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History

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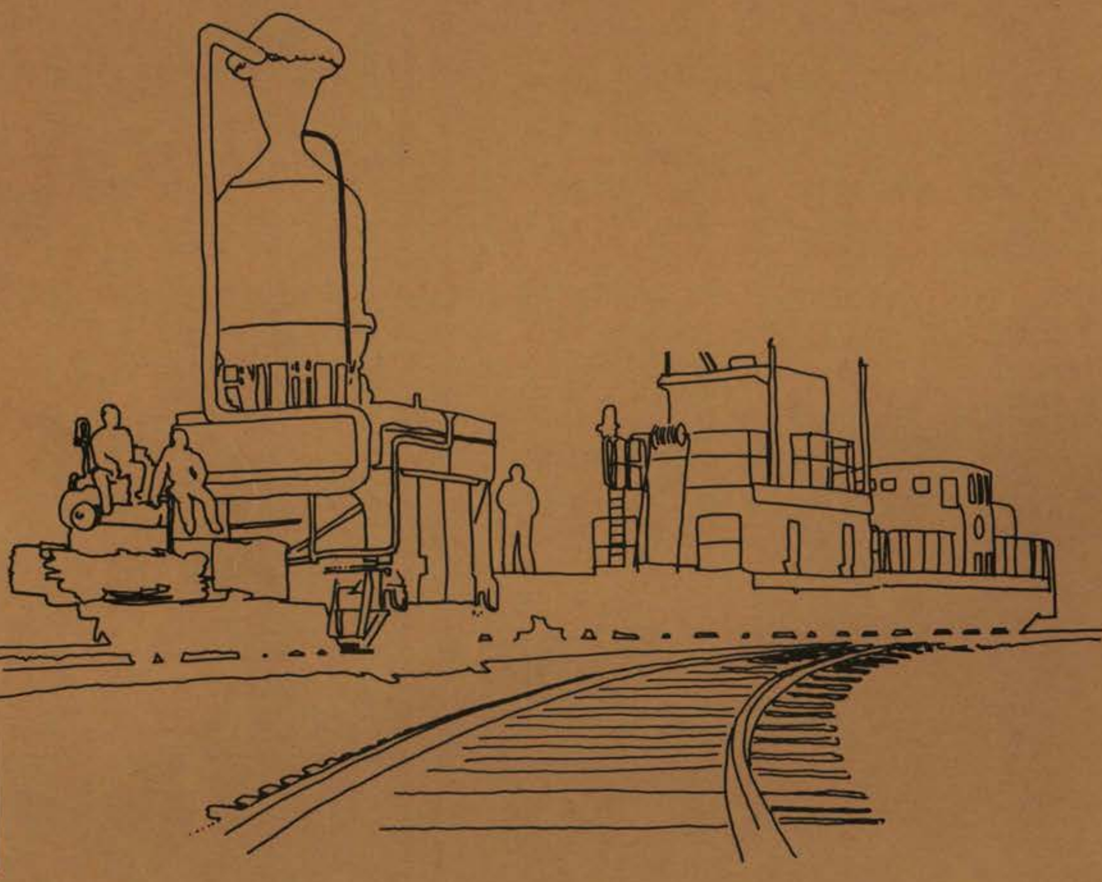

Experience Gained from the Space Nuclear Rocket Program (Rover)

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by

Daniel R. Koenig

ABSTRACT

In 1955 the United States initiated Project Rover to develop a nuclear rocket engine for use in defense systems and space exploration. As part of that project, Los Alamos developed a series of reactor designs and high-temperature fuels. Three high-power reactor series culminated in Phoebus, the most powerful reactor ever built, with a peak power level of 4080 MW. Two low-power reactors served as test beds for evaluation of high-temperature fuels and other components for full-size nuclear rocket reactors. Los Alamos developed and tested several fuels, including a fuel consisting of highly enriched UC_2 particles, coated with pyrolytic graphite, and imbedded in a graphite matrix and a composite fuel that formed a continuous web of uranium zirconium carbide throughout the graphite matrix. The program produced the design of the Small Engine, with a possible lifetime of several hours in space.

The Astronuclear Laboratory of the Westinghouse Electric Corporation, having responsibility for developing a prototype reactor based on the Los Alamos design, conducted an extensive and successful test series that culminated with the NRX-6 reactor test that ran continuously for 60 minutes at design power.

Aerojet-General Corporation, prime contractor for development of a complete rocket engine, developed two engine test series, the NRX/EST and the XE', to evaluate startup, full-power, and shutdown conditions in a variety of altitude and space simulations.

The United States terminated Project Rover in January 1973 at the point of flight engine development, but testing had indicated no technological barriers to a successful flight system. Conceptual studies also indicated that nuclear rocket engine technology could be applied to the generation of electric power in space.

I. OVERVIEW

In 1955 the United States embarked on a program to develop a nuclear rocket engine. The program, known as Project Rover, was initiated at Los Alamos National Laboratory, then called Los Alamos Scientific Laboratory. The concept to be pursued was a solid-core, hydrogen-cooled reactor in which the exiting gas expanded through a rocket nozzle and discharged in space. The motivation for the development of such a rocket engine was that it could provide about twice the specific impulse[1] of the best chemical rockets and, correspondingly, a reduction by a factor of 5 in the ratio of take-off mass to final mass at earth-escape velocity, as exemplified in Fig. 1. In January 1973, after a total expenditure of approximately one and a half billion dollars, the program (although judged a technical success) was terminated because of changing national priorities.

The expected application for nuclear rocket engines changed several times during the course of the program. At first, nuclear rockets were considered a potential back-up for intercontinental ballistic missile (ICBM) propulsion. Later they were mentioned as a second stage for lunar flight. A more durable possibility was their use in manned Mars flights. After plans for manned

Mars flights were abandoned as too ambitious, the final possibility advocated for nuclear engines was earth orbit-to-orbit transfer.

When analysis showed chemical rockets to be more economical for orbit-to-orbit missions, the need for a nuclear engine for rocket vehicle application (NERVA) evaporated, and the program was canceled before achievement of a flight demonstration. The design and the objectives of the NERVA are shown in Fig. 2. Most of the design objectives were met or exceeded during the course of the program.

The NERVA in Fig. 2 is attached to a slightly pressurized, liquid hydrogen tank. During operation, the hydrogen is fed to the engine by a turbopump. The high-pressure fluid first regeneratively cools the nozzle and the reactor reflector as shown in Fig. 3, then passes through the reactor core. Not shown in Fig. 3 is a parallel coolant circuit to cool the core-support tie rods; in the circuit the coolant is heated sufficiently to drive the turbopump before the coolant rejoins the main flow at the reactor inlet. The core contains solid hexagonal fuel elements banded together by lateral support springs. Longitudinal holes in the fuel elements provide coolant channels for the hydrogen propellant, which is heated to 2400-2700 K and finally expanded through a thrust nozzle. Rotating drums in the reflector containing neutron absorber material provide reactivity control of the reactor, which has an epithermal neutron energy spectrum.

The aim of the Rover program, besides designing and demonstrating a practical rocket engine, was to achieve the highest-possible propellant temperature (specific impulse is proportional to the square root of the temperature) for the duration of potential missions (several hours). This goal implied a strong technology development program in reactor fuels.

Los Alamos National Laboratory was given the role of establishing a basic reactor design and of leading the fuels development effort. Other key players were the Aerojet-General Corporation, the prime contractor to develop the complete rocket engine system, and the Astronuclear Laboratory of the Westinghouse Electric Corporation (WANL), the principal subcontractor to develop the NERVA nuclear reactor.

A series of reactors and engines was tested at the Nuclear Rocket Development Station (NRDS) in the test site at Jackass Flats in Nevada where major testing facilities were built for the Rover program (Fig. 4). These included an assembly and disassembly facility and two testing facilities for the research and engine reactors. The testing program for the nuclear rocket reactors is summarized in Fig. 5. It was initiated with a family of research reactors named Kiwi (for the flightless bird of New Zealand). The program objectives were first to demonstrate the proof of principle, then to establish the basic reactor technology and develop sound design concepts. These reactors were the first to demonstrate the use of high-temperature fuels and to operate with liquid hydrogen. The Kiwi testing series culminated with the Kiwi-B4E reactor, which operated for 11.3 min at a coolant exit temperature above 1890 K and for 95 s at 2005 K and a power level of 940 MW. These tests led to the Nuclear Reactor Experiment (NRX) series of developmental reactors. Their goal was to demonstrate a specific impulse of 760 s (7450 m/s) for 60 min at a thrust level of 245 kN (55 000 lb) in a 1100-MW reactor. These objectives were exceeded in the last test of that series, the NRX-6 reactor, which operated for 62 min at 1100 MW and a temperature of 2200 K, with only an $0.11 reactivity loss.

Another series of research reactors, called Phoebus, was developed with objectives to increase the specific impulse to 825 s, increase the power density by 50%, and increase the power level to the range of 4000-5000 MW. These capabilities were demonstrated in the Phoebus-1B and Phoebus-2A reactors. The latter, the most powerful reactor ever built, ran for 12 min at 4000 MW and reached a peak power of 4080 MW. The last two families of research reactors, Pewee and the Nuclear Furnace (NF), were tested only once each. They were lower-power reactors, 500 and 44 MW respectively, designed primarily as test beds to demonstrate the capabilities of higher-temperature fuel elements. Pewee-1 ran for 40 min at 2555 K, and NF-1 operated for 109 min at an average coolant exit temperature of 2450 K.

An engine development test program was part of the technology demonstration. Its objectives were to test nonnuclear system components, determine system characteristics during startup, full-power, and shutdown conditions, evaluate control concepts, and qualify the engine test-stand operations in a downward-firing configuration with simulated altitude and space conditions. These objectives were met or exceeded in the Nuclear Reactor Experiment/Engine System Test (NRX/EST) and Experimental Engine (XE) programs. A prototype flight engine system, XE, consisting of a flight-type reactor with nonnuclear flight components, was tested in a space-simulated environment, performing some 28 starts and restarts.

A chronology of the major tests conducted during the Rover program is shown in Fig. 6.

The major emphasis of the reactor development program was to increase the reactor coolant exit temperature because the specific impulse is proportional to the square root of that temperature and to increase the operating time of the reactor. The success of this part of the program is illustrated in Fig. 7. Coolant exit temperatures above 2500 K and operating time over 2 h were demonstrated. The cumulative time-at-power for the entire Rover program is shown in Fig. 8. The major performances achieved during the program are summarized in Fig. 9.

The Rover program was terminated before all of the NERVA objectives could be demonstrated, in particular, before showing that an engine could be operated for 10 h with up to 60 starting cycles with a reliability of 0.995.

Toward the end of the program, emphasis was being placed on smaller engines for the orbital transfer mission. A comprehensive design study was done on a 367-MW, 72-kN (16 000-lb) engine, the so-called Small Engine.[2,3] The total mass of this engine was 2550 kg, and its overall length was 3.1 m with the nozzle skirt in a folded position. The engine, together with a hydrogen tank containing nearly 13 000 kg of propellant, could be carried on the space shuttle. For comparison, the mass of several Rover reactors is plotted versus power in level in Fig. 10.

It was also recognized that the design of a nuclear rocket engine could be altered so as to provide continuous station-keeping power for the missions. Design studies for such dual-mode rocket systems were initiated in 1971-72 where one mode was the normal propulsion and the second, a closed-loop, low-power electrical mode.[4,5]

The Rover program was terminated in January 1973 at the point of flight engine development. For a flight system, it would be necessary to verify the flight reactor and engine design and to perform life and reproducibility testing. But there are no apparent barriers to a successful nuclear rocket.

The technology developed during the Rover program is directly applicable to the generation of electrical power in space, especially large (multimegawatt) bursts of electrical power. For an open-loop converter system, one would simply replace the rocket nozzle with a power conversion system. Some redesign of the core parameters would be involved because the power converter, unless it were a magnetohydrodynamic (MHD) system, could not operate at the high temperature of the Rover reactors. The startup time for such a power plant would be limited in part by the allowable rate of reactor temperature change, about 83 K/s. However, a more severe limitation is in the propellant feed system, which requires approximately 60 s to overcome pump cavitation before chill-down and to chill various parts of the engine. In addition, there would be time limitations imposed by the power conversion system.

A closed-loop system would require further redesign to incorporate the gas circulators, and the core design would have to be adjusted for the higher inlet temperatures.

As concerns dual electrical-power modes (a continuous, low-power mode and a short-term, high-power mode) much of the technology and many studies developed under the Rover program are applicable if the high-power converter is to be a gas system.

II. HISTORICAL PERSPECTIVES

This chapter summarizes the major events in the history of the Rover program. Information for the test summaries was obtained primarily from Refs. 6-8.

1945-1954

In 1945, at the suggestion of Theodore von Karman, the USAF Scientific Advisory Board studied the use of nuclear propulsion for rocket systems. However, because of a lack of a clear need for such systems, the shortage of fissionable materials, and the technical difficulties of developing such a propulsion system, no action was recommended. Nevertheless, paper studies of nuclear rocket systems were performed during this period.[9,10]

1954

Von Karman again suggests that, in view of the need for ICBMs and the good supply of fissionable material, the Scientific Advisory Board reconsider nuclear propulsion.

1955

October 18. In a final report, an ad hoc committee of the Scientific Advisory Board recommends that because of the potentially high specific impulses within the realm of immediate achievement from the nuclear rocket, substantial development work should be started on the nuclear rocket system.

November 2. The nuclear rocket propulsion program is established as Project Rover at Los Alamos and Lawrence Livermore Scientific Laboratories. Several conceptual nuclear rocket designs already had been under study.[11] But the concept chosen to be pursued was a solid-core, hydrogen-cooled reactor that would expand gas through a rocket nozzle.

1957

March 18. The Atomic Energy Commission (AEC) decides to phase Livermore out of the program as a result of budget restrictions and a Department of Defense recommendation for a more moderate level of support. The latter stemmed from the earlier-than-anticipated availability of chemical ICBMs, which reduced the urgency for development of nuclear propulsion.

1959

July 1. The first reactor test, Kiwi-A, is conducted successfully at the Nevada Test Site.[7,8] The reactor operated for 5 min at 70 MW and provided important design and materials information. The fuel was hot enough (2683 K) to melt carbide fuel particles. Vibrations in the core produced structural damage in the graphite elements. The reactor employed uncoated, UO_2-loaded, plate-type fuel elements and was cooled with gaseous hydrogen. The reactor core contained a central island of D_2O to reduce the amount of fissionable material required for criticality. Control rods were located in this island.

1960

July 8. Kiwi-A' is tested for nearly 6 min at 85 MW to demonstrate an improved fuel-element design. The reactor used short, cylindrical, UO_2-loaded fuel elements contained in graphite modules. The fuel element had four axial coolant channels coated with NbC by a chemical vapor deposition (CVD) process.

August 29. A Memorandum of Understanding defining NASA and AEC responsibilities and establishing a joint nuclear program office, the Space Nuclear Propulsion Office, is signed.

October 10. Kiwi-A3 reactor is operated in excess of 5 min at 100 MW. The fuel was similar to that used in the previous test. As with the earlier tests, core structural damage occurred, indicating that tensile loads on graphite structures should be avoided. This experiment was the third and last in the Kiwi-A series of proof-of-principle tests conducted by Los Alamos. The test series demonstrated that this type of high-power-density reactor could be controlled and could heat hydrogen gas to high temperatures.

1961

June-July. Industrial contractors, Aerojet-General for the rocket engine and Westinghouse Electric Corporation for the reactor, are selected to perform the nuclear rocket development phase. The reactor in-flight tests (RIFT) program was initiated at the Lockheed Corporation.

December 7. Kiwi-B1A reactor, first of a new series, is tested by Los Alamos. Kiwi-B reactors were designed for 1100 MW and used reflector control and a regeneratively cooled nozzle. This test was the last to be run with gaseous hydrogen coolant. After 30 s of operation, a hydrogen leak in the nozzle and the pressure vessel interface forced termination of the run. The planned maximum power of 300 MW was achieved, as limited by the capability of the nozzle with gaseous hydrogen coolant.

The core consisted of cylindrical UO_2-loaded fuel elements, about 66 cm long, having seven axial coolant channels NbC coated by a tube-cladding process. The fuel elements were contained in graphite modules.

1962

September 1. Kiwi-B1B reactor test is the first to operate with liquid hydrogen. The test met its primary objective of demonstrating the ability of the system to start up and run using liquid hydrogen. Following a smooth, stable start, the run was terminated after a few seconds at 900 MW when portions of several fuel elements were ejected from the reactor. The core employed the same type of fuel as Kiwi-B1A.

November 30. Kiwi-B4A, the first design intended as a prototype flight reactor, is tested. The power run was terminated at about the 50% level when bright flashes in the exhaust (caused by ejection of core material) occurred with increasing frequency. Subsequently, intensive analyses and component testing were conducted to determine the cause of the core damage. The core consisted for the first time of full-length, extruded, 19-hole, hexagonal fuel elements, loaded still with UO_2. The coolant channels were NbC coated by the tube-cladding process.

1963

December. The RIFT program is cancelled. It was decided to revise the nuclear rocket program to place emphasis on the development of ground-based systems and defer the development of flight systems.

1963-1964

Several cold-flow tests of Kiwi-B-type reactors are carried out to determine the cause and find solutions for the severe structural damage that was observed in the previous reactor tests. The cold-flow designation referred to reactor tests that contained fuel elements identical to the power reactors except that they had no fissionable material and, therefore, produced no power. These tests were performed with gaseous nitrogen, helium, and hydrogen, and they demonstrated that the structural core damage was due to flow-induced vibrations. Based on results of these tests and analyses, design changes that were completely successful in eliminating core vibrations were made.

May 13. Kiwi-B4D, the first test at full design power, is carried out with no indication of core vibration. This was also the first time a completely automatic start was accomplished for a nuclear rocket reactor. The test was terminated after 60 s at full power when several nozzle tubes ruptured. The core consisted of full-length, UO_2-loaded, 19-hole, hexagonal fuel elements with bores NbC coated by the tube-cladding process.

August 28. Kiwi-B4E, the eighth and final Kiwi reactor, is tested by Los Alamos. The reactor was operated for more than 12 min, of which 8 min were at nearly full power. The reactor operation was smooth and stable. Its duration was limited by the available liquid hydrogen storage capacity. On September 10, the reactor was restarted and ran at nearly full power for 2.5 min. This was the first demonstration of the reactor's ability to restart.

The core consisted of full-length, 19-hole, hexagonal fuel elements, loaded for the first time with UC_2 particles. The bores were NbC coated by the tube-cladding process.

September. Measurements, at zero power, of the neutronic interaction of two Kiwi reactors positioned adjacent to each other verify that there is little interaction and that, from a nuclear standpoint, nuclear rocket engines may be operated in clusters similar to chemical engines.

September 24. NRX-A2 is the first NERVA reactor tested at full power by Westinghouse Electric.[7] The reactor operated in the range of half to full power (1100 MW) for about 5 min, a time limited by the available hydrogen gas supply. The test was successful and demonstrated an equivalent vacuum specific impulse of 760 s. The reactor was successfully restarted on October 15 to investigate the margin of control in the low-flow, low-power regime.

1965

January 12. Kiwi-TNT (Transient Nuclear Test) is successfully completed by Los Alamos. In this flight safety test, a Kiwi-B-type reactor was deliberately destroyed by placing it on a fast excursion to confirm the analytical models of the reactor behavior during a power excursion.[12,13]

April 23. NRX-A3 reactor is operated for about 8 min with about 3.5 min at full power. The test was terminated by a spurious trip from the turbine overspeed circuit. The reactor was restarted on May 20 and operated at full power for over 13 min. It was restarted again on May 28 and operated for 45 min in the low- to medium-power range to explore the limits of the reactor operating map. The total operating time of the reactor was 66 min with over 16.5 min at full power.

June 25. The aims of Phoebus-1A, the first test of a new class of reactors, were to increase the specific impulse, the power density in the core, and the power level. The test is run successfully at full power (1090 MW) and core exit temperature (2370 K) for 10.5 min. The reactor was subsequently damaged when the facility's liquid hydrogen supply was exhausted. This course of events was in no way related to any defect in the reactor. The core consisted of full-length, 19-hole, hexagonal fuel elements loaded with coated UC_2 particles. The bores were NbC clad by the chemical vapor deposition (CVD) process.

1966

February 3 to March 25. The NRX/EST, the first NERVA breadboard power plant, is operated during 5 different days for a total of 1 h and 50 min, of which 28 min were at full power (1100-1200 MW). These times were by far the greatest achieved by a single nuclear rocket reactor as of that date.

June 8. NRX-A5 is operated successfully at full power for 15.5 min. It was restarted and operated again at full power (~1100 MW) on June 23 for 14.5 min to bring the total operating time at full power to half an hour. The liquid hydrogen capacity of the test facility was not sufficient to permit 30 min of continuous operation at design power.

1967

February 23. Phoebus-1B is operated for 45 min of which 30 min, the maximum time planned, were at design power of 1500 MW. The primary purpose of the test was to determine how the higher-power operation affected the reactor. The fuel was the same as that used in Phoebus-1A.

December 15. NRX-A6 test exceeds the NERVA design goal of 60 min at 1100 MW in a single run.

1968

June 26. Phoebus-2A, the most powerful nuclear rocket reactor ever built, runs for 12.5 min above 4000 MW. The duration of the test was determined by the available coolant supply. Designed for 5000 MW, the test was limited to 80% of full power because the aluminum segments of the pressure vessel clamp band overheated prematurely. The reactor was restarted on July 18 and operated at intermediate power levels.

December 4. Pewee reactor testing is successfully completed. Pewee, designed to be a small test-bed reactor, set records in power density and temperature by operating at 503 MW for 40 min at a coolant exit temperature of 2550 K and a core average power density of 2340 MW/m^3. This power density was 50% greater than that required for the 1500-MW NERVA reactor. The peak power density in the fuel was 5200 MW/m^3. The coolant exit temperature corresponds to a vacuum specific impulse of 845 s. The core contained the same type of fuel elements as Phoebus-1A.

March. XE', the first down-firing prototype nuclear rocket engine, is successfully operated at 1100 MW. The reactor was operated at various power levels on different days for a total of 115 min of power operation that included 28 restarts. Individual test times were limited by the facility's water storage system, which could not support operations longer than about 10 min at full reactor power. This test series was a significant milestone in the nuclear rocket program and demonstrated the feasibility of the NERVA concept.

In this year, the production of the chemical rocket Saturn V was suspended. It would have been the prime launch vehicle for NERVA.

1972

June 1. NF-1 test is successfully accomplished. The reactor was operated for 109 min at the full design power of 44 MW, demonstrating fuel performance at a coolant exit temperature to 2500 K and a near-record peak power density in the fuel of 4500-5000 MW/m^3. NF-1 was designed with a remotely replaceable core in a reusable test bed, intended as an inexpensive approach to multiple testing of advanced fuel materials and structures. Another special feature of this test series was evaluation of a reactor effluent cleanup system. The system performed as expected in removing radioactive contaminants from the effluent reactor gas.

Two types of fuel elements, (UC-ZrC)C "composite" fuel and the pure (U,Zr)C carbide fuel, were tested in NF-1.

1973

January. The Rover nuclear rocket program is terminated. It was judged a technical success, but changing national priorities resulted in the decision to cancel the program.

III. REACTOR DEVELOPMENT

The concept of a nuclear rocket engine is simple. As shown schematically in Fig. 11, it consists of a cryogenic propellant tank, a turbopump to feed the propellant through the system, a nuclear reactor to heat the propellant to the highest temperature possible, and a thrust nozzle through which the hot gas is expanded. The propellant is hydrogen because a gas with the lowest-possible molecular weight is most desirable.

The reactor design goals presented a real challenge in reactor design and materials development. The core exit temperature of the coolant had to be maximized to achieve the highest-possible specific impulse. The core power density also had to be maximized to minimize reactor mass. To achieve a practical engine longevity (initially 1 h, then 10 h), it was necessary to minimize hydrogen corrosion of the fuel and breakage of the core from vibrational and thermal stress. Only a few materials, including the refractory metals[11] and graphite, are suitable for use in reactors designed to run at high temperatures (up to 2700-2800 K). Graphite was selected because in contrast to the metals, it is not a strong neutron absorber, and it does moderate neutrons leading to a reactor with a smaller critical mass of enriched uranium. Graphite has excellent high-temperature strength, but its great disadvantage is that it reacts with hot hydrogen to form gaseous hydrocarbons and, unless it is protected, it rapidly erodes. Consequently, one of the greatest challenges of the nuclear rocket program was to develop fuel elements of adequate lifetime in high-pressure hot hydrogen.

