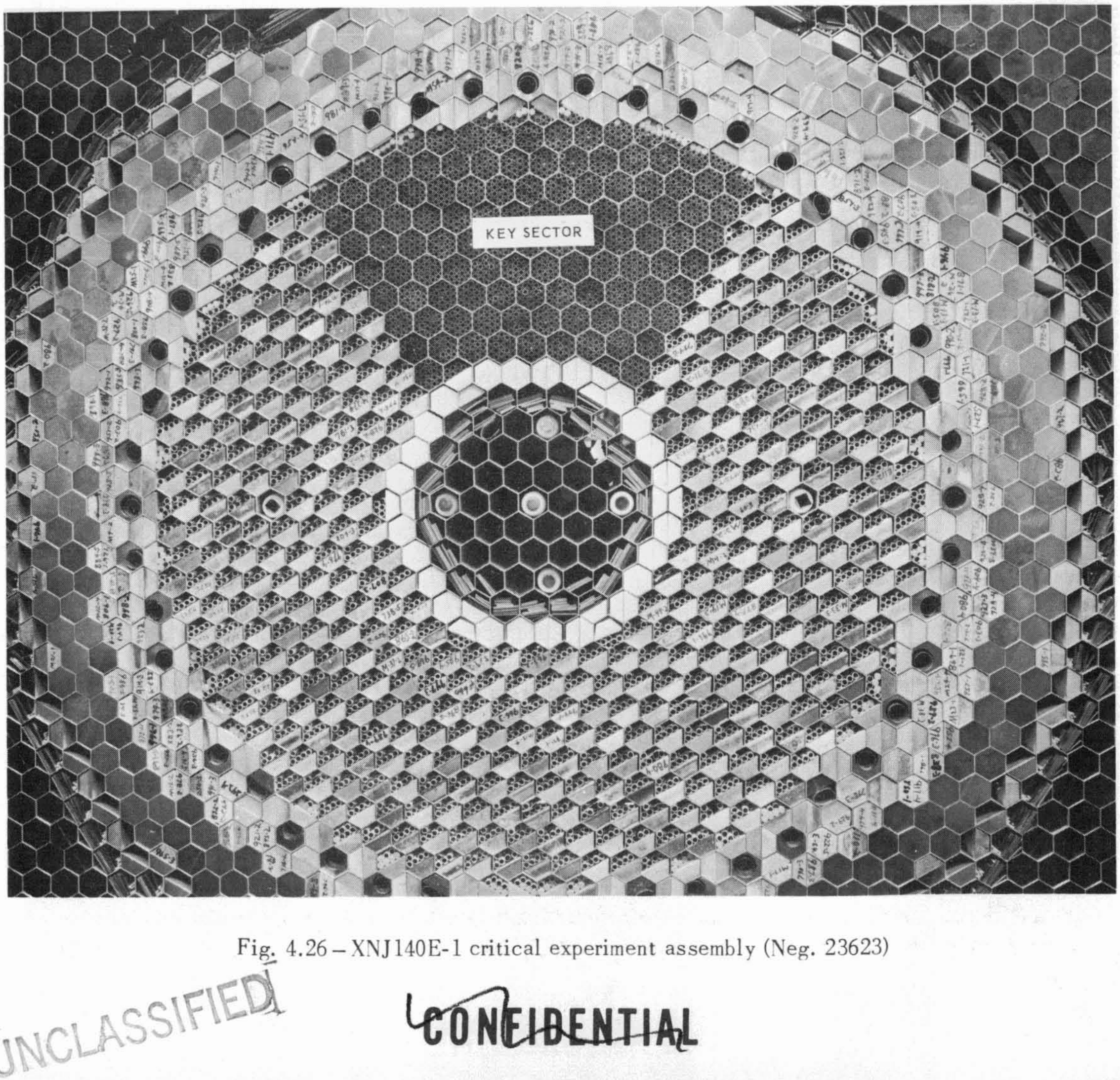

Fig. 4.26 – XNJ140E-1 critical experiment assembly (Neg. 23623)

Critical experiments had been performed to establish fuel element loading. (See Figure 4.27.)

Ceramic Fuel Elements

The ceramic fuel elements had been subjected to very detailed and careful development testing. Fuel element assemblies had been thoroughly evaluated in in-pile test programs: 3768 hours in the MTR, 5877 hours in the Oak Ridge Research Reactor, and 1038 hours in HTRE No. 2 (for prolonged periods at temperatures up to 2800°F and briefly at temperatures above 3000°F). These tests indicated that the fuel element was capable of operating at temperatures above the 2500°F maximum design temperature.

The retention of fuel in the fuel elements exceeded requirements by substantial margins. For example, tests made at 2600°F showed that less than 0.03 percent of the fuel was lost by volatilization in a 1000-hour period. A loss up to 10 percent could have been tolerated from the standpoint of reactivity. Even at 3000°F, a loss of only 3 to 5 percent was experienced in a 10-hour period.

The possibility that the ceramic tubes would bond together at high temperatures and over long periods when subjected to the radial forces imposed on the fuel element bundle by the radial support system was carefully investigated. Using forces between the tubes equal to approximately twice the maximum expected and at a temperature of 2675°F (about 150°F above the maximum predicted operating temperature), no sticking or bonding occurred in 500-hour tests.

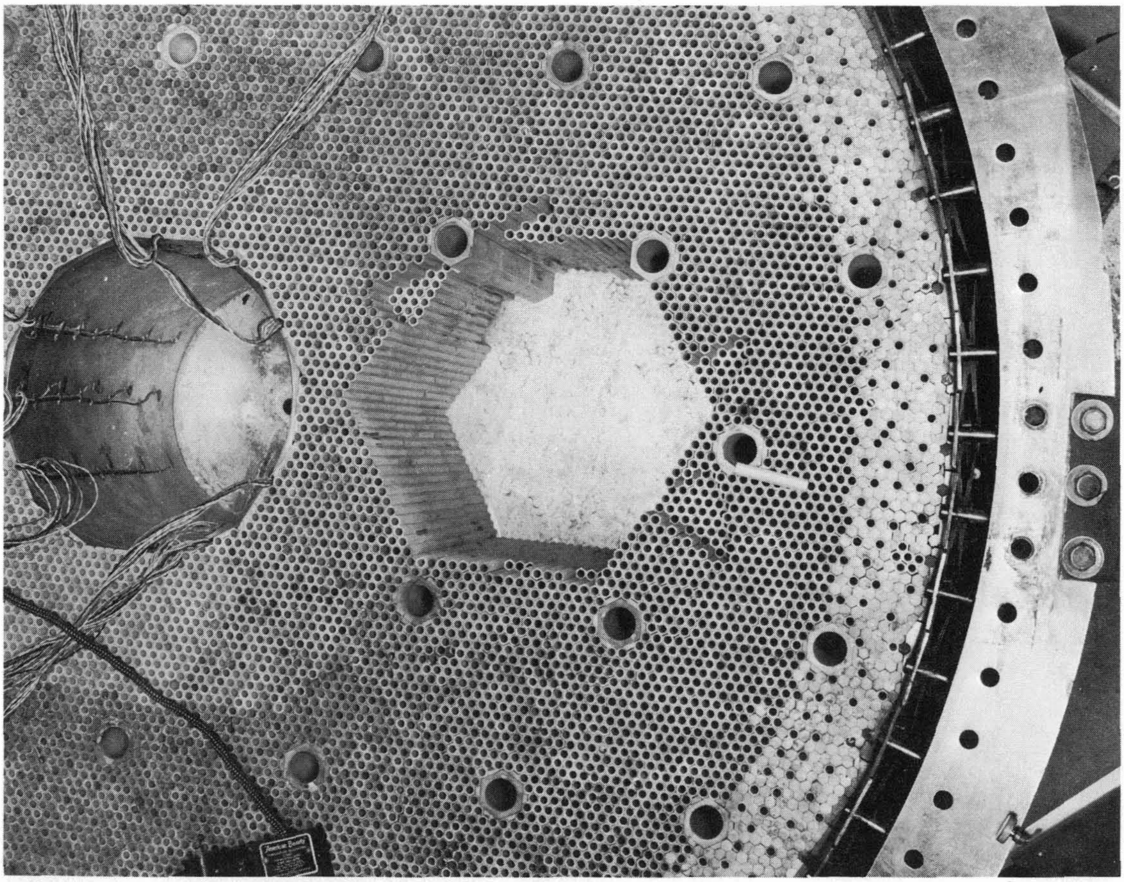

Fig. 4.27 – Three-tier mockup with 10-inch cavity after completion of 5-G shock load (Neg. C-23461)

The effectiveness of the zirconia coating in protecting beryllium oxide against water-vapor corrosion was measured extensively in tests of 1000 hours duration at temperatures of 2500°F and in air at a dewpoint of 25°F. In these tests the roughening of the inner surfaces of the fuel elements due to corrosion produced an increase in the friction factor of less than 1 percent. An increase of 10 percent or more could have been tolerated without adverse effects on engine performance or distribution of the cooling air.

Fuel material temperatures were high enough that creep under compressive forces might be expected. As much as 0.5 percent creep at 2200°F average core temperature was permissible without adverse effects. A large number of creep tests at various temperatures were performed and showed that an adequate margin existed in the design of the reactor. For example, at 2200°F and 2500°F the creep rate was immeasurable (less than 0.1 percent) in 1000 hours. At 2675°F, the maximum creep measured was about 0.7 percent in 1000 hours.

The effect of nuclear burnup on the strength of the fuel material was investigated. In a test in which the amount of burnup (fissions per unit volume) was roughly twice the maximum expected in the reactor during its entire operating life, the minimum crushing strength of fuel elements after irradiation was not detectably different from that of the fuel elements before irradiation.

Extensive testing was performed to determine the rate of release of fission products from the surfaces of the fuel. This work indicated that, at temperatures below 2500° to 2600°F, no diffusion of the fission products formed within the fuel material to the surface was detectable, although at higher temperatures (2750°F) diffusion became noticeable. Since the maximum operating temperature in the reactor was calculated to approximately 2530°F, fission products released from the reactor were expected to be limited to those that were ejected into the airstream as a result of direct recoil from the surfaces of the fuel elements. A large number of tests indicated that the rate of fission product release from the operating reactor would be considerably less than 0.1 percent of the fission products formed in the fuel. A preliminary study concluded that fission-product escape rates of a few tenths of a percent could be tolerated during operation of substantial numbers of nuclear aircraft.

The technology of ceramic reactor materials used in the design of the XNJ140E-1 is described in APEX-914, "Ceramic Reactor Materials," and the development of the power plant itself is described in APEX-908, "XNJ140E Nuclear Turbojet," of this Report.

4.3 REFERENCES

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5. ADVANCED POWER PLANTS

5.1 ADVANCED XNJ140 POWER PLANTS

Although the beryllium oxide reactor was the last reactor put into a hardware design status in the GE-ANPD program, simultaneous design and development studies were being made of reactors for use in advanced models of the XNJ140 power plant. Work was also proceeding on the development of X211 engine components for use at supersonic speeds at higher temperatures than in the XNJ140E.

Reactor development was proceeding along three lines: advanced BeO reactors, fast spectrum reactors, and folded-flow reactors. (A fourth, the rotating disc reactor, is an advanced version of the folded-flow reactor.)

5.1.1 ADVANCED BeO REACTORS

Design studies and component testing indicated that the XNJ140E BeO fuel tubes had already been sufficiently developed to operate at temperatures in excess of design requirements. Even higher operating temperatures were possible with further development. Advanced BeO reactors, therefore, were of primary interest for use in higher performance versions of the XNJ140 nuclear turbojet.

5.1.2 FOLDED-FLOW REACTORS

Another major advanced reactor design activity was concerned with folded-flow reactors utilizing metallic or ceramic materials. Potentially, folded-flow reactors could be smaller than straight-flow reactors providing the same air temperatures, but at the expense of more complex aerothermal and nuclear designs. A folded-flow reactor is shown in Figure 5.1. Extensive aerodynamic design, experimental studies, and nuclear critical experiments had been performed in support of the design. A proposed XNJ140 power plant designated the P122C, using a folded-flow reactor with niobium fuel elements, was in preliminary design. The P122C, shown in Figure 5.2, was being studied for possible supersonic use in the B-70 aircraft.

The rotating disc, or solid circulating fuel reactor, was an advanced design variation of the folded-flow concept that gave promise of heating large quantities of air with a small active core volume and lightweight shielding. In this configuration, an assembly of rotating fueled discs form a critical fissioning region at the shield center and a direct heat transfer region outside the active region. The discs convey the heat from the active region of the core to the shield periphery where it is removed by the engine airflow. This permits the use of a high density active core in a smaller reactor volume; thus, less shielding is required. The power plant using the rotating disc concept was designated the A115.

5.1.3 FAST SPECTRUM REACTORS

Unmoderated fast spectrum reactors were not developed in the early phases of the GE-ANPD program, primarily because uranium was not yet available in sufficient quantities.

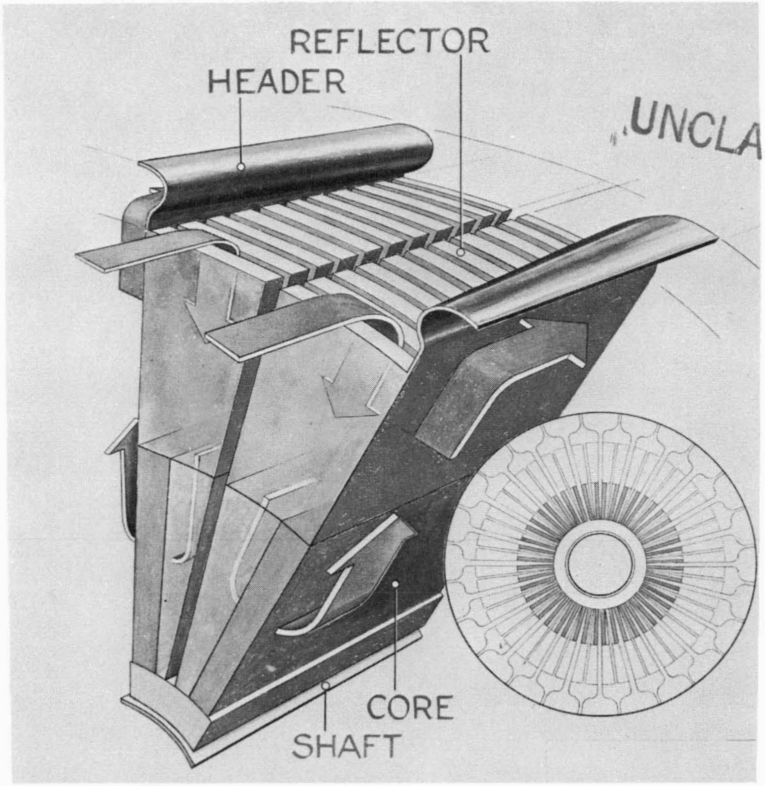


Fig. 5.1 – Folded-flow reactor concept

As this situation eased, studies were initiated for both ceramic and metallic unmoderated reactors. The combined oxides of thorium, plutonium, and uranium, or the oxides of yttrium and uranium were used in the ceramic reactors. Fast spectrum metallic reactors used refractory metals with surface protection against oxidation. The use of fast spectrum reactors introduced the possibility of adding alloying agents that could improve the high temperature oxidation resistance of metallic fuel elements but which were too poisonous from a nuclear standpoint to be considered for moderated systems.

A power plant designated the A132 using a fast spectrum reactor in an XNJ140 configuration is shown in Figure 5.3. In addition to the fast spectrum reactor, this concept also used a fast spectrum shield, consisting entirely of heavy materials of high temperature capability such as Inconel and thorium oxide. This choice was prompted by the fact that the neutrons that contribute most to the radiation dose after air scattering and crew compartment penetration are those that leave the reactor shield at energies in the vicinity of 10 Mev. On a thickness basis, heavy elements are more effective against these neutrons than lighter materials. The reduced shield thickness, the small diameter of the fast spectrum reactor, the effectiveness of the shield as a fast neutron reflector, the high gamma absorption of the heavy materials, the improved ability to remove shield heat at high temperature, and the ability of the Inconel to double as structure, all combined to produce a shield envelope that could conceivably fit entirely within the envelope of the X211 turbomachinery. This concept required further development before feasibility could be established. Nevertheless, it introduced the prospect of developing a nuclear engine which, in external appearance, was indistinguishable from a chemical jet engine but which could operate for the full life of the turbomachinery without a visible supply of fuel.

The A133 power plant was a variation of the A132 in which only a portion of the compressor discharge air was heated to extremely high discharge temperature. The discharge air was then mixed with the balance of the compressor discharge air before entering the

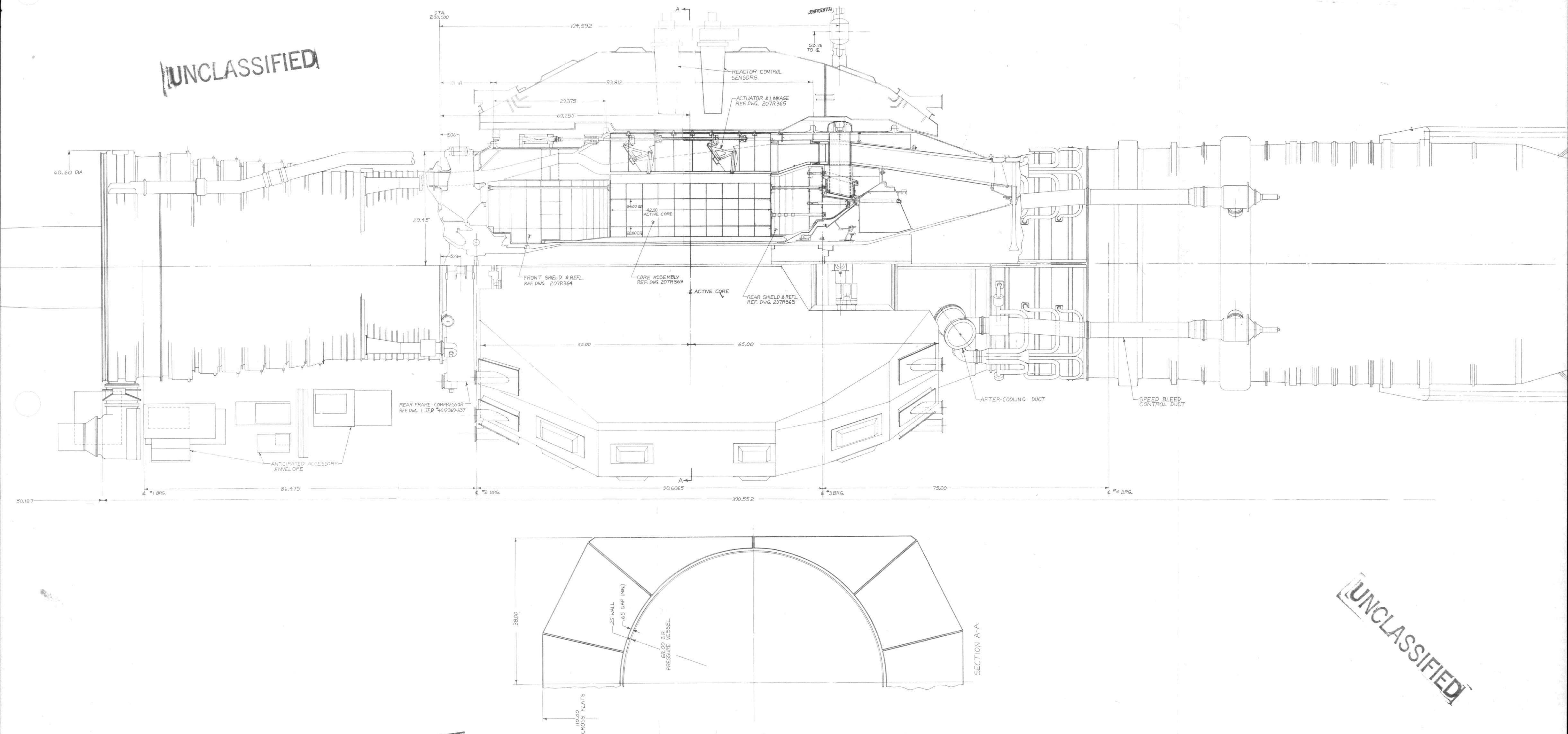


Fig. 5.2 – Supersonic XNJ140 using folded-flow reactor (Dwg. 207R366)

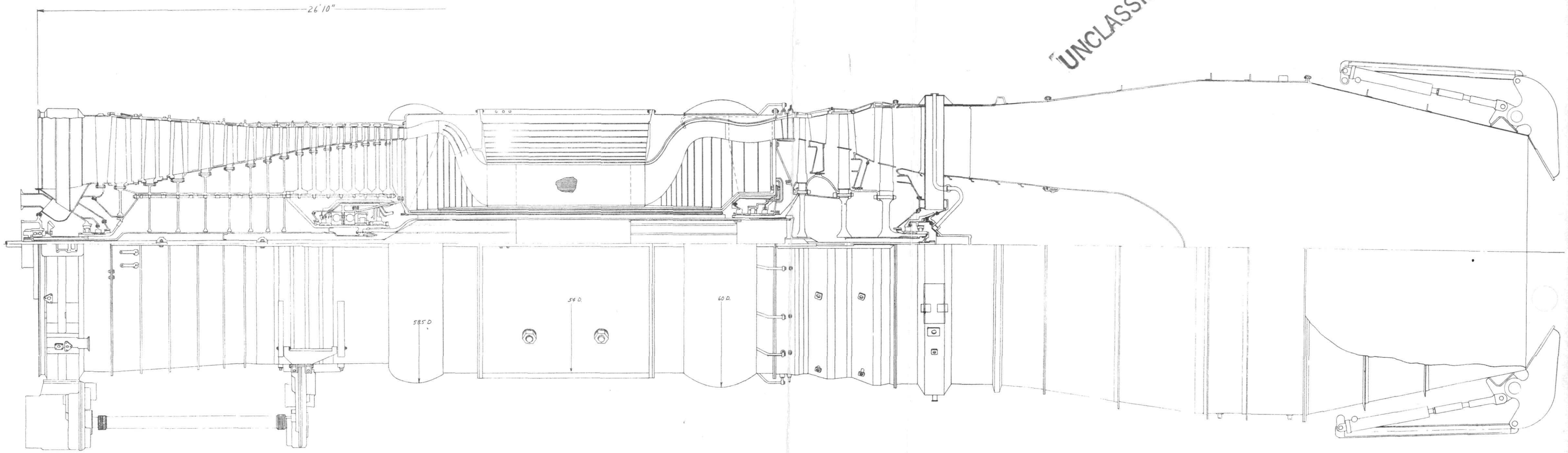


Fig. 5.3—Supersonic XNJ140 using fast spectrum reactor and shield

turbine at a lower average temperature level. The bypass air was taken from the compressor at an earlier stage at a somewhat lower pressure level than the air directed through the reactor to compensate for the pressure drop through the reactor. Thus, both streams, on mixing, were close to the same pressure. This type of cycle is similar in some respects to conventional chemical turbojets, because in chemical burning the initial flame is at a very high temperature and is reduced in temperature by the addition of air through the sides of the burner can before entering the turbine.

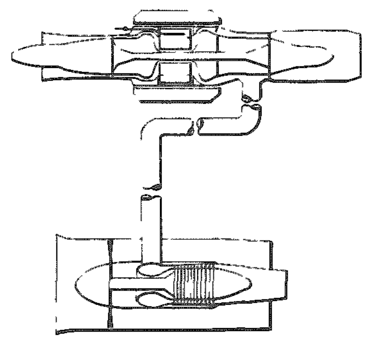
5.2 NUCLEAR TURBOPROP AND TURBOFAN POWER PLANTS

The nuclear turboprop combined the advantages of the gas turbine and the conventional propellor to provide superior low-speed, subsonic performance.

The nuclear turbofan, another modification of the basic turbojet engine, permitted the induction of additional ambient air which was compressed by the fan and exhausted to provide improved forward thrust. The nuclear turbofan power plant, like the turboprop, was studied for application in long range, subsonic missions requiring heavy payloads. The high takeoff thrust, permitting the use of short runways was an especially attractive feature of both the turboprop and turbofan power plants.

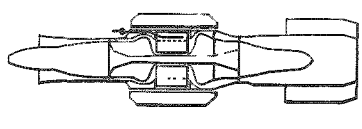
Typical configurations studied in the program are shown below. The feasibility of the turboprop was established using reactor materials which had been used in the HTRE tests. The XNJ140E reactor could also be used.

Bleed Turbofan.



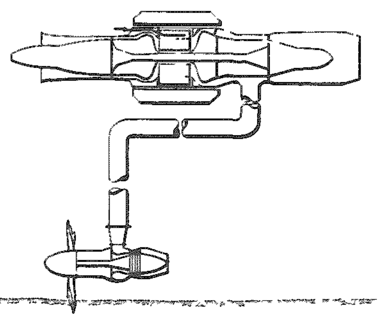
In bleed-turbofan adaptations of the turbojet, a portion of the reactor discharge air is ducted to the separate turbine driving a ducted fan.

Aft Turbofan.



In aft turbofan adaptations of the turbojet, the fan turbine is located aft of the main turbine. An illustration of an aft turbofan in an XNJ140E configuration is shown in Figure 5.4.

Bleed Turboprop.



Bleed turboprop versions of the turbojet were similar to the turbofan except that a full diameter propeller was used instead of the smaller diameter ducted fan.

5.3 NUCLEAR RAMJETS

The nuclear ramjet operated on a thermodynamic cycle basically the same as that of the turbojet. Air was 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 was provided by inlet diffusion at high flight speeds, the requirement for rotating turbomachinery was eliminated. The nuclear ramjet was studied for applications

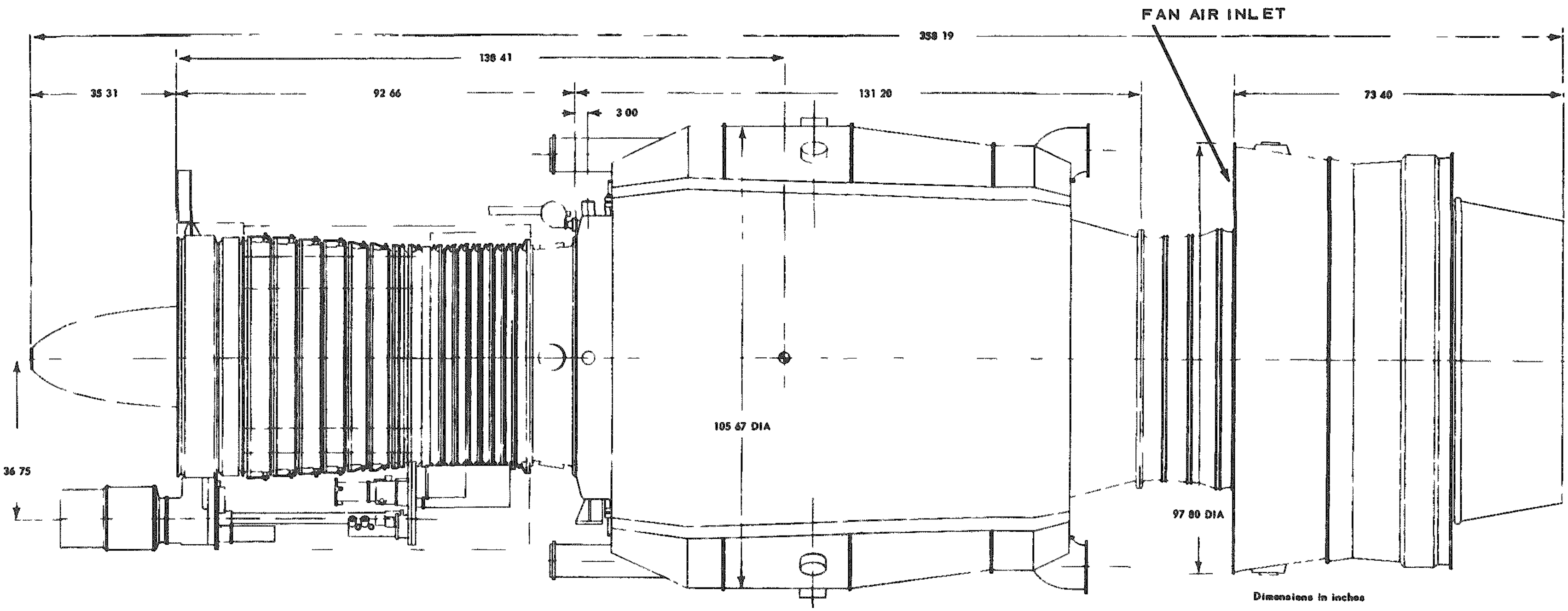


Fig. 5.4 – Aft turbofan version of XNJ 140 configuration

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requiring long endurance at high supersonic flight speeds in the atmosphere.

