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AN OVERVIEW OF TESTED AND ANALYZED NTP CONCEPTS

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ABSTRACT

If we buy into the goals of the Space Exploration Initiative (SEI) and accept that they are worthy of the hefty investment of our tax dollars, then we must begin to evaluate the technologies which enable their attainment. The main driving technology is the propulsion system; for interplanetary missions, the safest and most affordable is a Nuclear Thermal Propulsion (NTP) system. This paper presents an overview of the NTP systems which have received detailed conceptual design and, for several, testing.

INTRODUCTION

Why should the government of the American people invest billions of dollars on an initiative for space exploration? The following visions attempt to answer why:

- To further understand the origin and history of our Solar System, the origins of life, and the ultimate fate of our universe
- To utilize vast untapped space mineral resources (gold, platinum, titanium, chromium) and zero-gravity material processing awaiting commercialization for the benefit of mankind
- To stimulate a wide range of technical innovations which have previously shown abundant application in the consumer marketplace
- To strengthen the U.S. economy by high technology investments which improves American competitiveness and global market share
- To provide a motivational stimulus and direction for the advancement of the U.S. science and engineering talent in new generations
- To re-establish and maintain American preeminence in technological innovation and space leadership

If we agree these are tangible goals worthy of the investment to achieve them, then we need to evaluate the technologies allowing their attainment in a safe, affordable manner.

To begin with, shorter transit times are desirable to reduce the impact of the inter-planetary journey on the crew and vehicle (radiation, zero-gravity, psychological isolation, equipment degradation). The technological limit on the minimization of trip time is the propulsion system. If further developed, nuclear propulsion technology allows significantly reduced travel durations and reduced vehicle weight (launch costs) to roughly 1/2 that of a current chemical rocket propelled vehicle. From a safety standpoint, the robustness of nuclear thermal propulsion systems allows for greater abort-to-Earth flexibility, and by reducing trip time, allows for a reduced crew inter-planetary radiation exposure.

Since 1946, many nuclear thermal propulsion systems have been conceived, evaluated, and some even tested. The following sections first cover the general design of nuclear propulsion systems and then describe the systems which received detailed analysis. The intent is to inform the reader sufficiently on the main precursor technology development needed prior to attainment of the space exploration vision.

NUCLEAR THERMAL PROPULSION OVERVIEW

In conventional chemical rocket engines, such as the SSME, turbopumps drive the propellants (oxidizer (LOX) and reactant (LH_2)) from the tanks into the combustion chamber, where the heat of reaction increases the mixture stagnation enthalpy; the high temperature mixture is then exhausted by a convergent-divergent nozzle (Figure 1). For a given mixture molecular weight, a higher stagnation enthalpy results in a higher exhaust velocity and thrust per unit flow rate (I_{SP}). With nuclear thermal propulsion (NTP) systems, the increase in stagnation enthalpy is achieved by pumping the propellant through a fission reactor core where it cools the reactor; hence, the combustion process is replaced and only a single propellant is required (Figure 1). Since I_{SP} is inversely proportional to molecular weight, a low molecular weight propellant, like hydrogen, increases I_{SP}. The advantage of NTP systems is that by using a single propellant with the lowest molecular weight, H_2, a more than two-fold increase in Isp can be realized over current chemical systems; then, the maximum Isp is limited only by the maximum core fuel temperature and heat transfer rate to the propellant. The operating principals for solid core nuclear fission rockets are presented in greater depth in References 1 and 2.

The goal of fission reactor design for NTP is to achieve a high power output per unit volume for low weight while providing for a high coolant passage surface area per unit volume for high heat transfer. Both thermal- (low velocity) and fast- (high velocity) neutron induced fission reactor designs using U^{235} fuel have been evaluated for NTP

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systems. A typical reactor consists of a core, reflector, radiation shield, and pressure vessel.

The core contains the U^{235} fuel elements, support structure, and moderator (thermal-reactor only). The moderator consists of light weight materials, such as graphite, beryllium-oxide (BeO) or zirconium-hydride (ZrH), placed in the core to slow the neutrons to thermal velocities. The moderator can be built into the support structure (heterogeneous core) or incorporated in the fuel element (homogeneous core). The latter design results in a lower core weight but a substantially higher fuel loading and core cost (ref. 3).