The designers of the nuclear rocket engine had to consider many factors such as neutronic and heat-removal requirements; high mechanical loadings; and the complex problems of startup, control, shutdown, and safety. To permit preliminary evaluation of the neutronic calculations, a mockup or critical assembly of each reactor type, known as Honeycomb, was built as shown in Fig. 12.[14,15] It consisted of graphite slabs, enriched uranium foils, plastic to simulate the propellant, and beryllium-reflector blocks. Later, during construction of each new type of reactor, a more exact mockup of the final reactor, known as Zepo (Zero Power), was built (Fig. 13) using actual fuel elements to determine the system's neutronics. Such testing facilities were built at Los Alamos and WANL. The actual reactor and engine tests were carried out at the NRDS.

A. Kiwi-A

The first reactor tested under the Rover program was named Kiwi-A. It was designed and built by Los Alamos as were all of the Kiwi series of reactors. The reactor design,[16] as shown in Fig. 14, was intended to produce about 100 MW of power. It was, in fact, tested for 5 min at 70 MW. The Kiwi-A core consisted of an annular stack of four axial layers of flat-plate, graphite fuel elements loaded with highly enriched UO_2 particles. The fuel elements were retained and supported in graphite structures called whims. The whims, shown in Fig. 15, were wheellike structures with 12 wedge-shaped boxes of fuel plates fitted between their spokes, each box containing 20 fuel plates. A fifth whim contained unloaded fuel plates and served as an end reflector for the outlet end of the core. The inlet and radial reflectors consisted of several continuous graphite cylinders. Power flattening was achieved by varying the fuel loading. The core was separated from the radial reflector by a carbon wool region. The hole in the center of the core contained a "D_2O island," the function of which was to moderate neutrons, thereby reducing the critical mass of ^{235}U, and also to provide a low-temperature, low-pressure container for the reactor control rods that were cooled by circulating D_2O. The entire reactor was encased in an aluminum pressure shell to which a light-water-cooled nickel nozzle was attached. The nozzle was designed for choked-flow outlet conditions for the core coolant (that is, sonic flow at the throat of the nozzle).

The hydrogen coolant flow through the reactor is as follows. Coolant is delivered to the plenum near the top of the pressure vessel. The gas then flows axially downward through holes in the reflector segments and into the plenum at the bottom of the pressure vessel where the flow reverses, passing upward through holes in the inlet reflector. The gas now continues upward between the fuel plates of each whim, through the unloaded plates of the top whim, and out through the nozzle.

The Kiwi-A experiment was a first step toward demonstrating the feasibility of a high-temperature, gas-cooled reactor for nuclear propulsion, and as such it provided important reactor design and materials information.[8,17]

Much higher fuel temperatures (up to 2900 K) than anticipated were reached during the test because early in the run the graphite closure plate, located just above the D_2O island, shattered and was ejected out of the nozzle along with the graphite wool between the center island and the core. The functions of this plate were to contain the carbon wool insulation and to serve as a gas seal that prevented gas from bypassing the annular core into the central region. Failure of the closure plate allowed a lot of gas to flow radially inward through slots in the inside wall of the whims (Fig. 15) and into the central part of the core, thereby bypassing the power-producing region of the core. This bypassed gas was not heated to full temperature. Because the test conditions demanded a prescribed average gas outlet temperature, it follows that the gas that did pass through the active core had to be heated to a higher temperature. The high fuel temperatures that resulted led to melting of the UC_2 fuel and high erosion of the graphite fuel plates.

For the next two reactors, the Kiwi-A core design was modified to replace the whims and fuel plates with graphite modules containing cylindrical fuel elements[18] as shown in Figs. 16 and 17. This modification entailed a complete change in the fuel fabrication process from pressing and molding to a new graphite extrusion process. The fuel cylinders were segmented in short lengths and six of them were stacked on top of each other in each hole of the graphite modules to make up a complete fuel module. The fuel cylinders contained four axial coolant channel holes that were coated by a CVD process with NbC to reduce hydrogen corrosion of the graphite. This modified core configuration was tested twice for 5 to 6 min in the power range 85-100 MW in the Kiwi-A'[19] and Kiwi-A3[20] tests. Fracture of fuel modules was experienced in both of these tests, but the general appearance of the fuel elements after each test was quite good even though several elements showed blistering and severe corrosion.

The Kiwi-A series of tests[21] demonstrated that hydrogen gas could be heated in a nuclear reactor to the temperatures required for space propulsion and that such a reactor could indeed be controlled.

B. Kiwi-B and NRX

Built on the experience gained with the Kiwi-A reactors, a new reactor design evolved that more nearly resembled what would be needed for a flight engine. The Kiwi-B test series was initiated with the Kiwi-B1A[22] test in December 1961 and culminated 2 years and 8 months later with the successful Kiwi-B4E test accomplished in August 1964. During this test series, improvements were made with the extruded fuel design and the protective NbC-coating technology. Severe structural damage to the core was experienced with the second test in the series (Kiwi-B1B)[23] when the hot ends of seven fuel modules were ejected from the core during the transient rise to full power. It took several subsequent full-power tests, in particular Kiwi-B4A, Kiwi-B4D,[24,25] and several coldflow tests to discover and confirm that core damage was caused by flow-induced vibrations and to demonstrate, after design modifications were applied, that a stable design had been achieved. This successful reactor configuration (Kiwi-B4E)[26-30] led to the NRX series[6] of NERVA developmental reactors from which emerged the final NRX-6 design[31,32] shown earlier in Fig. 3. The reactor was designed for a nominal power of 1100 MW. It was all graphite moderated, and it had an epithermal neutron spectrum. The extruded graphite fuel elements were hexagonal and contained 19 cooling channels. The channel walls and the exterior surfaces of the fuel element were coated with NbC to reduce hydrogen corrosion. The fuel was assembled in clusters of six elements supported by a tie rod in the central location as shown in Fig. 18. The tie rod was attached to an aluminum support plate at the cold end of the reactor. Irregularly shaped clusters were fitted on the core periphery to obtain a cylindrical core configuration. The core dimensions were 1.32 m in length and approximately 0.89 m in diameter. Lateral support for the core was obtained with a spring and a ring-seal arrangement as described in Fig. 19. Power flattening was achieved by varying the fuel loading, and the coolant flow distribution was controlled by orifices in the inlet end of each coolant channel, sized to provide approximately the same exit gas temperature for all channels. The core, which contained 182 kg of uranium (enrichment 0.9315), was surrounded by a graphite cylinder about 46 mm thick and a beryllium reflector 114 mm thick. Twelve rotating drums located in the reflector contained segments of boron carbide neutron absorber that could be swung toward or away from the core to provide reactivity control of the reactor. The entire reactor was encased in an aluminum pressure vessel to which the exhaust nozzle was attached. The pressure vessel was approximately 21 mm thick, 1.9 m in length, and 1.3 m in outer diameter.

The flow of hydrogen coolant through the reactor was as follows (Fig. 3): liquid hydrogen entered the aft end of the nozzle to cool the nozzle wall before entering the reflector plenum. From this plenum the hydrogen traveled forward through the reflector and control drums, also cooling the pressure vessel. It entered a plenum again before flowing forward through the outer region of the simulated shield. The flow discharged from the shield and entered the plenum region between the shield and the dome of the pressure vessel. Here the flow reversed, and the gas flowed aft through the inner region of the shield, then through a fine mesh screen and the core support plate. Most of the coolant then flowed through the channels in the fuel elements where it was heated to a high temperature. A small part of the flow cooled the periphery region between the core and the beryllium reflector, and some coolant also flowed past the tie rods in the core. These coolant flows were mixed in the nozzle chamber at the reactor exit before expulsion through the nozzle.

One aim of the developmental series of tests conducted by Westinghouse Electric was to reduce the fraction of coolant flow that did not pass through the fuel in order to obtain the highest-possible gas temperature in the nozzle chamber. This aim was achieved by applying design modifications described below for the Phoebus reactors. The duration of full-power runs was gradually increased with each NRX reactor until the test in December 1967 in which the NRX-A6 ran continuously for 60 min at 1125 MW with an exit coolant temperature at or above 2280 K, corresponding to a vacuum specific impulse of 730 s.[33-35] The test duration and power level exceeded the NERVA design goals at that time.

C. Phoebus

Following the successful performance of the Kiwi-B4E reactor, the Los Alamos Scientific Laboratory devoted its attention to a new class of reactors similar in design to Kiwi-B but having greater coolant exit temperatures, power densities, and power levels. Power density was to be increased mainly by enlarging the diameter of the coolant flow channels in the fuel elements from 2.54 mm to 2.79 mm to reduce thermal stress and core pressure drop. The temperature increase was to be obtained by some minor design modifications in the fuel elements but mostly by reducing the amount of coolant flow that bypassed the core. The coolant flow along the core periphery was reduced, and the single-pass cooling of the metal tie rods in the core was reduced and eventually changed to two-pass regenerative cooling by replacing the tie rods with tie tubes. These tubes were cooled by diverting 10% of the flow to the core support and returning this flow to the main core coolant flow at the inlet of the fuel elements. These coolant flow modifications greatly reduced the mixing of cold coolant with the core exit gas in the nozzle chamber. The power level was increased simply by increasing the number of fuel elements in the core.

Three tests, Phoebus-1A,[36] -1B,[37-39] and -2A, were carried out in this series. The first two tests were essentially vehicles for experiments leading to the Phoebus-2A design. Phoebus-2A (Fig. 20) designed for 5000 MW was the most powerful nuclear rocket reactor ever built. It was intended originally to be a prototype optimum-thrust nuclear propulsion engine for ambitious planetary missions. The reactor had a nominal thrust of 1110 kN (250 000 lbf) and a specific impulse of 840 s, corresponding to a nozzle chamber temperature of 2500 K.

The Phoebus-2A reactor[40-42] incorporated all of the features mentioned above. The core, which contained about 300 kg of uranium, consisted of 4068 fueled elements plus 721 regeneratively cooled support elements. The active core dimensions were 1.39 m in diameter and 1.32 m in length. The 19-hole fuel elements were similar in geometry and had the same external dimensions as those of earlier reactors (Kiwi-B4E to NRX-A6), but the coolant channel diameter was increased from 2.54 mm to 2.79 mm. The channels were coated with NbC of tapered thickness and were overcoated with a layer of Mo to reduce hydrogen corrosion of the graphite. As before, the fuel was assembled in clusters of seven elements where the central element was unloaded graphite containing the tie-tube axial support assembly. Details of construction and coolant flow paths for the regeneratively cooled tie tubes are shown in Fig. 21. The core-reflector interface had an aluminum interface cylinder assembly that separated the high-pressure reflector system region from the lower-pressure core periphery, transmitted the axial pressure-drop load to the nozzle, and contained the seal rings. This assembly was cooled by reflector bypass coolant. The 203-mm-thick beryllium-reflector assembly contained 18 control drums rather than 12 as used for the earlier smaller reactors. The reactor was contained in an aluminum pressure vessel 2.54 mm thick with an outside diameter of 2.07 m and an approximate length (excluding the nozzle) of 2.5 m. Reactor mass including the pressure vessel was 9300 kg. A two-dimensional model of the reactor that was used in neutronic calculations is shown in Fig. 22. Reactor control was obtained by adjusting two basic parameters, namely, the coolant flow rate and the control-drum position.

The successful full-power test of Phoebus-2A took place in July 1968 and lasted for 12.5 min, a time limited by the available hydrogen coolant supply (coolant, driven by two Rocketdyne Mark-25 turbopumps operating in parallel, flowed through the reactor at a rate of 120 kg/s). The maximum power level reached during the test was 4080 MW.

The reactor could not be operated up to the design power level of 5000 MW because part of the aluminum pressure vessel assembly was overheating prematurely as a result of unexpected poor thermal contact with an LH_2-cooled clamp ring. The maximum fuel-element exit-gas temperature attained was 2310 K, and the maximum nozzle chamber temperature, nearly as high, was 2260 K. This small temperature difference is an indication of the effectiveness of the measures taken to reduce mixing of cold coolant with the core exit gas. At design power, the core power density would have been nearly twice that of the Kiwi-B reactors.

The Phoebus-2A test revealed some neutronic discrepancies when compared with pretest calculations and zero-power criticality experiments.[43,44] Specifically, the full-scale reactor test resulted in larger cold-to-hot changes in reactivity than had been predicted. The anomalies were eventually resolved and attributed mainly to the combined effect of low beryllium-reflector temperatures and the presence of cold high-density hydrogen in the aluminum interface cylinder and in the reflector. The result was to produce a large negative change in reactivity and a substantial reduction in control-drum worth. Neither of these effects had been correctly accounted for in pretest analysis.

The successful conclusion of the Phoebus-2A tests was a milestone in nuclear rocket technology because of the high-power capability that the test demonstrated. Some problems remained, particularly in the area of fuel longevity and temperature capability, but the feasibility of practical nuclear space propulsion had been convincingly demonstrated by this stage of the Rover program. Phoebus-2A was the last reactor design in direct support of the NERVA development that was tested by Los Alamos. Two smaller reactor designs were subsequently tested by Los Alamos, but they were primarily test beds for improving the fuel technology.

D. Pewee

Pewee[45] was a small reactor designed to serve as a test bed for the evaluation of full-size Phoebus and NRX fuel elements and other components. The general design was directed toward providing a realistic nuclear, thermal, and structural environment for the fuel elements in a core containing one-fourth the number of elements in these reactors, and one-tenth the number of elements in Phoebus-2A.

Most of the basic design features of Pewee were similar to those of the preceding reactors. The fuel elements were similar, and they were held in place by support elements; the control drums were incorporated in the beryllium radial reflector; and liquid hydrogen was used as the working fluid. There were, however, significant differences that distinguished Pewee from earlier reactors. The core diameter was reduced from 1400 mm in Phoebus-2A to 533 mm to reduce the number of fuel elements. Sufficient reactivity with the smaller core was obtained by inserting sleeves of zirconium hydride around the tie rods in the support elements as shown in Fig. 23. The hydrogenous material moderated the core neutrons and reduced the critical mass of uranium in the core to 36.4 kg. The ratio of support elements to fuel elements was increased from 1:6 to 1:3, as illustrated in Fig. 24, to increase the amount of ZrH_x moderator to the desired level. This eliminated the traditional clusters-of-seven concept; each fuel element was supported redundantly by two pedestals. The core contained 402 fuel elements and 132 support elements. Because Pewee was designed as a test bed for fuel elements, no attempt was made to maximize the specific impulse by maintaining a high temperature in the nozzle chamber; the support-element coolant was discharged directly into the chamber. This discharge reduced the nozzle temperature significantly because the hydride moderator required a larger amount of coolant than a graphite support element without moderator and because a conservatively low coolant discharge temperature was chosen.

The small size of Pewee required a thick (205-mm) beryllium reflector that was built in two concentric parts as shown in Fig. 25. The inner part consisted of beryllium rings that replaced the interface cylinder of previous designs. The outer part was made from Phoebus-1-type sectors and contained nine control drums.

The mass of the Pewee reactor, including the aluminum pressure vessel, was 2570 kg.

The Pewee test series conducted in November-December 1968 was successful, and it set several records for nuclear rocket reactors. The primary objective was to demonstrate the capability of this new reactor as a fuel-element test bed. Pewee ran for a total of 192 min at power levels above 1 MW on two separate days. The full-power test consisted of two 20-min holds at design power (503 MW) and an average fuel-element exit-gas temperature of 2550 K. This temperature was the highest achieved in the Rover program. It corresponds to a vacuum specific impulse of 845 s, a level in excess of the design goal set for the NERVA. The peak fuel temperature also reached a record level of 2750 K. The average power density in the core was 2340 MW/m^3, also a record high and greater than that required for the NERVA. The peak power density in the fuel was 5200 MW/m^3. The fuel elements were similar to those of Phoebus-1A except for a few elements CVD-coated with ZrC instead of NbC. The ZrC-coated fuel elements performed significantly better.

The reactor performed close to design conditions except for an unexpected, large, radial variation of 220-310 K in the fuel-element exit-gas temperature and a heat pickup in the reflector 14% greater than predicted at full power. But the successful performance of the Pewee reactor design was important because it demonstrated that small reactors, with low-temperature moderating materials inside the core, could be operated in the configuration and in the extreme temperature environment of a rocket engine. A second test of the Pewee reactor had been planned, but Pewee-2 was never built.

E. Nuclear Furnace, NF-1

The last reactor test of the entire Rover program was that of the NF-1,[46,47] a reactor ten times less in design power than Pewee. The NF-1 was devised to provide an inexpensive means of testing full-size nuclear rocket reactor fuel elements and other core components in a reactor having a low fuel inventory. It was never meant to be a candidate concept for a rocket engine. The reactor, described in Figs. 26 and 27, consisted of two parts: a permanent, reusable portion that included the reflector and external structure; and a temporary, removable portion that consisted of the core assembly and associated components.

A major objective of this design was to have a reusable test device that would reduce both the time between reactor tests and the cost of testing. After completion of a test series, the core assembly would be removed and disassembled for examination, whereas the permanent structure would be retained for use with a new core. Actually, the NF-1 was tested only once before termination of the program, but the removable feature of the design was demonstrated.

The NF-1 core was a 34-cm-diameter by 146-cm-long aluminum can that contained 49 fuel elements as compared to 402 in Pewee. This core was surrounded by a 27-cm-thick beryllium radial reflector that accommodated six rotating control drums. The fuel inventory was about 5 kg of uranium (93% enriched). Sufficient reactivity for critical configuration with such a small fuel inventory was obtained by designing the core as a heterogeneous water-moderated thermal reactor. Each fuel cell contained a standard 19-hole, hexagonal fuel element encased in an aluminum tube as described in Fig. 28. The cell tubes were inserted inside aluminum sleeves, and water flowed through the core in two passes, first between the sleeves and the cell tubes to the aft end of the core, where the flow turned around and went back between the elements. The hydrogen coolant, after making several passes in the reflector assembly, made a single pass through the core within the fuel coolant channels.

The hydrogen exhaust gas was handled differently than in previous reactors. Instead of being exhausted through a convergent-divergent nozzle directly to the atmosphere, the hot hydrogen was first cooled by injecting water directly into the exhaust gas stream as shown in Fig. 29. The resulting mixture of steam and hydrogen gas was then ducted to an effluent cleanup system to remove fission products before release of the cleaned gas to the atmosphere.

The primary objectives of the NF-1 test series were to verify the operating characteristics of the NF-1 and associated facilities and to operate at full power with a fuel-element exit-gas temperature of 2440 K for at least 90 min. All primary objectives were attained during the test series. A wealth of data was obtained on the dynamic and static characteristics of the NF-1 and the facility, and no major NF-1 design deficiencies were found.

The reactor was operated at the design power of 44 MW and a fuel-element exit-gas temperature of approximately 2440 K for a record time of 109 min and at or above 2220 K for 121 min. The maximum exit temperature reached was about 2550 K. Two new types of fuel elements were tested in NF-1. They were the (U,Zr)C graphite (composite) elements that comprised 47 of the 49 fuel cells in the core and two cells containing (U,Zr)C (carbide) elements. The carbide elements withstood peak power densities of 4500 MW/m^3 but experienced severe cracking. These elements were small (5.5 mm across the flats), hexagonal elements with a single 3-mm-diameter coolant hole. Redesign, by reducing the web thickness by 25%, would substantially decrease the temperature gradients and reduce the cracking. The composite elements withstood peak power densities in the fuel of 4500-5000 MW/m^3 and achieved better corrosion performance than was observed previously in the standard, graphite-matrix, Phoebus-type fuel element.

F. Fuel Development[8]

The major technology effort of the Rover program was expended on developing fuels. All of the Kiwi reactors except the last one, Kiwi-B4E, used highly enriched UO_2 fuel in a graphite matrix. The UO_2 particle size was 4 μm and the particle density was about 10.9 g/cm^3. At high temperatures (1873-2273 K) during processing, the UO_2 reacted with the carbon surrounding it and was converted to UC_2 with evolution of CO and consequent loss of carbon from the element. The fuel melting temperature was 2683 K, the melting temperature of the UC_2-C eutectic.

The fuel plates for the original Kiwi-A reactor were molded and pressed at room temperature, then cured to 2723 K. The plates had no coating to protect the carbon against hydrogen corrosion. All subsequent reactors used fuel elements that were extruded and coated, initially with NbC, to reduce hydrogen corrosion. The fuel element for the early reactors through Kiwi-B1B were extruded cylinders with first four, then seven coolant channels. The cylinders were contained in graphite modules. Kiwi-B4A was the first Kiwi-B design intended as a prototype flight reactor; and it used 19-hole, one-piece, hexagonal fuel elements, 19 mm across the flats. This fuel element shape became the adopted standard for all the remaining reactor designs.

The Kiwi-B4E test was the first use of coated UC_2 particles in place of UO_2 particles in the fuel. The major problem with oxide-loaded fuel elements was the so-called back-reaction. Micrometer-size UC_2 particles are extremely reactive and revert to oxide in the presence of air, particularly humid air. Thus, oxide-carbide-oxide reactions occurred during each heating and storage cycle, including graphitizing, coating, and reactor operation; and each cycle caused loss of carbon by CO gas evolution and degraded the element. Dimensional changes also were noted in stored elements. Oxidation of the UC_2 loading material caused the element to swell as much as 4% so that the final dimensions could not be controlled.

The solution to this problem was the introduction of UC_2 particles that were considerably larger, 50-150 μm diameter, and coated with ~25 μm of pyrolytic graphite. The first coated particles had a low-density pyrocarbon coat that could not withstand high temperatures. At 2273 K, the UC_2 core would migrate through the coating, thus destroying the protection against the back-reaction. Consequently, the graphitizing temperature had to be held lower at 2173 K. This temperature gradually increased with improved coated particles to 2573 K. Coated particles were eventually developed that could withstand 2873 K for 0.5 h. This work subsequently led to the development of TRISO fuel beads used in commercial high-temperature gas-cooled reactors. The coated particles in the nuclear rocket engine were not intended as a containment for fission products, the principal requirement in commercial reactors, but to provide stability during fuel-element processing and storage and to eliminate reaction with humid air and coating gases.

Coating technology evolved greatly during the Rover program. As mentioned earlier, the fuel elements of all the reactors tested in the program, except for Kiwi-A, were coated with NbC (or ZrC late in the program) to reduce hydrogen corrosion. It had been realized early that hydrogen and graphite, at the anticipated high temperatures of a rocket engine, would react to form methane, acetylene, and other hydrocarbons. Further, the graphite loss from hydrogen corrosion during reactor operation would seriously affect the reactor neutronics. So a fuel-element coating effort was undertaken in 1959 for the Kiwi-A' reactor to develop thin (0.025- to 0.05-mm-thick) NbC or ZrC coatings to act as a barrier to hydrogen attack for the length of time the reactors were to operate. Niobium carbide was selected initially because it has a higher eutectic temperature (3523 K) with carbon than does ZrC (3123 K). Much later, attention was shifted to ZrC because it adhered better to graphite and was more desirable neutronically. The coatings were applied initially with CVD techniques for the fuel elements in Kiwi-A' and -A3. These fuel elements were short: 216-mm cylinders containing four axial coolant channels. The cylinders were designed to nest into one another end to end to build up the total element length. The Kiwi-B elements were much longer, and CVD deposition of NbC on fuel-element bores had not developed to the point where they could be coated successfully and reproducibly. Consequently, a different cladding technique was used for these reactors. Simply stated, this technique was to insert niobium tubes into the fuel-element bores and heat the lined elements to the temperature (2623 K) at which the niobium in contact with carbon was converted to NbC.

Meanwhile, during the Kiwi-B testing series, the CVD technology was improving and becoming a sophisticated process in which 19 full-length, 1321-mm-long, 2.4-mm-diameter bores could be coated with NbC tailored in thickness over the full length of the elements. And so all the fuel elements for the reactors from Phoebus-1A through the last one, and including the NRX series of reactors, were CVD coated. The early CVD coatings had a useful life of about 10 min, but by the end of the program, NbC and ZrC coatings had been tested for as long as 5 h. Pewee was the first nuclear test to employ some fuel elements coated with ZrC. They performed significantly better than elements with NbC. The progressive improvements achieved in fuel performance during the NRX and Pewee series of tests are shown in Fig. 30, where corrosion measured in terms of mass loss has been normalized to one for the NRX-A2 and -A3 tests. No major change in fuel-element design, fabrication method, or characteristics occurred in the NRX series. The elements were all made from coated UC_2 beads dispersed in a graphite matrix, extruded with 19 coolant channels in a hexagonal prism, and coated with NbC. The main contributing factors for the improvements were the use in NRX-A6 of a Mo metal overcoat over the NbC bore coat in the first 1-m length of the elements (this overcoat reduced the midband corrosion, which will be discussed below, by a factor of 10); the use of thinner NbC coatings, which reduced their tendency to crack; tighter control of processing and tighter control of the fuel-element external dimensions to reduce interstitial gaps between elements; adjustments in flow orificing and fuel loading to improve the radial power and temperature profile across the core. The corrosion at the end of the NRX series was reduced to 30% of that at the beginning of the NRX series, and improvements planned for Pewee-2, which was never built, would have reduced this to 10%.