Active studies of nuclear ramjets started early in the GE-ANPD program. Design and development was carried to the point that feasibility was apparent for low- and high-altitude supersonic missions. A separate program was then established within the National Laboratories leading to ground test reactor experiments. Ramjet work was subsequently de-emphasized at GE-ANPD. Development of the ceramic reactor was concentrated almost entirely on turbojet application, whereas it had been previously planned for possible use in both the ramjet and turbojet. Shortly before program termination, a ramjet design study was made utilizing the ceramic reactor technology developed for the XNJ140E nuclear turbojet applied to the ramjet reactor for the PLUTO program. An illustration of the resulting configuration is shown in Figure 5.5. An installation configuration is shown in Figure 5.6.

The principal configuration deviation from the XNJ140E reactor was in the use of a ceramic arch for aft structural support rather than a cooled metallic tube sheet. Also four large central control rods were used instead of reflector rods.

The fuel tubes used in the TORY II-A-1 test were fabricated by GE-ANPD under contract to the AEC Lawrence Radiation Laboratory.

5.4 CLOSED-GAS-CYCLE NUCLEAR POWER PLANTS

Several closed-gas-cycle aircraft propulsion systems were studied. In a closed-gas-cycle turbojet, the gas is heated in the reactor and pumped in a closed loop, through a heat exchanger located in the engine. This is directly analogous to the liquid-metal system except that a gas is used instead of the liquid metal in the closed loop.

A closed gas cycle can also be used in a turbofan in which case the gas is used to drive the turbine and waste heat is rejected in a heat exchanger located in the fan discharge. This then becomes a "heated fan" or, in effect, a "compressor jet." This system takes advantage of a major difference between the indirect gas cycle and the indirect liquid-metal cycle, in the gas cycle the closed-loop working fluid can drive the turbine, whereas this is not the case with the liquid-metal system unless the liquid metal is vaporized. A closed gas cycle could also be used in a turboprop system.

The design studies indicated that closed gas cycles might be particularly attractive if very high reactor discharge temperatures could be achieved, which appeared to be feasible, at the time of program termination, with fast spectrum reactors using an inert gas as a coolant. A derivative from the aircraft closed-gas-cycle studies is shown in Figure 5.7. This is a highly integrated, shaft power package proposed for maritime propulsion. The reactor is located between the compressor and turbine as in the XNJ140E. The waste heat exchanger is wrapped around the unit. Reduction gearing is also included in the power package.

5.5 NUCLEAR ROCKETS

Propulsion in the nuclear rocket is attained by heating a propellant in the reactor and discharging the expanding fluid through a jet nozzle. The nuclear rocket is capable of a higher specific impulse (thrust per pound of propellant flow per second) than chemical rockets because the propellant for the nuclear rocket may be exclusively lightweight hydrogen, whereas the chemical system requires a high-molecular-weight oxidizer.

5.5.1 GRAPHITE REACTORS

Active design studies of nuclear rockets using graphit and carbide reactors started

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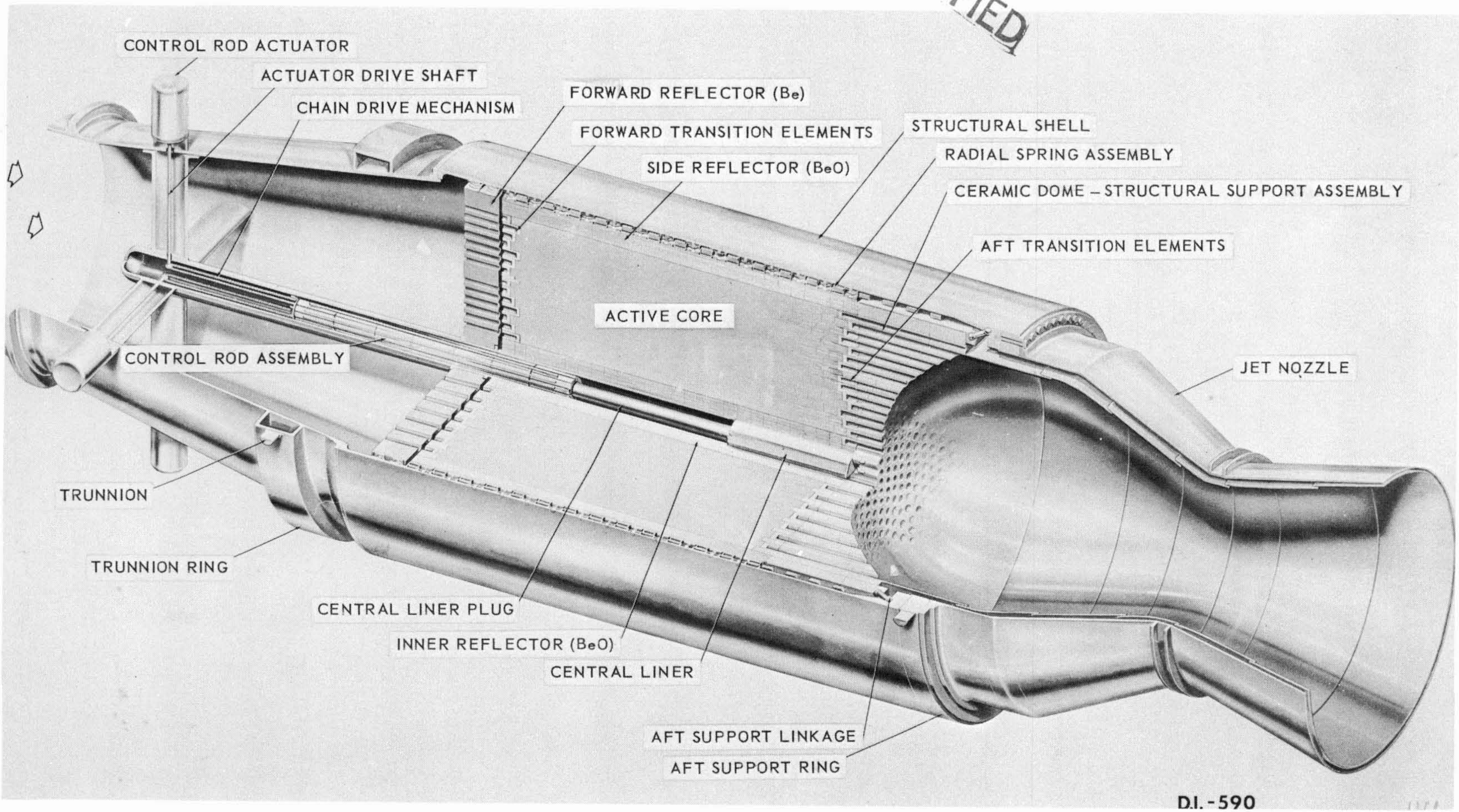

Fig. 5.5 – Nuclear ramjet reactor

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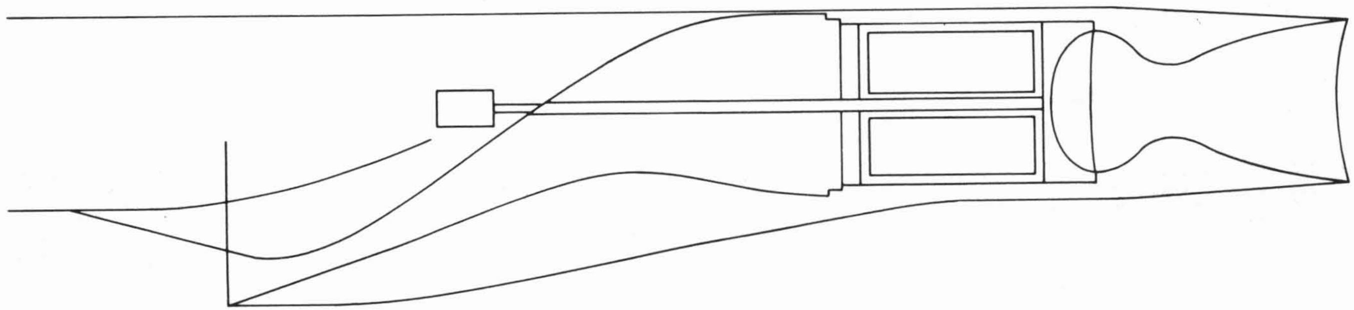

Fig. 5.6 – Ramjet power-plant configuration, bottom inlet

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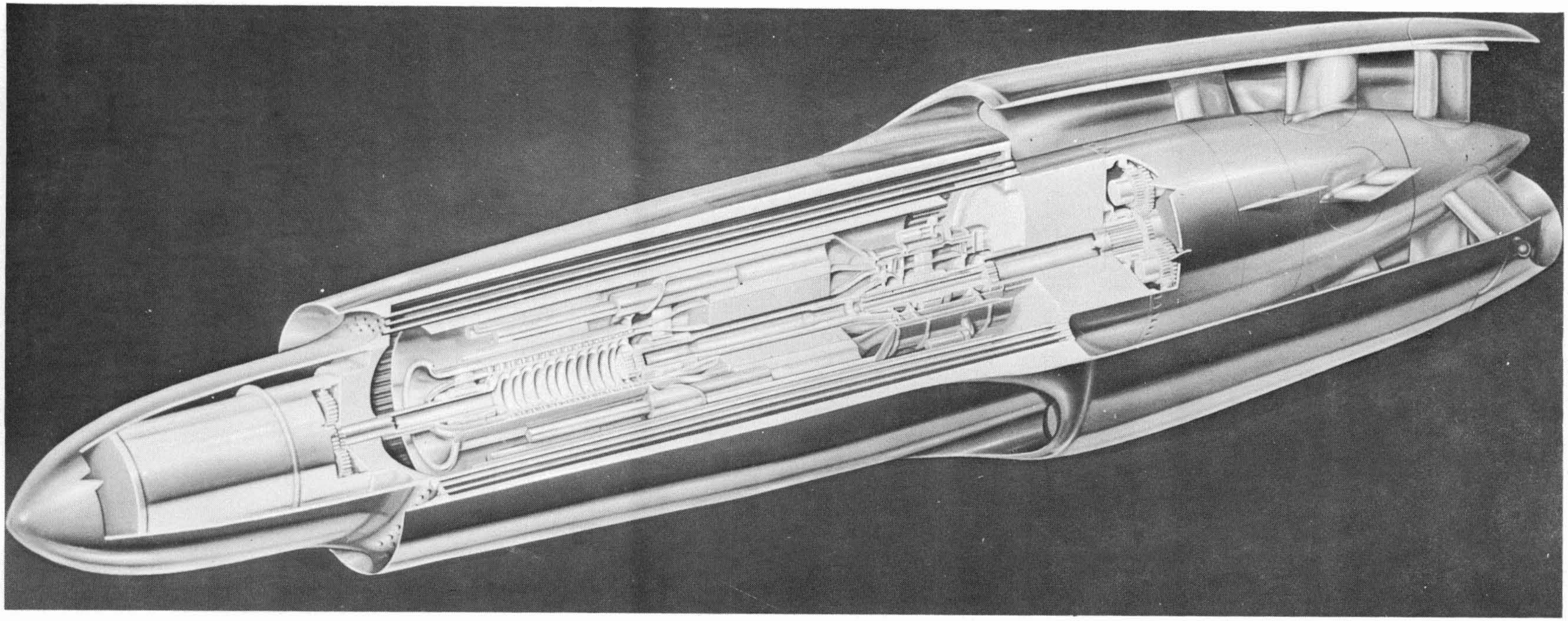

Fig. 5.7 – Artist's concept of 601A power package

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in 1954. Initial emphasis was placed on an application to the Atlas XB-65 ballistic missile. The system was envisioned as a hydrogen-cooled, nuclear-first-stage rocket reactor booster with a chemically powered second-stage missile. The scope of the work included a mechanical design of the nuclear reactor with nuclear and aerothermal performance estimates, design of related nuclear power plant components, an estimate of the characteristics of the chemically powered second stage, and a detailed weight estimate of the over-all missile.

The nuclear rocket configuration shown in Figure 5.8 consisted of a hydrogen storage tank, a turbine-powered, 5-stage centrifugal pump, a reactor, and a converging-diverging nozzle. The net thrust of the power plant was 300,000 pounds. The reactor core (see Figure 5.9) consisted of hexagonal graphite element tubes stacked parallel to the core axis. The study of graphite and carbide reactors, primarily for use in upper stages, continued throughout the program, but at a low activity level because of the increasing nuclear rocket activity in the National Laboratories.

5.5.2 HYDROGEN-MODERATED REACTORS

A brief investigation was made of hydrogen-moderated nuclear rocket reactors. The investigation consisted of preliminary nuclear sizing and performance studies on a modified HTRE-1 reactor and a water or pentalene moderator cooled by hydrogen. The fuel elements were similar in configuration to the HTRE-1 elements but were made of refractory metals.

The power plant configuration is shown in Figure 5.10. Liquid hydrogen was pumped from the storage tank, preheated while cooling the liquid moderator, heated to required temperatures in the reactor, and expelled through a converging-diverging nozzle. A secondary circulating system pumped moderator water from the reactor through the preheater and back to the reactor. Both pumps were powered by a hydrogen peroxide-driven turbine.

The study indicated that even with Ni-Cr fuel elements of the type already developed, significant improvements over chemical systems were achievable.

5.5.3 REFRACTORY METAL REACTORS

At the time of program termination the major GE-ANPD interest in advanced nuclear rocket reactors centered on the use of small, fast-spectrum, refractory metal reactors.

5.6 COMPOSITE CYCLES

A number of composite cycles combining features of the basic power plant types were also studied. These are summarized briefly below.

In the nuclear compressor jet cycle (also referred to as the "turbo-supercharged ramjet), the reactor discharge airflow was divided so that part was used to drive the turbine and the remainder was passed directly through the jet nozzle. Two designs of this type, the A113 and A136, are reported in APEX-909, "Aircraft Nuclear Propulsion Systems Studies," of this Report. This system would have been useful only if the reactor were able to deliver air temperatures well in excess of turbine temperature. The reactor discharge air would therefore be diluted with cold compressor air while the bypass air would exit at full reactor discharge temperature.

The nuclear turbojet-ramjet was a system in which the reactor reheated the exhaust gases from a turbojet engine at low flight speeds and then operated as a ramjet at high

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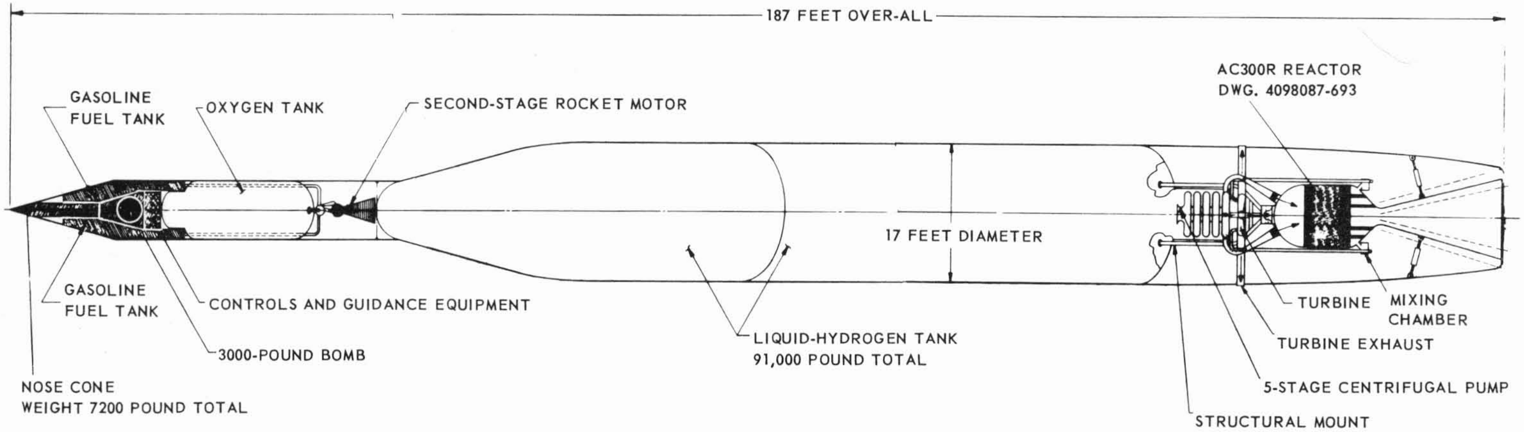

Fig. 5.8 – Ballistic missile with AC300R power plant as first stage

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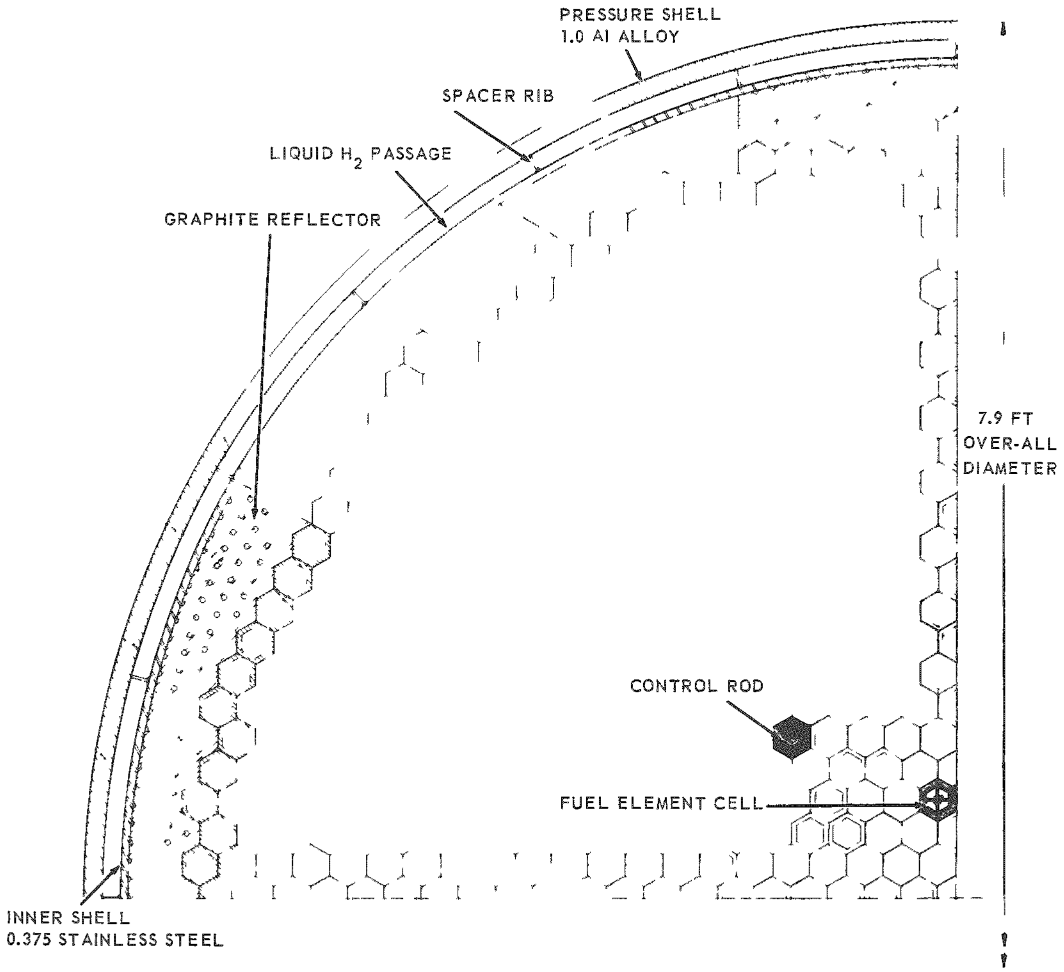

Fig. 5.9 – Section of AC 300R reactor

flight speeds. In low speed missions of short duration, the turbojet could be operated on chemical fuel. The turbojet was bypassed at high speeds so that the air entered directly into the reactor, as in a ramjet. Thus the turbojet-ramjet was studied to solve the problem of the simple ramjet which could not operate effectively in the subsonic and low supersonic flight regime. A design of this type, the A128, proposed for operation at Mach 4.5 at 70,000 feet, is reported in APEX-909. The A128 configuration is shown in Figure 6.7.

The nuclear turbojet-rocket embodied the basic features of both the turbojet with nuclear afterheat and the nuclear rocket cycles. In this configuration, when sufficient speed and altitude were reached, the air entry ports were closed off and a propellant was utilized for the nuclear rocket portion of the cycle. The nuclear turbojet-rocket was to be capable of operating to intermediate supersonic speeds as an air-breather in the lower atmosphere and to hypersonic velocities in the upper atmosphere and outside the earth's atmosphere. The reactor would be required to operate in both oxidizing and nonoxidizing atmospheres. Yttria-urania ceramic or coated refractory metals appeared to have this potential. Feasibility had not been established, however.

The nuclear turbojet-ramjet-rocket was a combination of all three basic propulsion cycles. In this system, low-speed propulsion would be provided utilizing a turbojet engine

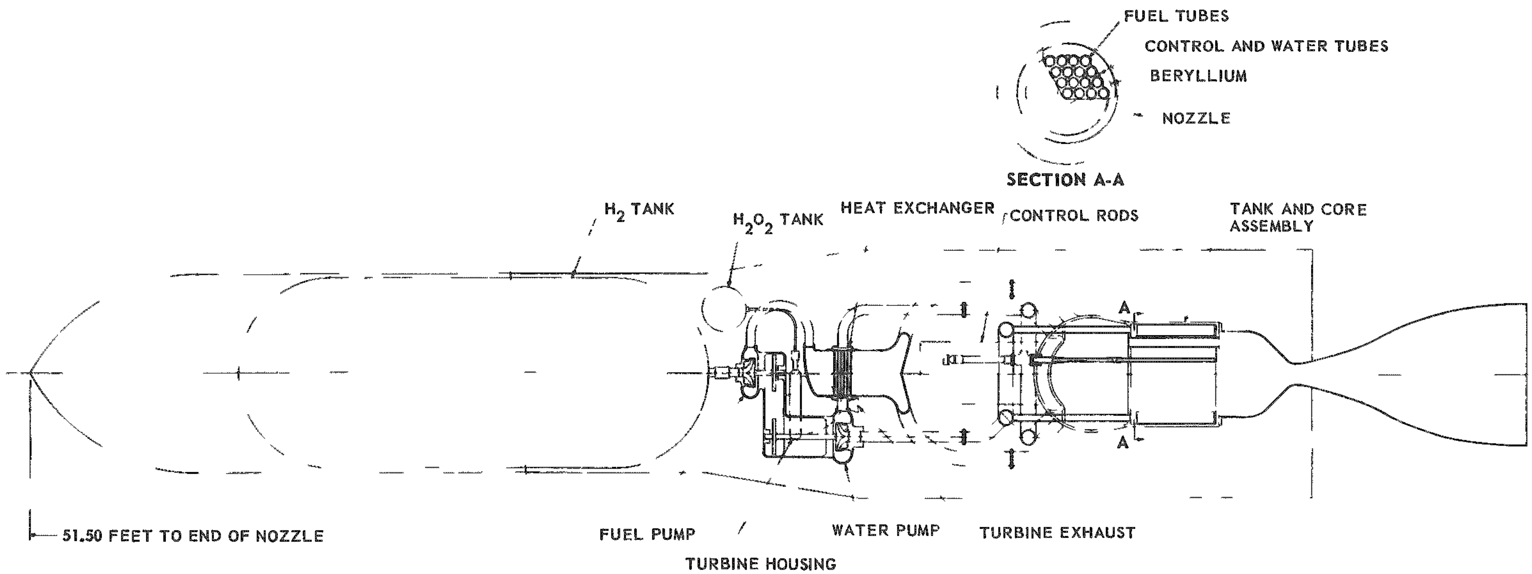

Fig. 5.10 – HTRE No. 1 nuclear rocket (Dwg. 207R297)

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with nuclear reheat as in the turbojet-ramjet. At higher speeds the turbomachinery would be bypassed to provide nuclear ramjet propulsion. At even higher flight speeds the atmospheric inlet to the engine would be closed and a propellant supplied so that the system would function as a nuclear rocket.

All of the various systems studied by GE-ANPD are described more fully in APEX-909, "Aircraft Nuclear Propulsion Systems Studies," of this Report.

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6. APPLICATION STUDIES

6.1 SUBSONIC MISSILE-LAUNCHING AND PENETRATION AIRCRAFT

Application studies were performed in 1959 to develop the aircraft and power plant characteristics required to meet the Department of Defense guidance for cruise at Mach 0.8 to 0.9 at 35,000 feet. Aircraft characteristics were based on the Convair Model 54. An advanced Convair Model 54 aircraft using four direct-air-cycle nuclear engines is shown in Figure 6.1. Air launch ballistic missiles were to be carried under the wings; an additional bomb load was to be carried in the fuselage. The Model 54 could cruise at continuous airborne alert for approximately 5 days, could launch ballistic missiles with nuclear warheads from outside the target area, and could penetrate the target area at sea level and high subsonic speed at altitude for reconnaissance or to drop additional weapons.

Both three-engine and four-engine aircraft were examined in the application study. Aircraft optimized for the cruise condition, as well as others optimized for low level penetration were considered. Power plant performance was based on the single- and dual-engine data developed in the Advanced Configuration Study (section 4). The aircraft performance using the P140B is summarized in Table 6.1. Aircraft takeoff was assumed to be entirely

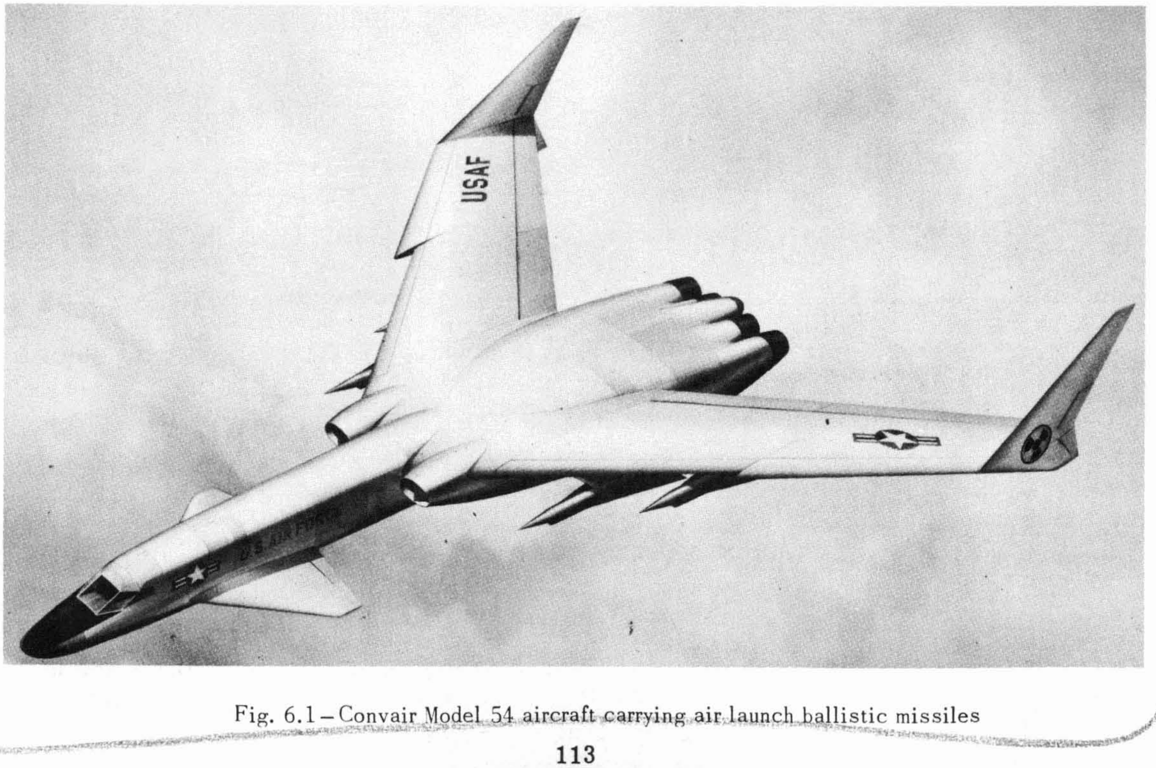

Fig. 6.1 – Convair Model 54 aircraft carrying air launch ballistic missiles

TABLE 6.1

PERFORMANCE OF SUBSONIC MISSILE LAUNCHING AIRCRAFT USING P140B NUCLEAR TURBOJET ENGINES

	Three-Engine	Four-Engine
Performance		
Cruise with full payload		
Speed, Mach No.	0.8	0.85
Altitude, ft	32,800	33,000
Rate of climb with reactor out at 5000 feet, hot day, ft/min		
With payload	130	340
No external payload	380	580
Ground run at 5000 feet, hot day, ft	6,900	7,200
Critical field length at 5000 feet, hot day, ft	8,200	8,300
Rate of climb with engine out, flaps set for takeoff, and gear down at 5000 feet, hot day, ft/min	1,500	1,500
Aircraft Gross Weight, lb	474,270	576,000

on chemical power using both chemical interburning and afterburning in the nuclear engines. Additional takeoff thrust was provided by two wing-mounted chemical turbojets which were not used in flight. As an alternative, water-alcohol injection could be used to augment the chemically powered takeoff thrust of the nuclear engines. This reduced the aircraft weight, but increased the critical field length by approximately 4000 feet for the three-engine aircraft and 2000 feet for the four-engine aircraft. This was still within the 15,000-foot limitation.