The reflector is an additional region of moderating material that surrounds the core to reduce neutron leakage by reflecting neutrons back into the core. The reflector assembly may also contain a rotating drum elements used to regulate the number of neutrons reflected back into the core to cause additional fissions, and hence, reactor power. These control drums are made from reflector material except a portion which is covered by a neutron absorbing material such as boron-carbide. The absorber on the drum can be rotated close-to or away-from the core to decrease or increase the number of reflected neutrons, respectively.

A protective radiation shield is normally placed between the reactor and sensitive engine components to decrease radiation heating and material damage from the substantial neutron and gamma field of the fission reactor. Light weight materials with low atomic weights, such as lithium-hydride, are used for neutron attenuation, while the more penetrating gamma-rays are better handled by a denser material, such as tungsten. The shield may be positioned outside the core and reflector, to intercept the largest possible solid angle, as seen by the reactor, for weight and size minimization. The core, reflector and shield are contained within a pressure vessel onto which the exhaust nozzle is attached. Nozzle thrust is transferred through the nozzle and pressure vessel to the tank thrust structure and spacecraft.

Fuel Element

For NTP systems, the goal of fuel element design is to achieve the highest possible propellant exit temperature while maintaining structural integrity under design loads. The fuel element must incorporate sufficient fissile fuel to provide the required power and to maintain reactivity across the design life; for reference, thermal-reactors consume 1.24 grams of U_{235} per megawatt in one day. Ultimately, the fuel element design is a compromise between fabricability, corrosion resistance, and strength at high temperature. The fuel elements in NTP systems may be categorized according to the uranium fuel compound, the matrix material, and the fuel element form.

Fuel Compound. The fuel elements in NTP systems usually contain the fissile fuel, U^{235}, in either nitride, oxide, or carbide compounds of micrometer particle size. These compounds have higher melting points and better strength characteristics than pure U^{235} (Table 1). Another important consideration of high temperature fuel compounds is their vaporization rate (Figure 2). The top curve is UO_2 which has the highest rate of all materials shown; it would not be practical to make a fuel element out of plain UO_2 because of its excessive vaporization rate. To reduce the fuel vaporization rate, the fuel compound should be completely contained within a matrix material that has a lower vaporization rate and is compatible with the coolant.

Note, UC_2 particles are extremely reactive and revert to oxide in the presence of air, particularly humid air. Oxidation of UC_2 loaded fuel elements could cause swelling up to 4%. This problem maybe solved by coating the UC_2 particles with pyrolytic graphite.

Matrix Material. The fuel elements usually contain fine particles of the fuel compounds suspended in either a refractory-metal or carbon-based matrix material. The latter matrix material lends itself to thermal reactor designs since carbon is a neutron moderating material; hence, no separate moderator structure would be necessary and the resulting homogeneous reactor would have a lower weight. Carbon-based matrix materials are advantageous because of their high melting temperatures, low density, low neutron absorption, and high strength at high temperature. However, carbon reacts with hot hydrogen to form methane and other hydrocarbons; this corrosion may be reduced with coatings, such as NbC or ZrC, otherwise carbon mass loss can affect reactor neutronics and life. The carbon-based matrix materials include graphite, carbide, and a composite of both. Figure 3 shows the structure of a graphite matrix where coated fuel particles are embedded in a continuous matrix. Also shown is a comparison with the graphite-carbide composite matrix; in the composite matrix, uncoated fuel particles are dispersed as to form a continuous webbed phase of carbide. For the graphite matrix coated with NbC or ZrC, once the coating between the matrix and the propellant cracks, carbon is lost indefinitely through the cracks since the graphite is continuous. With the composite matrix, carbon is lost through the cracks until the carbide web is reached; carbon stops escaping except a small amount diffusing through the carbon. The difference in the carbon loss rates between carbon-based matrix materials is shown by Figure 4.

The refractory-metal matrix materials lend themselves to fast-neutron reactor designs (ref. 4). Refractory-metal matrix fuel elements have been developed, such as with $Mo\text{-}UO_2$ and $W\text{-}UO_2$ dispersions (i.e. "cermet" matrix material). Moreover, a braided tungsten-wire tube has also been used as a matrix material to contain tungsten coated UN particles (ref. 5); tungsten vapor deposition and subsequent swaging is used to seal the fuel in the tube matrix. Tungsten is an ideal refractory-metal matrix material for the following reasons: 1) tungsten has the highest melting point of any element (Table 1), 2) tungsten has a low vapor pressure (much less than graphite), 3) tungsten does not react with hydrogen, and 4) tungsten has a high thermal conductivity.