Much of the fuel testing was done in a hot gas test furnace, which simulated the operating conditions, without radiation, of the nuclear reactors. The high-pressure furnace, which is shown in Fig. 31, provided a reasonable simulation of reactor power density, temperature and thermal stress, and the effects of flowing hydrogen. These tests provided, of course, no information about radiation damage, but it was felt that the high temperatures and the small burnup in actual reactor operations would minimize radiation effects. The fuel element under test was resistively heated with dc current. The volume heat generation produced by ohmic heating was not an accurate simulation of nuclear heating, and changes in fuel-element composition during the test affected the electrical conductivity of the element potentially causing problems. But in general, furnace testing was valuable in the development of new fuel-element technologies and also in quality-control sampling during manufacture of fuel elements for a specific reactor.

A major problem alluded to earlier throughout the fuel development program was the midrange corrosion, as exemplified in Fig. 32. It was the region about one-third the length from the cold end of the core where corrosion was greatest. The inlet end of the core had low corrosion rates because the temperatures were low. The fuel operated at much higher temperatures toward the nozzle chamber end of the core, but the fuel was processed during fabrication to accept the high-end temperatures. Also the neutron flux, and hence the power density, was low, resulting in low thermal stresses and consequently minimal cracking. There, mass loss was mostly due to carbon diffusion through the carbide coating. However, in the midrange, the power density was high and the temperature was now appreciable, yet still much lower than that at which the fuel was processed; the carbide coatings would crack because of mismatched expansion coefficients, and high mass losses would occur through the cracks. The improved performance of the ZrC coating is clearly shown in Fig. 32, as is that of a new type of fuel called composite fuel.

The composite fuel was developed near the end of the Rover program and tested in the Nuclear Furnace along with pure carbide fuel as an attempt to reduce midband corrosion.[48,49] The structure of the composite fuel is compared to that of the standard graphite-matrix fuel in Fig. 33. The composite fuel is made from uncoated (U,Zr)C particles in such a way as to form a continuous phase of carbide, as a web throughout the graphite matrix. The structure of the standard fuel shows coated UC_2 particles embedded in a continuous graphite matrix. When the carbide coating lining the coolant channels cracks in this fuel, carbon is lost indefinitely through the cracks because the graphite matrix is continuous. With the composite fuel, carbon is lost through coating cracks until the carbide dispersion phase is exposed to the cracks, and then carbon stops escaping except for a small amount diffusing through the carbide. As is evident in Fig. 32, the composite fuel did indeed perform better than the graphite fuel. However, for reasons that have not been fully determined, the midrange corrosion was still greater than expected. This unexpected corrosion was attributed in part to cracking from excessive thermal stress that resulted from a decrease in thermal conductivity during the power run. This decrease, which was measured, is thought to have been caused by fission fragments. Presumably such an effect would not occur in the standard, coated-particle, matrix fuel because the fission fragments do not penetrate through the particle coatings to damage the matrix.

Pure (U,Zr)C carbide fuels were also tested in NF-1 as another approach to reducing corrosion. The fuel elements were fabricated as small hexagonal rods with one coolant channel at their center. The fuel experienced minimal corrosion, but it cracked extensively as a result of its low fracture resistance and low thermal conductivity. However, by increasing the strain-to-fracture characteristics of the elements and redesigning their shape to reduce their cross-section and web thickness, the performance of the carbide elements could be substantially improved. Yet another advanced fuel was being developed near the end of the program. This fuel was similar to the standard UC_2-coated particles in graphite-matrix fuel, but the graphite matrix was made with POCO carbon-filler flour to yield a matrix having a higher coefficient of thermal expansion (CTE) that closely matched that of the NbC or ZrC channel coatings. This fuel, referred to as the high-CTE graphite-matrix fuel, was fabricated into fuel elements, and it exhibited better strain-to-failure characteristics than the standard fuel. It was intended for the NF-2 test that, unfortunately, did not take place. But the promising results obtained before cancellation of the program should be seriously considered in any future graphite fuel-element development.

The demonstrated operating performance of the standard graphite-matrix fuel was 1 h at a coolant exit temperature in the range of 2400-2600 K. This performance was obtained mainly from the NRX-A6 and Pewee tests. The demonstrated performances of the advanced composite and pure carbide fuels were nearly 2 h (109 min) at 2450 K and at a peak power density in the fuel of 4500-5000 MW/m^3, as obtained in the NF-1 test. Based on the extensive fuels work achieved during the Rover program, projections of endurance limits were estimated as shown in Fig. 34. These projections indicate that the composite fuel should be good for 2-6 h in the temperature range of 2500-2800 K. Similar performance can be expected at 3000-3200 K for the carbide fuels, assuming that the cracking problem can be reduced through improved design. For 10 h of operation, the graphite-matrix fuel would be limited to a coolant exit temperature of 2200-2300 K, the composite fuel could go to nearly 2400 K, and the pure carbide to about 3000 K.

And so the program was terminated with three promising fuel forms at hand, the carbide-carbon composite, the pure carbide, and the high-CTE graphite matrix. As discussed above, much testing was performed on these fuels, but their corrosion behavior was not completely understood. Most of the work was done in the temperature range of 2000-2800 K. This range would have to be extended to lower temperatures (below 1500 K) and testing done with gases other than hydrogen to evaluate the performance of these fuels for electrical-power production applications.

Fuel structures were also a major problem in the reactor development. Reactor cores in the early Kiwi series essentially fell apart from vibrations; these were induced thermal-hydraulic interactions. By the end of the Kiwi series, an adequate structural support system had been demonstrated. Improvements in the structural system continued to be made, with the results summarized in Table I. It should be emphasized that what structural anomalies existed were determined after the tests and did not cause a termination of power. At the end of the NRX-A6 test, some cracking in the beryllium-reflector ring, support blocks, peripheral composite cups, and one tungsten cup was found. These cracks were believed to be the result of excessive thermal gradients.

IV. ENGINE DEVELOPMENT

A. Engine Tests

An engine development program was carried out as part of the nuclear reactor research and development test series of Phoebus and NRX. Prime responsibility for this effort rested with the Aerojet-General Corporation. The primary objective of this test series was to further extend nuclear rocket technology in preparation for a flight system. This involved incorporating the advances made in reactor development into an engine that comprised the nonnuclear components of a complete flight system. Of particular interest were the investigation of engine startup, shutdown, and restart characteristics for different initial conditions; the evaluating of various control concepts; and testing the performance of nonnuclear engine components in the nuclear environment. Two full-power nuclear test series can be categorized in this program. These are NRX/EST,[50] which was carried out in February-March 1966, and XE',[51,52] which took place in March through August 1969. Both of these tests employed 1100-MW NRX-type reactors. In addition, a cold-flow test series Experimental

Engine Cold Flow (XECF) was conducted in February-April 1968.

The NRX/EST displayed in Fig. 35 was the first operation of a NERVA breadboard power plant with engine components connected in a flight-functional relationship. The test demonstrated the stability of the power plant under a number of different control modes while the engine operated over a broad area of its performance map. The endurance capability of the reactor and other engine components was demonstrated by operating the power plant at significant power during 5 different days for a total of 1 h and 50 min, of which 28 min were at full power. These tests served to demonstrate the multiple restart capabilities of the engine, including the feasibility of restarting the engine without an external power source.

Operation of the XE' Engine (Figs. 36 and 37) was the first test of a down-firing nuclear rocket engine with components in a flight-type, close-coupled arrangement. The test stand provided a reduced atmospheric pressure (about 1 psia, or 60 000 ft altitude) around the engine to partially simulate space conditions. The engine was successfully operated at full power. It ran at various power levels on different days for a total of 115 min of power operation that included 28 restarts. The bootstrap startups (without external power) were accomplished over a range of pump inlet suction pressures and with reactor conditions spanning the range that would be encountered in flight operations. Completely automatic startup was demonstrated. The capability of the engine to follow demanded temperature ramp rates up to 56 K/s was demonstrated, and based on this information, assurance was gained that rates up to 83 K/s could be achieved without exposing any of the engine components to a transient condition that would exceed its design limitations.

Figure 38 shows some of the characteristics of starting an engine of this sort.[52-54] The engine components must be conditioned before high power can be reached. The turbopump, nozzle, reflector, and core inlet are all designed to operate at low temperatures. When the pump shutoff valve is first opened, the pump tends to vaporize the fluid until sufficient fluid passes through it to chill down the pump to cryogenic conditions. The nozzle also tends to be a choke point, as are the core and reflector inlets. Therefore, a certain amount of fluid must be passed through the system to remove the stored heat in the lines, valves, and reflector. Once this is accomplished, the pump can be started and will operate normally. Approximately 1 min of fluid flow is necessary to accomplish this function. During this time period, the reactor can be brought up to a low-power level. Reactor drums are programmed out rapidly, almost to the cold critical point, and then put on a slow transient. Once appreciable temperature rise is sensed in the chamber, the reactor can be switched to closed-loop temperature control. This scheme does not require any neutronic instrumentation. When appreciable power has been achieved and the turbopump is running, the engine can be accelerated at the rate of 83 K/s. Experience on the NRX/EST and XE' engine programs showed that the engine system can be controlled in a predictable and safe manner.

B. Engine Design Improvements[55]

Other goals of the engine development program, besides demonstrating engine feasibility and control, were first of all to maximize specific impulse, which is proportional to the square root of the nozzle chamber temperature; to meet various design thrust levels that are proportional to flow rate and that demonstrate the capability to throttle the engine down and operate at a reduced thrust; to minimize engine size and weight; and to increase longevity from an initial 1 h to 10 h. In fact, system operating life is really determined by the amount of propellant that can be transported to space in a reasonable payload to perform the mission.

Increases in the chamber temperature were made first of all by improving the reactor fuel performance to permit raising the operating temperature, as discussed in the preceding chapter. A number of design changes also were made to improve reactor performance. The cores of the early reactors were supported axially with tie rods attached to the cold-end support plate. The rods were cooled by hydrogen that discharged into the nozzle chamber, and this cooling lowered the rocket specific impulse because the tie-rod coolant exit temperature was much lower than that of the coolant exiting the main fuel elements. In later reactors and in engine systems designed for flight, tie tubes were substituted for tie rods. The tubes were regeneratively cooled with the coolant discharging into the core inlet rather than the core exit. Also, in the early reactors, a large core peripheral flow rate was used to protect the core-reflector interface, and this cooler flow was discharged into the nozzle chamber. This flow was steadily decreased to almost nothing to increase the nozzle chamber temperature. A final optimization would be to employ a regeneratively cooled heat exchanger for the periphery assembly.

The engine flow cycle was also changed to increase specific impulse. The XE engine employed the "hot-bleed" cycle to drive the coolant turbopump. In this cycle, some coolant is extracted from the chamber and mixed with coolant from the reflector outlet, and the combined coolant is used to drive the turbine. Because the turbine exhaust pressure is low, this coolant could not be reintroduced into the main flow stream; it was discharged into space at a low temperature relative to the nozzle chamber conditions, thereby reducing the overall specific impulse of the engine.

Final engine designs evolved to employ a full-flow or topping cycle in which the turbine receives fluid from the tie-tube outlet and discharges it into the core inlet. This cycle, never tested experimentally, significantly raises the specific impulse of the engine. The turbine inlet temperature in the full-flow cycle is much lower than in the hot-bleed cycle. Consequently much greater turbine flow rates and discharge pressures are required than in the former cycle.

C. NERVA and Small Engine Designs

Design characteristics for several engines are listed in Table II. Only the experimental XE' engine[56,57] and Phoebus-2A were tested. Mass estimates are detailed in Table III. A comparison of just the reactor masses was shown earlier in Fig. 10. Both the NERVA and Small Engine designs took advantage of the full-flow cycle idea. The NERVA [58-61] was designed for the 337-kN (75 000-lb) thrust level using a 1570-MW reactor, and the Small Engine was designed for a 72-kN thrust level with a 367-MW reactor. The equivalent vacuum specific impulse of the XE' engine was 710 s. This was improved to 825 s for the NERVA flight engine and 875 s for the Small Engine. The increasing specific impulse levels are reflections of the chamber temperatures that went from 2270 K as demonstrated in the XE' test to a design value of 2695 K for the Small Engine.

The Small Engine[2,3] depicted schematically in Fig. 39 really represents an accumulation of all of the knowledge gained in the nuclear rocket program. It used hydrogen as the propellant, the full-flow engine topping cycle, and a single-stage centrifugal pump with a single-stage turbine. It had a regeneratively cooled nozzle and tie-tube support elements. A radiation shield of borated zirconium hydride was incorporated above the reactor. This was mainly to reduce heating to the propellant tank above the engine, although it also provided shielding for the payload and crew. Reactor control was done with six actuators for the 12 control drums in the beryllium reflector. The engine employed only five valves and their actuators, including a propellant tank shutoff valve (PSOV) located at the bottom of the propellant tank to provide a tight seal against propellant leakage when the engine is not in use; a nozzle control valve (NCV) to adjust the flow split between the nozzle coolant tubes and the tie tubes; a turbine series control valve (TSCV) that could be used to isolate the turbine during preconditioning and cooldown to remove after-heat and to extend the control range; a turbine bypass control valve (TBCV) to regulate the amount of flow to the turbine and thus the turbopump speed and flow rate; and a cooldown control valve (CCV) to regulate hydrogen flow for decay heat removal following engine operation and together with a small pump, to provide prepressurization for the tank.

A regeneratively cooled nozzle was used out to the area ratio of 25:1. It was followed by an uncooled skirt section. This section extended the nozzle out to an area ratio of 100:1. The uncooled nozzle skirt was hinged to facilitate packaging in the launch vehicle. This arrangement provided room for a larger propellant tank in the launch vehicle. The overall engine length was 3.1 m with the skirt folded, or 4.4 m with the skirt in place. The total mass of the system was 2550 kg.

The reactor core, pictured in Fig. 40, was designed to produce about 370 MW_t. There were 564 hexagonally shaped, (UC-ZrC)C composite fuel elements containing a total of 52.4 kg of uranium (0.9315 enrichment). Each fuel element had 19 coolant channels. There were 241 support elements, containing zirconium hydride, ZrH_2, as a neutron moderator. The core periphery included an outer insulator layer, a cooled inboard slat section, a metal wrapper, a cooled outboard slat section, and an expansion gap. The core was supported on the cold end by an aluminum alloy plate with the support plate resting on the reflector system. The reactor was contained in an aluminum pressure vessel. A beryllium barrel with 12 reactivity control drums surrounded the core. The reactor was designed for 83 K/s temperature transients.

Figure 41 provides more details on the fuel modules. The fuel provided the heat transfer surface and the energy for heating the hydrogen. It consisted of a composite matrix of UC-ZrC solid solution and carbon. The channels were coated with zirconium carbide to protect against hydrogen reactions. The tie tubes transmitted the core axial pressure load from the hot end of the fuel elements to the core support plate. They also provided an energy source for the turbopump and contained and cooled the zirconium carbide moderator sleeves. They consisted of a counterflow heat exchanger of Inconel 718 and a zirconium hydride moderator with zirconium carbide insulation sleeves. The elements were 0.89 m long and measured 19.1 mm from flat to flat. The effective core diameter was 570 mm and the overall reactor diameter was 950 mm.

As shown in Table III, the overall Small Engine weight was 2550 kg, with the reactor (minus the shield) almost 1600 kg of this. Figure 42 summarizes the Small Engine state points at design conditions. The chamber temperature was nearly 2700 K.

A study was made to see how the Small Engine would mate with the propellant tank so as to fit within the space shuttle. The results are summarized in Fig. 43. The nuclear stage with the Small Engine would weigh close to 18 000 kg, of which almost 13 000 kg would be propellant. If additional propellant modules were sent up separately, the stage would then weigh 23 000 kg, with over 21 000 kg of hydrogen available for propulsion. At a flow rate of 8.5 kg/s, a single nuclear stage would operate for approximately 1500 s in space. An additional propellant module would add 2500 s to the operating time. Thus, a possible lifetime of several hours would allow performing many significant missions.

D. Component Development[60]

A vigorous program for the development of nonnuclear engine components accompanied the reactor and engine test programs. The principal components were the main coolant turbopump, the valves and actuators, the nozzle assembly, the reactor pressure vessel, radiation shielding, and the controls and instrumentation.

Figure 44 is a picture of the turbopump developed for the XE' engine, including a listing of some key parameters related to the turbopump. The pump performs the function of pressurizing the propellant for the engine feed system. The low flow rates required by the Small Engine made it possible to run the shaft speed at a much higher rate than for the XE' engine or NERVA. The XE' turbopump was a single-stage, radial exit flow, centrifugal pump with an aluminum propeller, a power transmission that coupled the pump to the turbine, and a two-stage turbine with stainless steel rotors. On NRX/EST, this pump performed eight starts and operated 54.4 min at high power. In the XE' engine, it performed

28 starts and restarts, including runs to rated power. Potential problem areas were with the shaft system binding at the bearing coolant, a difficulty that was experienced in the XE tests. The solution was to increase clearance and to improve alignment. Bearings are probably one of the few life-limiting components in the non-nuclear subsystem. Results of life tests are listed in Table IV. The solution to the bearing problem seemed to depend on maintaining adequate cooling to reduce wear.

Various valves were required in the system to control the hydrogen flow.[52,60-62] These valves were binary valves, except for the in-out control valves and check valves, such as shown in Fig. 45. Valve operating experience with a reactor was obtained in both the NRX/EST and XE' engine tests. Table V lists the valve and actuator characteristics. The major potential problems appeared to be from seal damage by contaminants, erroneous position indicators, and leakage from poor lip-seal tolerances.

The number of valves in the Small Engine was five. However, there was some desire in the NERVA flight engine to increase system reliability by having two turbopumps, either of which could provide full flow and pressure to the engine systems, and redundant valve configurations. In order to provide switching between the turbopumps and to provide high reliability by backing up each valve in case of a failure, some 26 valves would be needed, as seen in Fig. 46.[61] Indeed it becomes questionable whether the redundancy gained is worth the added system complexity.

The nozzle assembly is used to expand the heated gas from the reactor in order to provide maximum thrust. A nozzle is pictured in Fig. 47.[63] The design conditions for the NERVA flight engine were a thrust level of 337 kN, an area ratio of 100:1, a service life of 10 h, reliability of fewer than four failures in 10^4 flights, chamber pressure of 3.1 MPa, chamber temperature of 2360 K, flow rate of 41.6 kg/s, and coolant channel temperature between 28 and 33 K. The regeneratively cooled nozzle part used an aluminum alloy jacket and stainless steel for the coolant channels. The graphite nozzle extension was uncooled out to an area ratio of 100:1. The cooled section used U-tube constructions and a divergent section. The major unresolved problems in achieving a 10-h life were associated with some remaining stress problems in the aluminum alloy. Fabrication problems appear to have been resolved.

Figure 48 depicts the pressure vessel and enclosure.[64] Its functions were to support the components of the reactor assembly, to form a pressure shell for the hydrogen propellant, and to transmit thrust to the thrust structure. The design conditions, as specified for the NERVA, were a maximum flow rate of 37.6 kg/s, maximum pressure of 8.66 MPa, temperature range from 20 to 180 K, reliability of fewer than three failures in 10^6 flights, and a service life of 10 h. Similar designs were demonstrated in the five NRX tests and the XE' engine. The pressure vessel was constructed of a cylinder that had a top closure with bolts and seals. A one-piece extruded forging of aluminum alloy 7075-773 was used, with a surface coating of Al_2O_3. The major items still being designed were the best ways of assuring bulk preload and of finalizing the surface coatings.

A radiation shield was located between the reactor core and the propellant tank. The shield was intended to prevent neutron heating of the propellant, and it also provided biological shielding for the crew as the tank emptied of its propellant. It did not present any difficult design problems.[65]

Controls and instruments were another major area of development during the Rover program.[35] For control-drum actuators, pneumatic-type actuators were developed. These were demonstrated on the XE' engine without apparent degradation or anomalies.

Instrumentation was also a major development. Thermocouples demonstrated performance at 2667 K for 1 h without degradation. Thermocouple displacement, pressure, and vibration sensors were developed for several hours of operation. A 1% measurement accuracy will require some further development.

Control logic reached a high degree of automation with demonstration of automatic control systems in XE' for all operational phases. Feedback control loops and drum position, power, temperature, turbine control valve position, and pressure were developed.

E. Testing Facilities

Testing facilities were another major development item. During the nuclear rocket program, major test facilities were developed at the NRDS at Jackass Flats in Nevada (Ref. 2, Vol. III). These included reactor test facilities, engine test facilities, and assembly and disassembly facilities (shown earlier in Fig. 4). The reactor test facilities were designed to test the reactor in an upward-firing position. These used facility-type feed systems for providing the hydrogen to cool the reactor and to support the reactor tests. The first downward-firing facility, which also included some atmosphere simulation, was the engine test facility. This was used in the XE cold-flow tests and also the XE' full-power test.

For maintenance, assembling, and disassembling of the reactor and engine systems, there were buildings called Maintenance Assembly and Disassembly (MAD), which provided the necessary facilities, hot cells, and other equipment for putting together and taking apart the reactors. The MAD buildings were linked (Fig. 49) to their respective test facilities by railroads that were used to transport the reactors by remote control if necessary.

V. FUTURE DEVELOPMENTS

A. Flight Engine

The basic research and technology development required for a nuclear rocket flight engine were essentially completed during the Rover program. Power levels in the range of 500-4100 MW were demonstrated in the NRX, Phoebus, and Pewee test series. A thrust level of 930 kN (200 000 lb) was reached in Phoebus-2A with a hydrogen flow rate of 120 kg/s. A specific impulse of 710 s (6060 m/s) was obtained in the XE experimental engine, but the highest equivalent specific impulse achieved was 845 s in Pewee, which operated at a peak coolant exit temperature of 2550 K and a peak fuel temperature of 2750 K. A core average power density as high as 2340 MW/m^3 and peak maximum fuel power density of 5200 MW/m^3 were obtained as well in Pewee. The NF-1, which experienced nearly as high peak power densities, ran at full power and an average coolant exit temperature of 2445 K for an accumulated time of 109 min. The experience gained from these tests indicates that the composite fuel would last for 6 h under these conditions before appreciable fuel loss occurs.[49]

The nonnuclear components of the engine proved to be capable of achieving the endurance required for testing. Fully automatic control and bootstrap startup were demonstrated for a wide range of operating conditions in NRX/EST and XE', the latter experiencing 28 engine starts and restarts with temperature ramp rates up to 56 K/s, which could be raised with confidence to 83 K/s.

This section describes the Rover technology base. The program was terminated at the point of flight engine development. For a flight system, it would be necessary to verify the flight reactor and engine design, perform duration testing, and verify reproducibility. There appear to be no technological barriers to the construction of a successful nuclear rocket.

B. Space Power Generation

The nuclear rocket engine technology base is directly applicable to the generation of electric power in space, particularly for high-power (so-called multimegawatt, 10-100 MW_e), open-cycle systems.[66] A schematic drawing of such a power plant is shown in Fig. 50. The plant is similar to a rocket engine in which the engine nozzle has been replaced by a turbine to generate electricity. The coolant gas is then exhausted in such a way as not to produce any thrust. The core exit temperature of the hydrogen coolant would have to be much lower than that of the rocket engine because of material limitations on the turboalternator. This means that for a given coolant flow rate the power would be reduced, or for a given power level, the flow rate would have to be increased as compared to that in the rocket engine. A multimegawatt, open-cycle power plant would necessarily have a duration at power measured in hours, being limited by the amount of coolant that can be reasonably transported to space. Because of the low operating temperatures of such a power plant, hydrogen corrosion in the reactor would be greatly reduced and would not be a life-limiting factor.

The Rover reactor designs are also applicable to closed-loop multimegawatt space power systems. The reactor inlet and outlet conditions would be different from those of the nuclear rocket engines, and the coolant would be helium or a mixture of helium and xenon instead of hydrogen. The three reactor fuels developed prior to termination of the Rover program would certainly be promising candidates for such power plants although much work would need to be done to map out their corrosion and erosion behavior as a function of time and temperature with these gases. The lower operating temperature would permit use of the TRISO-design fuel beads in fuel fabrication. The TRISO beads, developed and used in commercial gas-cooled reactors, would provide the advantage of reducing possible fission product contamination of the working fluid to low levels.

The simplest power plant would be a direct Brayton cycle power conversion system[67-69] in which high-pressure gas enters the reactor where it is heated, expands through a turbine generating electric power, and passes through a recuperator and heat-sink heat exchanger where it is cooled. The gas is then repressurized in the compressor and partially heated in the recuperator before re-entering the reactor.