This study was instrumental in selecting the three-engine aircraft and the P140B power plant configuration to meet the Department of Defense guidance. The auxiliary turbojet engines were adopted for takeoff augmentation in preference to water-alcohol injection because of the added flexibility. The aircraft was designated the "NX2" and the power plant the XNJ140E. The NX2 aircraft is shown in Figure 1.2. Figure 6.2 shows three XNJ140E engines installed in the NX2 aircraft. Flight conditions are given in section 4. Details of the NX2 and generic Model 54 aircraft may be obtained from Convair reports.

A study was performed of a B-52G aircraft as a flight test bed to evaluate and further develop the XNJ140E-1 nuclear engine prior to availability of the Convair NX2 aircraft. The aircraft would carry the engine in an external nacelle, side-mounted on the aft fuselage as in the Sabreliner and Caravelle configurations. The study indicated that the B-52G could be used for this purpose.

The first phase of the proposed B52 flight program was to be conducted with a chemically powered version of the power plant. Following the chemical-operation phase, a single nuclear power plant was to be installed, and nuclear flight testing initiated. Installation of a second XNJ140E-1 power plant on the other side of the fuselage, to create a twin-pod configuration, was assessed and the aircraft stressed for this condition. A performance calculation based on a configuration of two XNJ140E-1 nuclear power plants and eight J57 chemical engines showed that the modified B-52 aircraft was capable of demonstrating all-

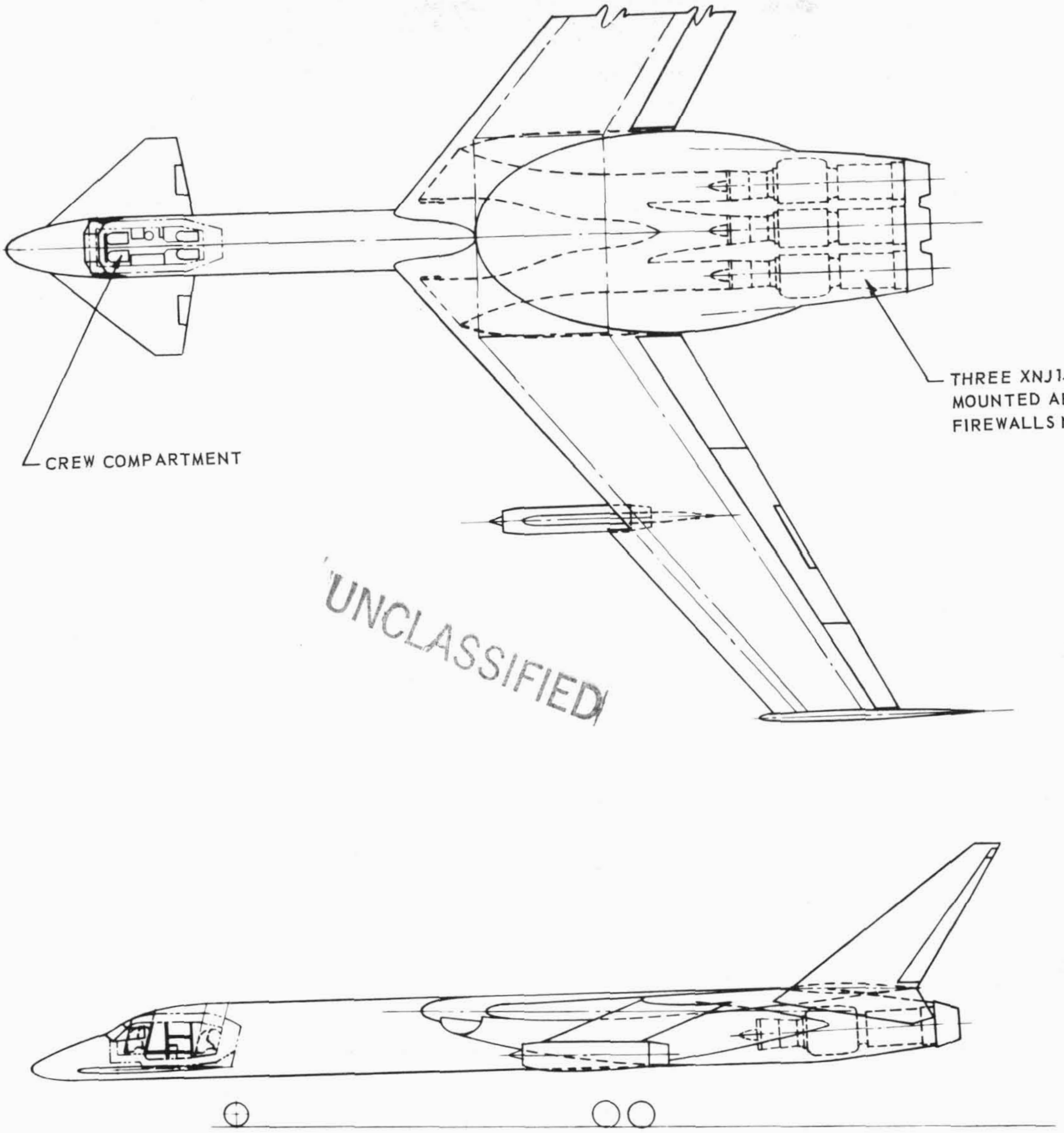

Fig. 6.2 – Installation of XNJ140E engines in Convair NX2 airframe (Dwg. 848D110)

nuclear flight. Improved nuclear performance could be obtained by removing the chemical engines not required for takeoff.

6.2 PRINCESS FLYING BOAT

In 1958, the United States Navy sponsored a study of the application of nuclear propulsion to the Saunders-Roe "Princess" flying boat. The companies participating in the study were Saunders-Roe, the Martin Company, and Convair-San Diego for the aircraft and General Electric and Pratt and Whitney for the nuclear power plants. Several of these flying boats had been built in England for trans-Atlantic passenger service but then had been moth-balled. The aircraft and engine configuration proposed by Convair is shown in Figure 6.3. The purposes of the program were (1) to develop an early, simplified aircraft system to demonstrate the feasibility of a nuclear-powered aircraft, and (2) to produce, by successive, well-planned stages of development, operational aircraft to fulfill the Navy's assigned missions.

The initial flight test operation was to be performed at an altitude of 10, 000 feet at a speed of 220 knots on a standard day. Follow-on power plants would provide improved performance.

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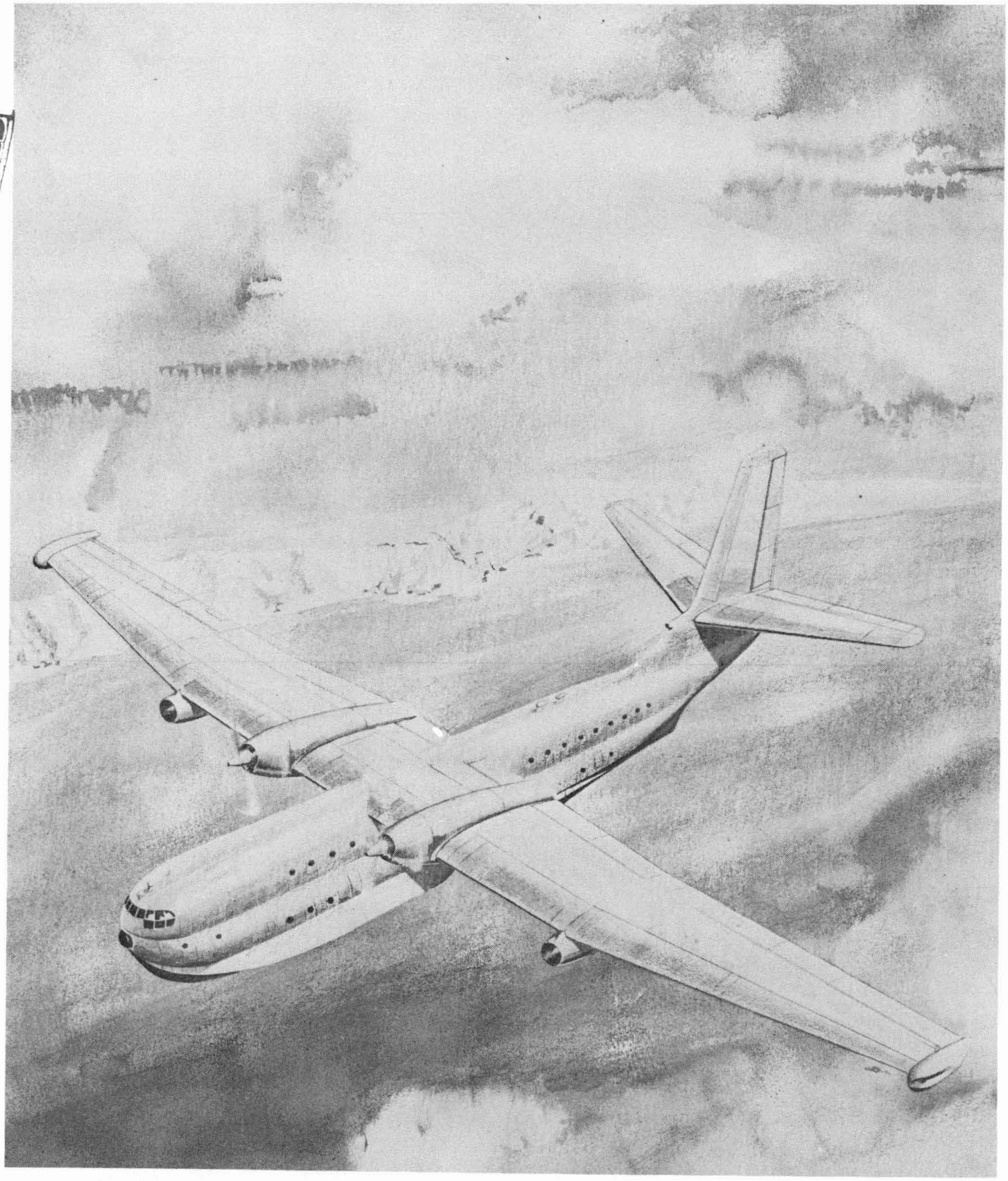

Fig. 6.3 – Convair – San Diego modification of the Princess aircraft

The study was based on existing GE-ANPD technology and employed, wherever practical, existing components and subsystems. A turboprop nuclear propulsion system was proposed using technology developed in the HTRE reactors which had indicated early availability of 1400°F turbine inlet temperatures. The aircraft structural weight as modified was targeted at 115,000 pounds. For takeoff and landing, the maximum weight was set at 345,000 pounds, leaving the difference for the nuclear system and for auxiliary chemical power plants and fuel.

The study indicated that the initial flight objective could be met using the HTRE technology. Technology later available in the XNJ140E program would have allowed substantially improved performance.

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6.3 HEAVY PAYLOAD SUBSONIC LOGISTIC AIRCRAFT

Studies were performed early in 1961 of heavy payload aircraft for possible use as airborne command stations, antisubmarine patrol, airborne early warning radar systems, logistic carriers, etc. Turbojets, turbofans, and turboprops based on XNJ140E technology were used in the study. The turbojet was an advanced XNJ140E with a turbine inlet temperature of 2000°F. The turboprops and fans were driven by bleed air from a turbojet gas generation (see section 5). The factors influencing the design and performance of a particular application (including payload capability) were the power-plant weight and performance capabilities, the cruise conditions assumed, the emergency performance required, and the desired takeoff characteristics. In particular, the emergency (engine-out) cruise and emergency go-around after takeoff each required the aircraft to maintain a 100-foot-per-minute rate of climb capability at 5000 feet altitude under Air Force Hot Day (AFHD) conditions with payload on board. The takeoff restraints included a maximum takeoff velocity of 170 knots and a maximum critical field length of 15,000 feet, also assuming an AFHD atmosphere. Cruise altitudes and flight speeds were considered from 20,000 to 40,000 feet and from Mach 0.5 to 0.9 respectively.

The comparative statistics of the three types of power plants are indicated in Table 6.2. The cruise altitudes represent the altitude that corresponds to the best payload for the given aircraft configuration and flight speed. The highest payload capabilities were associated with medium-subsonic flight speeds at altitudes ranging from 20,000 to 30,000 feet. The

TABLE 6.2

COMPARISON OF NUCLEAR TURBOJET, TURBOAFT-FAN, AND TURBOPROP FOR HEAVY PAYLOAD SUBSONIC AIRCRAFT

	Three-Engine Turbojet[a]	Three-Engine Turboaft-Fan[b]	Three-Gas-Generator Turboprop[a]
Gross weight, lb	844,000	973,300	1,340,000
Cruise weight, lb	829,000	958,300	1,327,000
Payload, lb	225,900	304,800	434,400
Crew shield weight, lb	100,000	87,000	100,000
Propulsion system weight, lb			
Nuclear	188,200	193,500	188,200
Chemical auxiliary	17,000	17,000	-
Bleed-prop assembly and accessories	-	-	150,000
Fuel and oil weight, lb	22,400	22,400	18,000
Equipment weight, lb	32,270	39,850	46,430
Crew weight, lb	900	900	900
Structural fraction, %	29.7	31	31

[a]Design point: Mach 0.7 at 20,000 feet.
[b]Design point: Mach 0.7 at 25,000 feet.

turboprop provided the best load carrying capability. Extremely heavy payloads could be carried, ranging from 200,000 to over 400,000 pounds.

6.4 SUPERSONIC RECONNAISSANCE-STRIKE AIRCRAFT

Advanced models of the XNJ140 power plant were studied in 1960 and early 1961 for use in a supersonic reconnaissance-strike system capable of reconnoitering enemy territory and destroying enemy installations. The assumed mission requirements defined an airplane with the following basic capabilities:

1. Airborne alert for extended periods of time
2. Penetrate enemy territory either at sea level or at high altitudes and supersonic speeds
3. Carry large payloads (50,000 to 100,000 pounds)

This system was an extension of the B-70 concept providing the additional advantages of airborne alert, longer range, sea level as well as altitude penetration, higher payloads and independence from fuel supplies in time of emergency.

The supersonic lift-to-drag ratios used in the airplane studies are shown in Figure 6.4. These are design-point values and are based on a fuselage of the fixed B-70 type. The effect of design speed on the ratio of structural weight to gross weight is shown in Figure 6.5. Typical wing loadings are shown in Table 6.3. The performance of the resulting airplane for payloads of 50,000 and 100,000 pounds is given in Table 6.4 and 6.5.

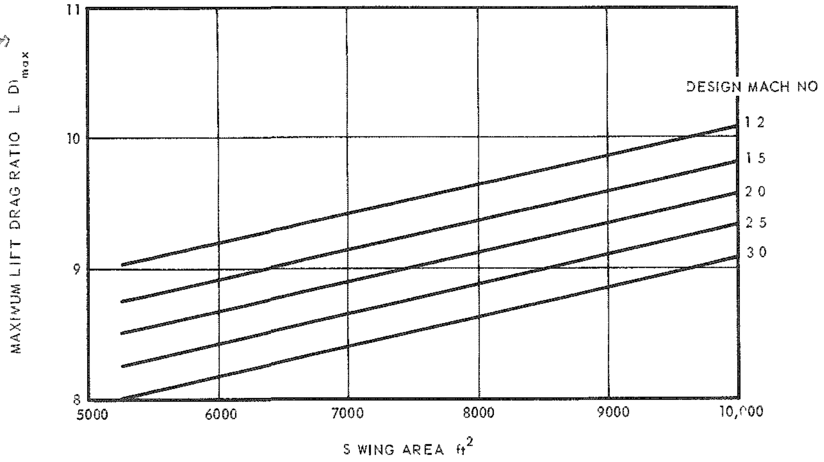

Fig. 6.4 – Maximum design lift-to-drag ratios

TABLE 6.3

WING LOADINGS FOR SUPERSONIC AIRPLANES

Design Speed, Mach No.	Wing Loading, lb/ft^2
1.5	100
2.0	94
2.5	87
3.0	79

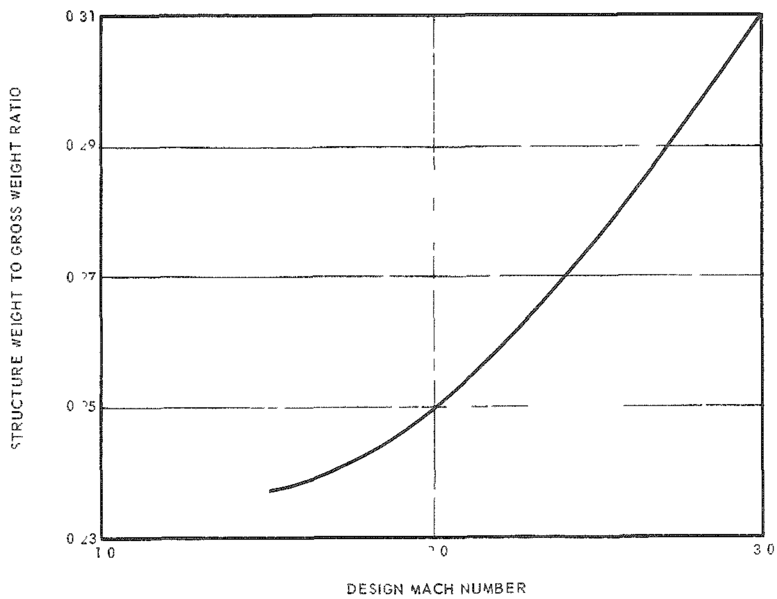

Fig. 6.5—Effect of design Mach number on airframe structure weight

TABLE 6.4

SUPERSONIC RECONNAISSANCE-STRIKE AIRPLANE PERFORMANCE WITH 50,000 POUND PAYLOAD

Power Plant	Number Of Power Plants	Gross Weight, lb	Sea Level Speed, Mach No.	Speed At Altitude Mach No.	Speed At Altitude Altitude, ft
XNJ140E	4	550,000	0.89	0.85	32,000
Advanced XNJ140E[a]	4	578,000	0.91	0.85	35,000
	5	654,000	0.9 - 0.95	1.5	35,000
XNJ140(P122C3)[b]	4	548,000	0.95	2.5	40,000
	5	645,000	0.95	2.7	40,000
A115D[c]	4	587,000	0.95	3.1	58,000

[a]Using advanced BeO reactor with 2000°F turbine inlet temperature - normal continuous.
[b]XNJ140 power plant with folded-flow reactor.
[c]Advanced reactor.

TABLE 6.5

SUPERSONIC RECONNAISSANCE-STRIKE AIRPLANE PERFORMANCE WITH 100,000 POUND PAYLOAD

Power Plant	Number Of Power Plants	Gross Weight, lb	Sea Level Speed, Mach No.	Speed At Altitude Mach No.	Speed At Altitude Altitude, ft
Advanced XNJ140E[a]	4	638,000	0.88	0.85	31,500
XNJ140(P122C3)[b]	4	580,000	0.9 - 0.95	1.2	25,000
	5	698,000	0.95	2.5	40,000
A115D[c]	4	660,000	0.95	3.0	55,000

[a]Using advanced BeO reactor with 2000°F turbine inlet temperature - normal continuous.
[b]XNJ140 power plant with folded-flow reactor.
[c]Advanced reactor.

The power plants used in the study included the XNJ140E with a BeO reactor operating at temperatures shown in section 4; the XNJ140E with an advanced BeO reactor operating at higher temperature levels; and an advanced model of the XNJ140, designated the P122C3, using the folded-flow reactor. In addition, the A115D variation of the folded-flow system and advanced XNJ140 configurations using a fast spectrum reactor were considered. These gave approximately equivalent performance. However, since the feasibility studies of the A115D had been carried into greater depth, only it is reported in the performance summary.

All of the power plants considered were assumed to have no chemical interburners and only enough chemical fuel on board for use in the takeoff afterburners. The cruise dose rate was set at 0.02 rem per hour at the subsonic loiter condition.

6.5 AIRBREATHING MISSILES

6.5.1 TURBOJET MISSILES

An application study was made in 1954 using turbojet propulsion for an unmanned guided missile to perform photographic reconnaissance missions. An illustration of such a missile, the ACA-8, is shown in Figure 6.6. The missile was studied at speeds up to Mach 2 at 40,000 feet on nuclear power alone and at speeds of Mach 2.5 at 60,000 feet with chemical reheat.

Several similar studies were performed in 1958 to evaluate the use of a nuclear turbojet in the Northrup SNARK missile at a speed of Mach 0.9 at 30,000 feet. The study indicated that nuclear propulsion was feasible for the SNARK for missions of several hundred hours duration using reactors developed in the HTRE program.

6.5.2 RAMJET MISSILES

Active ramjet missile studies started in 1953. Beryllium oxide reactors were used in most studies. Application studies were performed of high-altitude systems at speeds of Mach 4.25 at 90,000 feet and low-altitude systems at Mach 2.8. The ramjet demonstrated the greatest advantage at sea level because of the ability to penetrate enemy defenses at high supersonic speeds in the lower atmosphere. This was not feasible with chemical systems because of the extremely high rate of energy utilization. The application studies of low altitude systems resulted in establishing the feasibility of the beryllium oxide reactor for low level ramjet propulsion. This was instrumental in establishing the PLUTO program.

6.5.3 TURBO-RAMJET

A study was made of the functional feasibility and application to a manned supersonic aircraft of the A128 turbo-ramjet (Figure 6.7) intended for operation at Mach 4.5 at 70,000 feet altitude. The application associated with this system was the propulsion of a two-man crew, extended-range, supersonic bomber. The A128 power plant incorporated the use of a chemical turbojet cycle (J93-5 engine) with a nuclear afterburner. The chemical turbojet with nuclear afterburning provided thrust for that part of the flight regime that included both takeoff and boost to Mach 3 at 55,000 feet altitude. At this speed, the turbomachinery was bypassed, and the afterburner operated as a nuclear ramjet for the flight regime from Mach 3 at 55,000 feet to Mach 4.5 at the design altitude of 70,000 feet. The chemical J93-5 turbojet engines were assumed to operate along the engine operating limit line of Mach number versus altitude during the takeoff and boost portion of the flight regime.

The studies were terminated when the flight performance estimates indicated the A128 was deficient in thrust for the specific flight profile which was studied. It was considered potentially useful, however, for use in modified flight regimes or unmanned missiles.

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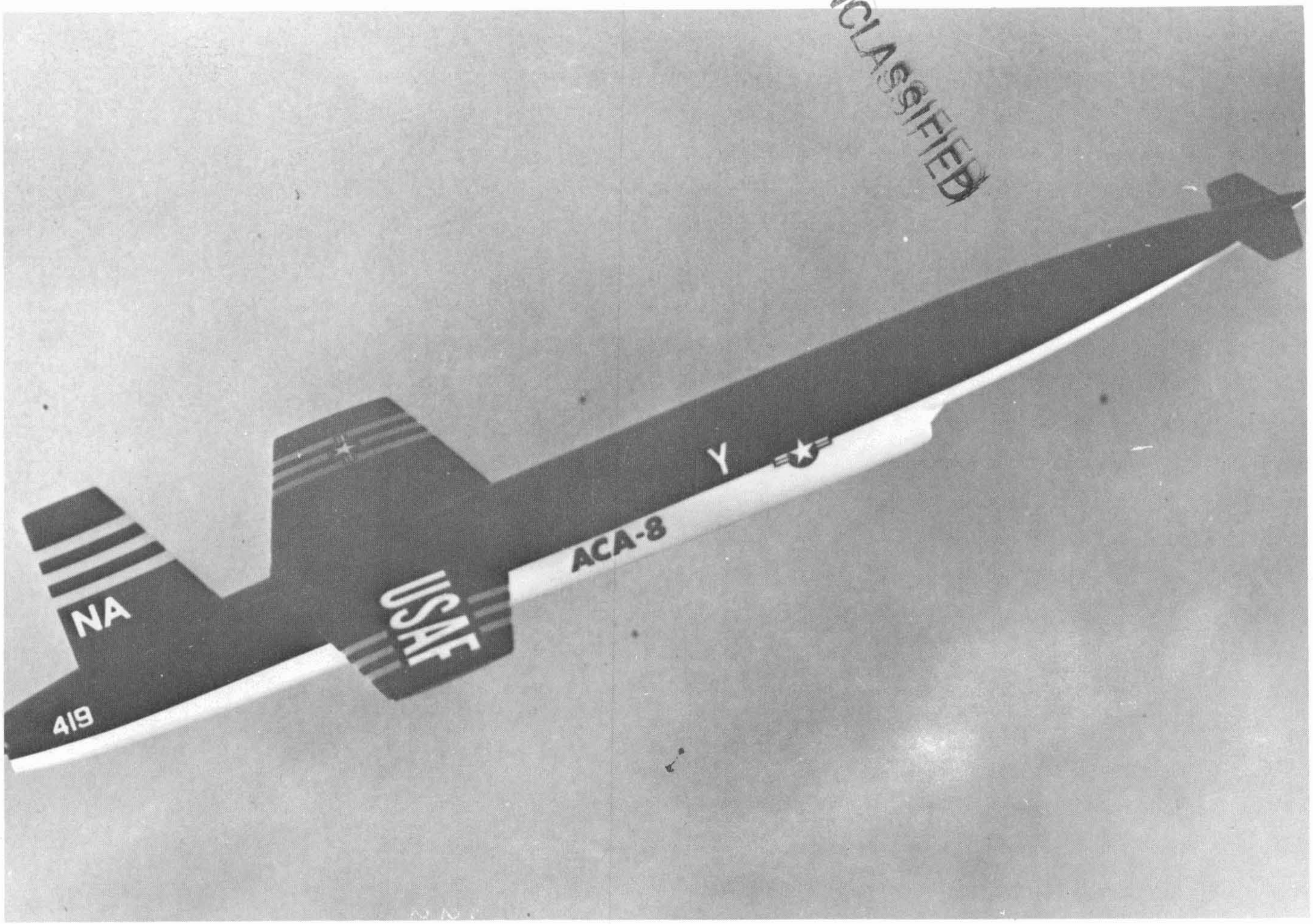