Closed-cycle power plants are attractive because their mission life at full power is not limited by the coolant supply. They are limited, however, by burnup of fuel in the core. Several small reactor designs were studied during the Rover program including Pewee, Nuclear Furnace, the Small Engine, and others. The fuel used in these studies was always some form of uranium carbide embedded in a graphite matrix, except for NF-1 where pure UC fuel was tested also. The pure UC fuel was promising but not fully developed, as extensive cracking of the fuel occurred in NF-1. For a variety of materials considerations, the density of UC in the composite fuels is limited to 700-800 mg/cc so that small reactor size can be achieved only by incorporating moderating material such as zirconium hydride in the core as was done in Pewee and the Small Engine, or water as was done in NF-1. The critical mass for the Pewee and Small Engine reactors was in the range of 35-60 kg of ^{235}U. The fractional burnup that can be tolerated in these reactors is not limited by the capabilities of the fuel but rather by the reactivity margin available from the reflector control system. A practical burnup limit based on reactivity control margin is probably in the neighborhood of 10%. Thus, assuming an energy equivalent of 1 g of ^{235}U per megawatt day or 0.36 kg/MW yr, the integrated power capability of this class of reactors is in the range of 10-20 MW yr. Possibly this capability could be increased through the use of burnable poisons or breeding of less-enriched fuel.

C. Dual-Mode Reactors

Near the end of the Rover program, it was realized that it would be convenient to use the nuclear rocket engine to provide long-duration, auxiliary electrical power for station keeping. Studies were carried out to modify the Small Engine design to perform two modes of operation as shown in Fig. 51.[4,5,70,71]

The normal, rocket propulsion mode of the Small Engine was unchanged. But a separate mode was incorporated into the design, where the hydrogen coolant, which normally flowed through the tie-tube core support structure and subsequently through the turbopump, could be diverted and isolated to flow continuously in a closed-loop system as shown in Fig. 52. In this mode, 10 to 25 kW of electric power could be generated continuously.

This dual-mode design employs an organic Rankine cycle using thiophene as the working fluid. For a 10-kW_e system, the particular design case shown in Fig. 52, the radiator would be incorporated into the surface of the propulsion module coolant tank. For higher powers, additional radiator area would be needed. The major engine modifications required for the auxiliary power plant were the addition of isolation valves for the flow through the tie-tubes support structure; a change of material from aluminum to stainless steel for the reactor dome, core support plate, and tie-tube supply lines; and a material change in the control-actuator windings from polyimide to ceramic. The material changes were needed to accommodate the wider temperature range of dual-mode operation. The reactor inlet and the reflector are uncooled during electric-power generation, and the tie tubes run hotter than during the propulsion engine operation mode.

The incremental mass penalty of a 10-kW_e power plant was calculated to be about 50 kg/kW_e, including the main engine modifications. This penalty could be reduced to 35 kg/kW_e for an optimized 25-kW_e system. The lifetime of this dual-mode power plant is limited by the radiation degradation limit of the thiophene working fluid. The study indicated that in 2 yr of operation at 10 kW_e, the radiation dose to the thiophene would be approximately a factor of 10 below that limit. In the same time, the burnup of fuel in the core would be insignificant. A feature of a dual-mode engine not yet mentioned is the possibility for reducing the amount of propellant used to remove the reactor after-heat following an engine shutdown by running the auxiliary power plant to cool the reactor. For the 10-kW_e system, a maximum thermal power of 140 kW could be removed in this manner, limited simply by the radiator configuration.

The dual-mode rocket engine described here can be altered to become a bimodal power plant, as shown in Fig. 53, in which the nozzle has been replaced by an open-cycle turboalternator. This configuration provides a high electric-power generation mode for a duration limited by the amount of stored coolant, plus a continuous, low electric-power mode for station keeping. The open-cycle electric-power system could be incorporated without eliminating the rocket engine thrusting potential by using a poppet valve, as shown in Fig. 54.[72] The valve, normally retracted into the convergent section of the nozzle, is moved to close off the rocket nozzle and open a bleed line to divert the heated hydrogen to the turboalternator in the power-generating mode.

In summary, the Rover nuclear rocket technology is relevant to the generation of electric power in space, particularly for multimegawatt, open-cycle, single- or dual-mode systems. Conceptual studies performed during the program show that a station-keeping auxiliary power plant in the range of 10-25 kW_e can be incorporated into the nuclear rocket engine, with only minor modifications to the engine already developed, to produce a dual-mode system. But extensive redesign would be required to provide greater, continuous power. Likewise, relatively minor modifications would be required to convert the nuclear rocket engine to the power source for an open-cycle Brayton power plant. Other than conceptual studies, however, little work was done in the program to develop the electric conversion systems themselves.

VI. SUMMARY

In this report we presented first a summary of milestone achievements obtained during the Rover nuclear rocket propulsion program. This summary was followed by a brief chronological history of the program, a generally chronological description of the reactor technology development with emphasis on the fuels program, and a description of the engine testing program, engine design, evolution, and component technology status as of the end of the Rover program. We ended by discussing a few of the ideas and studies that were generated near the end of the program in recognition of the need to apply the demonstrated nuclear rocket technology to the generation of electric power in space. A valuable part of this report is the significant but unavoidably incomplete (over 100 000 volumes of data were generated during the program) bibliography that we have compiled. We have made an attempt to list only the most pertinent references that we could find.

Today, a dozen years after termination of the Rover program, as we ponder future requirements for nuclear power in space, we can place the relevancy of the Rover technology to our current needs in the following perspective. The NERVA nuclear rocket reactor, as exemplified by the NRX-A6 reactor, was not the fruit of a paper study. It was the result of a long series of experimental tests and was demonstrated to work and to operate under conditions of temperature and power densities more severe than are likely to be encountered in space reactors designed specifically for power production. The NERVA reactor design was demonstrated to be sound structurally, thermohydraulically, and neutronically. Although it was optimized for thrust production rather than power generation, it can serve as a real reference point against which large, future space reactor designs of any sort can be judged and evaluated, if the differing operating conditions are properly extrapolated.

The class of Rover reactors was tested over an enormous power range from 0 to 4000 MW, thereby providing valuable scaling relationships for size, mass, and other parameters, based on real data.

The hardware technology acquired through the Rover program and the basic reactor design itself are directly applicable to future space powerplant designs based on either closed- or open-loop gas cycles. Probably the outstanding question here is how to extrapolate the erosion and corrosion behavior of the best Rover reactor fuels, obtained at hydrogen gas exit temperatures of 2500 K, to gas temperatures likely to be below 1500 K, and for gases other than hydrogen such as He or He-Xe mixtures.

Finally, many of the design and testing methodologies, as well as tools, computer codes, and facilities, developed during the Rover program can be adapted for use in the development of future nuclear power plants in space.

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TABLE I

SUMMARY OF NRX - REACTOR STRUCTURAL ANOMALIES

Source	Significant Structural Anomalies	Phenomenon Causing/ Allowing Occurrence	Resulting Technique, Criteria of Geometry Change (Corrective Action)
NRX-A1	1) Control-drum binding.	1) Deflection of control drum and outer reflector due to thermal gradients.	1) Complete deflection analysis of all interacting components for all reactor conditions indicated. Increase bore diameter in outer reflector.
	2) Pressure vessel-outer reflector support ring interference.	2) Deflection due to thermal gradients.	2) Deflection analysis for startup, steady state, and shutdown required.
	3) Reflector cracks.	3) Contraction due to thermal gradients on nozzle end ring.	3) Finite element analysis technique was final technique before A6.
NRX-A2	1) Aluminum barrel tab bent.	1) Combination of barrel pretest axial movement, shrinkage, and inner reflector axial contraction.	1) None indicated, not considered serious.
NRX-A3	1) Support block cracks, aft c' bore fillet to inner flow holes missed on A2, same cond.	1) Stresses due to thermal gradients and mechanical loads at termination of startup ramp.	1) Initiated first finite-element computer analysis study. (This technique was developed extensively.)
	2) Control-drum rubbing.	2) Deflection of sector and drum due to thermal gradients during cooldown after flow initiation.	1) Transient analysis required, drum material removed.
NRX-A4	1) Cracking of support-block fillets, also peripheral and axial cracks.	1) Stresses due to thermal gradients.	1) Reduction of start-ramp on A5, also planned test holds, also partial lobes removed, c' bore radius was increased.
	2) Flaring and splitting of liner tube ends.	2) Excessive growth due to over-heating.	2) Reduce liner tube length.
	3) Light rubbing of control drum.	3) Deflection due to thermal gradients.	3) Analysis techniques improved, drum longitudinal slots added.
NRX-A5	1) Peripheral and axial support-block cracks.	1) Stresses due to thermal gradient and mechanical loading.	1) Serious problem at end of reporting period.
	2) Liner tube flaring.	2) Cluster components' overheating caused liner tube expansion that coupled with tapered sleeve restraint to cause flaring.	2) Reduced line tube length.
	3) 1/3 of fuel elements broken.	3) Reduced strength due to excessive corrosion plus inflow gradients and end of life.	3) Increase emphasis on coating technology.
	4) Hot buffer filler strip extensive breakage.	4) Stresses due to thermal gradients.	4) Eliminate hot buffer.
NRX-A6	1) Beryllium-reflector ring cracked.	1) Stresses due to thermal gradients.	1) Components tests verifying 3-D finite element analysis technique.
	2) Support-block cracks.	2) Stresses due to thermal gradients.	2) A predictable problem requiring extended efforts for similar components in the flight engine era of NERVA.
	3) Cracking of peripheral composite cups and one tungsten cup.	3) Stresses due to high non-symmetric transverse temperature gradient.	3) Nonexpected, protection cup performance not impaired.

TABLE II

ENGINE DESIGN CHARACTERISTICS

Dimensions	Small Engine	XE'	NERVA	Phoebus-2A
Thrust (kN)	72	245	337	1123
Specific Impulse (s)	875	710	825	840
Thermal Power (MW)	367	1140	1570	5320
Turbopump Power (MW)	0.93	5.1	6.9	19.4
Turbopump Speed (rpm)	46 950	22 270	23 920	34 000
Pump Discharge Pressure (MPa)	6.03	6.80	9.36	8.54
Engine Flow Rate (kg/s)	8.5	35.9	41.9	129
Chamber Temperature (K)	2695	2270	2360	2500
Chamber Pressure (MPa)	3.10	3.86	3.10	4.3
Core Diameter (m)	0.89	0.84	0.57	1.39
Core Length (m)	1.32	1.32	0.89	1.32
Be Reflector, o.d. (m)	1.25	1.12	0.95	1.99
Be Reflector Thickness (mm)	117	114	134	203
Pressure Vessel o.d. (m)	1.3	1.3	0.98	2.07
Pressure Vessel Length (m)	1.9	1.9	1.7	2.8
Pressure Vessel Thickness (mm)	21	25.4	25.4	25.4

TABLE III

ENGINE MASS ESTIMATES (kg)

Dimensions	Small Engine	XE'	NERVA	Phoebus-2A
Reactor Core and Hardware	868	1270	3 367	5511
Reflector and Hardware	569	1467	1 660	2676
Internal Shield	239	1316	1 583	--
Pressure Vessel	150	422	863	1075
Turbopump	41	182	243	--
Nozzle and Skirt Assembly	224	558	1 051	--
Propellant Lines	15	450	500	2008
Thrust Structure and Gimbal	28	480	663	--
Valves and Actuators	207	930	1 262	--
Instrumentation and Electronics	159	400	508	--
Contingency	50	225	600	--
TOTAL	2550	7700	12 300	

TABLE IV

BEARINGS

- One of the few life-limiting components in nonnuclear subsystem
- Special bearing tests showed with Polyhenzemedazol-graphite fabric composite (PBI/graphite):
  - 23 h and 13.8 h for two tests in nonirradiated environment
  - 6 h of successful operation in three tests irradiated to 4.8×10^{10} and 6×10^{10} ergs/gm
- More radiation tolerant than armalon (teflon-glass fabric)
- Adequate cooling essential to reduce wear

TABLE V

VALVE AND ACTUATOR CHARACTERISTICS

	COOLDOWN CONTROL VALVE (CCV)	NOZZLE CONTROL VALVE (NCV)	PROPELLANT SHUTOFF VALVE (PSOV)	TURBINE BYPASS CONTROL VALVE (TBCV)	TURBINE SHUTOFF CONTROL VALVE (TSCV)	TURBINE VALVE ACTUATOR (TVA)	CONTROL DRUM ACTUATOR (CDA)
Actuator size	small	small	large	large	large	large	small
Seat diameter, cm	2.54	2.54	15.24	3.175	11.43	--	--
Seat area, cm^2	5.06	5.06	182	7.93	103	--	--
Stroke	1.52 cm	1.52 cm	5.08 cm	1.52 cm	3.18 cm	$^+$0.44 rad	3.14
Load max. velocity	0.15 cm/s	0.15 cm/s	0.5 cm/s	2.5 cm/s	0.2 cm/s	0.007 rad/s	0.2 rad/s
Seating force, N	3550	3558	13 340	2800	10 000	--	--
Plus max. % at rated speed, N cm^{-2}	760	275	20.7	103	103	--	--
Lead screw diameter, cm	1.27	1.27	1.9	1.9	1.9	--	--
Lead screw advance, cm rad^{-1}	0.0809	0.0809	0.0809	0.0809	0.0809	--	--
Max. stepping rate, steps s^{-1}	62	62	200	210	75	143	122
Motor gear ratio	50	50	50	10	50	380	917
Torque at speed N-m	--	--	--	--	--	3250	34

CCV and NCV are identical.

PSOV and TBCV have same actuator.

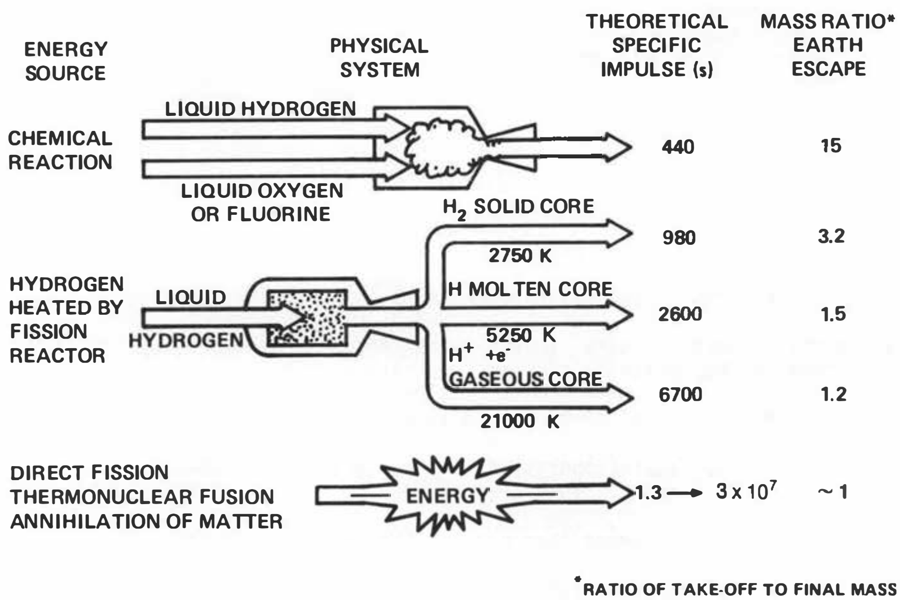
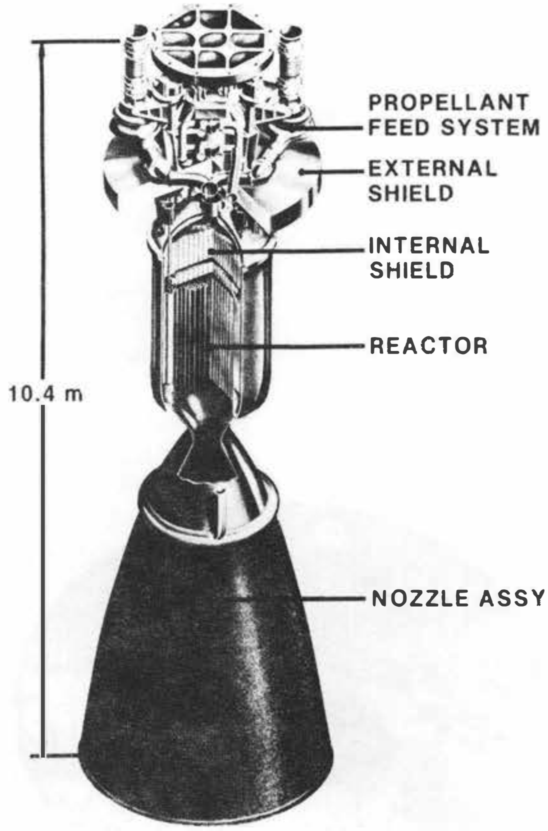


Fig. 1. Comparison of various energy sources for rocket propulsion. Hydrogen heated by a solid-core fission reactor offers the potential for more than doubling the specific impulse and for reducing by nearly a factor of 5 the ratio of take-off to final mass for earth escape, relative to the performance of chemical rocket engines. Specific impulse, I, is also expressed in meters per second. [I(m/s) = 9.8 I(s)].



SPECIFICATIONS

THRUST (N)	334,000
SPECIFIC IMPULSE (s)	825
POWER (MW)	~ 1500
CHAMBER PRESS (MPa)	3.10
CHAMBER TEMP (K)	2360
EXPANSION RATIO	100:1
WEIGHT (kg) (W/O EXT SHIELD)	11,250
TOTAL OPERATING TIME (min)	600
NO. OF CYCLES	60 (TO 2360 K)
RELIABILITY	0.995

Fig. 2. The NERVA and its design specifications.

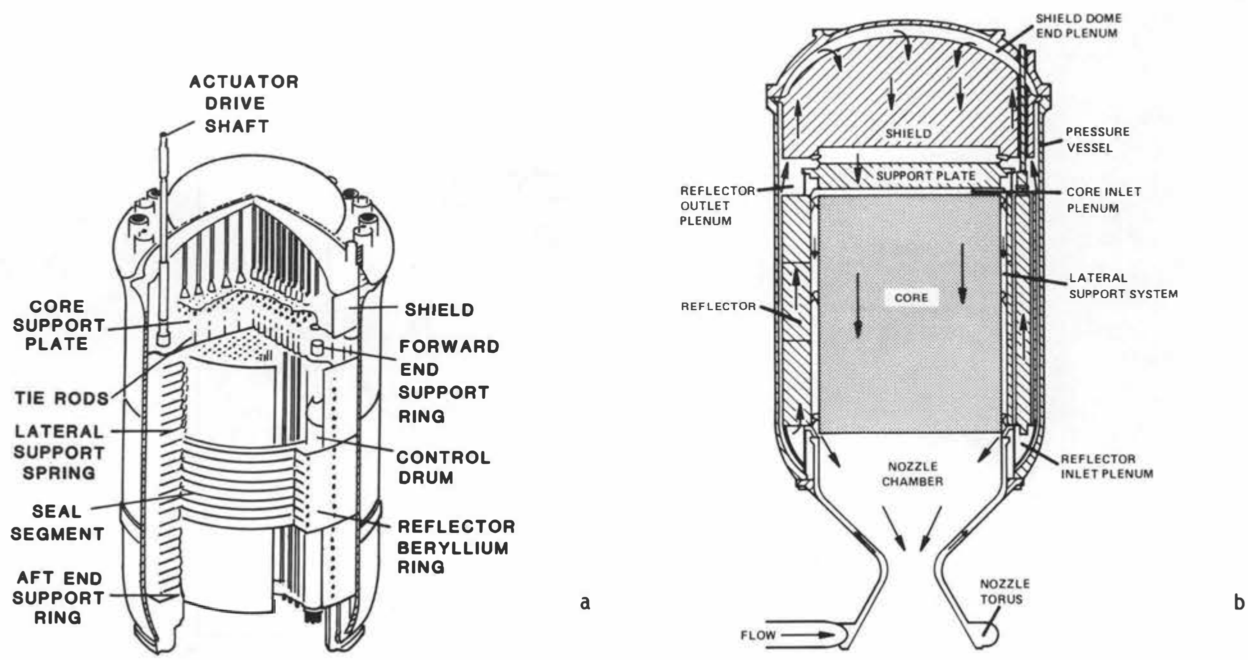
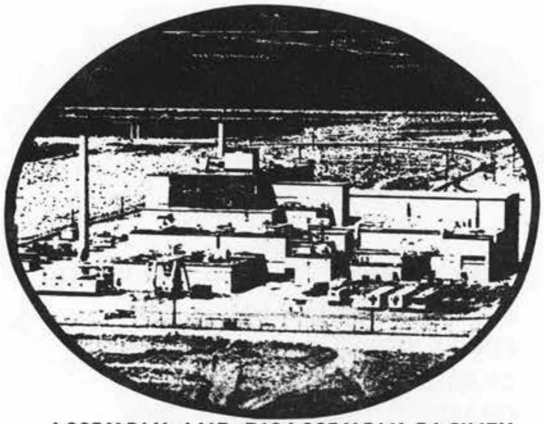
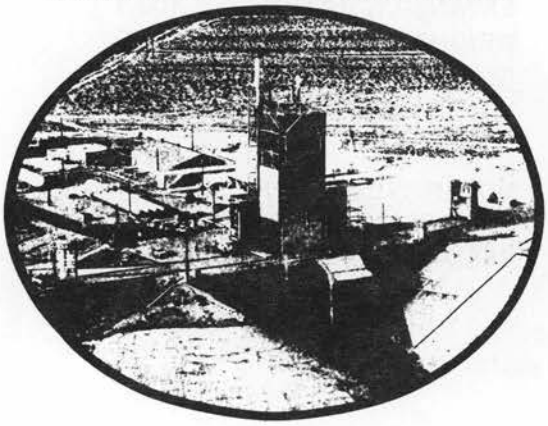


Fig. 3. Cut-away and schematic flow description of the NERVA reactor.



ASSEMBLY AND DISASSEMBLY FACILITY



ENGINE TEST FACILITY

REACTOR TESTING FACILITY

Fig. 4. Photographs of some of the major testing facilities built at NRDS at Jackass Flats, Nevada, for the Rover program.

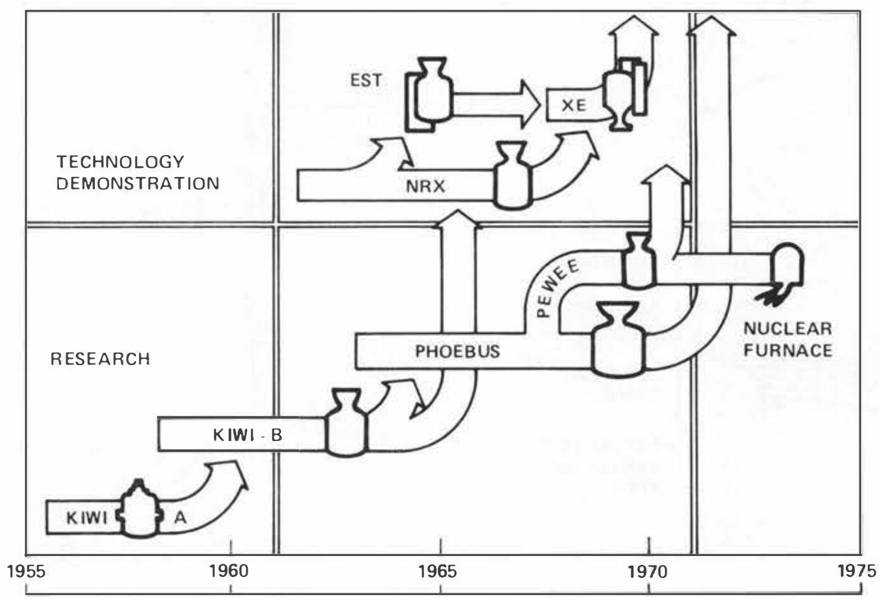


Fig. 5. Summary of the testing program for the nuclear rocket reactors. The two top arrows point to what would have been the development of a flight engine had the program proceeded through a logical continuation.

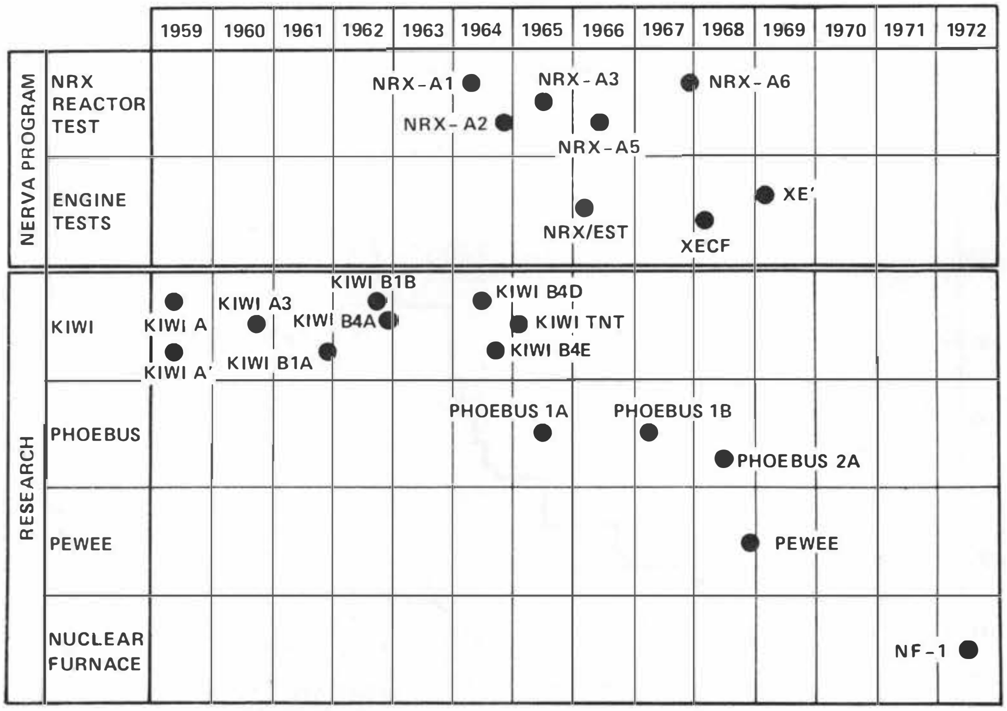


Fig. 6. Chronology of major nuclear rocket reactor tests.

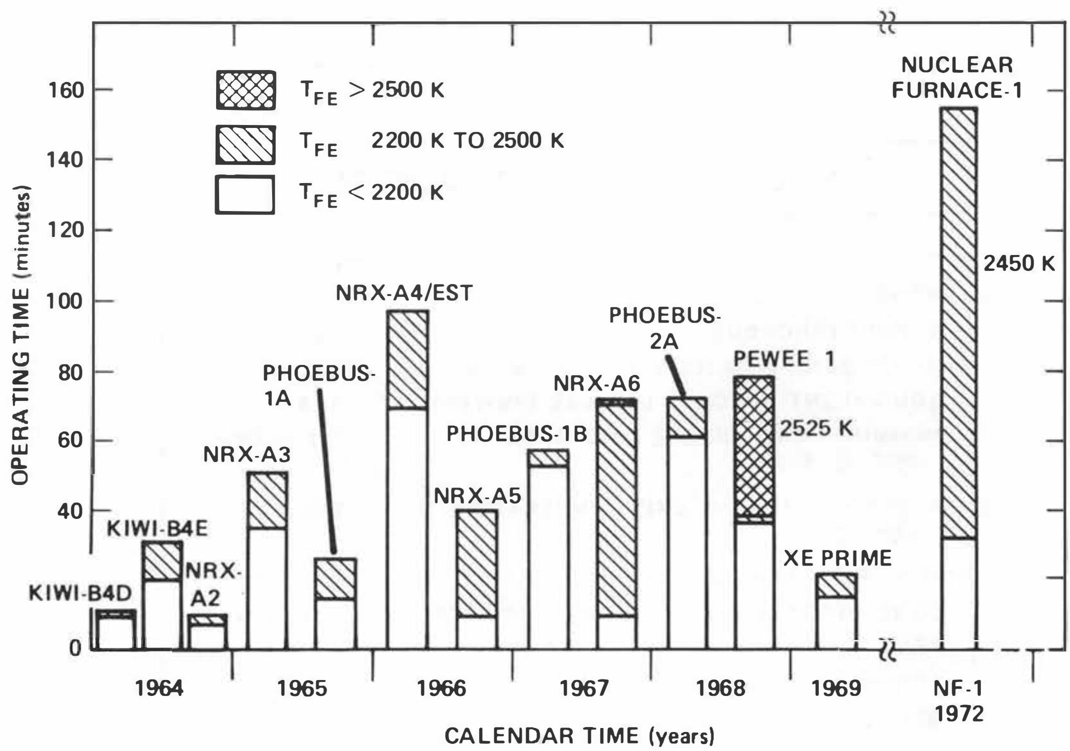


Fig. 7. Operating time versus coolant exit temperature for the full-power reactor tests conducted after development of a successful core design (Kiwi-B4D). Note the general trend toward an increase in both running time and temperature.

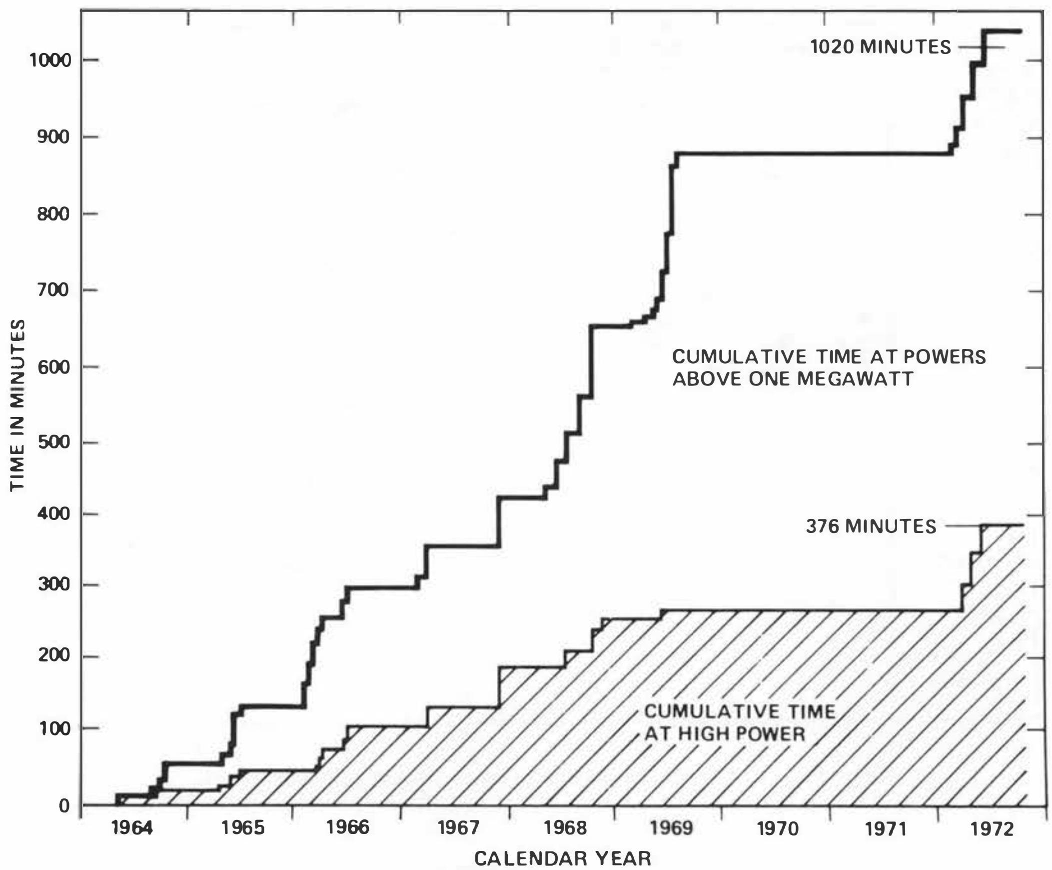


Fig. 8. Cumulative testing time for all Rover reactor and engine system tests.

RECORD PERFORMANCES

POWER (PHOEBUS-2A)	4100 MW
THRUST (PHOEBUS-2A)	930 kN
HYDROGEN FLOW RATE (PHOEBUS-2A)	120 kg/s
EQUIVALENT SPECIFIC IMPULSE (PEWEE)	845 s
MINIMUM REACTOR SPECIFIC MASS (PHOEBUS-2A)	2.3 kg/MW
AVERAGE COOLANT EXIT TEMPERATURE (PEWEE)	2550 K
PEAK FUEL TEMPERATURE (PEWEE)	2750 K
CORE AVERAGE POWER DENSITY (PEWEE)	2340 MW/m^3
PEAK FUEL POWER DENSITY (PEWEE)	5200 MW/m^3
ACCUMULATED TIME AT FULL POWER (NF-1)	109 MIN
GREATEST NUMBER OF RESTARTS (XE)	28

Fig. 9. Summary of major performances achieved in actual tests during the Rover program.

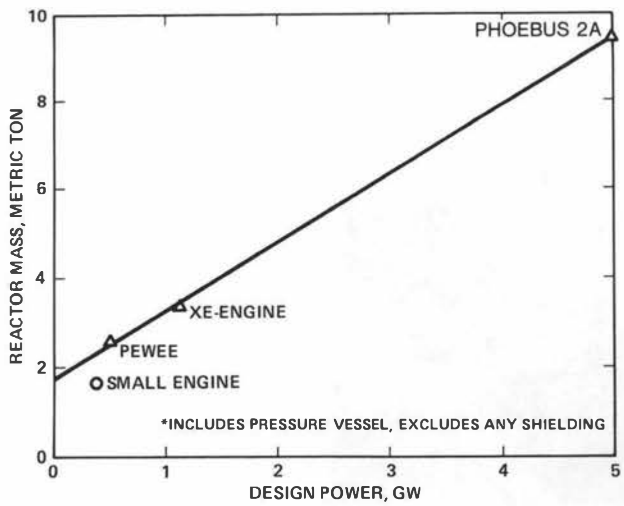


Fig. 10. Reactor mass versus design power for several Rover reactors. The Small Engine was not built but was the result of a design study done near the end of the program.

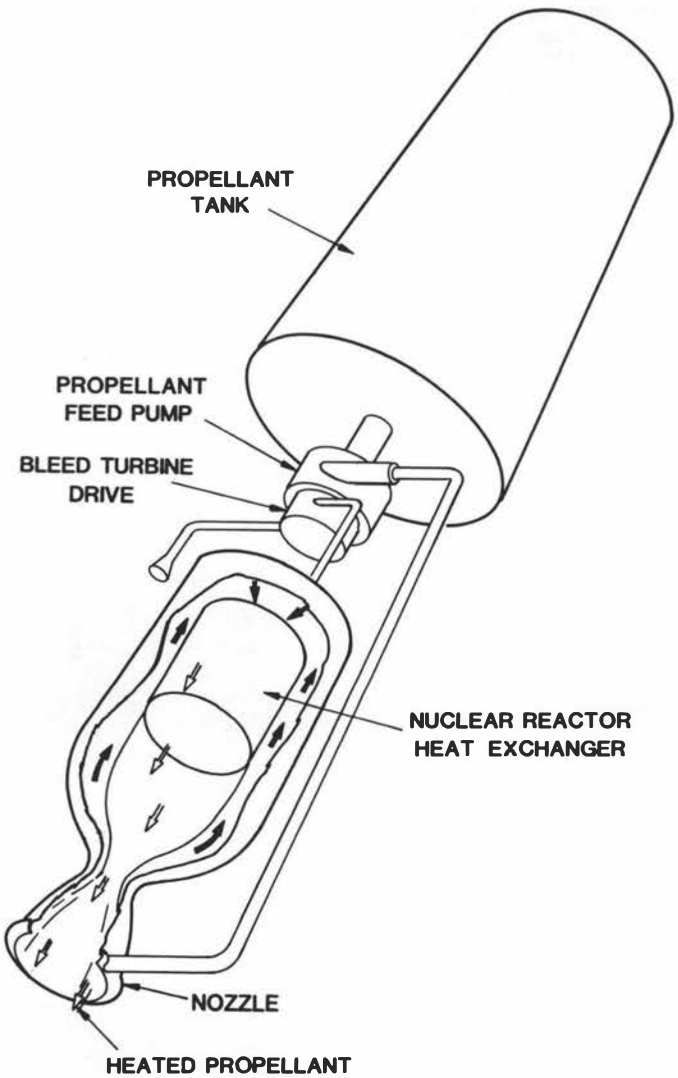


Fig. 11. Schematic description of a nuclear rocket propulsion engine.

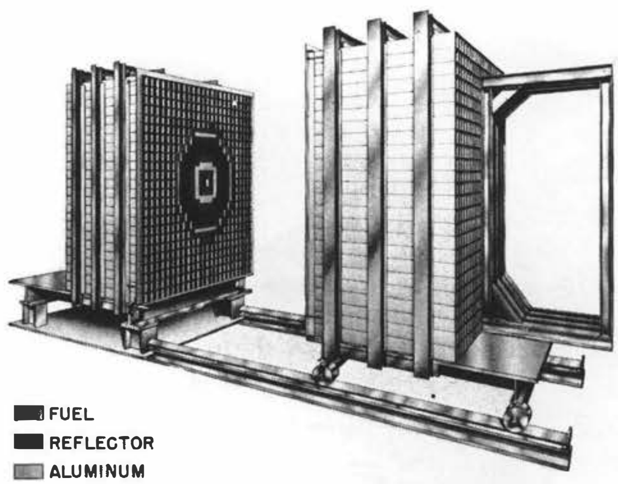


Fig. 12. Honeycomb assembly machine used to model the reactors for neutronic criticality experiments.

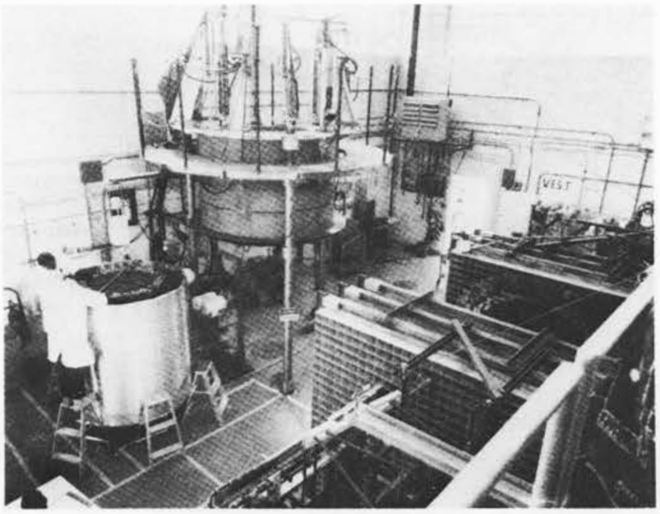


Fig. 13. The core being assembled in the zero-power mockup of the Phoebus-2A reactor. Figure 12 shows the much cruder simulation of the same reactor in the Honeycomb machine.

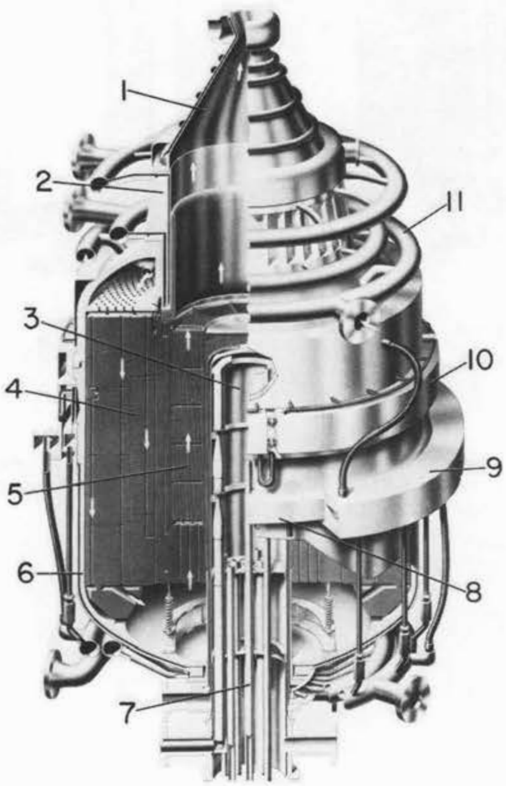


Fig. 14. Cut-away description of Kiwi-A, the first Rover reactor design tested.

1. Nozzle
2. Core support liner
3. Center island
4. Graphite reflector
5. Fuel plates
6. Pressure shell
7. Control rod
8. Lift band
9. Test stand support ring
10. Lock ring
11. Main coolant inlet manifold

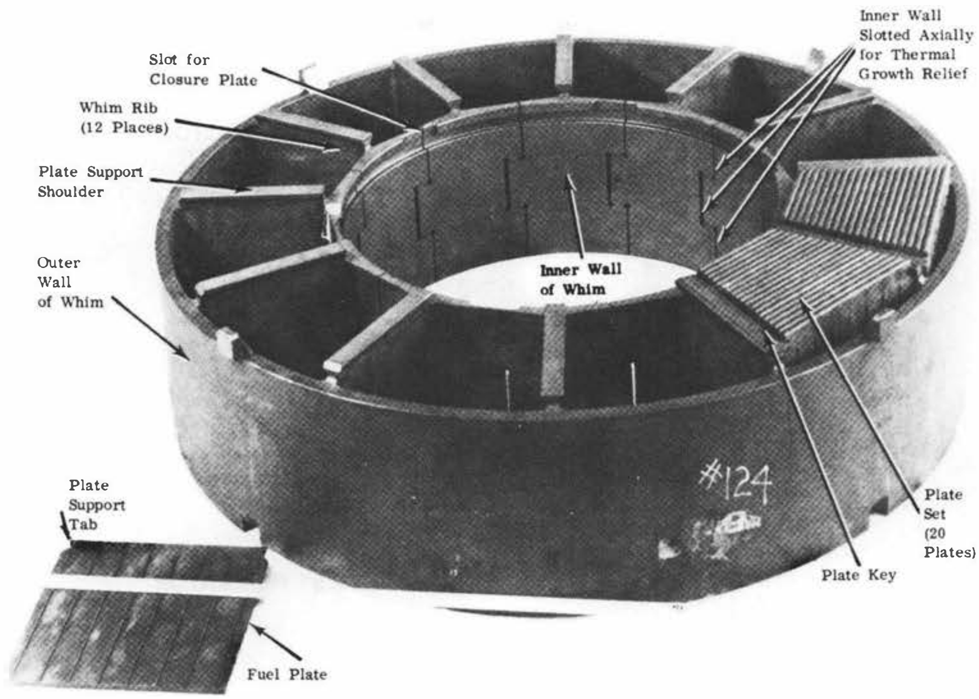


Fig. 15. Fifth whim of Kiwi-A during assembly. The Kiwi-A design was the only one that employed fuel elements in the form of plates.

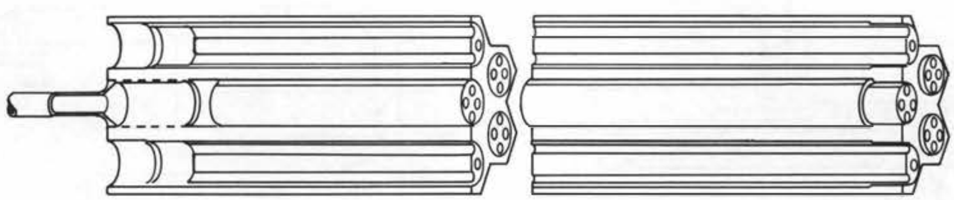


Fig. 16. Cylindrical fuel elements in a graphite module employed in Kiwi-A' and Kiwi-A3. Six cylinders were stacked end to end in each hole of the graphite module to form a complete fuel module.

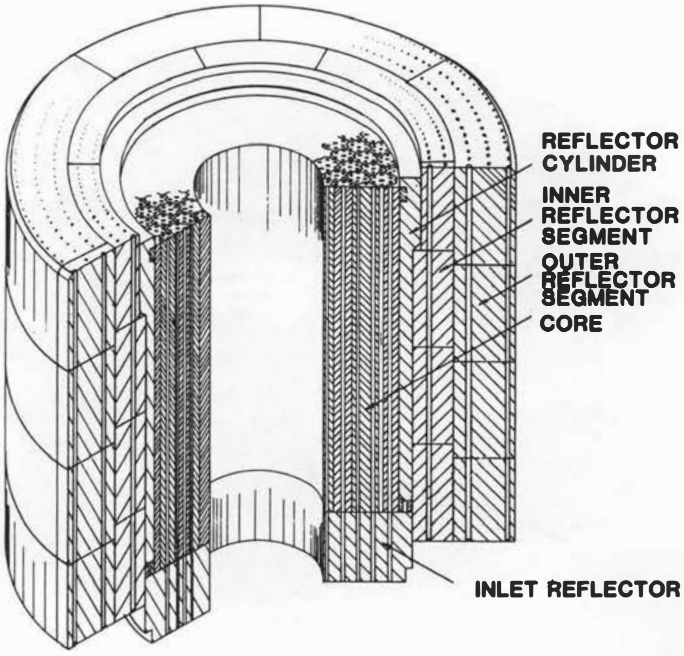


Fig. 17. Details of Kiwi-A' core design. The central island has been removed.

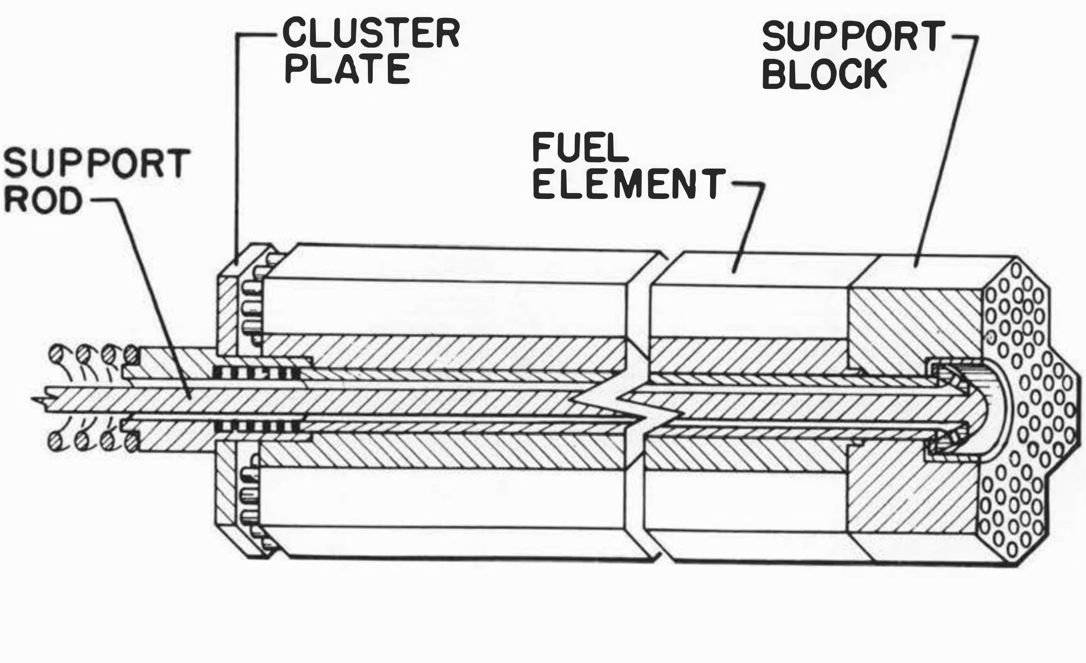


Fig. 18. Fuel-element cluster employed in most of the later Rover reactor designs. It consists of six, full-length, hexagonal fuel elements supported by a centrally located tie rod. Each extruded graphite fuel element had 19 cooling channels.

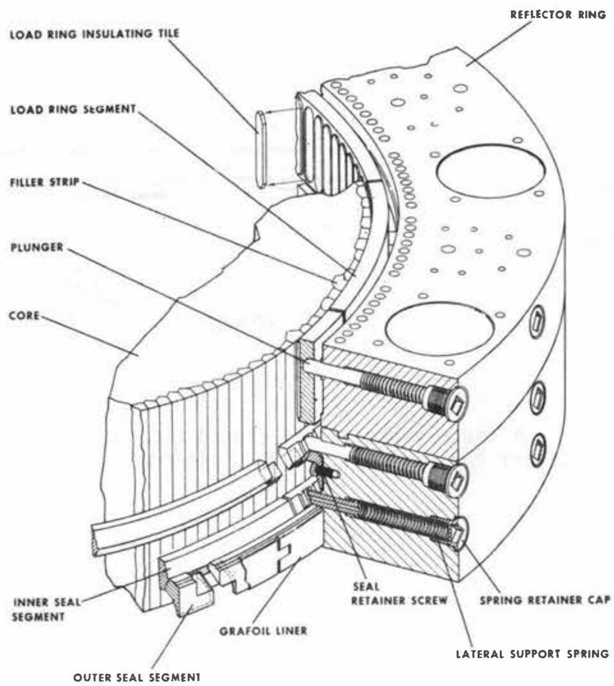


Fig. 19. Detail of core periphery and lateral support for the NRX-A6 reactor.