Fig. 6.6 – ACA-8 missile

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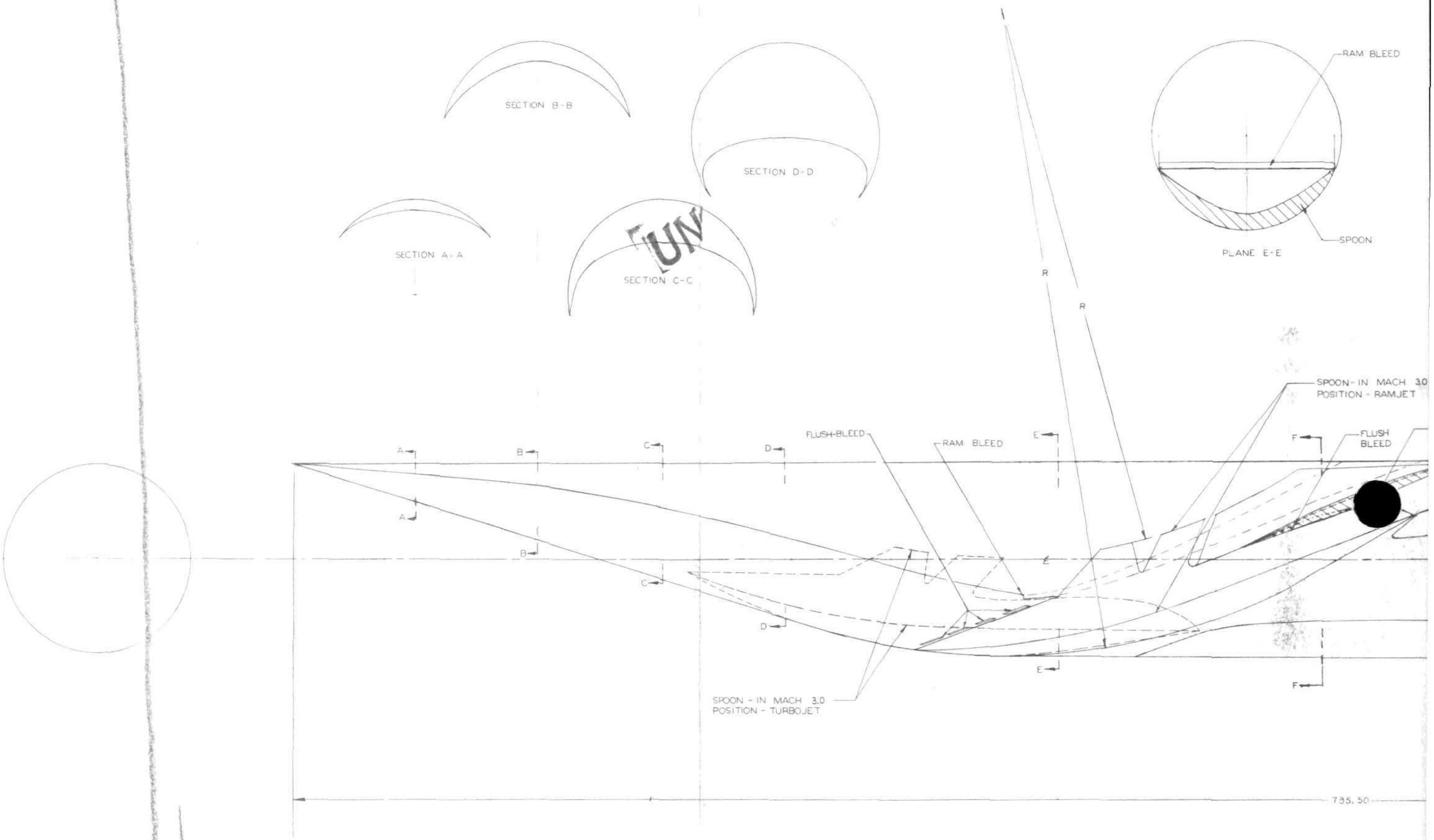
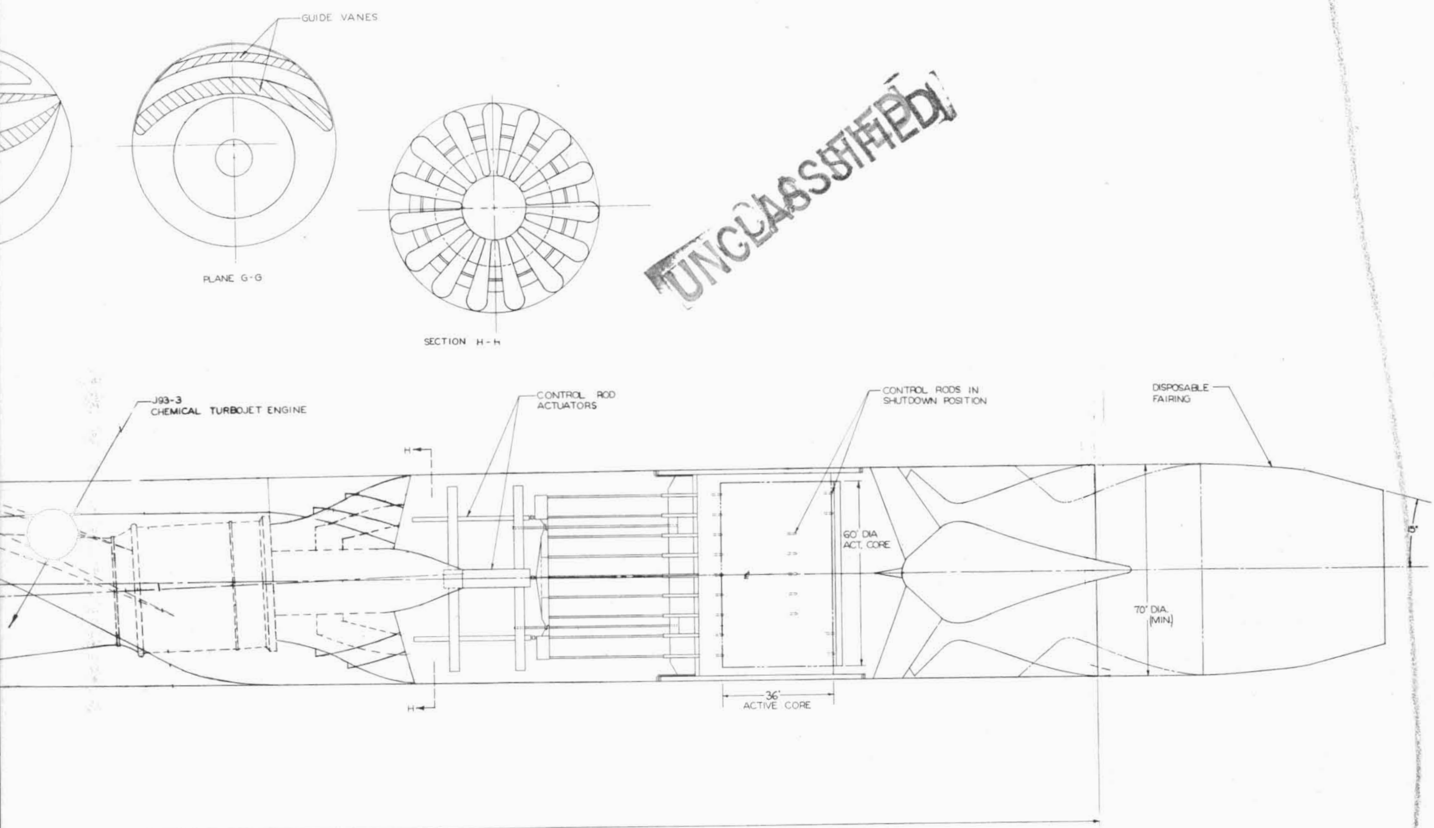


Fig. 6.7 – A128 power



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6.6 SPACE PROPULSION APPLICATIONS

A number of potential applications of nuclear space propulsion were studied. Effort was concentrated on high thrust heat transfer nuclear rockets. Early studies examined systems such as first stage boost of an ATLAS missile (section 5). Later, emphasis was placed on nuclear upper stages.

6.6.1 MANEUVERABLE ORBITAL VEHICLES

The use of a relatively small nuclear rocket in a manned maneuverable orbital vehicle appeared to have military potential. Such a vehicle would be launched into orbit and, when called upon to do so, would rendezvous with and possibly destroy other orbital vehicles. The vehicle had to be capable of changing both orbital altitude and plane and be able to modify its position in the new orbit to overtake another orbital vehicle. The propulsion system required a lower thrust than that required for boost into orbit but far greater than that required to change vehicle attitude. Nuclear rocket propulsion systems of relatively small size, possibly using small, fast spectrum reactors were considered promising for this application.

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6.6.2 INTERPLANETARY EXPLORATION

Consideration was also given to space vehicle concept using high-thrust propulsion units that could operate in the atmosphere of other planets as well as on earth. Such a system would use ramjets or turbo-ramjets in the planetary atmospheres and nuclear rockets in space. A concept of the main nuclear power plant for such a system is shown in Figure 6.8. Auxiliary atmosphere-breathing turbo-ramjet units of the type shown in Figure 6.9 would be attached to the main power plant and assembled with extra propellant tanks in an orbital

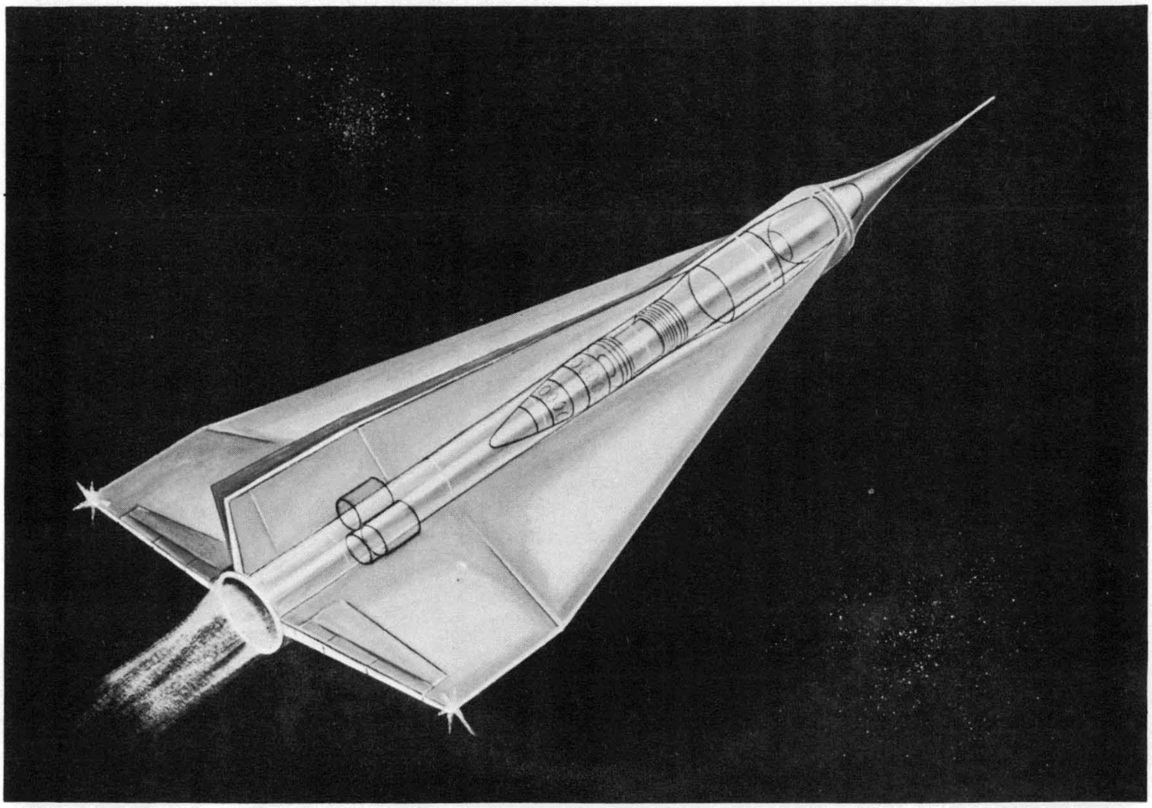

Fig. 6.8 – Main nuclear power plant for turbo-ram-rocket

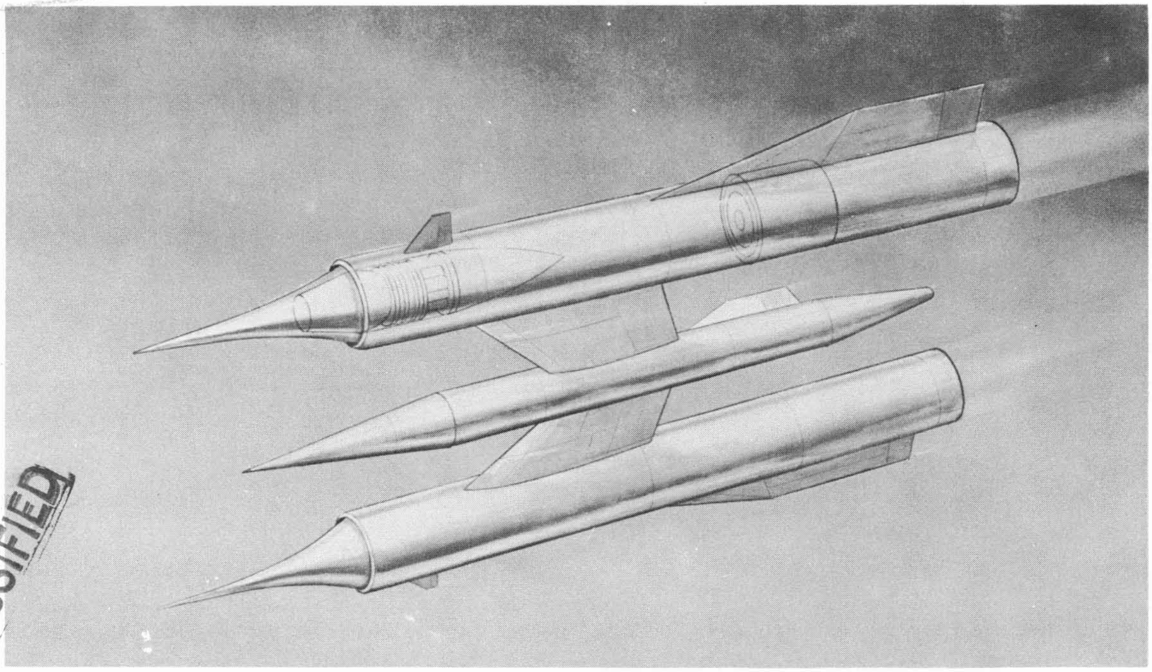

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Fig. 6.9 – Recoverable unmanned nuclear-powered turbo-ram booster

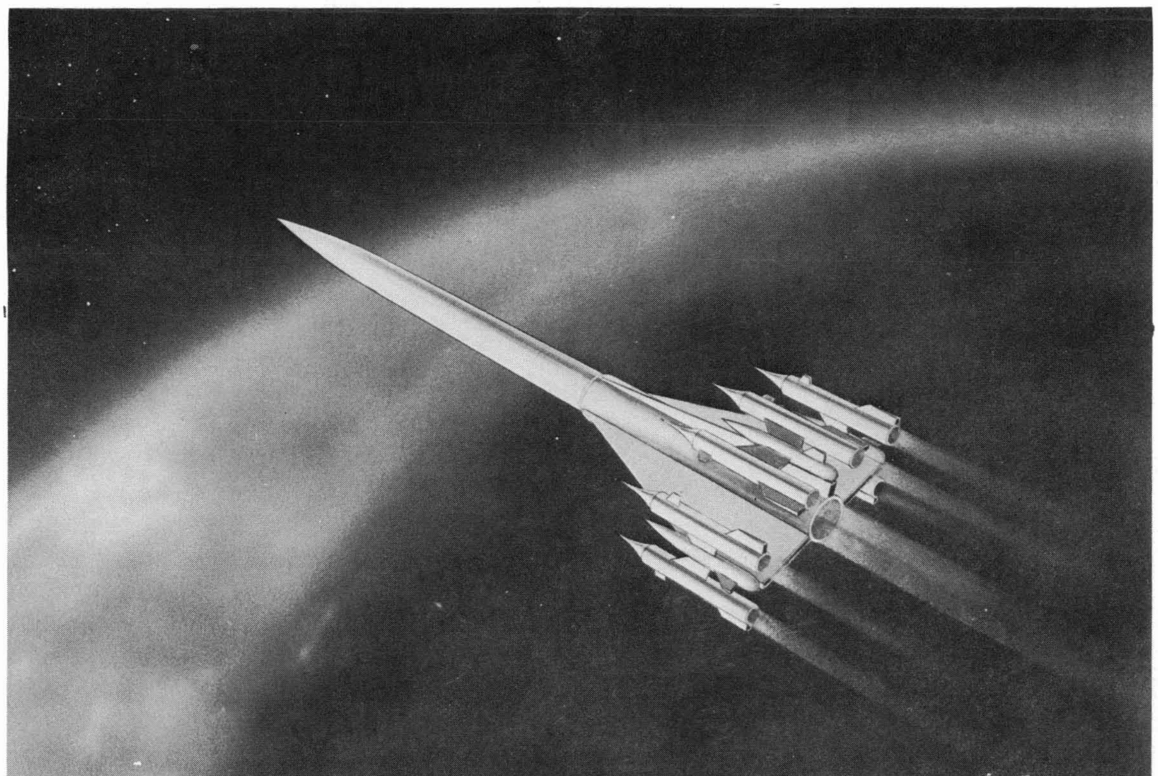

Fig. 6.10 – Nuclear-powered turbo-ram-rocket

rendezvous, as shown in Figure 6.10. Upon arriving in orbit at another planet, a turbo-ramjet booster would be detached for exploratory flight in the planetary atmosphere, after which it would return to the mother ship and then to earth.

For atmospheric exploration of other planets, nuclear propulsion has even greater advantages over chemical propulsion than it has on earth, since the fuel and possible the propellant for a chemical exploration vehicle would have to be transported from earth, thus consuming transport power and imposing severe payload limitations. The turbo-ramjet units could double as air-breathing boosters in the earth's atmosphere rather than relying entirely on rocket boost.

Materials were under development in the GE-ANPD program which were potentially insensitive to corrosion at relatively high temperatures in almost any probable planetary atmosphere. Mission analysis had not been performed to the point that system feasibility or firm propulsion system requirements had been established.

6.7 DERIVATIVE APPLICATIONS OF AIRCRAFT PROPULSION SYSTEM TECHNOLOGY

Several derivatives of the GE-ANPD nuclear turbomachinery technology were studied briefly in the program. These included applications of gas-cooled reactors to helicopters, ground-effects machines, hydrofoil propulsion, portable electric power plants, airship propulsion, and marine propulsion. Generally speaking, the studies indicated that application of gas-cooled nuclear power to these systems would be feasible if a need were established.

The possible use of closed-gas-cycle power packages of the AC601 type (section 5) for later-generation naval propulsion is illustrated in Figure 6.11. The power units are mounted as pods in accordance with aircraft practice. Ocean water provides the bulk shielding, although some heavy shielding is located close to the reactor. Installations of this type can potentially be provided at weights in the vicinity of 10 pounds per shaft horsepower. Inboard installations can also be made with weights which are substantially lower than present-generation naval propulsion units.

Additional information on the GE-ANPD application studies is contained in APEX-910, "Aircraft Nuclear Propulsion Application Studies," of this Report.

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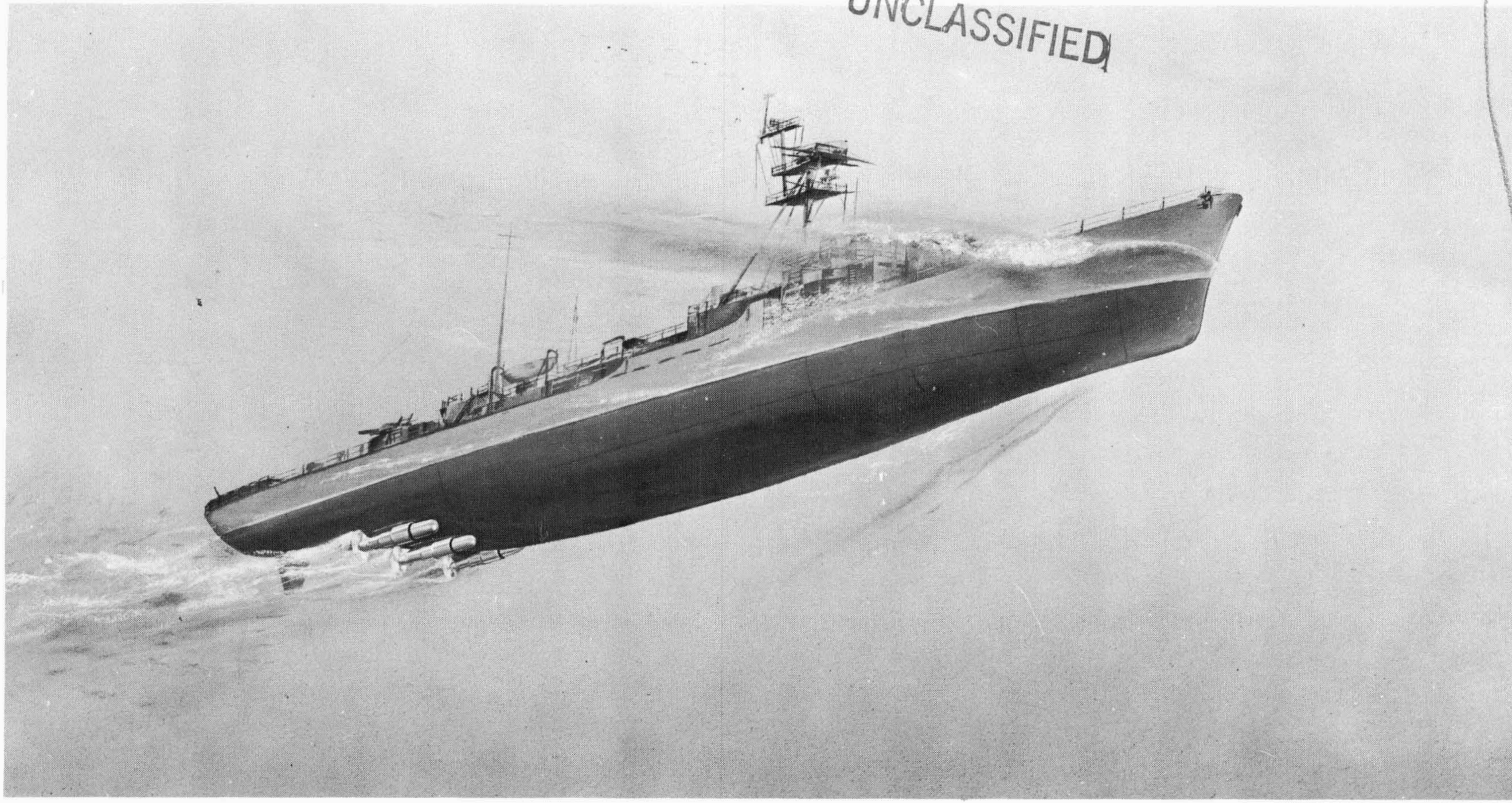

Fig. 6.11 – Installation of AC601 power package on destroyer escort

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7. BASIC TECHNOLOGICAL DEVELOPMENTS

The requirements of an aircraft nuclear propulsion system imposed conditions more severe than any previously encountered by nuclear designers. Entirely new technologies were required, in addition to major advances in many existing technical fields, to achieve the necessary high performance within the size and weight limitations of a practicable aircraft power plant. A vast amount of the total effort of the Aircraft Nuclear Propulsion program was channeled into the development of suitable materials, analytical techniques, manufacturing procedures, etc. A brief summary of some of these efforts is contained in this section. The companion volumes of this Report, referenced at the appropriate places, present the information in much greater detail.

7.1 ENGINEERING ANALYSIS AND EXPERIMENTAL EVALUATION

The design of aircraft nuclear reactor systems followed a conventional iterative procedure of preanalysis, design, reanalysis, and redesign, until performance requirements and design criteria were met within acceptable materials temperature limitations.

The design that existed at any point in this process was in the form of a mechanical configuration drawing and specification defining the arrangement, materials composition, and dimensions of the components and air passages. The purpose of the analysis was to predict the temperature, airflows, etc., that would be observed if the system were built to the design specification and operated.

The primary purpose of the nuclear analysis was to predict the heat generation rates resulting from neutrons and gamma interactions within the uranium and other reactor and shield materials. The nuclear analysis also determined whether or not the specified quantity and distribution of uranium would be sufficient to provide an adequate margin of criticality. External radiation levels were also predicted.

The relative size and shape of the various air passages in the design effectively predetermined the airflow distribution and pressure drops which would occur during the operation. The airflow conditions and local heat generation rates combined to establish the materials temperatures and consequently the air temperatures that would exist. The aerothermal analysis predicted what these airflows and temperatures would be. The aerothermodynamic behavior of the system and the performance of the resulting propulsion system could then be determined.

The vibrations and stress analysis predicted materials stress and strain due to mechanical and aerodynamic loads on system components and temperature differentials within the materials.

The most efficient design and the minimum reactor and shield size and weight were achieved when the analysis indicated that each material would operate at a uniformly high temperature and stress level with an adequate safety margin. To achieve this objective it was necessary to flatten gross and fine radial power distributions and temperatures within the reactor by local variations in uranium loadings, flux levels, materials thickness, and the location and size of air passages. Longitudinal flattening was also desirable but less important. Practical considerations, such as the component sizes, shapes, and compositions which can be economically manufactured, limited the ability to achieve completely flat conditions. Nevertheless, the achievement of the most efficient feasible design was of exceptional importance in aircraft reactor and shield design because of the combined requirements of high temperature and low weight. Consequently, the analysis methods that were developed were unusually rigorous. Computer programs were developed that were capable of handling a wide variation of input conditions and a large number of data points. Some of the more significant methods used in the analyses are described below together with references to those volumes of this report where greater detail is available.

7.1.1 REACTOR NUCLEAR PHYSICS

Critical Mass Calculations

Calculations of critical mass were carried out by means of synthesized one-dimensional diffusion theory calculations. The synthesis, in terms of end and side reflector savings, was completely mechanized. This feature was particularly valuable in preliminary design calculations, where many proposed and alternative configurations could be analyzed quickly with maximum accuracy. It was also useful in providing rapid evaluation of proposed changes later in the design sequence.

Heating Rate Calculations

Fission power distributions were computed using one-dimensional multigroup codes, with the transverse dimension represented in terms of the matched savings from the previous calculations. Standard multigroup methods were applicable for isothermal or nearly isothermal reactor designs in which the core and near-core regions were approximately the same temperature. For systems with large temperature variations, thermalization and rethermalization of neutrons were considered, since the power distribution near region interfaces is dependent on the thermal and near-thermal neutron spectrum. The nuclear data system included thermalization (up and down scattering) cross sections for all materials for four energy groups from 0.0253 to 1 electron volt. These cross sections included crystalline binding effects for the important moderator materials and were computed from a monatomic gas model for other materials. The convergence problems generally associated with this type of calculation were avoided, using a unique solution in which a direct inversion of the energy matrix was used for the thermalization region. Azimuthal fission power variations, caused by reflector poison variations, were calculated using two-dimensional (R, θ) codes. When necessary, three-dimensional distributions were generated by synthesizing various (R, θ) calculations with one-dimensional calculations along a desired axis. The one-dimensional calculations also provided automatic calculation of gamma ray source distributions in both space and energy.

Secondary heating distributions due to gamma scattering and absorption were calculated using one or two space dimension transport theory codes or Monte Carlo codes as required. Secondary heating analysis was particularly important since structural component cooling requirements must be considered early in the design sequence.

Control Evaluation

Evaluation of control configurations was determined using cross-section data and methods evaluated against danger-coefficient measurements and a large series of control rod experiments. The control requirements due to temperature were determined by analysis and then compared to elevated temperature critical experiments and operating power reactor experience. Control analysis methods generally involved representation of the absorber materials in a form suitable for diffusion theory calculations. Where convenient, this was done in terms of energy-dependent transmission fractions based on higher order transport theory calculations. The control material could also be directly represented as a reactor region, with the energy-dependent diffusion coefficient adjusted to preserve absorption rates as compared to transport theory calculations.

Reactor Kinetics

Reactor kinetic and runaway analyses were accomplished by one of two possible appropriate reactor shutdown models. The first model assumed a shutdown mechanism that was, at any time, proportional to the energy generated in the runaway. This model was appropriate for: (1) "slow" runaways in reactors with a prompt negative temperature coefficient, or (2) runaways in which fuel vaporization and direct expulsion is possible, e.g., critical experiments employing fuel foils. The second model assumed a destructive reactor runaway with shutdown due to vaporization of fuel or fuel carrier, assuming vapor flow through a simple convergent nozzle. Both models provided a six-delayed-neutron-group representation of the reactor kinetics and essentially arbitrary representation of reactivity insertion in terms of either constant, linear, or quadratic time functions.

Experimental Reactor Physics

Critical experiments were performed on nuclear mockups of the HTRE, XMA-1, and XNJ140E reactors, and a number of proposed configurations which did not reach final design or power testing. A limited number of experiments was run at elevated temperatures up to 1000°F. The HTRE operations provided further verification of nuclear characteristics at full temperatures. In the area of basic reactor physics, approximately 30 different reactor configurations were investigated in a flexible critical assembly.

The inner portion of the reactor shield was included in a number of reactor mockups. This made it possible to determine reactor-shield nuclear interactions, shield heating due to gamma absorption, and radiation levels which could be extrapolated to the outer shield surface.

A photograph of a typical critical experiment assembly, the XNJ140E-1 mockup, is shown in Figure 4. 27. Progressing radially from the center, the regions shown represent the shaft void, shaft and shaft island material, fueled core, reflector, pressure-pad spring and pressure-shell materials, and shield. Shown surrounding the core in the reflector are the 48 rod positions which represent the control and safety rod positions in the design core.

A detailed-measurement sector was located at the top of the assembly. In the core portion of this sector were fueled BeO tubes that closely simulated the design-core tubes. The remainder of the core was correct in composition but only approximated the individual fuel-tube geometry. In the shield portion of the sector, the matrix tubes were replaced by a can containing LiH shield material. The front shield region consisted of a mockup of the tapered borated stainless steel region nearest the core, backed up with approximately 8 inches of beryllium. The rear shield and grid plate were represented by appropriate amounts of boral, stainless steel, and beryllium inserted to approximate the correct shapes and locations. This mockup was also extended to a thickness of about 8 inches.

Because the design of aircraft reactors was carried on by a combined analytical-experimental approach, great emphasis was laid on the measurement of fission-power distributions. During design iterations, fission-power distributions were often measured at as many as 7000 points. This measurement sequence was repeated for the various conditions of power plant operation, i.e., control rods out, simulation of xenon buildup, or different phases of the design iteration. An automated 10-channel proportional counting system was used for reduction and reporting of this information.

Fission fragment catcher foils and uranium wire were used for measuring power, or uranium-sensed neutron flux. The usual flux-detector materials, such as gold, indium, copper, and uranium, were used either as foils or wires to measure thermal neutron flux distributions. Cross calibration with the standard flux in the sigma pile provided absolute flux values. Techniques used for reactivity measurements were primarily (1) positive-period evaluation of a control rod or other reactor components, and (2) comparison of differences in positions of the control rods with the reactor critical under varying conditions. Gamma ray heating was measured with calorimeters and with Bragg-Gray detectors whose walls were made of the appropriate reactor and shield materials. Measurements of the (n, α) reaction rates for use in determining heat production in the borated outer regions of the reactor assembly were made using a miniaturized BF_3 counter or a solid-state detector with the junction faced by a layer of boron.

Photographic film, air-equivalent, carbon-wall ionization chambers, and Hurst-type dosimeters were used to measure biological dose due to fast neutrons and gamma rays for use in shield evaluation. Determinations of gamma spectra were made by the use of a thallium-activated sodium iodide crystal. Lithium-loaded Ilford emulsion plates were used to obtain fast neutron spectrum.

APEX-918, "Reactor and Shield Physics," of this Report contains a more complete description of GE-ANPD reactor nuclear physics.

7.1.2 INTERNAL AERODYNAMICS AND THERMODYNAMICS

Airflow and Temperature Distributions in Reactor Cores

The prediction of materials and air temperatures was basically an iterative process in which, as a first step, airflow rates and pressure drops were calculated entirely on the basis of air passage geometry and inlet and outlet conditions. The materials and air temperatures were then predicted using the calculated airflow rates and the heating rates which had been previously established by nuclear analysis. The airflows were then recalculated to include the temperature effects on airflow, and the process was repeated. In practice, this analysis procedure was mechanized.

The expected fuel ring temperatures in reactors using concentric ring metallic fuel elements were calculated by the Heated Annuli and Ansector computer programs. These programs predicted the local airflow and pressure distribution, and the local surface temperature and air temperatures for an assumed total airflow and inlet total pressure distribution.

Another computer program was used to predict the temperature distribution in the core of a homogeneous reactor such as that in the XNJ140E, assuming an arbitrary assignment of inlet air pressure distribution across the face of the reactor, pressure distribution across the exit face of the reactor, radial and longitudinal power profiles, and radial variation of coolant channel diameter.

The Compressible Flow Network Program, called COMNET, was another method developed that was applicable to a more generalized reactor configuration. This program could assume any arbitrary collection of air passage junctions and branches and could predict the pressure drop and flow rate that would occur in each of the branches and through the overall system. The program would also compute air passage surface temperatures.

Transient Temperature Patterns

The Transient Heat Transfer Program would calculate either steady-state or transient temperatures for any arbitrary system. This was a nodal point program in which the component to be analyzed was divided into a number of connected nodes. The program would accommodate internal conduction, surface convection, surface thermal radiation, internal heat generation, and heat capacity. The program would predict time-dependent temperature variations due to power demands, reactor scrams, etc. Fuel element temperature changes were predicted as well as changes in other components such as reflector, structure, control rods, insulation, etc.

Development of Basic Aerothermodynamic Data

Basic aerothermodynamic data, such as heat transfer coefficients, was taken from the literature wherever possible. Data was developed experimentally where literature data was not available or applicable.

Experimental Aerothermodynamics

Because of the complex flow geometries, involving many parallel air passages, the analytical procedures were heavily dependent on experimental verification. Inlet and outlet conditions were especially difficult to predict analytically. Therefore, full-scale and reduced-scale models were used to determine airflow distributions and pressure drops within inlet and exit ducts, across the faces of the reactor, and within the reactor core, reflector, controls, and structure. Many of these tests were run under cold airflow conditions, while others utilized air which had been heated electrically or with a gas flame. In many cases, the reactor elements were electrically heated, thus simulating nuclear heat addition to the air. In-pile tests, in which the fuel elements were heated by the fission process and cooled by air, provided a direct verification of airflows, pressure drops, and temperatures. An ultimate evaluation was obtained in the HTRE experiments which were extensively instrumented to obtain aerodynamic and thermodynamic data under full power, airflow, and temperature conditions.

A more thorough discussion of the GE-ANPD work in this field is presented in APEX-919, "Aerothermodynamics," of this Report.

7.1.3 SHIELD NUCLEAR PHYSICS

Point Kernel Shield Calculations

The principal method used in predicting the radiation levels external to GE-ANPD reactor shields involved a point kernel approach in which the reactor was divided into a large number of small regions, each of which was treated as a point source of radiation. Machine programs were used to calculate the attenuation from each contributing point source and to add the individual contributions to obtain the total radiation level at the receiver point. These programs were based on the assumption that radiation attenuation depends only on the quantity of each material encountered along a straight-line path between a source point and a receiver point. The attenuation was the product of a material attenuation function and a geometrical attenuation function. Material attenuation functions were established by fitting data obtained by either theoretical or experimental methods.

A modification of a material attenuation function suggested by R. D. Albert and T. A. Welton was used for computing fast neutron dose rates from fission sources in mixtures of hydrogenous and heavy shield materials. This theoretical expression was based on the assumption that a single neutron collision in hydrogen was tantamount to absorption, and was derived by using empirical approximations of the hydrogen cross section and the fission neutron spectrum. The parameters of the function were fitted to measured dose rate data in water in order to correct for buildup of scattered neutrons.

Neutron dose rate calculations in nonhydrogenous materials also were made with the Albert-Welton function, but the parameters had to be fitted to penetration data computed for those materials by more exact methods or measured in bulk experiments. Such data were available for numerous materials, including beryllium, beryllium oxide, carbon, and lithium hydride. Material attenuation functions in the form of bivariant polynominals were used for computing neutron differential number spectra. Coefficients of these functions were fitted to computed penetration data for the materials mentioned above.

The point kernel method was well suited for determining fast neutron dose levels but did not accurately depict the intensity of lower energy neutrons which, upon being absorbed, produced both secondary gammas and heat within the reactor shield. For direct calculation of neutron absorption and energy deposition in the shield, various combinations of point kernel and multigroup diffusion methods and programs were used. The diffusion results usually were normalized at each point by the ratio of fast neutron dose rate computed by the point kernel method to that computed by the diffusion method. Two slowing-down models were used in the diffusion calculations. A Coveyou-Macauley slowing-down model was used for low atomic weight, nonabsorbing nuclei in a region where the perpendicular losses are small, and a modified-age model was used otherwise. In an alternative method, the uncollided flux per unit lethargy throughout a shield region was computed by a point kernel program and used to establish first-scattered sources throughout the region for a multigroup diffusion calculation. This method was more tedious than the other, but it gave absolute rather than normalized answers.

An exponential attenuation function multiplied by a buildup factor was used for computing gamma ray dose and energy absorption rates. The buildup factors were computed for single materials from cubic polynominals fitted to buildup factors that were computed by a moments solution of the Boltzmann transport equation. Empirical expressions established by Monte Carlo calculations were used to combine buildup factors for two materials. Bivariant polynominals were also used to compute gamma ray energy spectra.

Single-scattering programs were developed for approximate calculations of air-scattered neutrons and gamma rays. These were used for crew-shield calculations and to determine the increased radiation level caused by air scattering into regions where the direct radiation was low.

Monte Carlo Shield Calculation Methods

The point kernel programs were phenomenological or empirical methods based largely on bulk shielding experiments. Because of the many limitations of this approach, the rapid improvement of digital computers, and the expanding knowledge of basic nuclear data, development of Monte Carlo codes for use in shield analyses were also developed.

The transport of neutrons and gamma rays in a shield is an example of a physical process, probabilistic in nature, which can be conveniently handled by the Monte Carlo method. Average quantities for neutrons and gamma rays, such as current, absorption, flux, heating, dose rate, and leakage, were determined in this approach by tracing individual histories chosen from appropriate distributions for a sufficiently large number of histories. Monte Carlo methods were potentially better able to handle complex geometries and to provide more accurate data over the entire radiation spectrum. Five Monte Carlo programs were developed for shield analysis. These include a specialized code for analysis of reactor-shield assemblies; two general purpose codes for analysis of reactor or crew shields; a gamma ray air-scattering code; and a neutron air-scattering code.

In the specialized reactor-shield Monte Carlo program, the shield portion of a reactor-shield assembly was described by regions formed by rotation of a class of simply connected quadrilaterals about the reactor-shield axis. Each region was composed of a homogeneous mixture of the basic materials in the region. Output available from the program included the energy deposition in each shield region due to neutron and gamma ray reactions, the energy-angle leakage distributions for neutrons and gamma rays for a point source equivalent to the assembly, and the neutron and gamma ray currents across specified boundaries.

Although the two general purpose codes provided similar output data, the source-shield geometrical description was accomplished by describing each boundary of a region by a second degree equation which involves the three space variables. Most desired geometrical surfaces could be approximated by proper specification of the coefficients of the equation.

The Monte Carlo air-scattering programs calculated the energy spectrum and angular distribution of gamma rays and neutrons at a point detector due to single and multiple scattering in air from a monoenergetic, monodirectional point source.

Whenever neutron inelastic scattering (n, γ) and fission events were treated in the neutron portion of any of these programs, parameters of secondary gamma rays were generated and stored on tape for subsequent analysis.

Shield Nuclear Experiments

Many shielding experiments were performed using partial and complete shield mockups to develop or verify calculation methods and to provide empirical data which could be applied directly to the shield design. Other more basic experiments investigated duct leakage, secondary gamma rays, production material attenuation with and without hydrogenous backing, attenuation through combinations of materials, nuclear heating, and many other special problems.

These experiments were performed in several facilities, including the Bulk Shielding Facility (BSF), Lid Tank Shielding Facility, and Tower Shielding Facility (TSF) at Oak Ridge National Laboratory. Also used were the Outside Test Tank (OTT) at Convair, Fort Worth, Texas, the Source Plate Facility at Battelle Memorial Institute and the Shield Test Pool Facility (STF) operated by General Electric at the Idaho Test Station.

In addition to special shield experiments, shield measurements were regularly included as a part of the HTRE critical experiments and tests. The HTRE-3 test was particularly valuable because a full-scale aircraft reactor and shield configuration was used for the first time.

The Lid Tank Shielding Facility was used periodically over a span of several years for measuring thermal neutron fluxes and fast neutron and gamma ray dose rates after penetration through various arrays of liquid and solid shield materials. Some of these arrays were designed as partial mockups of design shields, and others were designed to determine radiation attentuation properties of shield materials and to investigate the importance of secondary gamma ray sources. Several partial mockups of annular and helical ducting systems were also tested to determine leakage properties. Several mockups were also tested in the BSF including partial mockups of ducting systems and a unit shield.

Numerous porous shield plugs were tested in the OTT. The OTT was also used for extensive in-air measurements of neutron and gamma ray penetration of several multiple material shield combinations. Many of these measurements provided penetration data for shield materials without hydrogenous backing. The Convair Ground Test Reactor (GTR) served as the radiation source in these measurements.

A complete mockup of the lead and water shield for the P-1 power plant reactor was tested at the BSF and the TSF. Measurements of thermal neutron flux and fast neutron and gamma ray dose rates were made in water surrounding the shield at the BSF. The TSF test included similar measurements in air in the vicinity of the shield, in a water-filled detector tank, and in a crew-shield mockup. Later, more fundamental experiments, the 2 Pi Shield experiments, were performed in the TSF to evaluate air scattering and crew-shield penetration, with particular emphasis on ground scattering.

In order to learn more about some of the physical phenomena associated with reactor-shield assemblies composed of depleted uranium and lithium hydride and employing realistic shield design geometries, a mockup called the Solid Shield Mockup (SS-1) was constructed at Oak Ridge and tested at Convair, using the GTR. This mockup closely simulated characteristics incorporated in the XNJ140E shield design.

APEX-918, "Reactor and Shield Physics," of this Report discusses GE-ANPD shield nuclear physics in greater detail.

7.1.4 MECHANICS

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Power Plant Structural Analysis

Whether nuclear or chemical, aircraft turbojet engines are complex mechanical structures which must be lightweight, reliable, and operate at high temperatures. An operating life of approximately 1000 hours is acceptable between major overhauls, at which time mechanical components may be replaced if necessary. This is a shorter operating life than required in stationary or maritime installations. However, the life requirement does not differ greatly from other high-performance lightweight power units, such as automotive engines which normally require major service between 1000 and 2000 hours of operation.

The basic structure of a turbojet engine consists of the compressor and turbine casings and interconnecting pressure shell, together with the bearing support frames carrying the engine rotor. All other components, such as burners and accessories, are supported by the pressure housing or the bearing support frames. The engine is normally attached to the airframe at mounting points connected to the bearing support frames. The GE-ANPD power plants including the XNJ140E nuclear turbojet were designed in accordance with this practice. The structure of the nuclear power plants was analyzed using conventional aircraft engine analysis methods. The entire power plant, including both the turbomachinery and nuclear components, was designed to conventional engine standards, specifying the G loads in the vertical direction (both up and down), horizontal loads, maneuver loads, and crash loads. The principal difference in the nuclear and chemical systems from the overall structural viewpoint was the presence of the high-mass nuclear heat source. However, since this load was carried directly to the engine mounting points, the component structure of the turbomachinery was relatively conventional.

Reactor and Shield Structural Analysis

Within the nuclear reactor the fuel elements operated at very high temperatures toward the rear of the core, and at lower temperatures in the front region. Structural components were operated at lower temperatures in order to provide sufficient strength to support the reactor mass under both horizontal and vertical G-load conditions. The predicted materials temperature differences for the XNJ140E are graphically illustrated in Figure 4.16. The structural analysis took into account the interaction of the components at their various temperatures under both steady-state and transient conditions of temperature, aerodynamic loading, and G loading. The reactor aft support and aft shield ducting to the turbine represented difficult structural problems because of the high operating temperatures. Special analysis methods were developed and utilized to predict the stress levels for both steady-state and transient operation.

Thermal Stress in Ceramic Fuel Elements

Stress levels due to temperature differentials within construction materials in a nuclear system constitute a more severe problem than in a chemical system because heat is being generated internally in the fuel elements and, to a lesser extent, in other components due to the absorption of nuclear radiation. A high power density in the fuel elements, for example, produces a high temperature differential between the center of the material and the surface. Thermal stress is a particularly critical problem in the design of ceramic reactor fuel elements.

The assumptions made in thermal stress analyses were the usual assumptions made in applying elasticity theory, together with the assumptions of a Maxwell-type model for representing creep behavior.

The ceramic fuel tube geometry represented a highly complex three-dimensional problem in thermoelasticity for which no completely rigorous analytical treatment was available. The differential equations of thermoelasticity are linear, and permit using the principle of superposition in the solution of thermoelastic problems.

There were three significant sources of thermal stresses in the XNJ140E-1 fuel elements:

1. Radial temperature gradients
2. Longitudinal temperature gradients
3. Circumferential temperature scalloping

Stresses due to the radial temperature gradient were the primary thermal stresses in the fuel tubes. The approach used to analyze these stresses was first to solve the problem for an equivalent circular tube and then to modify the solution with suitable correction factors for the actual tube with hexagonal outer boundary. Correction factors were obtained from a finite-difference computer solution of the thermoelastic equations. The computer program solved the plane-stress and the plane-strain thermoelastic problem (both with internal heat generation) for any arbitrarily shaped two-dimensional region. Because of its generality, and also because of the large number of mesh points required in the solution of the hexagonal tube geometry, the most economical way to utilize this program was to conduct a parametric study to determine correction factors which could be used in conjunction with the equivalent circular tube solution. The correction factors were then expressed as functions of the parameter, W/D_i, where W is the across-flats dimension of the hexagonal tube and D_i is the inner diameter of the tube.

Since the solution included the case of plane-strain (infinite length tube), end correction factors were applied in order to obtain the maximum stresses at the ends of the tube. No exact solution was available for the end stresses in hexagonal tubes, and an approximate solution, based on the solution for a thin-walled circular tube, was developed.

Thermal stresses due to axial temperature gradients are proportional to the second derivative of the temperature with respect to the axial coordinate (d^2T/dx^2). Under reactor operating conditions the value of d^2T/dx^2 was always so small that the resulting stresses were neglected.

Stresses due to circumferential temperature scalloping could result from both internal and external sources. Within an individual tube, the effects of the hexagonal outer surface of the tube and the lack of homogeneity of material properties were considered. Thermal stresses induced by the lack of homogeneity of materials were negligible. Asymmetry of reactor configuration, neutron flux distribution, and temperature distributions were considered as external sources of thermal stresses. The most probable effects of asymmetry external to an individual tube were linear temperature gradients across the diameter of the tube. These gradients would cause the tube to bow if it were not externally restrained.

Thermal stresses induced by the differential thermal expansion between the clad and the matrix were calculated by allowing independent free thermal expansions of the clad and the matrix. Surface tractions (at the clad-matrix interface) required to restore continuity at the clad-matrix interface then were determined, based on thick-cylinder equations. The resulting stresses were then calculated and superimposed on the thermal stresses due to radial temperature gradients in each of the components.

The major mechanical loadings on the ceramic fuel elements were due to the radial pressure produced by the springs of the radial support system. The induced beam loading resulted from a fuel tube acting as a simple beam supported at each end with a concentrated load applied between the ends and tending to restrain or force deflections. These deflections could be caused by the manufacturing tolerances for camber and external surface dimensions, thermal camber, and core barrelling. The loading condition produced bending stresses in the axial direction. Ring loading resulted from the pressure concentrations acting on opposite faces of the hexagonal tube. This loading condition also produced tangential bending stresses normal to the tube axis. Tensile stresses were of primary concern in ceramic materials since the compressive strength was much greater than the tensile strength.

APEX-908, "XNJ140E Nuclear Turbojet," and APEX-914, "Ceramic Reactor Materials," of this Report contain further information on thermal stresses in ceramic fuel elements.

Experimental Mechanics

The aircraft engine industry places heavy reliance on experimental mechanical development. This procedure was also followed with the nuclear systems.

Fuel element development relied heavily on experimental methods. Structural evaluations of metallic and ceramic fuel elements were made in gas dynamics test equipment. Reactor conditions of temperature and airflow were simulated with electrical or combustion heat. These tests were operated both to obtain completely empirical structural designs for fuel elements as well as to develop basic data which could be used in analytical design. An extensive in-pile test program was also used to evaluate mechanical behavior of the fuel elements.

The ceramic fuel elements were subjected to extensive testing to verify and further develop the analytical method of predicting fuel element stress levels, particularly under transient conditions. In addition to tests on individual fuel elements and small assemblies, full-diameter arrays of ceramic tubes were subjected to vibrational and impact tests to evaluate the structural integrity of ceramic reactor cores. Vibration frequencies used included those induced by the operation of turbojet engines. Tests of impact loads covered the expected G-load range. A photograph of such a test specimen is shown in Figure 4. 26.

Reactor components such as radial support springs, tube sheets, shield components, and ducting were also evaluated experimentally. Relaxation of materials at high temperatures was experimentally determined. This was an especially critical problem with the radial support springs used in the ceramic reactor design. Similar tests were performed on shield and ducting components, particularly those with complex shapes operating at high temperatures.

APEX-920, "Applied Mechanics," of this Report contains greater detail on this effort. APEX-902 through APEX-908 of this Report, describing the HTRE test assemblies and XMA-1 and XNJ140E power plants, also contain further information on the mechanical analysis and testing techniques developed by GE-ANPD.

7.2 MATERIALS DEVELOPMENT

7.2.1 BERYLLIUM OXIDE FUEL ELEMENTS

Because of the potential of ceramic materials for use in high-temperature gas-cooled reactors, an intensive program was performed to develop a ceramic fuel element to operate for 1000 hours in air at temperatures up to 3500°F.

The most significant accomplishments were achieved with beryllia-based fuel elements. Investigations were also made with zirconium and niobium beryllides. In addition to these potential self-moderating materials, studies were performed on fuel systems of urania-thoria, urania-yttria, plutonia-thoria, and plutonia-yttria.

Beryllia-based fuel elements were irradiation-tested at temperatures above 3000°F and yttria-urania bodies were heated to 3300°F in air for hundreds of hours with essentially no loss of uranium.

Complex fabrication techniques were developed for the coextrusion of tubular or hexagonal fuel elements with a 0.003-inch cladding on the base material. Also, vapor-deposition techniques were developed to deposit dense coatings of alumina and stabilized zirconia on the base material.

Fuel Stabilization

The initial work on BeO-UO_2 fuel elements showed that, when heated in air, this material was subject to large volume changes caused by the oxidation of the UO_2 to U_3O_8. In addition, the volatility of uranium at high temperature was excessive. This problem was solved by incorporating the urania with yttria and reacting these materials to form a stable solid solution.

The phase relationships of BeO-UO_2-Y_2O_3 were defined during the development of this stable fuel additive. An extremely stable compound ($3Y_2O_3 \cdot UO_{2+x}$) was discovered in the yttria-urania binary system. This compound has been heated in air for several hundred hours at temperatures up to 3300°F with essentially no loss of uranium.

The basic stability of beryllia-urania-yttria fuel elements was proved by testing in the LITR and MTR at temperatures up to 3000°F. This preliminary work culminated in a large-scale test of an insert in HTRE-2 as shown in Figure 3.11. About 6000 uncoated fueled tubes were tested, and fuel element temperatures higher than 2800°F were measured. Air temperatures over 2150°F were produced from the hottest part of the insert. The stability of the fuel element was excellent, with the exception of water-vapor corrosion of the beryllia. The solution to the corrosion problem was achieved later with the development of coating techniques.

Development of High-Purity BeO Powders

Early in the development work on the BeO-base fuel elements it became apparent that the purity, sinterability, and reproducibility of the available beryllia powders must be greatly improved if optimum fuel element properties were to be obtained. As a result of further development work, a high-purity, nuclear-grade, readily sinterable BeO powder became commercially available in large quantities.

Beryllia of even higher purity than that commercially available was developed in the laboratory and pilot-line quantities were used in research and development for advanced BeO-based fuel elements.

Coating Development for BeO-Base Fuel Elements

BeO is subject to water-vapor corrosion when heated in moist air at high temperatures.

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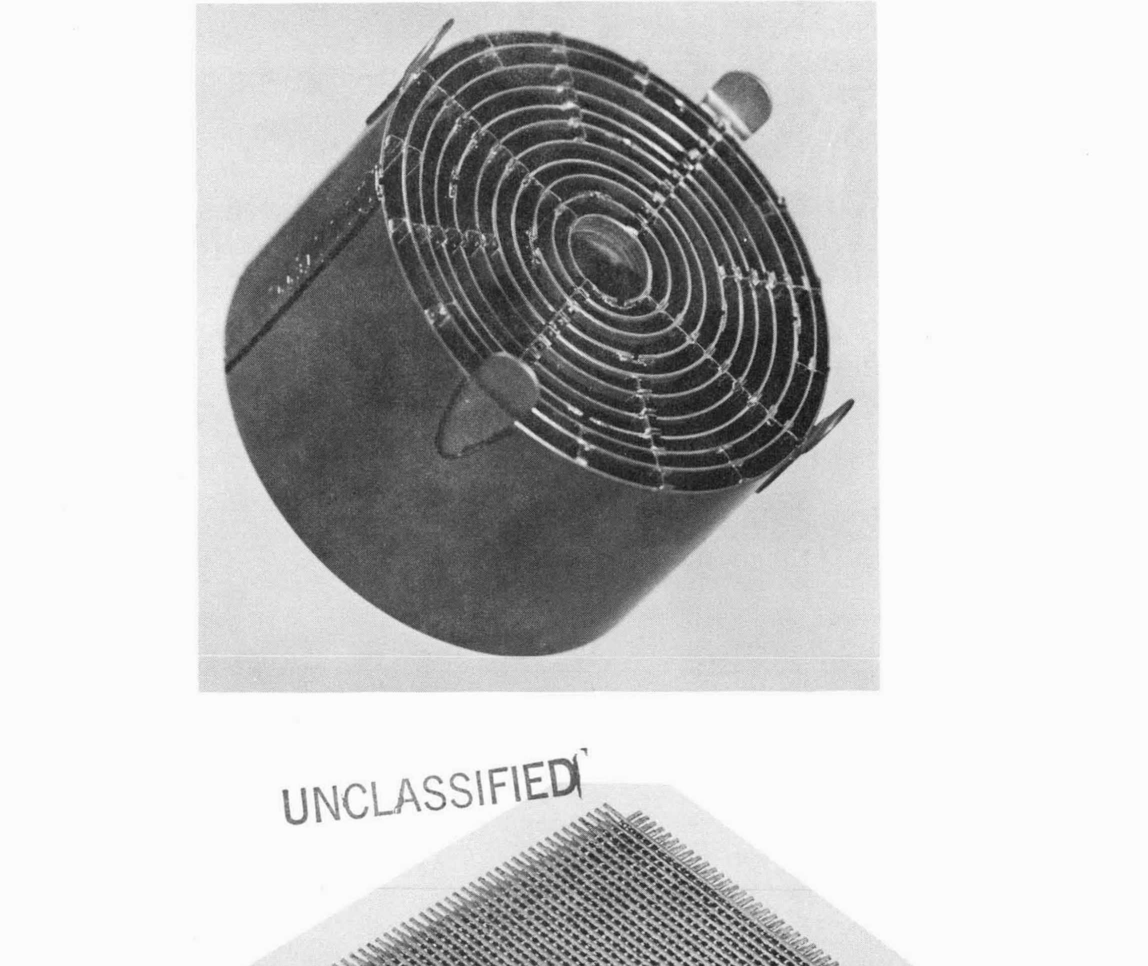


(Neg. U-38014A)

Fig. 7.1 – Concentric ring and wire matrix fuel elements

In theoretical consideration of this problem it was hypothesized that the corrosion rate is controlled by the diffusion of $Be(OH)_2$ through the boundary layer into the gas stream. On the basis of this hypothesis, equations for predicting the corrosion rate were developed and verified experimentally.

Thermodynamic consideration of this corrosion problem indicated that reacting the surface layer to form a stable BeO compound resistant to water-vapor corrosion was unlikely. In experimental verification, attention was given to the prevention of water-vapor contact with the BeO by coating the BeO with material not subject to water-vapor attack. As a result of these studies, techniques for applying alumina and yttria-stabilized zirconia to the BeO were developed. Vapor-deposition processes for applying the alumina coating were developed to apply uniform (0. 0005 to 0. 0020 inch) thicknesses with a ±10 percent variation in thickness for any 0. 0005-inch step between these limits. The density of these coatings was essentially theoretical.

The alumina coating provides excellent protection against water-vapor corrosion at temperatures to 2600°F and also significantly reduces fission product release at 2600°F.

Fabrication Development

Of particular significance in fabrication was the development of a coextrusion process in which the fueled matrix and bore clad are formed and extruded in one operation. Hexagonal fuel tubes of this type were to be used in the XNJ140E-1 reactor and are illustrated in Figure 7. 2 The clad had excellent compatibility with the BeO to at least 3100°F; tubes tested at 2700°F showed no evidence of attack by water vapor.

Fission Product Release Studies

Techniques were developed to measure fission product release rates from metallic and ceramic fuel elements. In-situ release rates from ceramic fuel elements were measured in dynamic irradiation tests in the LITR, MTR, and ORR reactors at temperatures to 3000°F. These techniques are applicable to much higher temperatures.

Three principal release mechanisms - recoil, diffusion, and corrosion - were investigated both theoretically and experimentally. Computer programs were developed and used to predict the recoil release from various sizes and shapes of unclad fuel elements.

Laboratory diffusion experiments were performed to determine the diffusion coefficients for I^{131} and rare gases such as Kr^{85}.

APEX-914, "Ceramic Reactor Materials," of this Report covers GE-ANPD ceramic fuel elements in detail. Further data on BeO as applied to the XNJ140E is provided in APEX-908, part B.

7. 2. 2 METALLIC FUEL ELEMENTS

Work on metallic fuel elements was directed principally toward development of metal-UO_2 composites. Two forms of metallic fuel were developed extensively: fuel sheet used in the concentric ring fuel elements, and fueled wire for use in the folded-flow reactor (Figure 7. 1).