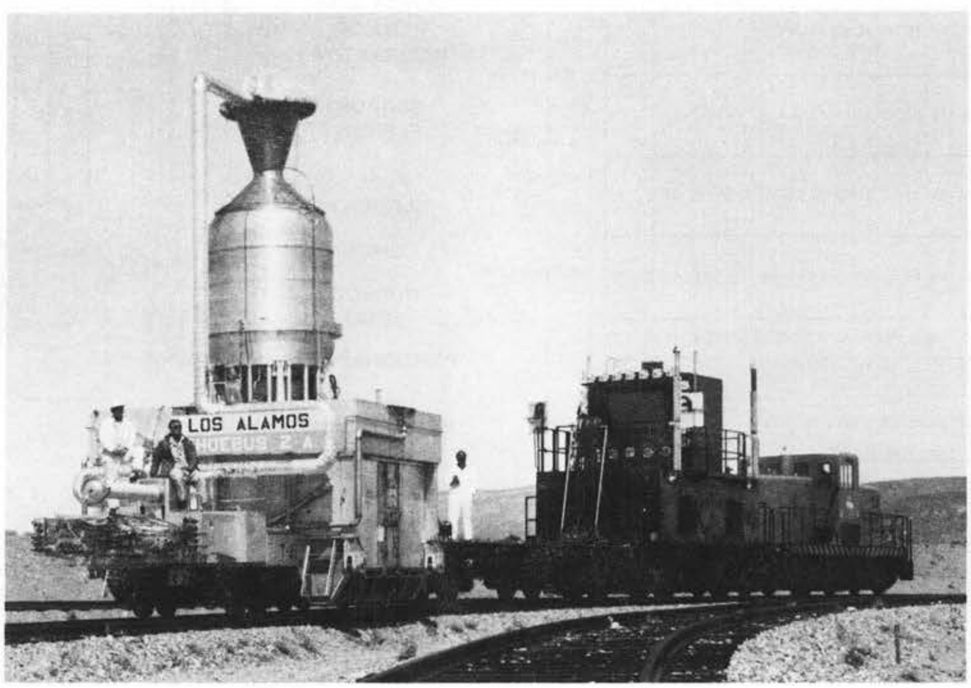


Fig. 20. Phoebus-2A being towed on rails to its testing station at NRDS. Designed for 5000 MW and tested at 4000 MW, it was the most powerful nuclear rocket reactor ever built.

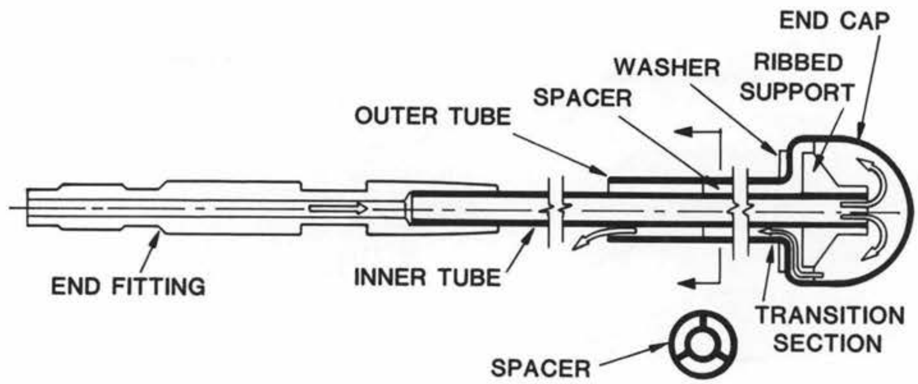


Fig. 21. Detail of the regeneratively cooled tie tube employed to support the Phoebus-2A fuel clusters.

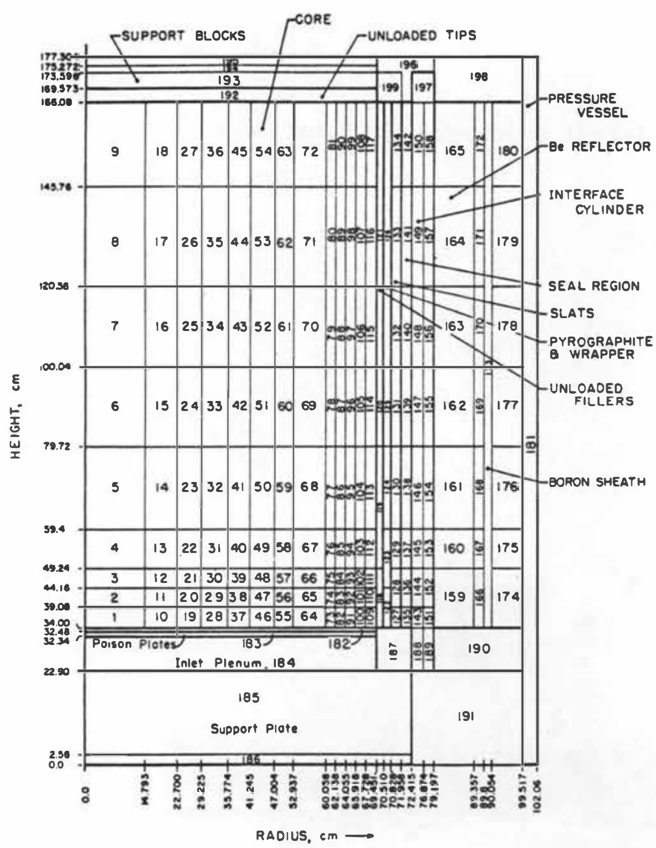


Fig. 22. Two-dimensional model of Phoebus-2A reactor.

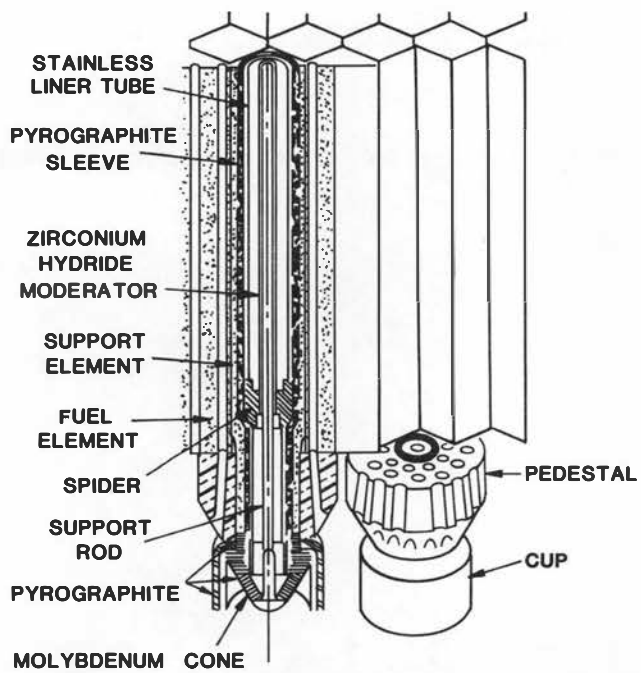


Fig. 23. Detail of the hot end of the Pewee fuel support system. Sleeves of zirconium hydride around the tie rods moderated the core neutrons and greatly reduced the critical mass of uranium in the core.

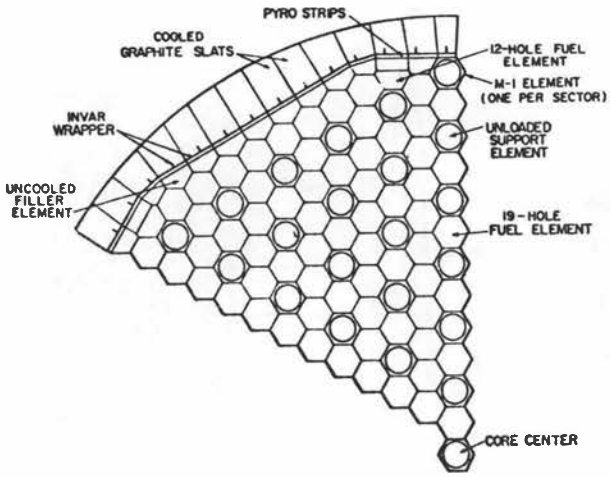


Fig. 24. A 60° sector of the Pewee core showing the three-to-one ratio of fuel and support elements.

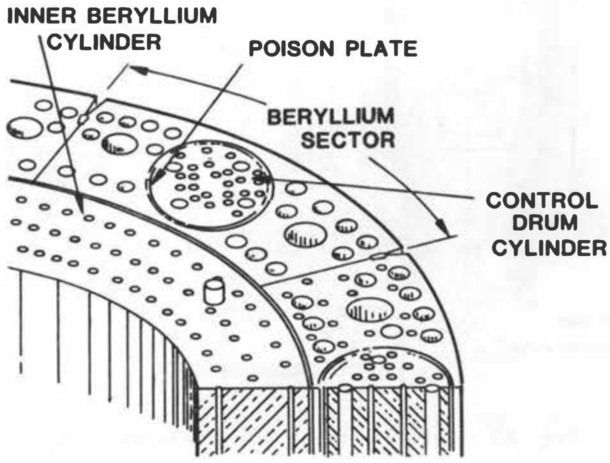


Fig. 25. Detail cross section of the Pewee beryllium-reflector assembly that contained nine control drums.

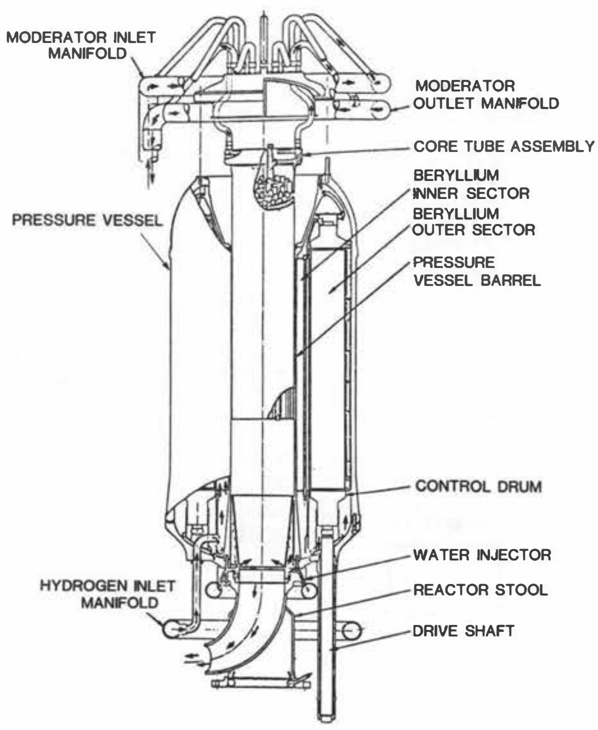


Fig. 26. Axial view of the NF-1 reactor. Designed strictly as a nuclear test bed for fuel elements, it was the last reactor tested in the Rover program.

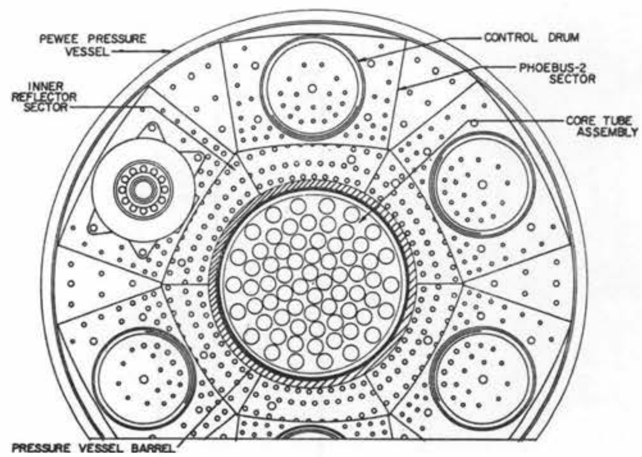
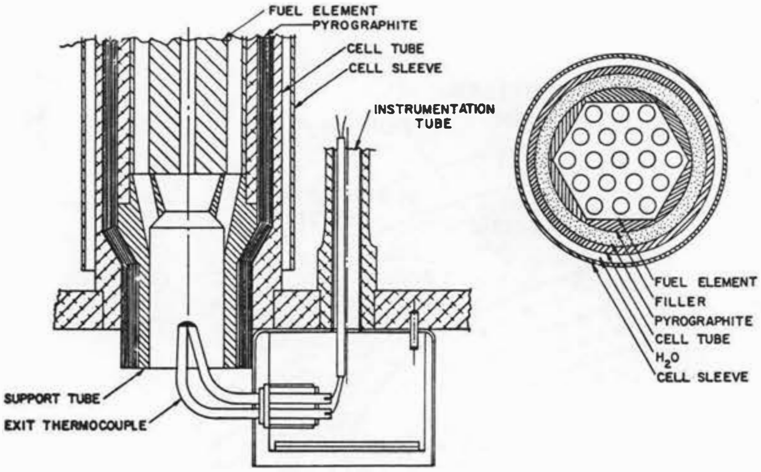
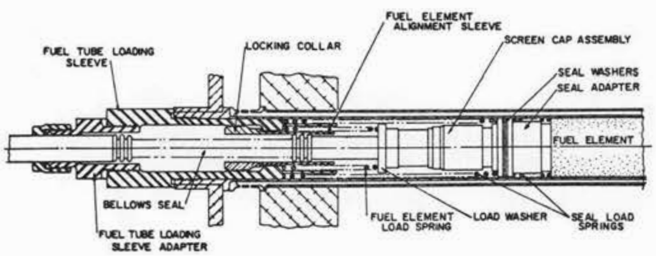


Fig. 27. Cross-sectional view of NF-1. The core assembly was water moderated to reduce the uranium critical mass to about 5 kg. The beryllium-reflector assembly was designed to be reusable.



a

Fig. 28. Details of the NF-1 fuel-cell assembly.



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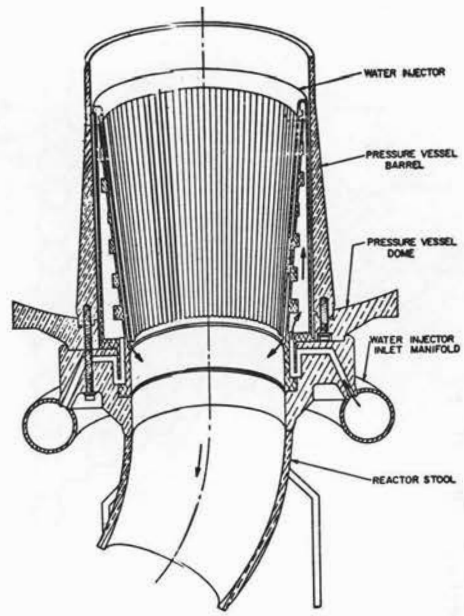


Fig. 29. Detail of the NF-1 water injection system (also visible at the bottom of Fig. 26) employed as part of the effluent cleanup system used to remove fission products from the exhaust hydrogen gas.

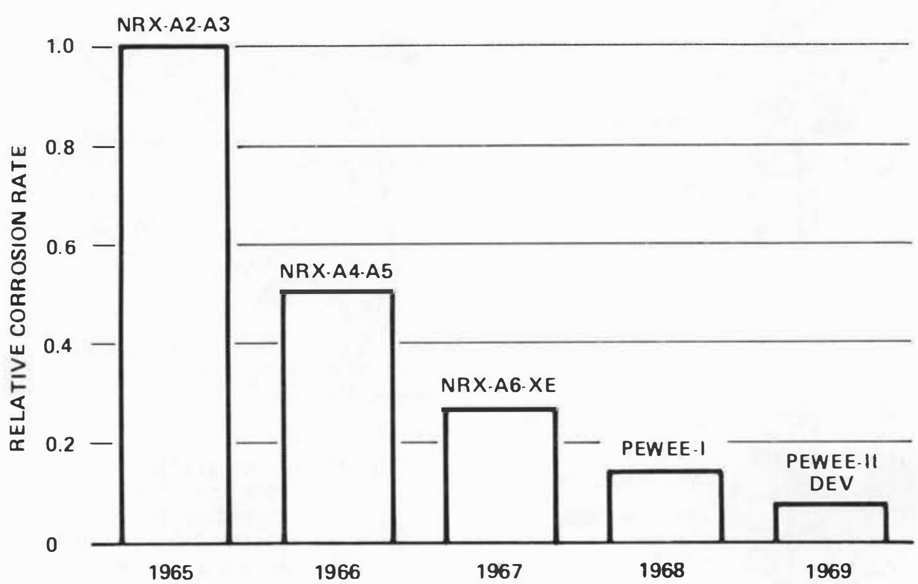


Fig. 30. Progress achieved in reducing hydrogen corrosion of fuel elements normalized to the corrosion rate observed in the NRX-A2 and -A3 tests. The corrosion rate shown for Pewee-2 is a projection because that reactor was never built.

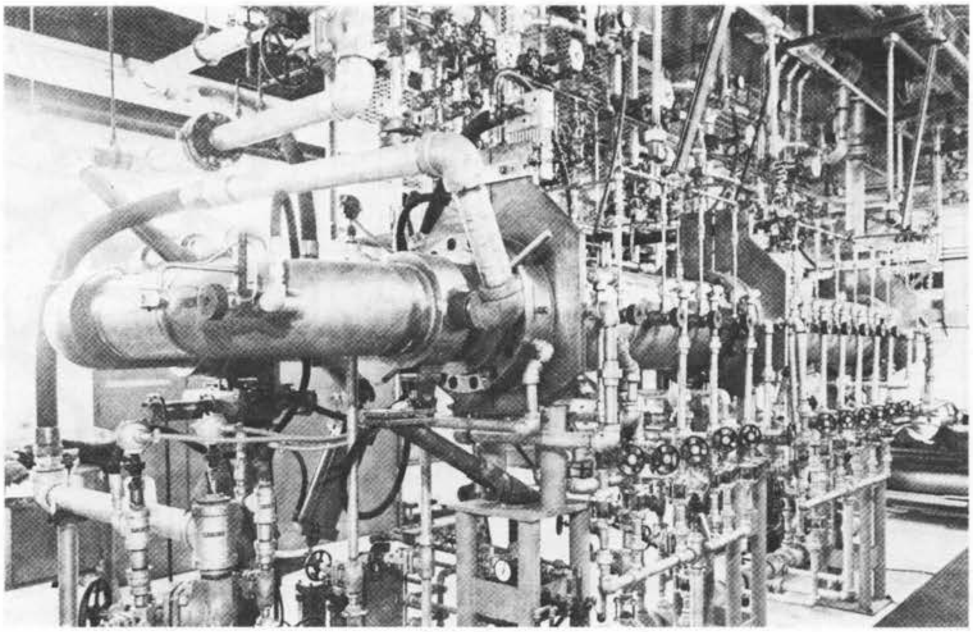


Fig. 31. The hot gas test furnace used for hydrogen corrosion tests of fuel elements in a nonnuclear environment.

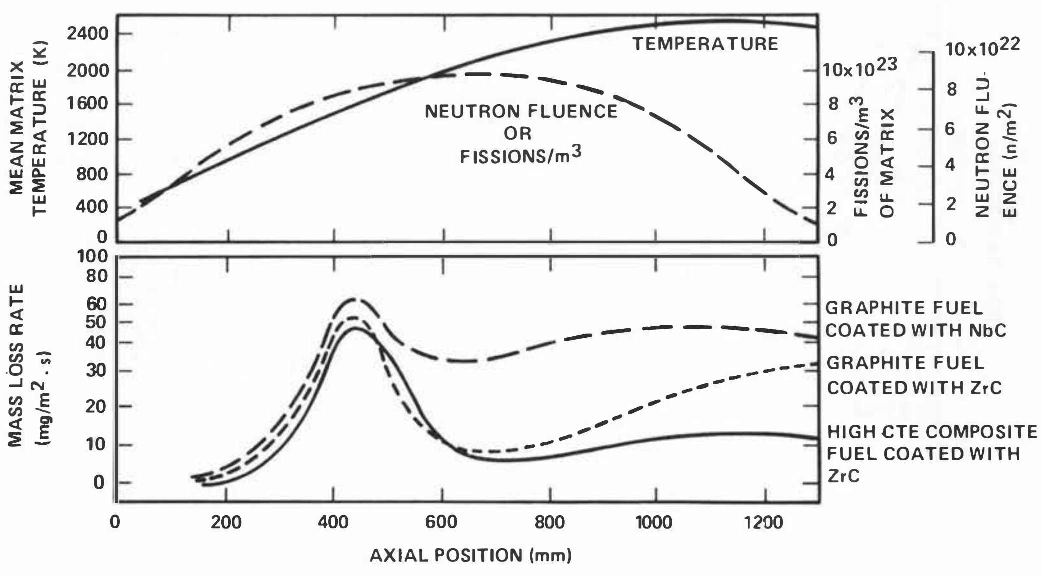


Fig. 32. Mass loss from Pewee and NF-1 fuel elements versus axial position and reactor environment. The peak in the mass loss curve is the so-called midrange corrosion. The best performance was obtained with the composite fuel coated with ZrC.

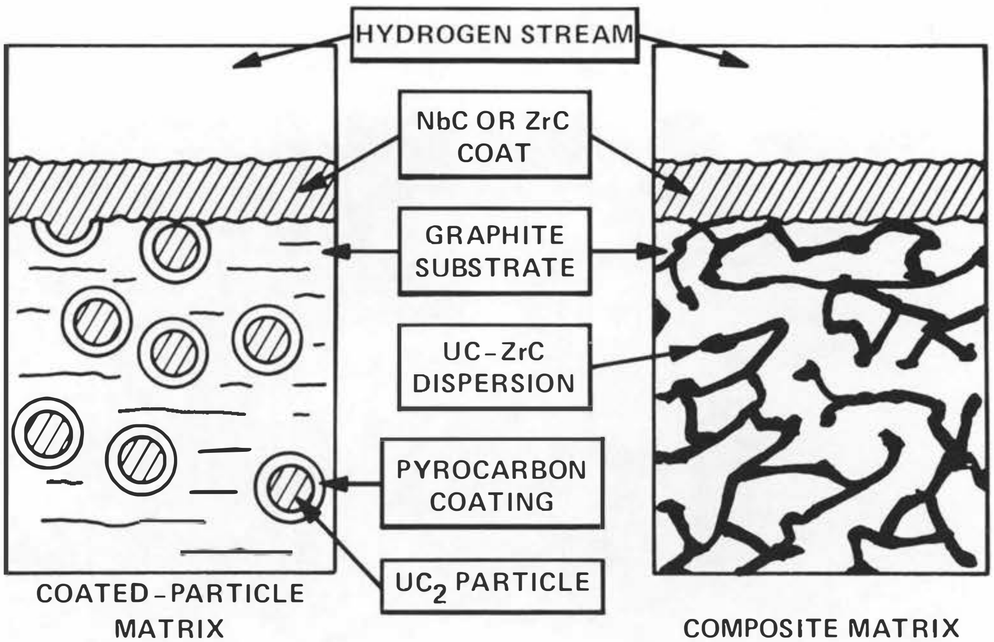


Fig. 33. Comparison of the fuel structure in the standard, coated-particle graphite matrix with the composite matrix fuel. The continuous, webbed UC-ZrC dispersion prevents hydrogen, entering through cracks in the top coating, from eating deeply into the graphite matrix.

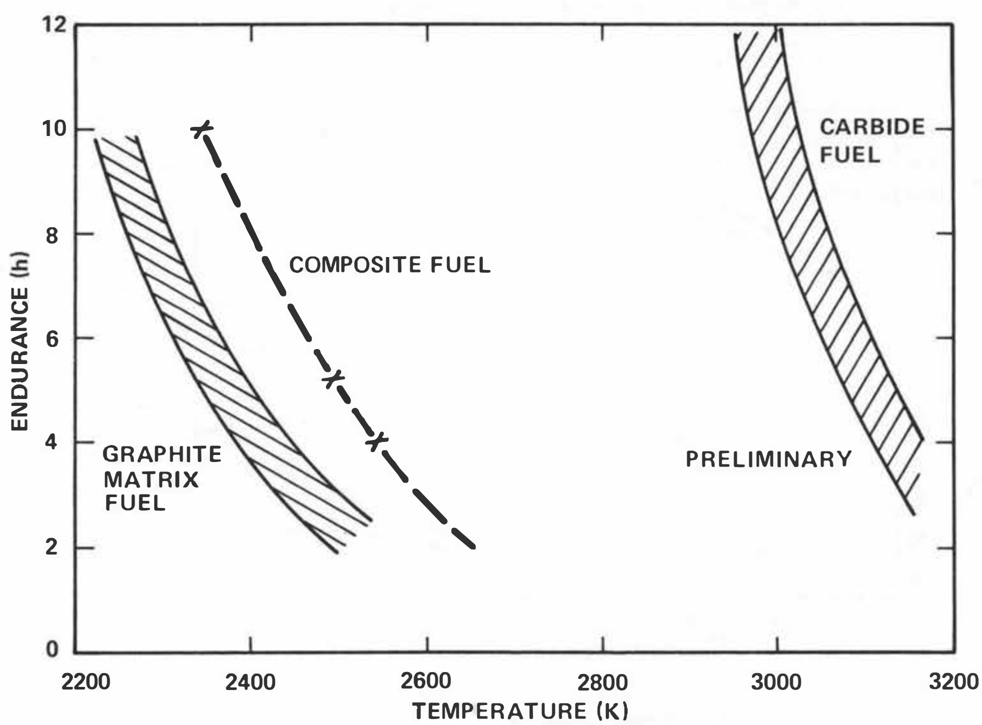


Fig. 34. Comparison of projected endurance of several fuels versus coolant exit temperature.