In developing the fuel sheet, it was necessary that the matrix and clad materials be metallurgically bonded and that the sheet have adequate strength and resistance to deformation. The clad material was required to be oxidation resistant, to resist the diffusion of fission products to the surface, and to be strong enough that the fuel element structural components could be attached without rupturing the clad at the attachment point or exposing the fueled matrix under imposed stresses.

The requirements for fueled wire were similar except that bonding between core and

Fig. 7.2 – Stacked hexagonal fuel tubes

clad was less critical. Furthermore, the wire fuel elements were intended for use in the folded-flow reactor in which the mechanical stress due to aerodynamic loading was low. Hence, resistance to deformation was less critical than in the fuel sheet for concentric ring elements.

Type 310 stainless steel, Ni-Cr, Fe-Cr-Al, Fe-Cr-Y, chromium and chromium alloys, and niobium were investigated for application as matrix and/or cladding materials.

Stainless Steel

The original work on uranium dioxide dispersions in high-temperature, oxidation-resistant, metallic materials involved sandwich-type plates consisting of Type 310 (or 309) stainless steel cladding with a core of uranium dioxide dispersed in stainless steel. The cladding thickness was 0.005 inch. The active core consisted of a mixture of 30 weight percent enriched uranium dioxide dispersed in 70 weight percent Type 310 stainless steel. The stainless steel fuel elements met many of the requirements but had relatively limited temperature potential.

Nickel-Chromium

After withdrawal of the P-1 power plant objective, research and development work shifted to an 80Ni - 20Cr alloy which had greater strength and better oxidation resistance than the stainless steels. This alloy was of the same general type used in electric furnace resistance-heater elements. Substantial improvements in the strength and oxidation resistance of the alloy were made in the course of the GE-ANPD development. Concentric-ring Ni-Cr fuel elements were successfully developed for use in the HTRE reactors. The fuel matrix contained 40 to 42 percent enriched UO_2 dispersed in the 80Ni - 20Cr alloy. The fuel sheet was clad with a nickel-chromium-niobium alloy and was fabricated in ribbon form. The fuel ribbon was cut to the required lengths, sealed at each end with brazed-coated wire, and formed into rings. It was discovered that hot finishing at 1400° to 1800°F resulted in increased resistance to deformation and a decrease in oxidation penetration. This development emphasized the importance of processing variables on the performance and reliability of components. The Ni-Cr fuel sheet proved stable, reliable, and amenable to fabrication into suitable configurations for direct air cycle nuclear reactors. There was an excellent metallurgical bond between matrix and clad. The clad resisted both oxidation and fission product diffusion. A 1000-hour life capability was demonstrated at 1900°F.

Iron-Chromium Alloys

Fe - 25Cr - 5Al - Fe-Cr-Al and other iron-chromium alloys have excellent oxidation resistance but low deformation resistance at high temperatures. Hence, they were most useful for low-stress applications such as existed in the folded-flow reactor. Fe-Cr-Al proved unsatisfactory because the aluminum reacted with the UO_2, resulting in uranium diffusion through the clad and in a brittle interface between the clad and the matrix.

Fe - 30Cr - 1Y - The use of yttrium instead of aluminum was found to improve the oxidation resistance of iron-chromium alloys. Alloys containing 25 to 35 percent chromium and about 1 percent yttrium were developed as possible cladding material for use in air at temperatures up to 2300°F. Alloys containing both yttrium and aluminum (e.g., Fe - 30Cr - 3Al - 0.5Y) were found to have outstanding oxidation resistance at temperatures up to 2500°F.

Chromium-Base Alloys

Cr-UO_2-Ti Clad With Fe-Cr-Y - An oxidation-resistant cladding material, 69Fe - 30Cr - 1Y, gave indications of meeting a service life of 100 hours or more at 2100°F. Adding 1 weight percent titanium to the Cr-UO_2 core material improved the strength of the basic fuel material. While the above requirements were met, new requirements for a longer service life were imposed. Consequently, this effort was de-emphasized in favor of ceramic materials.

Cr-UO_2 Clad With Cr - Using a chromium clad with a Cr-UO_2 matrix has been variously called all-chromium or chromium-base ribbon. Both ribbon and wire geometries were investigated by GE-ANPD. Investigation of the system was emphasized after discovering in 1956, that adding 1 weight percent yttrium greatly improved the oxidation resistance of chromium in the 2100° to 2500°F temperature range. Interest was further increased in 1958 with the discovery that the addition of 1 weight percent titanium increased the core strength. When emphasis was placed on ceramic materials, however, further work on this system was discontinued.

Niobium

The strength, high melting point (approximately 4400°F), relatively low thermal neutron cross section, and the relatively high melting points of its oxides, made niobium attractive as a potential core-matrix in fuel elements for reactors capable of higher operating temperatures. Unfortunately, niobium has very poor oxidation resistance and therefore, unlike Ni-Cr, cannot be used as both the clad and matrix material.

Niobium fuel elements clad with Fe-Cr-Al were investigated for application at temperatures up to about 2300°F. Interfacial reactions between the core and cladding materials at these high temperatures, and the diffusion of uranium through the clad, prevented the successful use of these fuel elements. The niobium fuel element development was de-emphasized after withdrawal of the 125A Weapons Systems objective.

Intensive development of niobium fuel elements was resumed in 1959 with the investigation of protective coatings to satisfy service requirements in oxidizing environments up to 2500°F for periods up to 1000 hours. Materials investigated as coatings included aluminum, zinc, silver, tin, and combinations of these elements. The aluminum coatings gave good oxidation resistance at 2500°F but were deficient at intermediate temperatures; adding Ti and Cr improved the characteristics of the aluminum coatings. The zinc coatings were effective below 1800°F but unsatisfactory at higher temperatures. A complex coating consisting of Ag - 22Sn - 15Al - 0.5Ti - 0.5Cr was operated successfully for 1000 hours at 2200°F and 100 hours at 2300°F.

A niobium coating was developed that withstood over 1000 hours of testing at 2500°F. Additional work was under way to perfect an alloy, Nb - 33Ti - 3Al, which showed evidence of even better oxidation resistance. This coated niobium fuel material was considered primarily for use in wire form in the folded-flow reactor. In the wire matrix fuel element design, the problem of coating penetration by structural appurtenances was minimized.

In addition to coating studies on niobium, an alloy development program was carried out to achieve greater oxidation resistance in the base system. Alloys investigated included compositions of Ni-Al, Ni-Al-Ti, and Ni-Al-Ti-Cr systems, with other additions in some cases. Some improvement in oxidation resistance was noted in many cases, but a completely satisfactory alloy was not developed.

With nonoxidizing coolants such as helium, neon, nitrogen, etc., the use of a clad is obviously not required. See APEX-913, "Metallic Fuel Element Materials," of this Report for further details.

7.2.3 SHIELD MATERIALS

Development of shield materials was directed toward materials that provide maximum neutron and gamma attenuation per unit weight while withstanding the elevated temperature environment to which the material would be exposed. Availability and fabricability were also important considerations in the choice of shield materials. An operating life of 1000 hours was required of these materials.

The proposed reactor shields used compressor discharge air at temperatures of 400° to 600°F as a coolant. This set the temperature and oxidizing environment for the shield materials. Ram air injected during the flight operation was used to cool the side shield. This lower temperature permitted consideration of many materials for the side shield that could not be used in the front and rear shields.

Neutron Shielding Materials

The major effort in the development of neutron shielding materials was directed toward lithium hydride, beryllium plus boron, and beryllium oxide plus boron.

1. Lithium Hydride

Lithium hydride is one of the best materials for use as a neutron shield for an airborne reactor. It combines low specific gravity (0.78 g/cc), a high hydrogen content for thermalizing fast neutrons, and a high absorption cross section for thermal neutrons. Its thermal conductivity is low and coefficient of expansion is high. The successful use of lithium hydride as a neutron shield required the development of methods of circumventing its engineering deficiencies while capitalizing on its nuclear advantages. A cold-pressing fabrication technique was developed in which a 0.002-inch-thick perforated stainless steel honeycomb foil was incorporated in the lithium hydride. Several studies were performed to evaluate this composite as a high temperature neutron shielding material: methods of machining; determination of the most corrosion-resistant container materials; basic physical property studies such as thermal conductivity, compressive strength, and coefficient of expansion; dynamic air oxidation tests; and radiation stability studies.

As an alternative fabrication method, techniques were developed for casting lithium hydride. The most useful of these was the reservoir technique, in which surplus lithium hydride was maintained in a sacrificial connecting reservoir above the desired cast container. Using this technique made possible the successful casting of prototype reactor shield segments up to 9 feet long and containing 950 pounds of lithium hydride.

No appreciable changes in dimensions or volume were observed when LiH was irradiated at 1.3×10^{19} nvt (thermal) at $1000^{o}F$.

A parallel corrosion study of various austenitic structural materials such as Types 301 and 316 stainless steel, and 19-9DL, was conducted to determine the most reliable container material for application with the casting development program. The highest strength levels in a lithium hydride environment at the anticipated operating temperatures were demonstrated by 19-9DL which was, therefore, recommended as a structural material for the casting process.

Specifications for bulk and cold-pressed lithium hydride were written and commercially prepared material was procured to these specifications.

2. Beryllium-Boron Bodies

   Beryllium was a particularly attractive shielding material for aircraft reactors because of its low weight, high thermal conductivity, and high temperature capability compared to that of most hydrogenous materials. Boron also possesses a unique nuclear property: A neutron collision with this element results in an (n, α) rather than an (n, γ) reaction. Although the (n, α) reaction results in high local heat generation, the low thermal neutron flux resulting from the reaction reduces induced radioactivity from (n, γ) structural materials such as nickel and iron. A body composed of both beryllium and boron has the nuclear advantages of both materials.

   Beryllium-boron bodies in which 1 weight percent natural boron was added were successfully fabricated utilizing the hot-press technique. These materials were combined so that the boron was uniformly dispersed as an unreacted phase and the usual metallic properties of beryllium were retained within the structure. Numerous determinations of the physical and mechanical properties of this material were conducted. In general, the characteristics of the composition are the same as pure beryllium, although with somewhat lower ductility and higher strength. A glass-ceramic spray-coating was developed which resulted in oxidation protection of the beryllium-boron bodies at $1400^{o}F$ for 1000 hours. A materials specification was written as the result of this work, and material meeting this specification was obtained from a commercial vendor.

3. Beryllium Oxide-Boron Bodies

   Two methods were developed for fabricating beryllium oxide-boron bodies containing up to 4 weight percent boron. Hot-pressing of a composition of BeO plus ZrB_2 was successfully used for bulk shapes. In the other process, small hexagonal bodies of BeO plus B_4C were extruded, prefired, and sintered to more than 90 percent of theoretical density. These materials were found to retain uniform boron distribution throughout the body after 2000 hours at 2000°F.

Gamma Shielding Materials Development

The most desirable materials for gamma shielding, from a nuclear standpoint, are the elements of high atomic number such as lead, uranium, and tungsten. Stainless steel is also a logical choice from a practical standpoint.

The shield design requirements for high temperature, air cooling, and long life immediately eliminate most materials unless they are modified in some manner. Although lead could not be considered because of its low melting point, tungsten, uranium, and other heavy elements with high melting points could be considered. Most of these materials are not oxidation-resistant, however, and require a coating or cladding for operation with air at high temperatures.

1. Tungsten and Uranium

   Initial development on gamma shielding materials was conducted on uranium (as the uranium-molybdenum alloy) and on tungsten (as the tungsten-nickel-copper alloy). The tungsten alloy program resulted in the development of a high-temperature, high-strength alloy and a protective coating capable of operating for several hundred hours at 1600°F in dynamic air. Boron additions (1 to 3 weight percent) were successfully made to the W-Ni-Cu alloy and also to a W-Ni-Fe alloy.

2. Stainless Steel

   Despite the nuclear desirability of the heavy elements, stainless steel was the principal gamma shield material used in propulsion system designs, because of its high strength and oxidation resistance under the environmental conditions in the shield. It was also readily available, easy to fabricate, and relatively inexpensive. Furthermore, stainless steel could also be used as a structural material; this double utility largely counteracted its disadvantage as a shield material compared to the heavier elements. Compositions of Type 304 stainless steel plus boron (as B^{10} and B^{n}) were evaluated for application to 1500°F in an oxidizing atmosphere.

GE-ANPD progress in the development of shielding is discussed in APEX-915, "Shield Materials," of this Report.

7.2.4 MODERATOR AND REFLECTOR MATERIALS

The moderator and reflector materials which were investigated included hydrogenous liquids, hydrided metals, beryllium oxide, intermetallic compounds such as zirconium beryllide, graphite, and carbides.

Hydrided Metallic Moderator Materials

One of the significant metallurgical achievements of the GE-ANPD program was the development of high-temperature hydrided metals for use as moderator materials. Zirconium hydride and yttrium hydride were the most attractive materials from the nuclear, metallurgical, and mechanical standpoint. Zirconium hydride can be used unclad, but yttrium hydride oxidizes rapidly at low temperatures and must be clad for all applications.

The useful temperature limits of unclad ZrH_x range from 800° to 1600°F depending on N_H,* required operating time, and temperature gradients. The useful temperature range of YH_x is from 1600° to 2100°F, depending on the composition and thickness of the cladding.

1. Hydrided Zirconium

Hydrided zirconium was developed as a moderator material because of the low nuclear cross section of zirconium and the high N_H of the stoichiometric compound, ZrH_2. Massive forms of reactor-grade zirconium metal were hydrided at temperatures from 1400° to 1700°F. The resulting bodies remained intact, and a hydrogen content as high as $N_H = 6.6$ was achieved. The material was metallic in character, with a thermal conductivity equal to or slightly superior to that of the parent metal. Because the thermal conductivity of the massive hydrided bodies was better than that of hydrided bodies prepared by powder metallurgy, the problems of removing moderator heat from the reactor were greatly alleviated.

Cold and hot extrusion techniques were developed to produce the forms to be hydrided. Zirconium in hollow hexagonal shapes 4.5 inches across flats, 36 inches long, with wall thicknesses of 0.50 to 0.75 inch, was extruded, hydrided, and used in the HTRE-3 reactor.

Metallic cladding was successfully bonded to hollow hexagons of zirconium hydride, as shown in Figure 7.3. Multiholed, internally cooled, biflute zirconium shapes (Figure 7.4) and triflute shapes were produced as a result of advanced manufacturing studies. It was demonstrated that hydrided zirconium in air retains hydrogen for acceptable time periods at temperatures up to 1600°F without cladding.

2. Hydrided Yttrium

Yttrium was also hydrided in massive metallic form. Whereas the hydrogen-retention capability of zirconium decreases markedly at temperatures above 1600°F, yttrium retains hydrogen to a remarkable degree even at temperatures as high as 2400°F. From this standpoint, yttrium is distinctly superior to zirconium in a nuclear power plant in which the moderator temperature exceeds 1600°F.

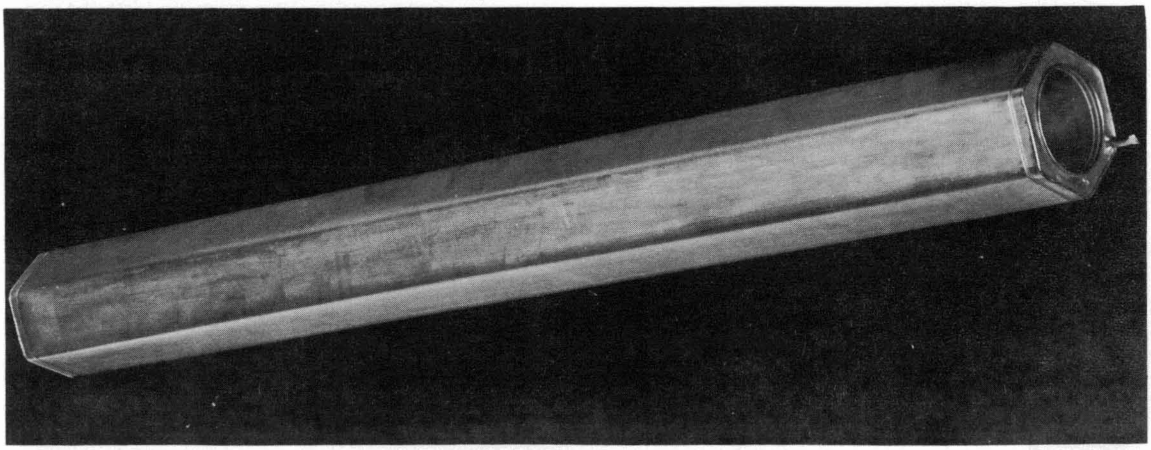


Fig. 7.3 – Completely bonded hollow hexagonal section of clad hydrided zirconium (C-20071)

*N_H = Number of hydrogen atoms per cubic centimeter of hydride x 10^{22}

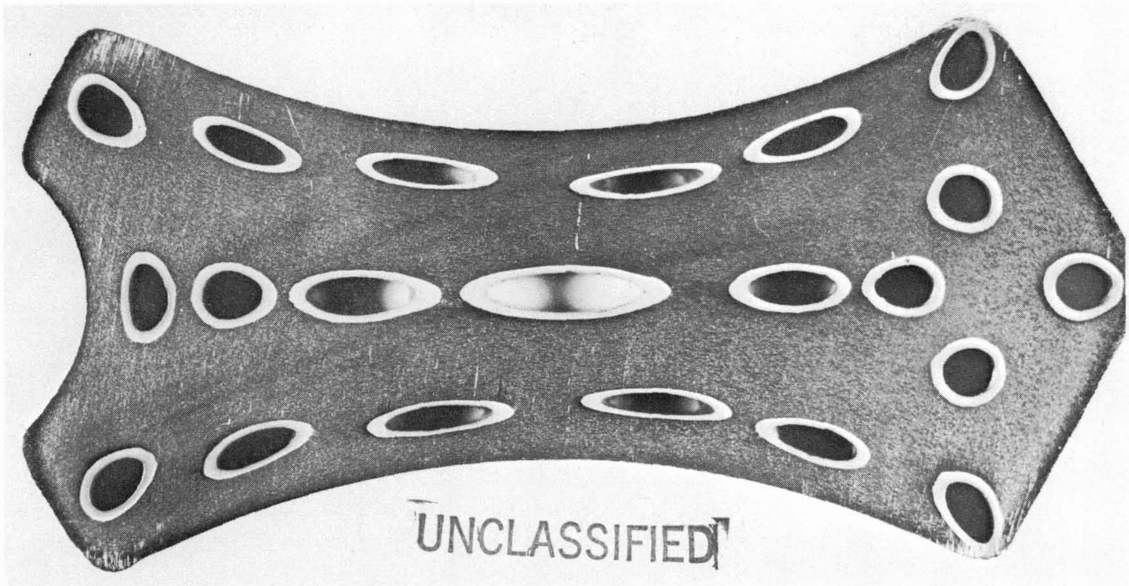


Fig. 7.4 – Section of coextruded ZrH_x biflute showing bonded 446 stainless steel cladding in holes. Steel cones have been removed by acid leaching and unbonded outer cladding has been stripped.

On the other hand, hydrided yttrium proved to be brittle and therefore sensitive to thermal gradients and stress. It was expected that the brittle nature could be altered with further metallurgical development. An interim resolution of this problem was attained by alloying yttrium with zirconium and chromium. Thermal-mechanical tests of hydrided 70Y - 30Zr alloys and 95Y - 5Cr alloys were conducted for periods of 1000 hours at temperatures from 1750° to 1900°F.

3. Beryllium Oxide

Because of its excellent refractory properties beryllium oxide is useful as a moderator, shield, reflector, or as part of a homogeneous fuel-moderator mixture. Uncoated BeO can operate at temperatures as high as 2100° to 2200°F before hydrolysis becomes a factor. Coatings of ZrO_2 and Al_2O_3 extended this range to 2600° to 2700°F with no appreciable hydrolysis of BeO.

In the ANP program, beryllium oxide was used primarily as a reflector, although consideration was also given to its use as an unfueled moderator material. Its application in shields and as a fuel-bearing material is described elsewhere in this section.

4. Beryllium

Beryllium was used primarily as a reflector and as a front shield plug material where cooler compressor air was available and ambient temperatures, usually from 600° to 1000°F, could be maintained. Some progress was made in developing coatings for beryllium that would permit surface temperatures of 1400° to 1600°F in an oxidizing atmosphere.

Hydrogenous Liquid Moderators

Water was used as the moderator in the first GE-ANPD reactor designs. Higher temperature organic moderators were also investigated extensively. Although this work demonstrated the advantages of Pentalene 290,* subsequent work by other investigators showed

*Registered trade mark of Sharples Chemical, Inc.

that the excellent radiation resistance of alkylbenzene 250 and 350 compensates for the somewhat lower N_H of these materials.

Intermetallics

Zirconium beryllide, $ZrBe_{13}$, and other beryllides were also investigated by GE-ANPD. These materials demonstrated thermal instability that precluded their use at the high-temperatures encountered in nuclear aircraft.

The development of moderator and reflector materials is described in detail in APEX-916, "Moderator Materials," of this Report.

7.2.5 ORGANIC, STRUCTURAL, AND CONTROL MATERIALS

Organic Materials

The principal effort with organic materials was devoted to evaluations of commercially available materials under the expected environmental conditions, particularly radiation. Early effort in the program was concerned with establishing a means of comparing dam-age by different types of radiation particles. This work led to the use of "equal absorbed energy - equal damage" as an approximate basis of comparison, and culminated in a summary and analysis of the existing data on plastics, elastomers, and organic liquids.

Subsequently, experimental investigations were performed on both materials and components under combined radiation, temperature, and atmosphere environments. Materials examined extensively included turbojet engine lubricating oil, reactor moderator-coolants, and elastomers of the types used in engine accessories and controls. Components tested included electric motors, semiconductors, elastomer seals and hoses, and several types of components of the turbomachinery subsystems and the reactor control systems. Entire subsystems of the power plant were also irradiated under operating conditions; these included control subsystems and the fuel, lubrication, and hydraulic subsystems of the X211 engine.

Structural Materials

The most significant of the GE-ANPD investigations of commercial metallic structural materials was that of the effect of radiation and irradiation temperature on the precipitation-hardened alloys A-286 and Inconel X. In post-irradiation examinations, these alloys exhibited the increased strength and hardness and decreased ductility typically found in the room-temperature properties of irradiated metals. At elevated temperatures, however, the ductility, notch sensitivity, and rupture strength were significantly reduced. The tests revealed that these property changes were of practical significance at dosages less than 10^{18} nvt ($\geqq$ 1 Mev) and that the magnitude of the changes was greater under irradiation at elevated temperatures.

Other work with structural metals included limited post-irradiation examination of Inconel X and Inconel 702 of different heat treatments, including some welded specimens, and measurements to define the design properties of non-irradiated Inconel X, Hastelloy X, and 19-9DL.

Control Rod Materials

Research and development with control rod materials ultimately centered on europium oxide which possessed the requisite thermal stability in the 1500° to 2000°F temperature range as well as a high nuclear absorption cross section in both the thermal and epithermal neutron energy ranges. Fabrication and evaluation studies were conducted on mixtures of europium oxide and various metals as matrix materials in combination with Inconel, Inconel X, and 80Ni - 20Cr as cladding stock. From this work, a Ni - $42Eu_2O_3$ (weight percent) core matrix with 80Ni - 20Cr* cladding was developed for use at 1650°F. Approximately 100 of these control rods were fabricated and used in HTRE-3. Thermal mechanical tests on various designs of this material combination were successful up to 1800°F for periods of 1200 hours. Additional work on the fabrication of pure Eu_2O_3 demonstrated its use as a potential control material in oxidizing or neutral atmospheres at temperatures above 2000°F.

Tungsten Alloy Thermocouples for 5000°F Operation

High temperature thermocouples were developed for measuring temperatures in various furnaces and test equipment used in the fabrication or evaluation of fuel element materials. Initial work was on thermocouples of pure tungsten and rhenium; however, it was found that important advantages in strength of signal at very high temperatures (> 4000°F) could be obtained by replacing the rhenium with a W - 25Re alloy. With this type of thermocouple, relatively convenient temperature measurements can be made in neutral or reducing atmospheres at temperatures up to about 5000°F. Special techniques and procedures were developed for sintering and fabricating the new alloy into wire and also for calibrating the thermocouples in the temperature range from 4000° to 5000°F. These processes for fabricating tungsten-rhenium alloys into wire and for calibrating the thermocouples are now applied regularly by commercial vendors.

The foregoing materials are described more fully in APEX-917, "Organic, Structural, and Control Materials," of this Report.

7.3 NUCLEAR SAFETY

In the HTRE tests air passed through a nuclear reactor operating at high temperature and subsequently discharged to the atmosphere. Consequently, the release of relatively small amounts of radioactive material to the atmosphere was a result of normal reactor operation, and field monitoring of air and vegetation samples was done routinely.

To evaluate the possible hazards resulting from a reactor accident during the nuclear testing, detailed dose calculations were made prior to each startup. These calculations, based on the actual meteorological parameters and reactor fission-product inventory existing at the time of testing, were intended to show whether startup could proceed without undue risk to persons in the area of the test site in the event of an accident. Contamination resulting from normal operation was so slight that no meaningful health physics data were available from routine measurements to verify the calculations. Therefore a unique program was instituted in which segments of reactors were intentionally damaged to reveal the radiological consequences of such damage. The data from these experiments showed the validity of the operating procedures based on meteorological control.

Dose-prediction equations can be thought of as equations containing three terms: the biological factors, the radioactive source, and the meteorology (spread of the cloud in the atmosphere).

While the whole-body gamma dose from an effluent cloud can be measured directly, the internal beta dose must be calculated from measured or calculated airborne concentration values. In performing these calculations, values must be assigned to the various processes involved in the uptake and elimination of the harmful material. These values are obtained from published, acceptable data.

When dose calculations are used for the control of reactor operation, the meteorological term is the most important of the three terms because the biological factors are fixed,

*Special heat with less than 0.20 weight percent silicon.

and the source term, which is determined by reactor design, materials, and operating conditions, is essentially fixed for a given system. The major variable, then, is the diffusion and trajectory of the cloud in the atmosphere.

This procedure of estimating the dose that humans would receive from the release of radioactive materials to the atmosphere after a reactor excursion and of preceding each test with the consideration for the meteorological conditions that existed at the time of the test proved to be very useful for the direct-cycle testing program. The adaptation of Sutton's diffusion equation and the refinement of environmental measurement techniques provided for constant monitoring of each reactor test on an instantaneous basis.

As a result of this control, the reactor testing program was conducted at the Idaho Test Station with a minimum of radiation dosage to the inhabited areas near the National Reactor Testing Station, and most certainly far below the allowable dose specified by nationally recognized authorities in the field of health physics.

The methods and results of effluent dose analysis are presented in APEX-921,"Nuclear Safety," of this Report. APEX-921 also describes the safety experiments that were performed and presents data pertaining to the safety of operational nuclear propulsion systems.

Appendix A
FACILITIES

The General Electric Aircraft Nuclear Propulsion Department (GE-ANPD) operated Government owned facilities at Evendale (near Cincinnati), Ohio and at the Idaho Test Station, located within the National Reactor Test Station about 50 miles from Idaho Falls, Idaho.

In addition to its own facilities and personnel, GE-ANPD had access to other research, laboratory, and development facilities. These included, within the General Electric Company, the Large Jet Engine Department in Evendale, where the X211 engine was developed, and the General Engineering Laboratory and Research Laboratory in Schenectady, New York. Facilities of the Atomic Energy Commission that were available to GE-ANPD included portions of the Materials and Engineering Testing Reactors at the National Reactor Testing Station in Idaho and the Low Intensity and Research Reactors at Oak Ridge. Considerable work was also performed in other locations.

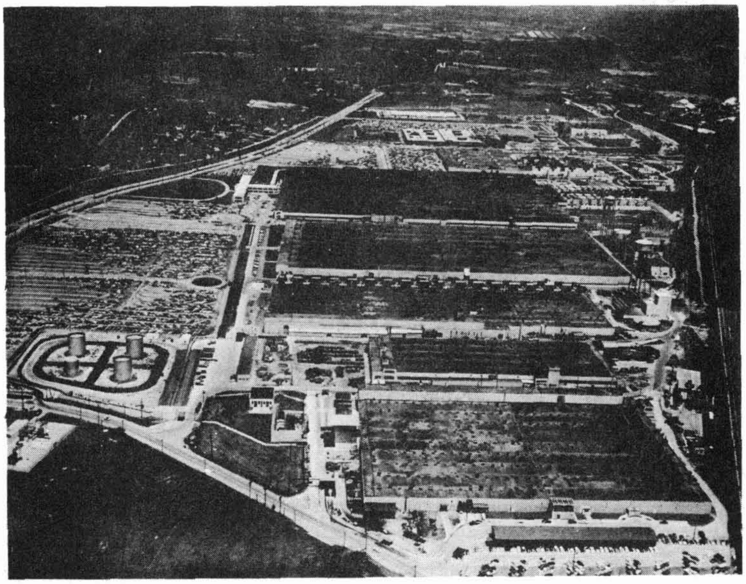
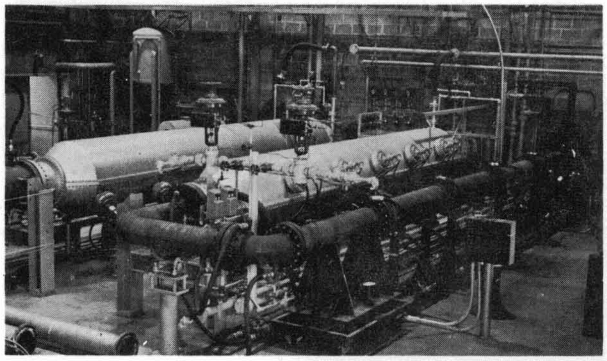


Figure A-1 - General Electric facility at Evendale, Ohio. Turbomachinery development, materials development, design, manufacture, non-nuclear testing, and preliminary critical experiments were performed here, as were reactor and power plant design. The chemical, metallurgical, and high-temperature laboratories in the building shown in the foreground were considered to be among the most advanced in the country. The laboratories contained a wide variety of equipment for producing and testing high-temperature, high-purity, high-density metals; shield and control materials; and in-core sensing devices. Two critical experiment cells were available and were instrumented to measure reactor parameters with a high degree of precision prior to commitment of the reactor to hardware. All reactor components were fabricated at the Evendale plant, in a shop area that included extensive brazing, roll-forming, and heat-treating equipment, in addition to specially enclosed machine tools. Complete production lines were in operation for the manufacture of both ceramic and metallic fuel elements and cartridges.



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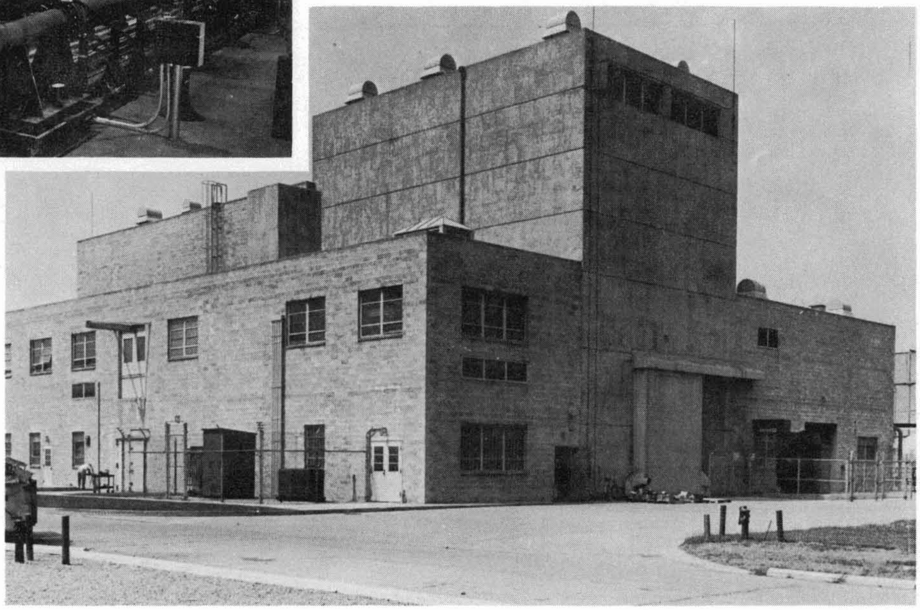

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Figure A-2 - Gas Dynamics Facility II (GDF II). Reactor components, materials, and instruments were evaluated in non-nuclear tests in GDF II under simulated conditions of pressure, temperature, and airflow. Two hot-gas test loops (see inset), a cold-flow loop, a burst chamber, a structural test facility, and a millisadic data-recording system were included. Fuel cartridges were proof-tested in the hot gas loop prior to final commitment to quantity production. In-pile tests of materials and components prior to reactor operation were conducted in the Engineering Test Reactor (ETR), the Materials Testing Reactor (MTR), and the Low Intensity and Research Reactors.

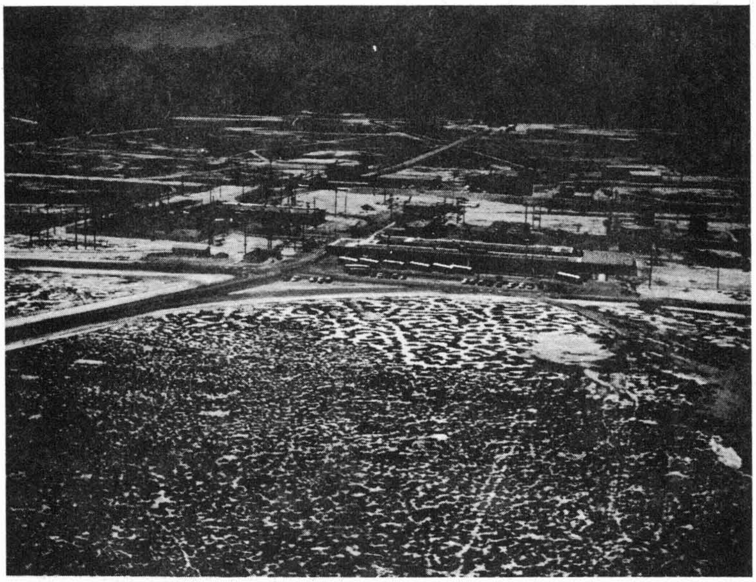

Figure A-3 - Idaho Test Station (ITS). Located at the north end of the 894-square-mile National Reactor Testing Station, the ITS had complete capability for assembling, testing, and maintaining high-temperature nuclear reactors and power plants, and performing exhaustive post-test analysis. Facilities included a complete machine shop, a cold assembly shop, a Hot Shop, a Low-Power Test facility, two high-power test sites, a swimming-pool shield test facility, laboratories, and an engineering and administration area. The initial buildup of the reactor test assemblies, with the exception of fuel element and reactor insertion, was made in the cold shop.

Figure A-4 - Low Power Test (LPT) and Shield Test Pool (STP) facilities. Fuel elements were inserted into the reactor in the LPT for an initial critical checkout. This was followed by low-power testing to evaluate nuclear characteristics prior to power operation. Operations were viewed by means of remotely operated television. (U-3343-55)

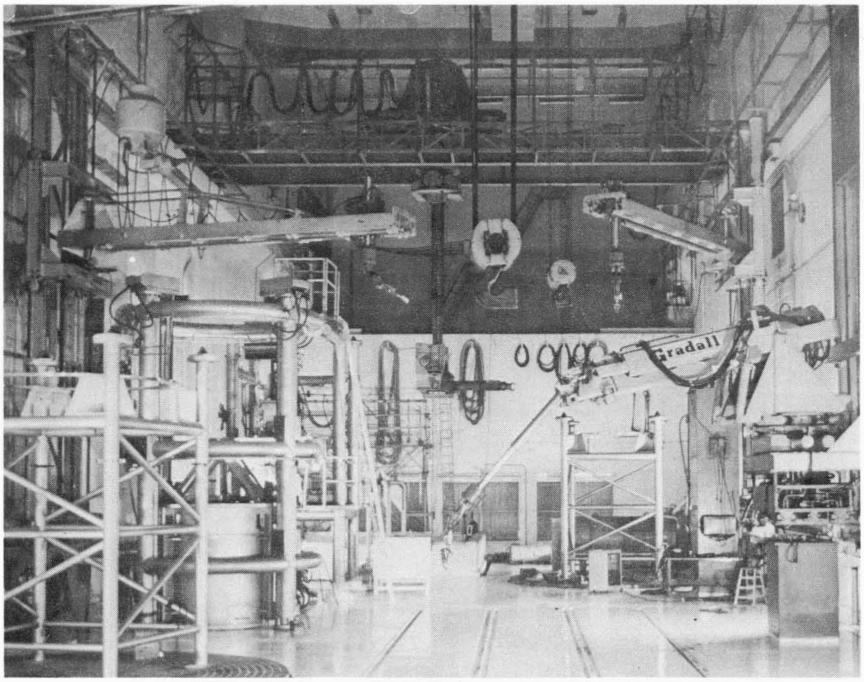

Figure A-5 - Hot Shop at the Idaho Test Station. After low-power testing, the reactor was returned to the Hot Shop, where it was inserted into the test assembly. The Hot Shop facilities included a 100-ton crane; a heavy-duty overhead manipulator called the O-man; four wall-mounted, boom-carried manipulators; a 15-ton crane and underwater dolly for the storage pool; remotely operable turntables; master-slave manipulators; and viewing windows. The storage pool that adjoins the Hot Shop was used for temporary storage of used fuel elements and other radioactive pieces.

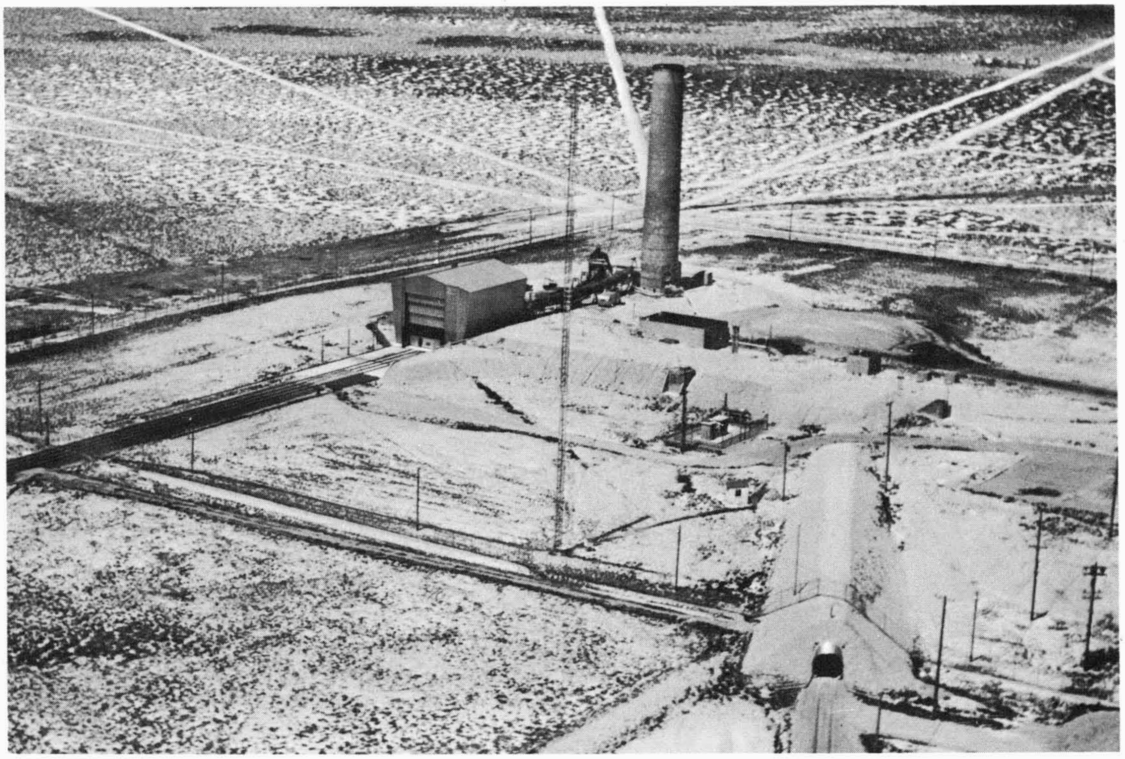

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Figure A-6 - Initial Engine Test (IET) facility. The reactor and engine test assemblies, mounted on a railroad dolly, were moved to the IET by a traction vehicle for power testing. The movable aluminum building served as weather protection, and the large unobstructed space which surrounded the reactor test assembly permitted shield and air-scattering tests to be conducted. Poured concrete, 3 feet thick, and 14 feet of compacted earth shielded operating personnel in the control and equipment building, which was accessible by means of the 450-foot-long access tunnel shown in the foreground. The 150-foot stack disposed of the exhaust gases, after they had been filtered to remove radioactive particles. A larger power test facility, the Flight Engine Test (FET) facility, was also built, but not fully completed.

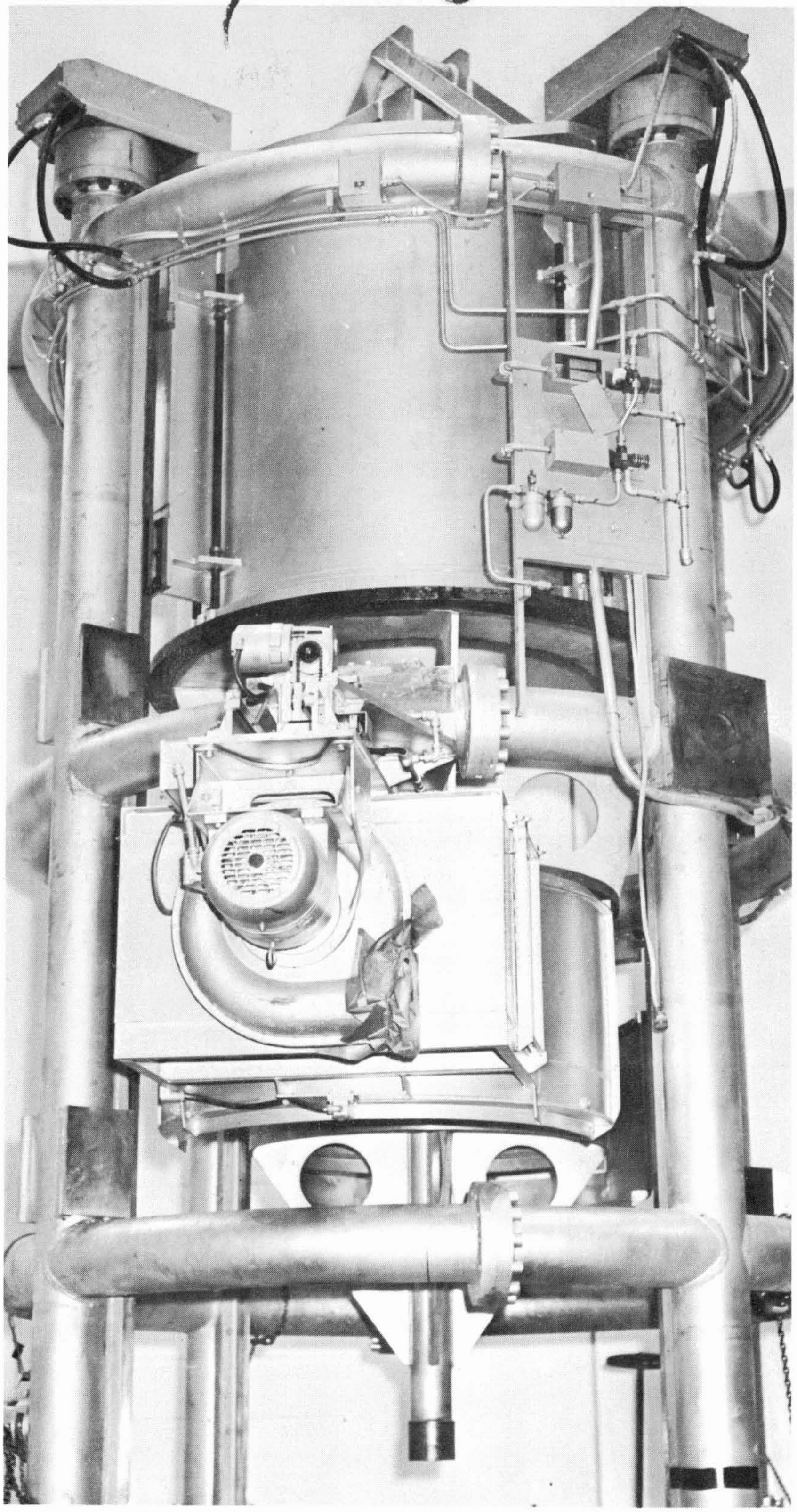

Figure A-7 - Core removal fixture. After full power tests had been run in the IET, the power plant was returned to the Hot Shop for repair or disassembly. The core removal fixture was used both for inserting and removing the reactor core and shield plug from a close-fitting pressure vessel.

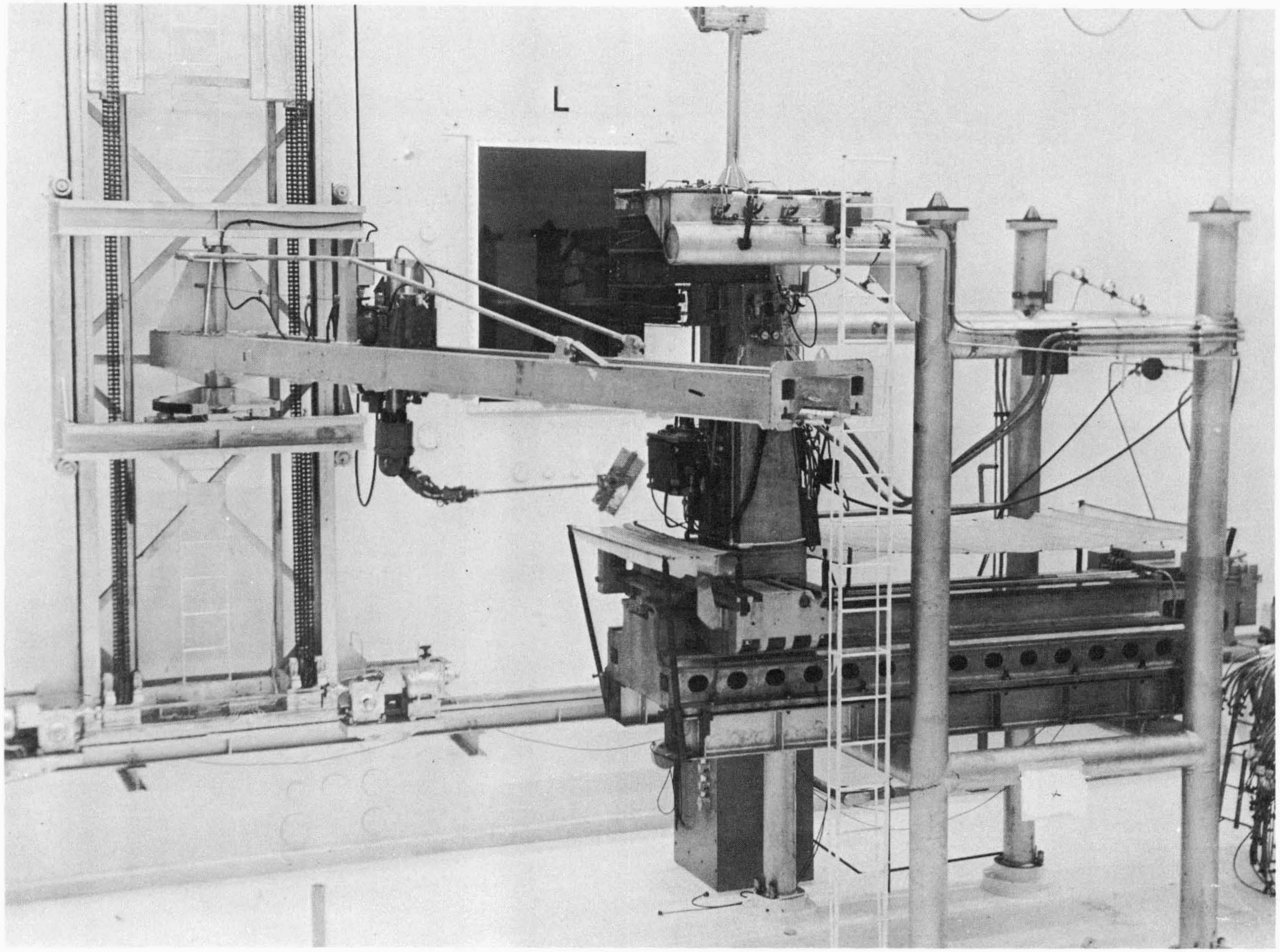

Figure A-8 - Fuel element loading machine. This special device remotely unloaded fuel elements after removal of the reactor from the test assembly. Following disassembly, fuel elements and other small samples were removed to a hot cell for close examination and inspection. Remote handling equipment was available in the test cell for this purpose. In some instances, where the radioactivity level was low enough, samples were returned to Evendale to be studied in a warm cell.

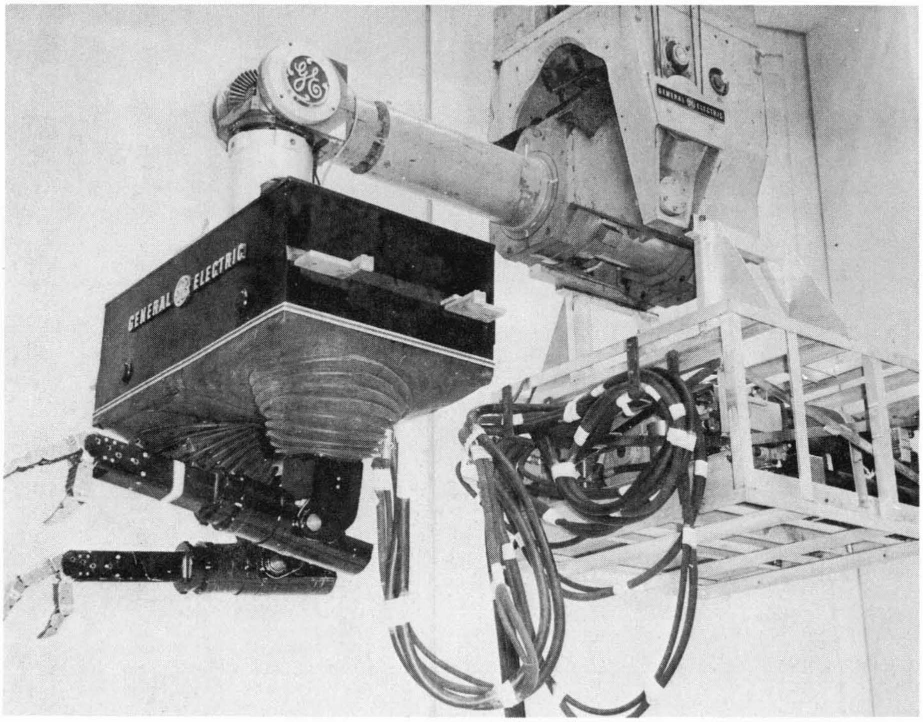

Figure A-9 - Handyman manipulator. This general purpose device was attached to the O-man and was located in the Hot Shop. It was developed by the General Engineering Laboratory of the General Electric Company and was able to duplicate the movements of the operator from the shoulders to the finger tips. A force-feedback system within the Handyman enabled the operator to "feel" the object being handled.

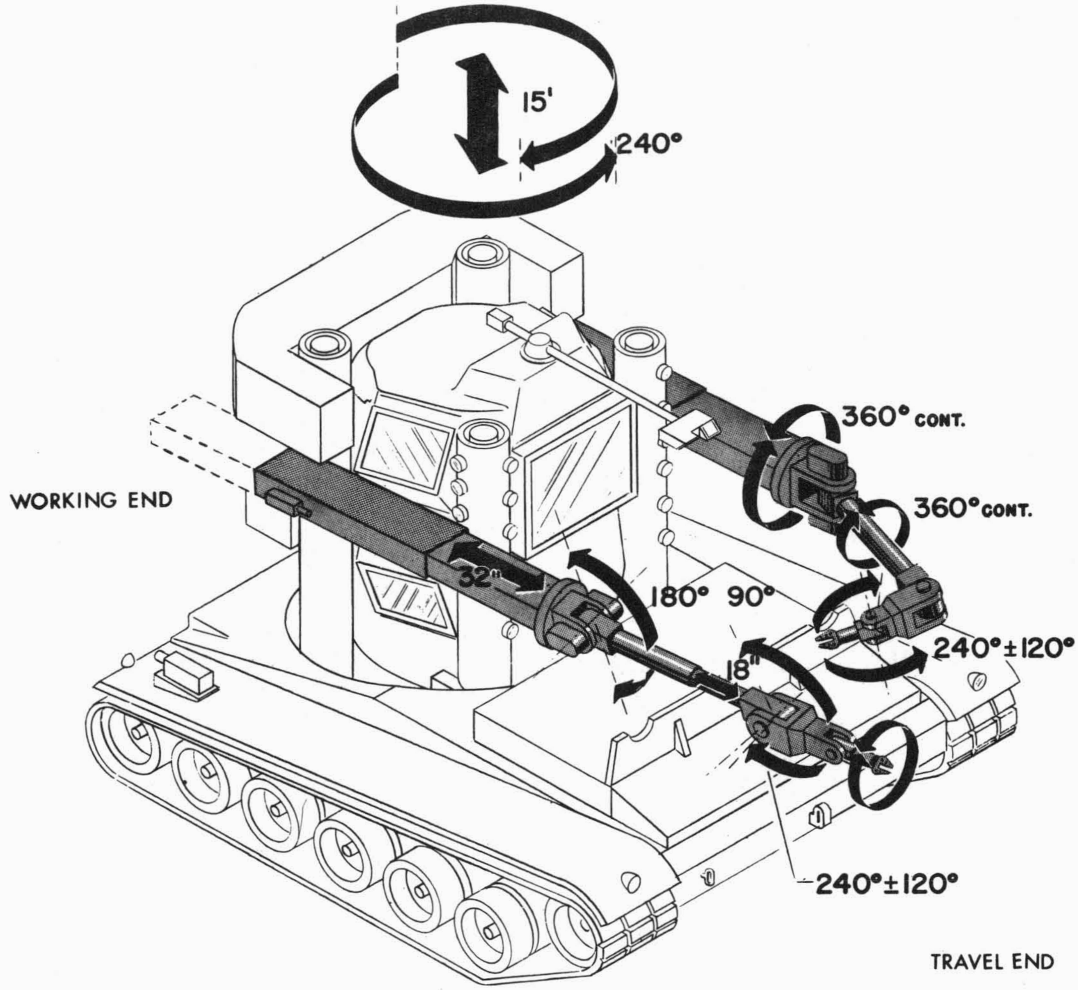

Figure A-10 - The "Beetle." This manned, shielded vehicle was designed for use in the Flight Engine Test facility. Its purpose was to provide a means of maintaining and removing the reactor and power plants from the aircraft mockup, and eventually, from an actual aircraft. It was nearing completion at program termination, and was subsequently completed for use in the nuclear rocket program.

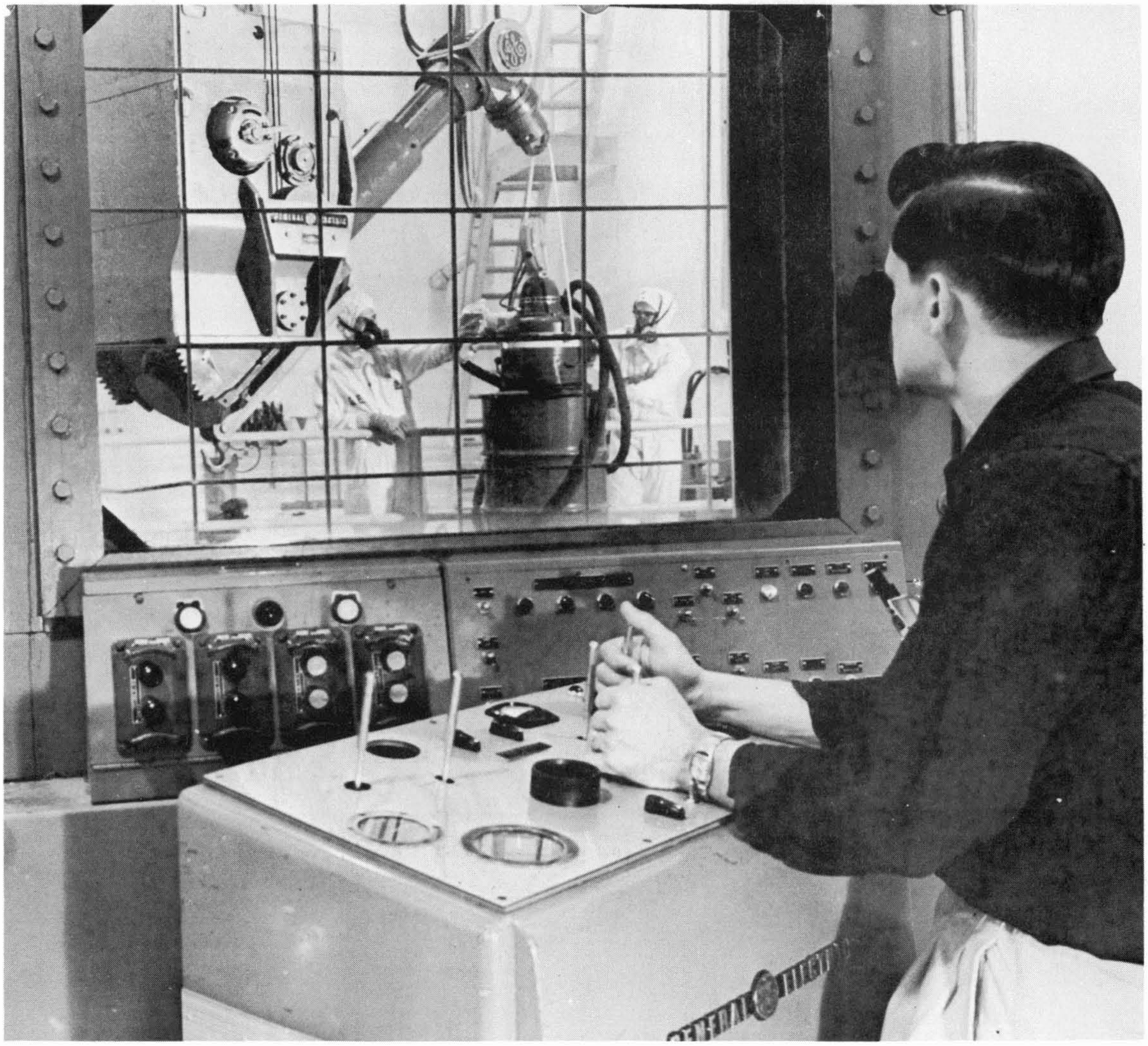

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Figure A-11 - View from Hot Shop gallery showing use of remote manipulators combined with limited manual operation.