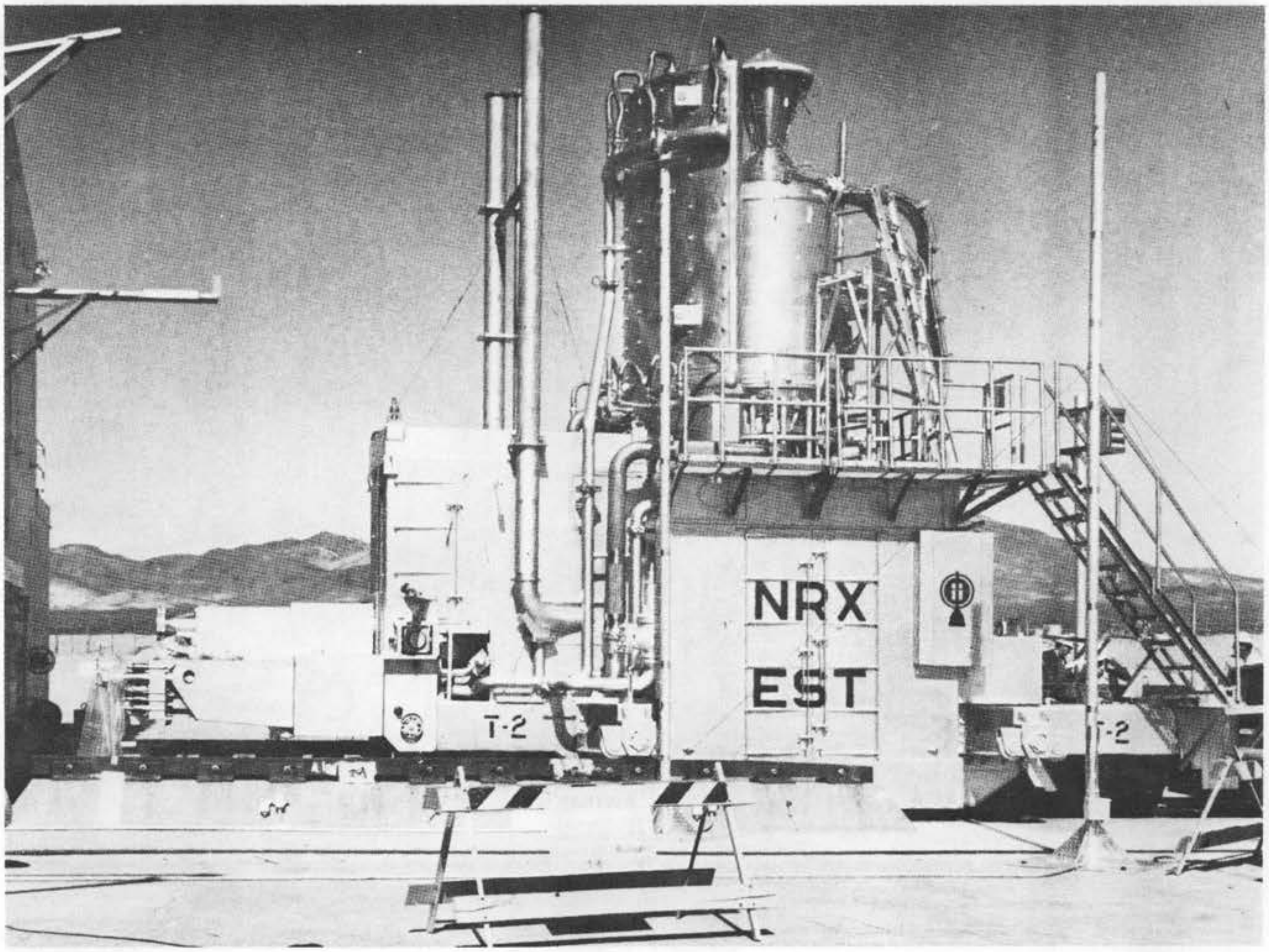


Fig. 35. View of NRX/EST test assembly on the test pad. This was the first test of a NERVA breadboard power plant with all the major flight engine components functionally connected.

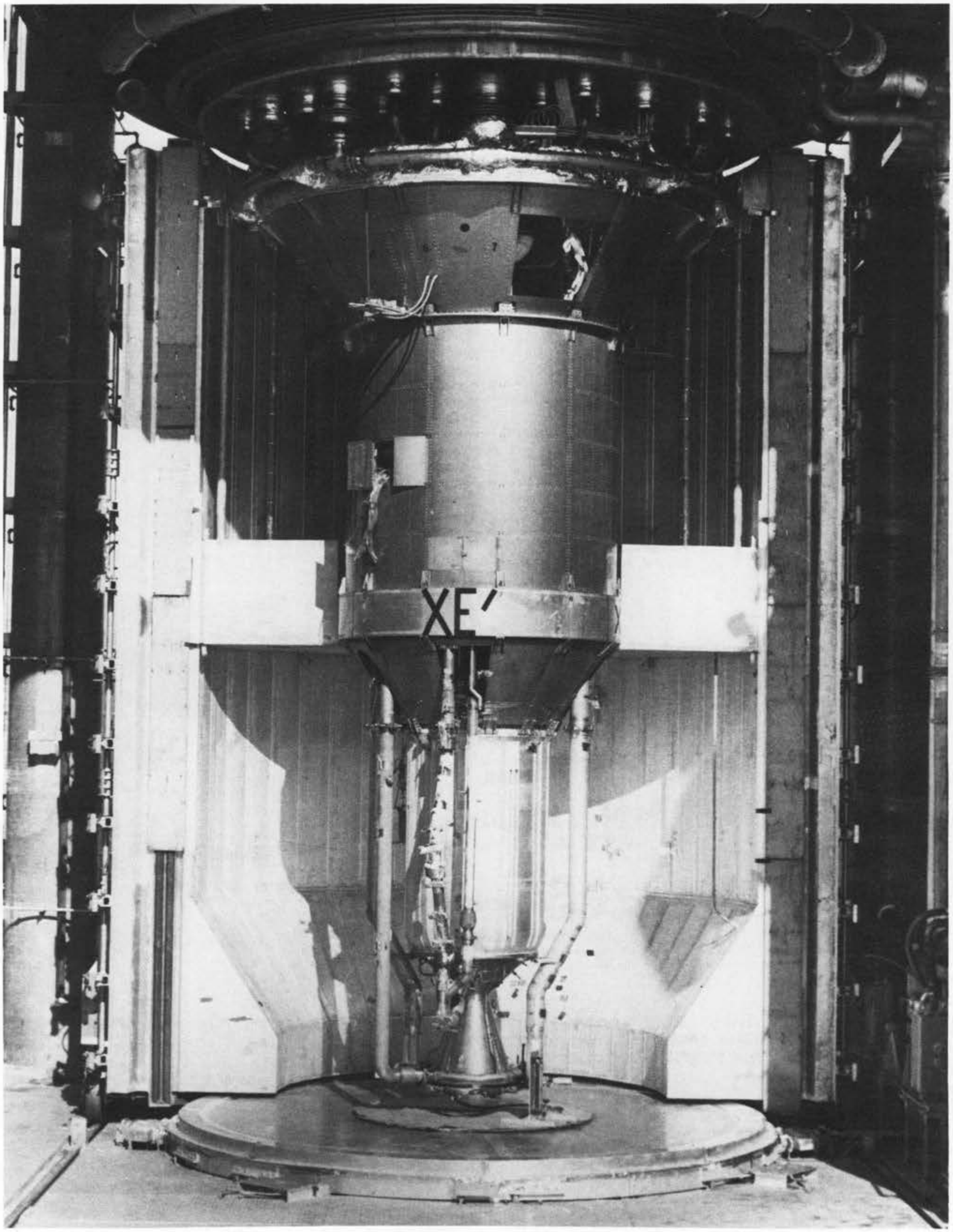


Fig. 36. XE' engine installed in its test stand. The engine is positioned in a downward-firing attitude. The test stand provided a partial simulation of outer space conditions.

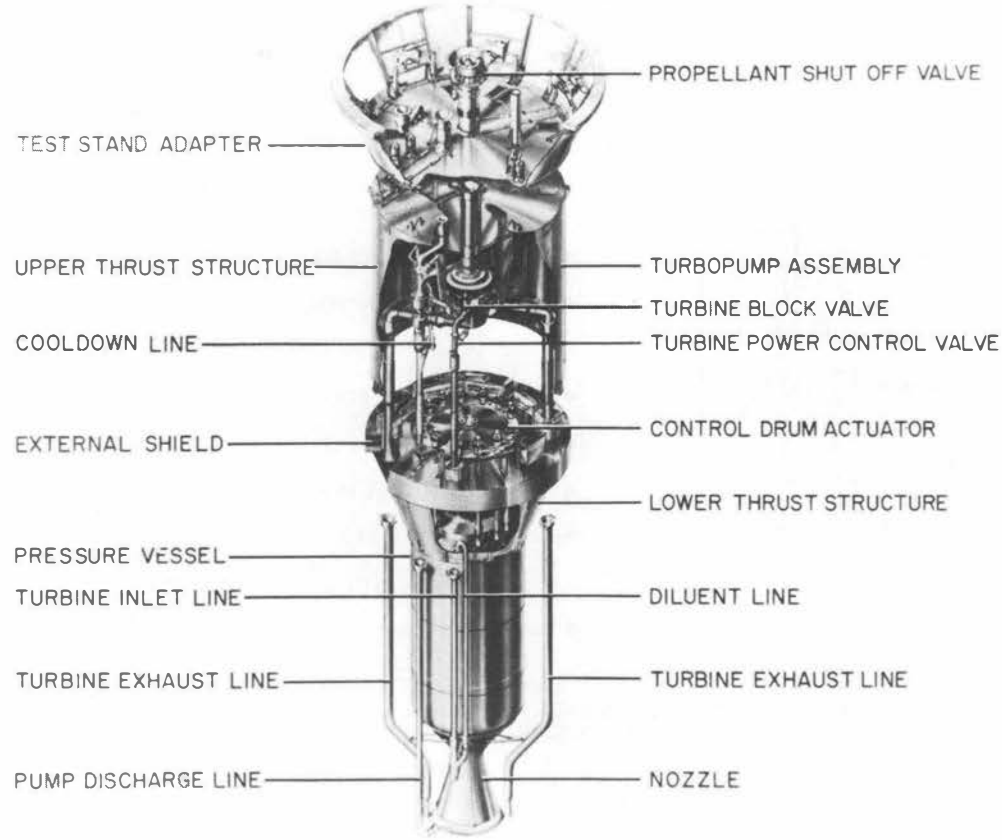


Fig. 37. XE engine design concept.

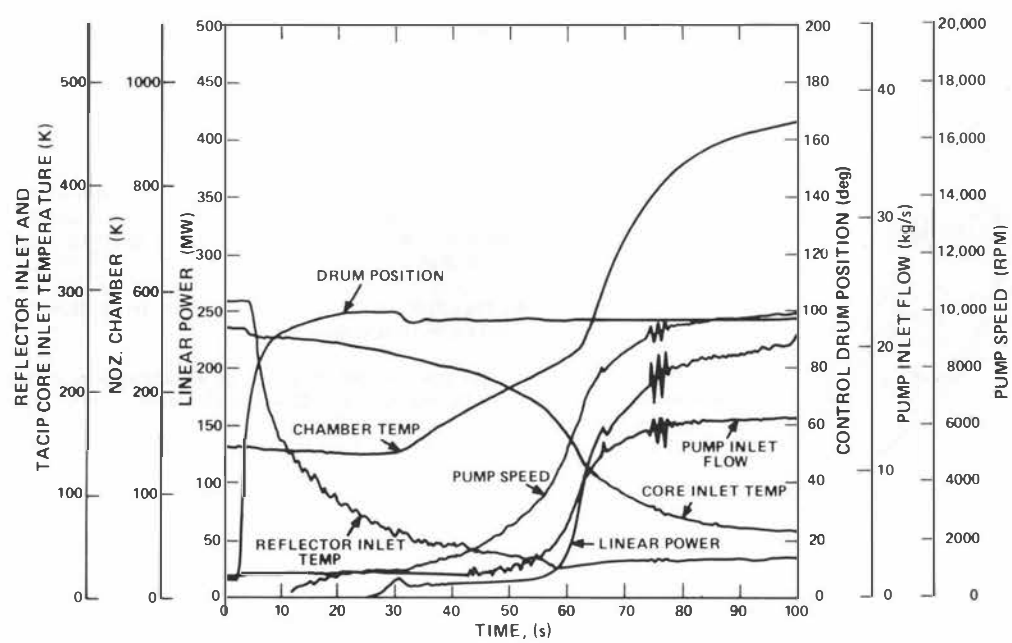
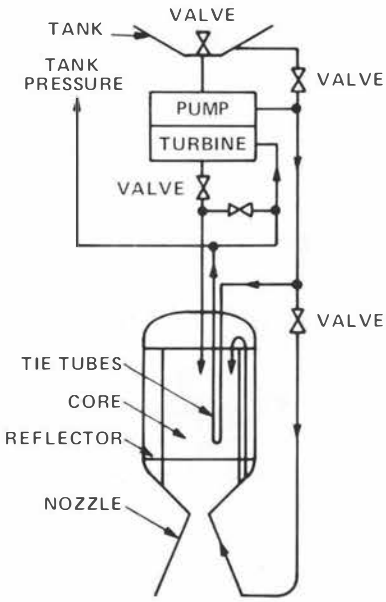
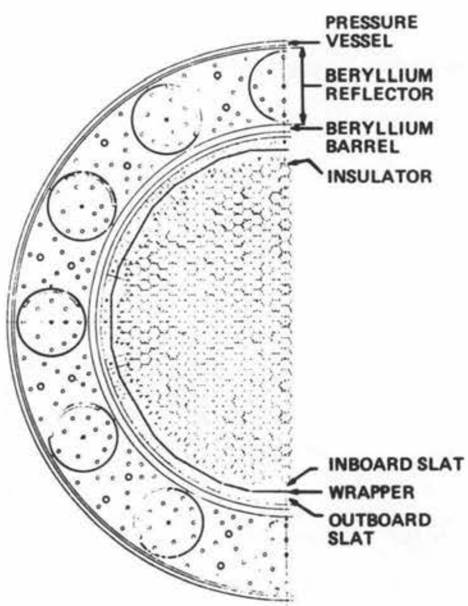


Fig. 38. Typical characteristics of the nuclear rocket engine startup. Note that chill-down of the various engine components takes about 60 s. Then the engine can be turned on to full power at a rate limited by thermal stresses in the core resulting from the temperature transient (not to exceed about 83 k/s).



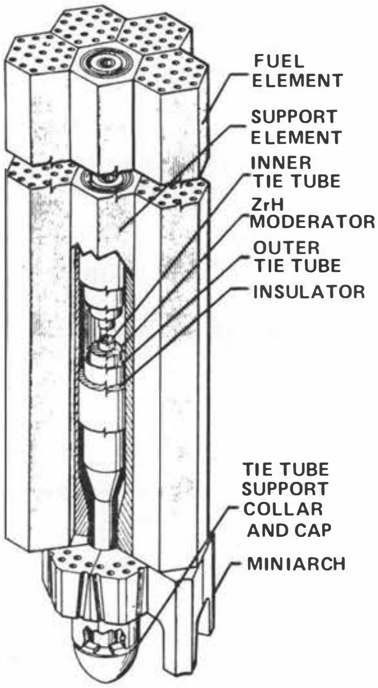
- HYDROGEN PROPELLANT
- FULL FLOW TOPPING CYCLE
- SINGLE-STAGE CENTRIFUGAL PUMP AND SINGLE-STAGE TURBINE
- REGENERATIVELY COOLED METAL-CORE SUPPORT ELEMENTS (TIE TUBES)
- RADIATION SHIELD OF BORATED ZIRCONIUM HYDRIDE
- 6 CONTROL-DRUM ACTUATORS
- 5 VALVES AND VALVE ACTUATORS
- REGENERATIVELY COOLED NOZZLE, AREA RATIO = 25:1
- UNCOOLED NOZZLE SKIRT, EXIT AREA RATIO = 100:1
- UNCOOLED NOZZLE SKIRT HINGED AND ROTATABLE
- OVERALL ENGINE LENGTH =
  3.1 m (123 in.) WITH SKIRT FOLDED
  4.4 m (174 in.) WITH SKIRT IN PLACE
- TOTAL MASS = 2550 kg (5620 lb)

Fig. 39. Schematic flow description of the Small Engine conceptual design. The engine, which required only five control valves, was designed to produce 72 kN of thrust from a 370-MW reactor.



- **PRODUCES 365 MW**
- **564 HEXAGONALLY SHAPED (UC-ZrC) C COMPOSITE FUEL ELEMENTS**
- **241 SUPPORT ELEMENTS CONTAINING ZrH NEUTRON MODERATOR**
- **19 COOLANT CHANNELS PER ELEMENT**
- **CORE PERIPHERY CONTAINS AN OUTER INSULATION LAYER, A COOLED INBOARD SLAT SECTION, A METAL WRAPPER, A COOLED OUTBOARD SLAT SECTION, AND AN EXPANSION GAP**
- **REFLECTOR IS BERYLLIUM BARREL WITH 12 REACTIVITY CONTROL DRUMS**
- **CORE SUPPORT ON COLD END BY AN ALUMINUM-ALLOY PLATE. SUPPORT PLATE RESTS ON REFLECTOR SYSTEM**
- **REACTOR ENCLOSED IN ALUMINUM PRESSURE VESSEL**
- **CAPABLE OF 83 K/s TEMPERATURE TRANSIENTS**

Fig. 40. Cross-sectional description of the Small Engine concept. The overall reactor diameter was 950 mm. The design used ZrH-moderated support elements, as was done in Pewee, to reduce the uranium critical mass.



FUEL

- FUNCTION
  - PROVIDED ENERGY FOR HEATING HYDROGEN PROPELLANT
  - PROVIDED HEAT TRANSFER SURFACE
- DESCRIPTION
  - ^{235}U IN A COMPOSITE MATRIX OF UC-ZrC SOLID SOLUTION AND C
  - CHANNELS COATED WITH ZrC TO PROTECT AGAINST H_2 REACTIONS

TIE TUBES

- FUNCTION
  - TRANSMIT CORE AXIAL PRESSURE LOAD FROM THE HOT END OF THE FUEL ELEMENTS TO THE CORE SUPPORT PLATE
  - ENERGY SOURCE FOR TURBOPUMP
  - CONTAIN AND COOL ZrC MODERATOR SLEEVES
- DESCRIPTION
  - COUNTER FLOW HEAT EXCHANGER OF INCONEL 718
  - ZrH MODERATOR
  - ZrC INSULATION SLEEVES

Fig. 41. Description of the Small Engine fuel module design showing the ZrH sleeve in the regeneratively cooled tie-tube support element.

"SMALL ENGINE" STATE POINTS AT DESIGN CONDITION

CHAMBER PRESSURE	310 N/cm^2
CHAMBER FLOW RATE	8.51 kg/s
CHAMBER TEMPERATURE	2696 K
REACTOR POWER	367 MW
SPECIFIC IMPULSE	8580 m/s
THRUST	7297 N
NOZZLE FLOW FRACTION	44.9%
TURBINE BYPASS FLOW FRACTION	11.8%
TURBOPUMP SPEED	4917 rad/s
TURBOPUMP SHAFT WORK	0.93 MW
PUMP EFFICIENCY	65%
TURBINE EFFICIENCY	80%
NOZZLE VALVE AREA	2.63 cm^2
TURBINE CONTROL VALVE AREA	3.02 cm^2

STATE POINT DESCRIPTION	FLOW RATE (kg/s)	PRESSURE (MPa)	TEMPERATURE (K)
PUMP INLET	8.51	0.12	17.0
PUMP EXIT	8.51	6.03	19.8
TIE TUBE MANIFOLD INLET	4.05	5.72	20.3
TIE TUBE FIRST PASS EXIT	4.05	5.38	56.9
TIE TUBE EXIT	4.05	5.02	428.9
SLAT MANIFOLD INLET	0.64	5.72	20.3
SLAT FIRST PASS EXIT	0.64	5.24	167.1
SLAT EXIT	0.64	5.02	431.5
TURBINE INLET	4.13	4.86	428.6
TURBINE EXIT MIXED	4.69	4.13	415.6
TURBINE BYPASS INLET	0.55	4.86	428.6
NOZZLE INLET	3.83	4.63	21.4
NOZZLE EXIT	3.83	4.21	240.4
REFLECTOR EXIT	3.83	4.21	294.9
SHIELD INLET	8.51	4.06	361.0
CORE INLET	8.51	3.96	370.1
FUEL ELEMENT EXIT	8.33	3.10	2728.0
CORE BYPASS EXIT	0.18	3.10	370.1
CHAMBER	8.51	3.10	2695.8

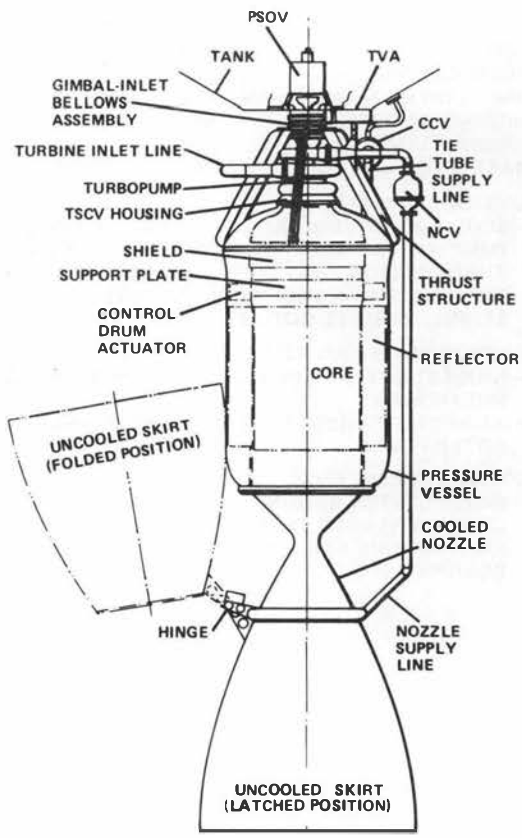


Fig. 42. Description of Small Engine operating parameters at design conditions.

	STAGE MASS, kg (lb)	USABLE PROPELLANT MASS, kg (lb)
NUCLEAR STAGE (INCLUDING SMALL ENGINE)	17783 (39205)	12814 (28250)
PROPELLANT MODULE (18.3 m)	23181 (51105)	21265 (46880)

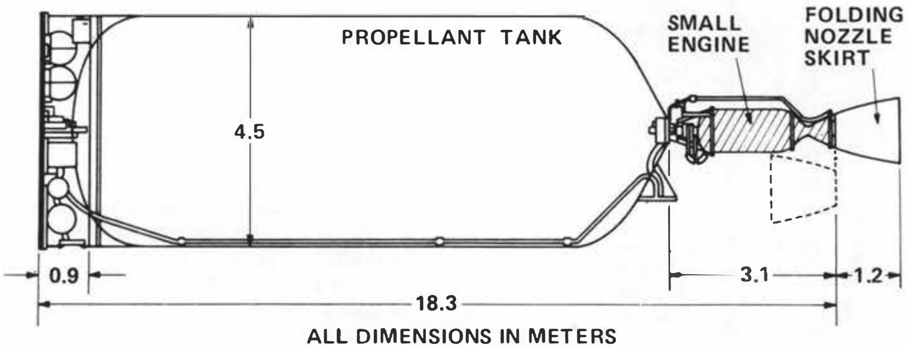


Fig. 43. Description of a nuclear rocket stage employing the Small Engine and compatible with the space and mass constraints of the space shuttle.

- FUNCTION
  - PRESSURIZE THE PROPELLANT FOR THE ENGINE FEED SYSTEM
- DESIGN CONDITIONS

	XE'	NERVA	SMALL ENGINE
PUMP DISCHARGE PRESSURE (MPa)	6.69	9.36	6.03
PUMP FLOW RATE (kg/s)	35.8	20.9 – 41.7	8.5
TURBINE TEMP (K)	648	154	429
TURBINE FLOW RATE (kg/s)	3.32	19	4.1
SHAFT SPEED (rpm)	21989	23920	46952

- CONSTRUCTION (XE)
  - SINGLE-STAGE, RADIAL-EXIT-FLOW-CENTRIFUGAL PUMP WITH AN ALUMINUM IMPELLER, A POWER TRANSMISSION THAT COUPLES THE PUMP TO THE TURBINE, AND A TWO-STAGE TURBINE WITH STAINLESS STEEL ROTORS
- REACTOR TESTS EXPERIENCE
  - NRX/EST 8 STARTS INCLUDING 54.4 MINUTES AT HIGH POWER
  - XE-28 STARTS/RESTARTS INCLUDING RUNS TO RATED POWER
- POTENTIAL PROBLEMS
  - SHAFT SYSTEM BINDING AT BEARING COOLANT LABYRINTH IN XE TESTS (INCREASE CLEARANCE AND IMPROVE ALIGNMENT)
  - BEARING LIFE

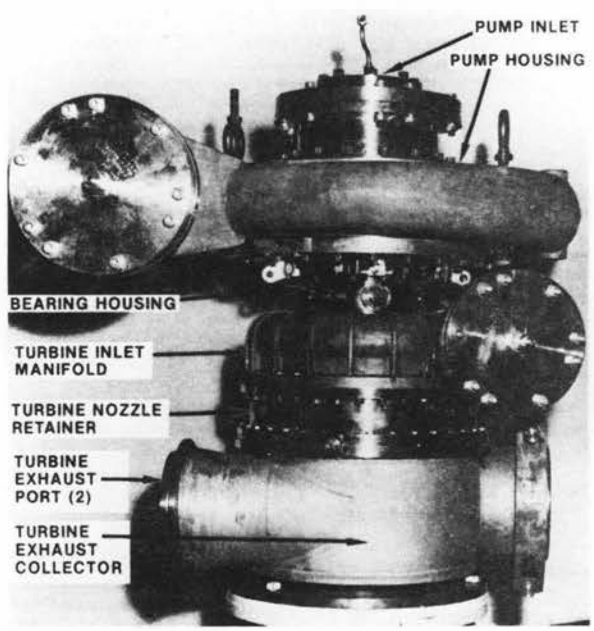


Fig. 44. Turbopump experience and design conditions for the nuclear rocket engine propellant feed system.