Appendix B
GE-ANPD EXPENDITURES AND PERSONNEL

Appendix B contains figures and tables which show the expenditures and yearly employment during the GE-ANPD program. Figure B-1 and Table B-1 reflect the cost figures; Figure B-2 and Table B-2 contain employment statistics.

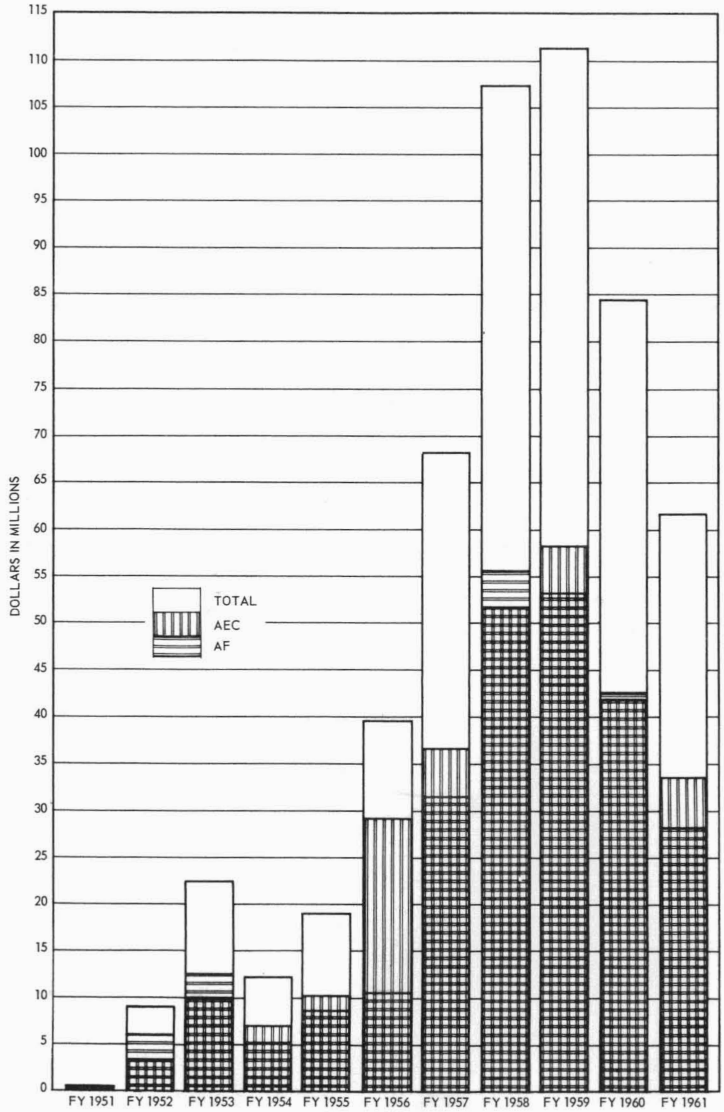

Fig. B-1 – GE-ANPD Fiscal Year expenditures

TABLE B-1

COSTS INCURRED BY USAF AND AEC UNDER CONTRACTS WITH GE-ANPD - (FISCAL YEARS)
($ IN MILLIONS)

Fiscal Year	Research and Development						Facilities and Equipment						Total					
	Yearly			Cumulative			Yearly			Cumulative			Yearly			Cumulative		
	AEC	AF	Total	AEC	AF	Total	AEC	AF	Total	AEC	AF	Total	AEC	AF	Total	AEC	AF	Total
1951	139	174	313	139	174	313	0	282	282	0	282	282	139	456	595	139	456	595
1952	2,971	3,655	6,626	3,110	3,829	6,939	168	2,431	2,599	168	2,713	2,881	3,139	6,086	9,225	3,278	6,542	9,820
1953	9,233	5,755	14,988	12,343	9,584	21,927	1,028	6,621	7,649	1,196	9,334	10,530	10,261	12,376	22,637	13,539	18,918	32,457
1954	5,370	3,851	9,221	17,713	13,435	31,148	1,731	1,253	2,984	2,927	10,587	13,514	7,101	5,104	12,205	20,640	24,022	44,662
1955	9,140	5,868	15,008	26,853	19,303	46,156	1,119	2,875	3,994	4,046	13,462	17,508	10,259	8,743	19,002	30,899	32,765	63,664
1956	18,026	9,896	27,922	44,879	29,199	74,078	11,179	743	11,922	15,225	14,205	29,430	29,205	10,639	39,844	60,104	43,404	103,508
1957	32,665	29,122	61,787	77,544	58,321	135,865	4,068	2,503	6,571	19,293	16,708	36,001	36,733	31,625	68,358	96,837	75,029	171,866
1958	41,939	48,573	90,512	119,483	106,894	226,377	9,890	6,986	16,876	29,183	23,694	52,877	51,829	55,559	107,388	148,666	130,588	279,254
1959	44,921	45,112	90,033	164,404	152,006	316,410	13,277	8,009	21,286	42,460	31,703	74,163	58,198	53,121	111,319	206,864	183,709	390,573
1960	36,924	40,315	77,239	201,328	192,321	393,649	5,091	2,269	7,360	47,551	33,972	81,523	42,015	42,584	84,599	248,879	226,293	475,172
1961	32,328	27,441	59,769	233,656	219,762	453,418	1,301	634	1,935	48,852	34,606	83,458	33,629	28,075	61,704	282,508	254,368	536,876

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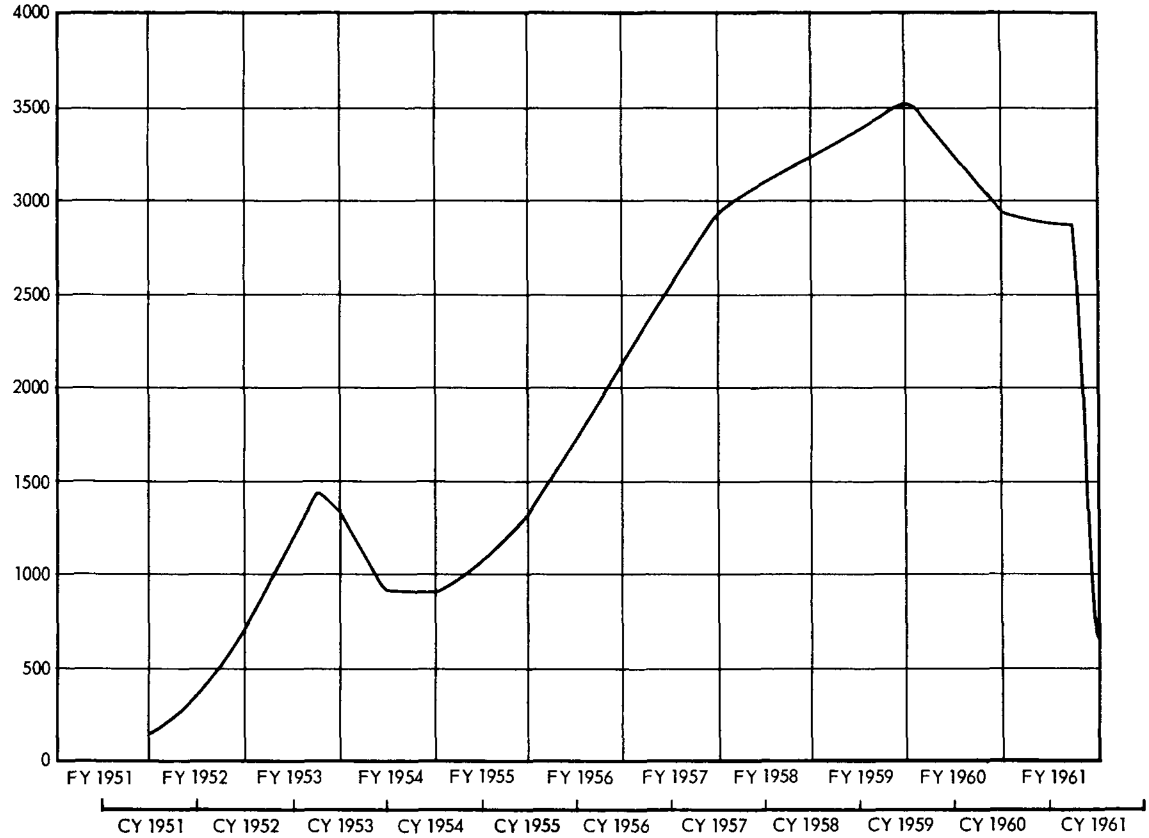

Fig. B-2 – GE-ANPD employment levels

TABLE B-2

DEPARTMENT PERSONNEL - FISCAL YEARS

	Managerial		Technical		Functional		Semi-Technical		Drafting		Fabrication Shop		Supporting		Total	
	Year End	Avg	Year End	Avg	Year End	Avg	Year End	Avg	Year End	Avg	Year End	Avg	Year End	Avg	Year End	Avg
FY 1951																
FY 1952	62	48	121	94	38	30	123	95	33	26	11	3	326	261	714	557
FY 1953	125	105	237	200	73	63	236	201	56	54	133	104	482	448	1342	1175
FY 1954	89	85	154	169	52	49	144	157	37	38	89	89	344	330	909	917
FY 1955	135	112	242	200	79	66	262	216	58	50	192	128	513	429	1481	1201
FY 1956	186	165	378	324	117	100	398	347	85	71	250	225	714	634	2128	1866
FY 1957	255	227	545	459	201	157	520	453	120	114	313	277	983	928	2937	2615
FY 1958	279	265	654	609	250	232	493	495	126	120	419	374	1024	1008	3245	3103
FY 1959	282	283	730	675	274	263	554	490	133	128	466	433	1090	1055	3529	3327
FY 1960	269	278	608	669	240	258	450	493	119	126	363	420	889	994	2938	3239
FY 1961[a]	283	278	557	564	256	256	392	423	125	123	363	359	916	895	2892	2897

[a]Fiscal Year 1961 through April only.

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Appendix C
REFERENCE INFORMATION CONTAINED IN APEX-902 THROUGH APEX-921 OF THIS COMPREHENSIVE TECHNICAL REPORT

This Appendix is a compilation of the abstracts, tables of contents, and reference lists of the other 20 volumes of the Comprehensive Technical Report of the General Electric Direct-Air-Cycle Aircraft Nuclear Propulsion Program.

The documentation numbers and titles of all 21 volumes are shown on the inside front cover of this volume for quick identification of a particular subject of interest. Reviewing the pertinent information on that volume in this Appendix will establish the approach used and the depth of treatment of the subject. It will also guide the reader in requesting more detailed reference information where required.

APEX-902

P-1 NUCLEAR TURBOJET

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This portion describes the design of the reactor for the first direct-air-cycle nuclear power plant undertaken in this Program. Presented are the bases for selection of the moderator, fuel element material and shape, basic reactor configuration, design point selection, and turbomachinery selection.

Details are presented of the methods used to flatten power radially and longitudinally, verify nuclear design (critical mockup and shield mockup), and fabricate fuel elements. The control system and its components are described, including the control console and instrumentation. The design of the shield and moderator cooling systems are reported.

The modifications of the conventional turbomachinery are included - in-line combustors, ducting and duct arrangement, valving, and the turbomachinery controls. The Propulsion Unit Test, for chemically simulating reactor operation, is also described.

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APEX-903
REACTOR CORE TEST FACILITY

ABSTRACT

This volume presents a summary description of the Core Test Facility (CTF) that was used in the first two Heat Transfer Reactor Experiments (HTRE No. 1 and No. 2), which were conducted at the Idaho Test Station (ITS). The CTF consisted of shielding, an air supply, and other necessary auxiliaries and services, which were combined into assemblies in order to test a succession of direct-cycle cores, fuel elements, controls, and other components.

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APEX-904

HEAT TRANSFER REACTOR EXPERIMENT NO. 1

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This portion describes Heat Transfer Reactor Experiment No. 1, believed to be the first successful operation of a turbojet engine on nuclear power. Design data are presented, including a general description of the test assembly, the nuclear characteristics of the reactor, fuel element thermodynamic characteristics, and the control system. The three series of test runs are also described and the test results summarized.

The general objectives of Heat Transfer Reactor Experiment No. 1 were to demonstrate the feasibility of the direct air cycle system by operating a turbojet engine on nuclear power, to demonstrate the adequacy of reactor design features, and to evaluate aerothermodynamic and nuclear characteristics of the reactor for use in the design of militarily useful aircraft power plants.

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APEX-905

HEAT TRANSFER REACTOR EXPERIMENT NO. 2

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume describes Heat Transfer Experiment No. 2, a test reactor used to evaluate ANP fuel elements and solid moderator materials.

The reactor, a modification of Heat Transfer Reactor Experiment No. 1, had the seven center cells of the core removed, providing a hexagonal hole for test inserts.

The reactor and the inserts tested are described and the results of the various tests presented.

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APEX-906

HEAT TRANSFER REACTOR EXPERIMENT No.3

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume describes Heat Transfer Reactor Experiment No. 3 (HTRE No. 3), a solid-moderated nuclear power plant with a horizontal reactor. The objectives and accomplishments of the program are presented in addition to nuclear, thermodynamic, and control system design data. The power plant and components, including the reactor, shield, turbomachinery, controls, and test support equipment are described, and the low-power and operational tests are discussed. Manufacturing techniques, component testing, and materials developments are also presented.

The objective of the HTRE No. 3 program was to provide the technical information needed for the design of a ground test prototype power plant and to test methods of design analysis and performance prediction.

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APEX-907

XMA-1 NUCLEAR TURBOJET

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This portion describes the XMA-1 which was the first developmental model of a power plant designed for operational applications.

The XMA-1 was designed to be a nuclear powered turbojet aircraft power plant consisting of a direct air cycle reactor, nuclear shielding, two parallel sets of X211 turbomachinery, and the required ducting, interconnecting structure and controls. Provisions were made for operation on either nuclear or chemical heat source.

Initially, the XMA-1 was to satisfy the requirements which included (1) cruise at Mach 0.9, 20,000 feet altitude; (2) low level attack at Mach 0.9, 500 feet altitude; and (3) sprint at Mach 2.5, 55,000 to 60,000 feet altitude. These were later changed to (1) cruise at 30,000 feet altitude at Mach 0.9 and (2) flight at Mach 0.9 at sea level.

The power plant proposed for first flight test was designated XMA-1A. The objective of the first power plant was a system that would represent as closely as possible the requirements and characteristics identified for the operational version. The operational version was designated XMA-1C.

This report presents a description of the XMA-1A design, design requirements, design data, and major test results, as well as the results of the studies directed toward selection of a reactor for the XMA-1C.

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APEX 908

XNJ 140E NUCLEAR TURBOJET

ABSTRACT

This volume is one of twenty-one summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. It is a comprehensive technical report of the design and development activities of the XNJ140E Project. Included are a presentation of the design objectives and requirements, an engineering description of the XNJ140E-1 nuclear turbojet engine, supporting analytical design data and methods of calculation, and a brief review of three design studies preceding, and directly applicable to the XNJ-140E program.

Beginning early in 1960, a major phase of the national effort leading to the achievement of nuclear powered flight was the design and development of the XNJ140E-1 nuclear turbojet engine to be utilized in an Advanced Core Test program. This program was to demonstrate the capabilities of a ceramic reactor coupled with the appropriate associated components of a direct-air-cycle nuclear turbojet engine. Descriptive material contained in this report is based upon the status of the XNJ140E Project at the time of contract termination.

The XNJ140E-1 engine was designed with a reactor of sufficient capability to provide engine performance equivalent to that specified in Department of Defense guidance, which required a speed of Mach 0.8 at an altitude of approximately 35,000 feet in a Convair Model NX 2 aircraft, or equivalent, and an engine life potential of 1000 hours. During this flight condition, the estimated minimum net thrust of the engine was 8120 pounds.

The engine contained a reactor-shield assembly coupled with a single set of X211 turbomachinery and arranged in an integral, in-line configuration. The compressor and turbine were separated, but connected by a long coupling shaft. An annular combustor system, using JP-4 jet fuel, was placed

in-line between the reactor rear shield and the turbine inlet, and was arranged concentrically around the coupling shaft.

The reactor-shield assembly was a prototype of comparable components to be used in subsequently planned flight versions of the engine. Turbomachinery components of improved design and an operational afterburner also would have been used.

The reactor fuel elements were made of a beryllium oxide matrix impregnated with enriched uranium dioxide (~93% U^{235}); the uranium dioxide was stabilized with yttrium oxide to limit the conversion of uranium dioxide to higher states of oxidation. Fuel element surfaces exposed to high velocity cooling air were coated with zirconium oxide stabilized with yttrium oxide; this coating eliminated water vapor corrosion of the beryllium oxide. The maximum operating temperature was 2530°F.

Beryllium oxide was used in the front, rear, and outer reflectors. Aluminum oxide was used as the inner reflector and served as thermal insulation between the core and the coupling shaft. Beryllium and stainless steel were used as shielding material in the end shields; each material was used both borated and unborated. Lithium hydride, sealed in stainless steel cans, was used as shielding material in the side shield.

This over-all report is divided into four parts. Part A contains section 1., a summary of the report and significant terminology; section 2., precedent studies leading to the selection of the XNJ140E power plant; and section 3., a description of the over-all power plant. Part B contains section 4., a description of the reactor. Part C contains section 5., a description of the shield; section 6., a description of the turbomachinery; and section 7., a description of the control system. Part D contains section 8., a description of test planning, special engineering data instrumentation, and test installations for the Advanced Core Test program; section 9., a discussion of remote handling and maintenance; and section 10., a discussion of on-site and off-site hazards associated with the operation of the engine during the Advanced Core Test program.

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APEX-909

AIRCRAFT NUCLEAR PROPULSION SYSTEMS STUDIES

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume describes the advanced systems which were studied and identifies those from which hardware-oriented programs were derived.

In the earlier periods, the objectives of the advanced system studies were the exploration of the numerous possible performance and design parameters in order to determine the most promising design approaches. The parameters investigated included flight speed and altitude, number of engines per reactor, number of reactors per propulsion system, core configuration, and materials and shapes of core components. Later, the objectives of the advanced system studies were to incorporate the technology advances in nucleonics, materials, and engineering into new system designs in order to obtain increased performance or new mission capability. An additional purpose was to identify the developmental data required to further a particular reactor or propulsion system concept to the final design phase.

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APEX-910

AIRCRAFT NUCLEAR PROPULSION APPLICATION STUDIES

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This portion describes the studies of advanced applications of nuclear reactors that were performed, including various types of aircraft, missiles, space vehicles, ships, and portable power plants. In part, the studies are based on the advanced power plants described in APEX-909, "Aircraft Nuclear Propulsion Systems Studies." Although most of the work was concerned with open-cycle, gas-cooled systems, other systems were also investigated, such as closed gas cycle, indirect liquid metal cycle, gas fission, and liquid circulating fuel systems. Air, helium, hydrogen, and neon were considered as coolants for the gas-cooled systems. Except for a portable nuclear system for power generation, all the studies were concerned with propulsion applications.

The application studies show the feasibility both of using reactors developed during the ANP program in advanced vehicles, and of the use of advanced reactors in various types of systems. Performance data, configurations, development program elements and schedules, and estimated costs are included in this summary of the results of the studies.

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APEX 911

REMOTE HANDLING

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume describes the remote handling tools and techniques developed in the course of the program.

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APEX-912

CONTROLS AND INSTRUMENTATION

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume is a compilation of reports on the more worthwhile technology and hardware development efforts for the control system of an aircraft nuclear power plant. However, the major part of this volume is devoted to the development efforts applied to the reactor control rather than to the turbomachinery control.

This volume omits those developments that were applied to specific power plants and their associated controls, since they are covered in individual volumes of this Comprehensive Technical Report.

The information presented is divided into four sections:

Control System Technology - This section covers the philosophy of control, analysis and simulation techniques, and some of the systems that were proposed and investigated. Power plant simulation is included in this section, although it was used as a design and analysis tool for three power plants.

Mechanical Components - This section describes the various actuator designs that were considered as possible solutions to such problems as high temperature, poison rods located behind the turbomachinery compressor, and backup safety actuators.

Electrical Components - This section covers all of the control hardware except for the actuators and sensors. Much of the electrical component effort was directed toward obtaining equipment capable of operating in the nuclear and temperature environment of a supersonic aircraft while meeting established reliability, size, and weight requirements.

Instrumentation - This section describes the efforts on the sensors, primarily for measurement of temperature and neutron power level, that were necessary for the control of the reactor.

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APEX-913

METALLIC FUEL ELEMENT MATERIALS

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This portion describes work on Metallic Fuel Element Materials. Information on properties, fabrication procedures, test data, and technical developments in high-temperature gas-cooled metallic nuclear fuel element technology are presented. Nuclear fuel element materials suitable for gas-cooled reactors with gas exit temperatures of from 1200°F to somewhat higher than 2000°F are discussed and test results summarized.

The general objective of the fuel element materials development program was to provide materials for the design of reactors for militarily useful aircraft power plants. Service lives from a few hours to a thousand hours were considered, depending upon the mission contemplated.

The report includes discussions and data on nuclear fuel elements containing dispersed uranium dioxide with a matrix and cladding of stainless steel, nickel-chromium alloy, iron-chromium-yttrium alloy, iron-chromium-aluminum alloy, niobium alloy, or chromium alloys.

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APEX-914

CERAMIC REACTOR MATERIALS

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume deals principally with research, development, and manufacture of ceramic core components developed for use in an annular, beryllium-moderated, air-cooled reactor for nuclear flight application. Feasibility studies on silicon carbide and molybdenum disilicide as fuel-bearing media are discussed. Fabrication techniques and properties of the various ceramic materials used in fuel elements, reflectors, shaft-housing shields, radial arches, and transition pieces are described. The technology of the preparation, fabrication, and properties of beryllium intermetallics which was undertaken to develop core components for advanced reactor concepts, is discussed. In-pile materials tests are summarized.

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APEX-915

SHIELD MATERIALS

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This portion discusses the shielding materials development, which was begun in 1956. Specialized materials, which would provide the most efficient neutron and gamma shielding per unit of weight and would be capable of high temperature (900° to 1600°F) operation under the thermo-mechano-nuclear environment associated with the direct-cycle reactor power plant, are presented.

Primary development effort was focused on three neutron shielding materials: (1) beryllia plus boron, (2) beryllium plus boron, and (3) lithium hydride. In addition, stainless steel plus boron and tungsten-base alloys with boron additions were evaluated as combined gamma-neutron shielding materials.

Fabrication technologies that were developed, physical and mechanical properties that were determined, and the environmental testing programs, including radiation stability studies used to prove the usefulness of the materials under simulated operating conditions, are discussed in their respective sections.

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APEX-916

MODERATOR MATERIALS

ABSTRACT

The development of high-temperature, highly efficient moderator materials, which include oxides, metals, metal hydrides, and intermetallics, is summarized. Operating temperatures generally vary from maximums of 1200° to 2700°F but some extend to 5000°F and higher. Fabrication methods, physical properties, and sources of additional materials information are presented. The development of massive zirconium hydride, massive yttrium hydride, and other moderator materials is described.

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APEX 917

ORGANIC, STRUCTURAL AND CONTROL MATERIALS

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This portion discusses the radiation effects on organic materials used in accessories and controls for the nuclear power plant. High-temperature control rod development, radiation effects on high-temperature structural material, and thermocouple development are also discussed.

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APEX-918

REACTOR AND SHIELD PHYSICS

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume describes the experimental and theoretical work accomplished in the areas of reactor and shield physics.

The reactor physics technology for all ANP reactor types is presented in its most advanced stage; i.e., no attempt is made to present chronologically the development of the technology.

The use of automated techniques for power-mapping critical experiments in the reactor physics program are discussed, with particular attention to the use of high speed computer programs employing the IBM 704 and IBM 7090 computing systems.

In the nuclear shielding program, efforts were concentrated in two main areas: (1) the cptimum placement of shield materials to reduce radiation levels, and (2) the calculation of specific nuclear data, such as nuclear heating and activation, which are important to the design of an efficient, safe power plant.

Methods were developed for determining, at any position in the reactor-shield assembly, the total flux and the angle and energy distribution of neutron and gamma rays, as well as the response of any detector used to measure radiation effects.

Important shielding computer codes described are the point kernel and single scattering codes and the more recently developed Monte Carlo codes.

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APEX-919

AEROTHERMODYNAMICS

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume summarizes the methods and techniques developed for use in the thermal design of nuclear reactors associated with that program.

Information and references are given on the analytical and experimental work required to design and evaluate the proposed high performance air cooled fuel elements. Methods of optimizing the thermal designs, particularly by the use of high speed electronic digital computing equipment, are discussed. The computer programs developed to provide accurate performance predictions, are identified and described. Details of the computing programs may be found in the referenced material.

Means for matching the coolant-flow to the predicted internal heat generation rates in the non-fueled components are discussed. Test methods and results are indicated and significant equipment and instrumentation information provided.

The relationships of reactor pressure losses and of localized perturbances to power plant performance are indicated, and the detailed analyses which were required to identify and predict these effects are discussed.

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APEX-920

APPLIED MECHANICS

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This report contains brief descriptions of the methods and techniques that were used by General Electric's Aircraft Nuclear Propulsion Department to achieve structural integrity in developmental nuclear flight power plants. Vibrational and impact methods were used to evaluate ceramic-core structural mockups under several conditions of mounting. Experimental and theoretical stress-analysis technology was used to extend existing information in high-temperature environments and in complicated stress fields. IBM programs were set up for computing thermal stresses. Progress was made in photoelasticity through the use of the techniques of photostress and photothermoelasticity. The state of the art of high-temperature strain gages was advanced by raising the temperature limits and by developing and manufacturing a precision device for fabricating strain gages. Standard procedures for installation of strain gages were developed, evaluated, and published. Reactor control elements were developed to regulate reactor power output. Relaxation characteristics of high-temperature alloys under stress were evaluated at elevated temperatures. The results of stress tests performed at elevated temperatures on molybdenum clad with protective coatings are reported. Investigational work was done on material friction and wear at elevated temperatures. Several different designs of heat and airflow sources are presented for use in reactor-component testing. Test data of the properties of several thermal-insulating materials are presented. Several configurations of metallic fuel elements were structurally evaluated under conditions of reactor temperatures and gas flow rates.

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APEX-921
NUCLEAR SAFETY

ABSTRACT

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. The highlights of the technical effort to evaluate, understand, and ameliorate aircraft reactor hazards are presented. The program included unique field-release studies, safety-fuse studies, experiments to measure fission-product release from molten fuel, analytical and experimental methods for estimating the consequences of a reactor runaway, criticality safety, and predicted consequences of a nuclear-powered-aircraft crash. Also included is an account of several analytical and experimental developments pertaining to reactor hazards which were employed during the course of the aircraft nuclear propulsion work at the General Electric Company.

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