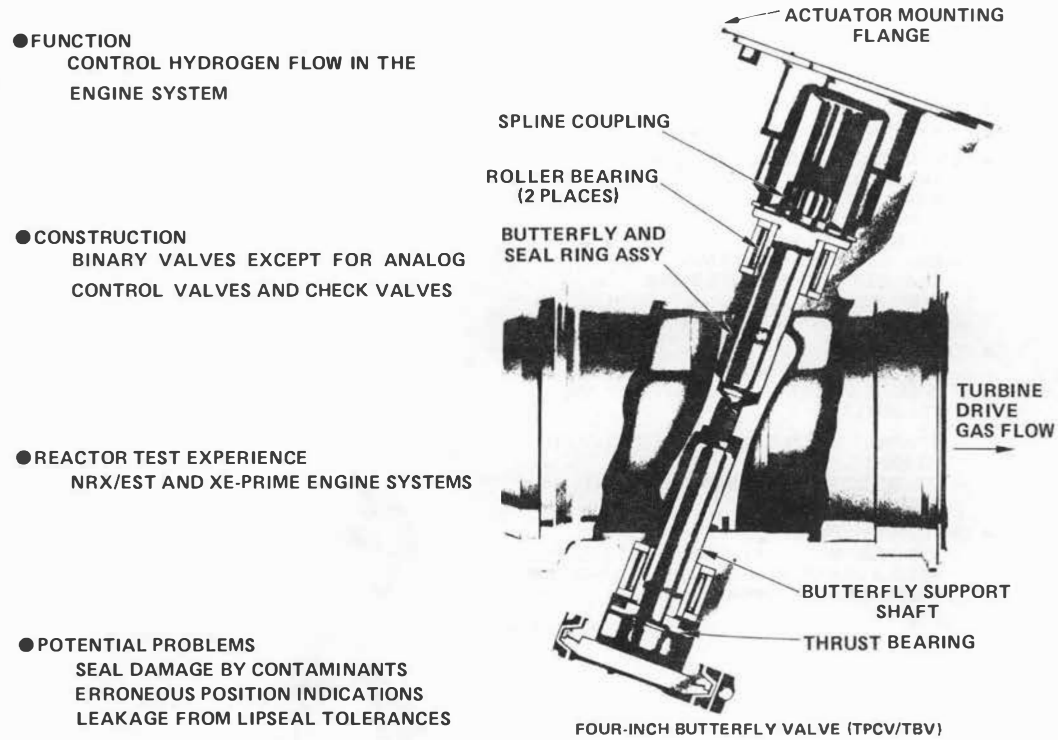


Fig. 45. Typical coolant valve employed in the nuclear rocket engine.

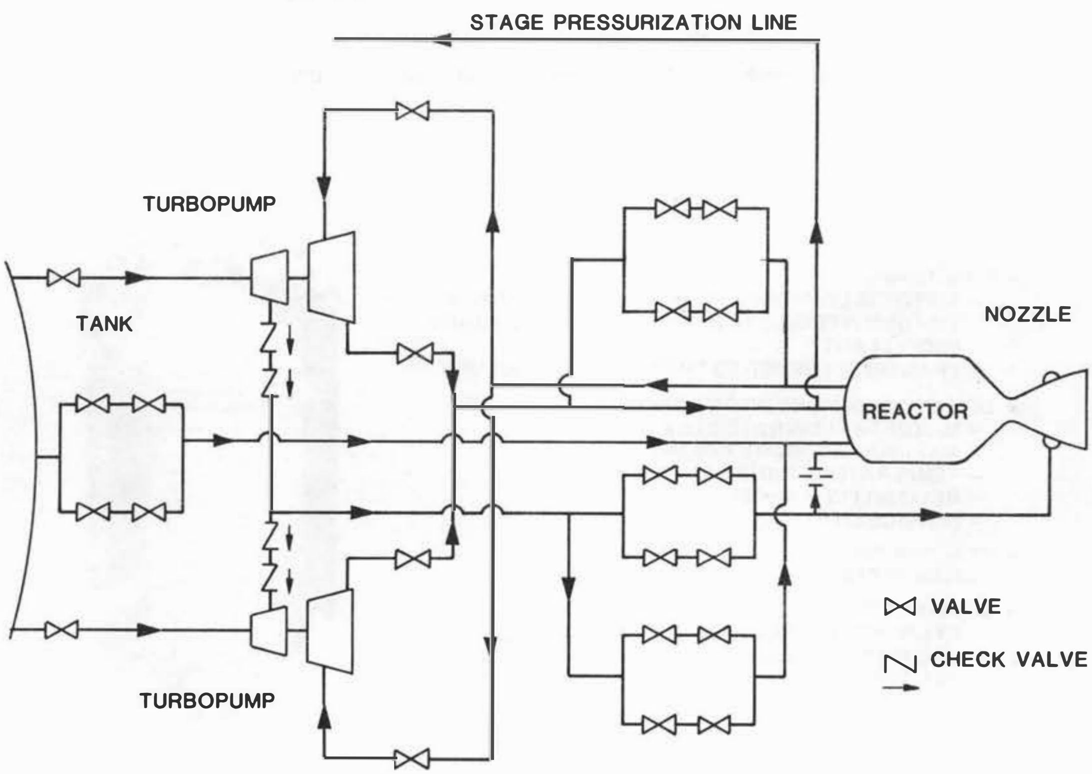


Fig. 46. Schematic engine coolant flow diagram showing the consequence of avoiding single-point failures in the coolant circuit described in Fig. 39 by adding redundant valves and turbopumps. Some 26 valves would be required.

- FUNCTION
  - EXPAND GAS TO PROVIDE MAXIMUM THRUST
- DESIGN CONDITIONS (NERVA ENGINE)
  - THRUST 334 kN
  - AREA RATIO 100:1
  - SERVICE LIFE 10 h
  - RELIABILITY 0.9998
  - CHAMBER PRESSURE 3.1 MPa
  - CHAMBER TEMPERATURE 2360 K
  - FLOW RATE 41.6 kg/s
  - COOLANT CHANNEL 28–33 K
- CONSTRUCTION
  - LH_2 COOLED SECTION TO 24:1 OF ARMCO 22–13–5 JACKET AND CRES 347 COOLANT CHANNELS
  - GRAPHITE NOZZLE EXTENSION UNCOOLED TO 100:1
  - U-TUBE CONSTRUCTION IN DIVERGENT SECTION
- UNRESOLVED PROBLEMS
  - USE OF ARMCO 22–13–5 APPEARS TO HAVE RESOLVED STRESS PROBLEMS. FABRICATION PROBLEMS ALSO APPEAR RESOLVED

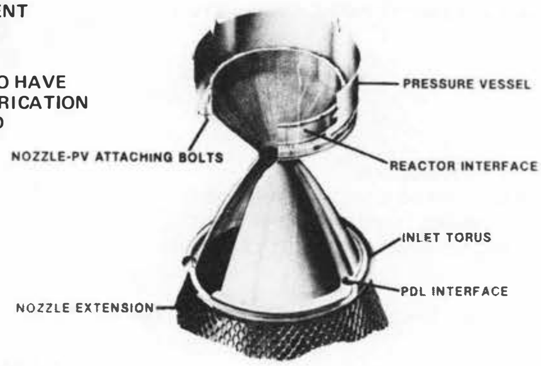


Fig. 47. Experience and NERVA design conditions for the nozzle assembly.

- FUNCTION
  - SUPPORTS COMPONENTS OF REACTOR ASSEMBLY TO FORM A PRESSURE SHELL FOR THE HYDROGEN PROPELLANT
  - TRANSMITS THRUST TO THE THRUST STRUCTURE
- DESIGN CONDITIONS (NERVA ENGINE)
  - MAXIMUM FLOW RATE 37.6 kg/s
  - MAXIMUM PRESSURE 8.66 MPa
  - TEMPERATURE RANGE 20 – 180 K
  - RELIABILITY 0.999997
  - SERVICE LIFE 10 h
- REACTOR TESTS
  - NRX–5 TESTS, XE′
- CONSTRUCTION
  - CYLINDER, TOP CLOSURE, BOLTS AND SEALS
  - ONE-PIECE EXTRUDED FORGING OF ALUMINUM ALLOY 7075–773
  - SURFACE COATING $A\ell_2O_3$
- UNRESOLVED DESIGN ITEMS ON NERVA ENGINE
  - METHOD TO ASSURE BULK PRELOAD
  - FINALIZE SURFACE COATINGS

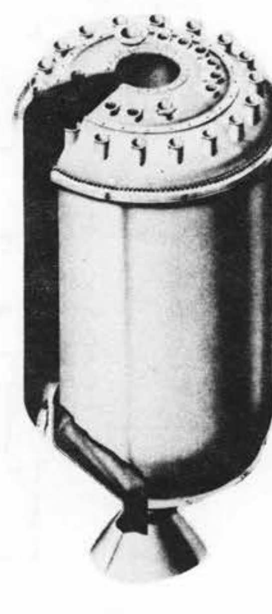


Fig. 48. Experience and NERVA design conditions for the reactor pressure vessel.

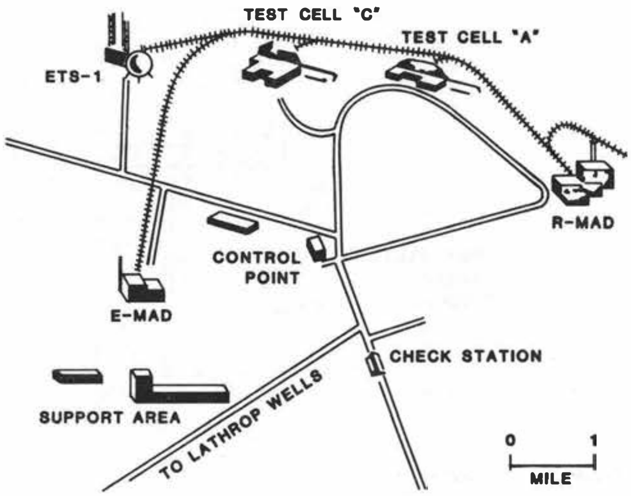


Fig. 49. Arrangement of facilities at the Nuclear Rocket Development Station.

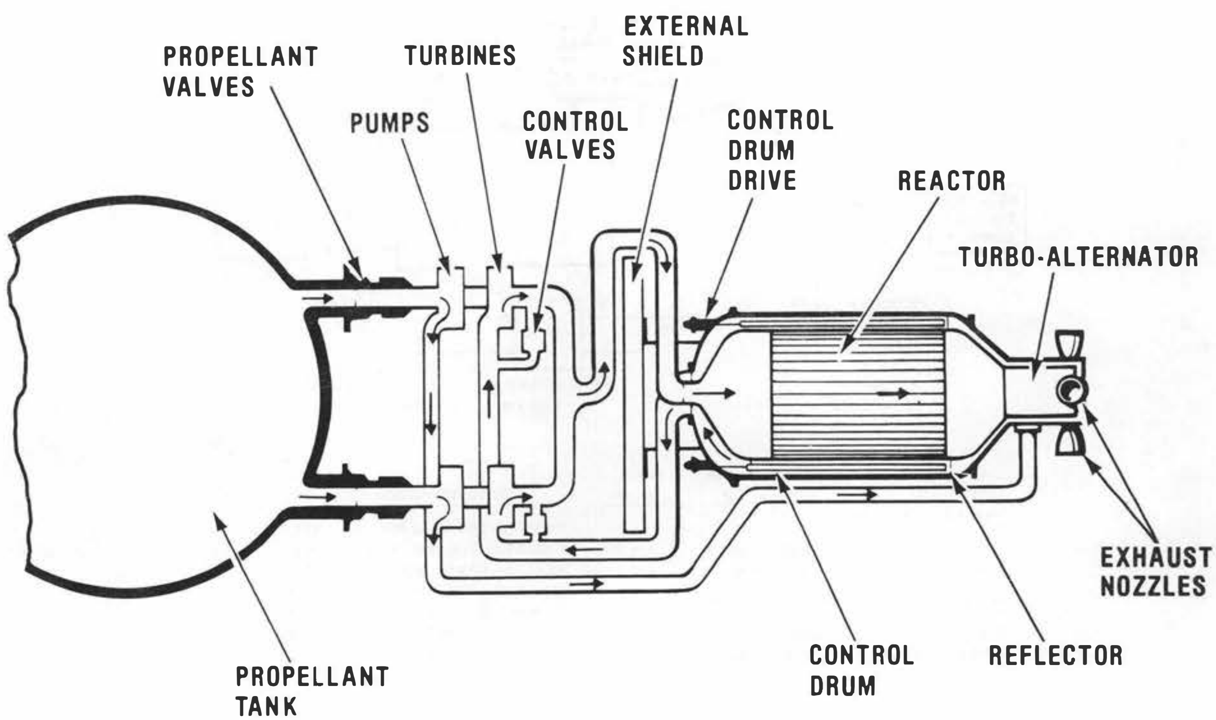


Fig. 50. Schematic coolant flow diagram of an open-cycle, gas-cooled, nuclear space power plant.

MODES

- HIGH-POWER ROCKET MODE OF 365 MW, 10 h LIFE, 2600-2660 K PROPELLANT TEMPERATURE, NEGLIGIBLE FUEL BURNUP
- LOW-POWER ELECTRICAL MODE OF 10-25 kW(e), I MW(t), ORGANIC RANKINE CYCLE DRIVEN BY THERMAL ENERGY FROM STRUCTURAL SUPPORT SYSTEM, USES PROPULSION MODULE SURFACE TO SUPPORT RADIATOR UP TO 10 kW(e)

ENGINE MODIFICATIONS

- STRUCTURAL SUPPORT SYSTEM ISOLATION VALVES FOR LOW-POWER
- CHANGE REACTOR DOME AND TIE TUBE CORE SUPPORT PLATE LINES FOR WIDER TEMPERATURE RANGE OPERATION TO STAINLESS STEEL FROM AI, AND ACTUATOR WINDINGS TO CERAMIC FROM POLYIMIDE

LIFETIME

- STUDY SHOWED TWO YEARS LIFE WILL NOT SIGNIFICANTLY AFFECT REACTIVITY CONTROL MARGIN

*STUDIED IN EARLY 1970's BY J. ALTSEIMER, L. A. BOOTH, "THE NUCLEAR ROCKET ENERGY CENTER CONCEPT" LA-DC-72-1262, 1972, BASED ON IDEAS OF JOHN BEVERIDGE

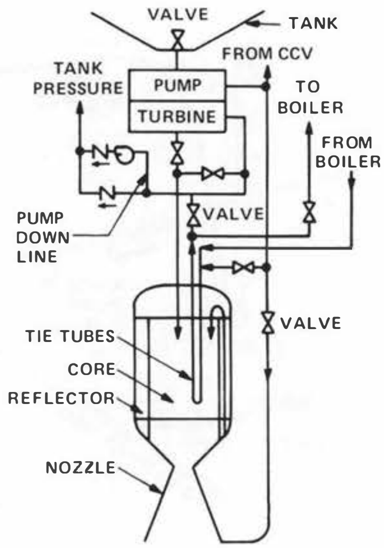
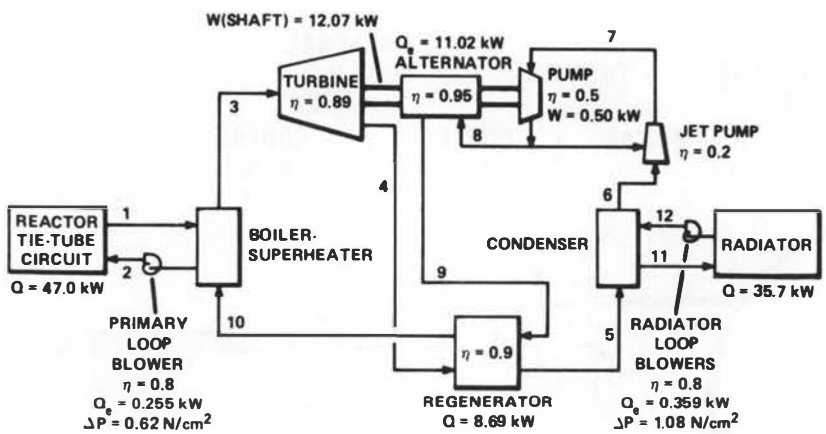


Fig. 51. Schematic coolant flow diagram for a dual-mode nuclear rocket engine based on the Small Engine design parameters. The engine provides the normal high-power rocket propulsion mode, plus a low-power, continuous electric-power generation capability using an organic Rankine cycle driven by a closed loop that cools the support elements in the core.



COMPONENT INLET	FLOW, (kg/s)	TEMP., (K)	PRESS., (N/cm²)
PRIMARY LOOP – GH_2			
1 BOILER	0.0622	555.6	419.7
2 REACTOR	0.0622	504.0	420.0
POWER CONVERSION SYSTEM – THIOPHENE			
3 TURBINE	0.0891	550.0	140.0
4 REGEN-VAPOR	0.0891	438.9	5.02
5 CONDENSER	0.0891	354.8	4.85
6 JET PUMP	0.0891	333.1	4.52
7 PUMP	0.178	334.5	14.90
8 ALTERNATOR	0.0891	342.1	155.6
9 REGEN-LIQUID	0.0891	346.2	154.8
10 BOILER	0.0891	411.6	150.0
HEAT REJECTION LOOP – GH_2			
11 RADIATOR	0.0798	355.2	419.5
12 CONDENSER	0.0798	304.6	420.0

POWER BREAKDOWN (kW)	
REACTOR THERMAL POWER	47.0
RADIATOR THERMAL POWER	35.7
ALTERNATOR OUTPUT	11.0
CONTROL SYSTEM	0.39
PRIMARY BLOWER	0.26
RADIATOR BLOWERS	0.36
NET ELECTRIC OUTPUT	10.0
TOTAL SYSTEM SPECIFIC MASS	48.6 kg/kW(e)

Fig. 52. Design operating parameters for a 10-kW_e organic Rankine power-generating plant to be used with the Small Engine.

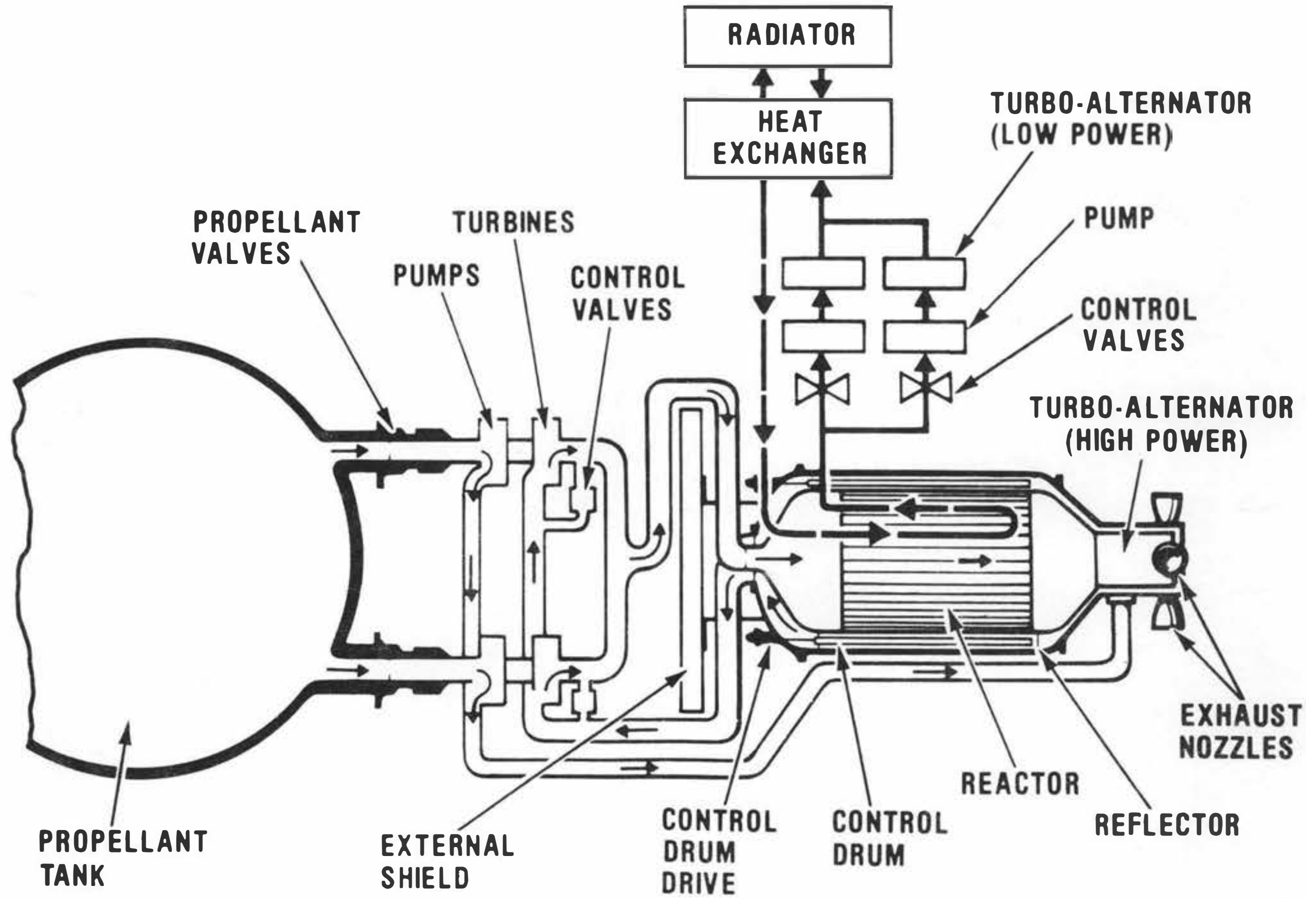


Fig. 53. Schematic diagram of a bimodal power-generating plant based on the Rover reactor design. It would have an open-cycle, high-power generation mode plus a continuous, closed-cycle, low-power generation capability.

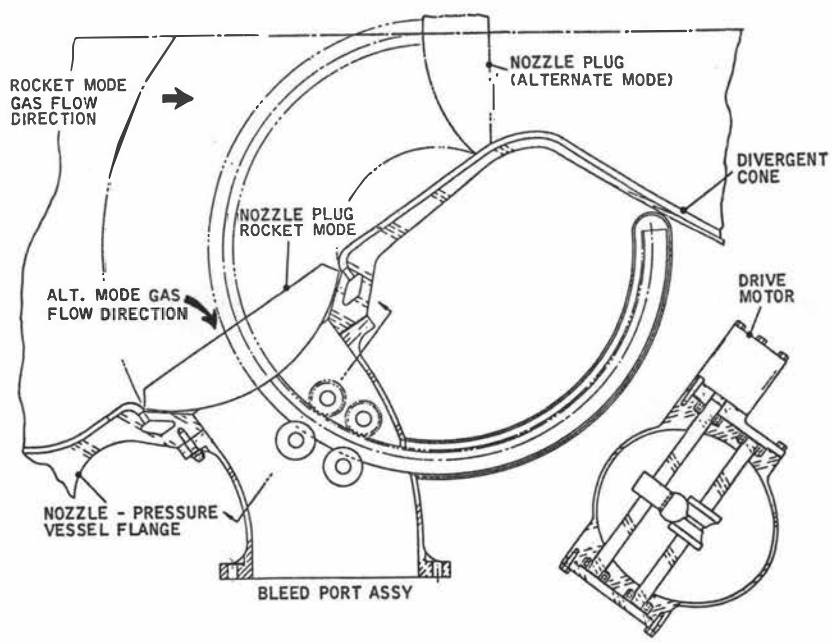


Fig. 54. Poppet valve design used to switch from a rocket propulsion mode to a power generation mode.

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