Note, some reactor designs call for no matrix to contain the fuel particles. These reactors contain either beds of particles (0.02-0.03 inches in diameter) between two porous frits (Figure 16) or thin wall refractory metal tubes filled with the fuel compound (fuel pins) around which coolant flows.

Element Form. The fuel elements in NTP reactors have been designed in many forms, depending on the matrix material. The various forms conceived include the following: plates, wires, cylinders, hexagonal (prismatic) tubes, particles, and pellets. The configurations which have received the most development include the following: UO_2-graphite plates, UO_2-graphite cylinders with 1, 4, or 7 coolant channels, UC-graphite hexagonal tubes with 19 coolant channels, UO_2-cermet hexagonal tubes with 19 metal tube coolant channels, UO_2-tungsten wire, and UC-coated particles and pellets.

Support Element

For axial flow prismatic (hexagonal) fuel elements, the core pressure drop is high and therefore, the axial loads on the fuel elements are also large. In the NERVA reactor design (discussed in latter sections), tie-rod support elements were incorporated in the core design; the tie-rod is similar in shape to the fuel elements but contains no fuel. Typically, there are either two or six fuel elements per support element. The tie-rod support element is cooled by either a single-pass of coolant, as are the fuel elements, or are cooled in a two-pass, regenerative mode (Figure 5). Since the tie-rods are usually unfueled, the single-pass tie-rod exit temperature is lower than the fuel element exit temperature; therefore, the mixed core exit temperature will be lower. The two-pass tie-rod exhausts back into the core inlet plenum, therefore, the core exit temperature remains high, and should result in a higher mixed mean exit temperature.

Engine Turbopump Drive Cycle

To avoid the need for an auxiliary power system for driving the propellant (H_2) pump, hot propellant is extracted from the system to drive a turbopump assembly. NTP concepts are mainly based on two cycle flow-path layouts which differ by the location of the hot hydrogen extraction.

Generally, liquid (or possibly slush) H_2 is pumped from the tank to a nozzle coolant manifold. The hydrogen flows through coolant channels to cool the nozzle walls and throat. The flow then cools the reflector and pressure vessel which receives radiation heating. In a "topping-cycle" (Figure 6), this heated hydrogen is routed to drive the turbine; the hydrogen then returns to the reactor vessel inlet. Next, the hydrogen cools the dome shield and core support structure. Finally, the hydrogen enters the core, cools the fuel elements, and increases in stagnation enthalpy. The flow through the core is either axial along the length or radial from an outer-to-inner plenum, or vis-a-versa (Figure 7). The hot core cooling gas exits into the nozzle plenum chamber and then through the nozzle to produce thrust.

In a "hot-bleed-cycle" (Figure 6), a small portion (~3%) of the hot hydrogen is extracted through a bleed port in the plenum to drive the turbine. The turbine inlet temperature is adjusted by mixing the hot gas with a quantity of cooler pump exit gas. The turbine exit drive gas is routed to auxiliary nozzles for roll control or is just dumped. The average I_{sp} of the hot-bleed-cycle is lower than for the topping-cycle since the auxiliary nozzles operate at a lower temperature; however, the topping cycle maybe more difficult to implement. Both cycles are started with a "boot-strap" technique which uses the reactor heat capacity and tank pressure for initial turbine drive fluid energy.

TESTED NTP SYSTEMS

Since 1955, several projects have been sponsored by the United States government to investigate gas-cooled, nuclear fission reactor-based space propulsion systems. These projects were conducted under two main programs, ROVER and NERVA, for research and development, respectively. Over 1.5 billion dollars (1968) were invested by the U.S. under these programs leading towards the development of a nuclear thermal propulsion system. These tests were conducted at the Nuclear Rocket Development Station (NRDS) at Nevada's Nuclear Test Site in Jackass Flats. Over 20 thermal-reactors for NTP systems were designed, built and tested (Table 2). Using reactor power levels of 1100, 1500, and 5000 thermal megawatts, thrust levels of 55, 75, and 250 klbf were to be demonstrated, along with restart and sustained burn capability. Discussion of individual reactor test objectives and results is presented in Reference 6. These rocket tests were conducted open cycle with hydrogen coolant/propellant exhausted into the atmosphere; however, current environmental standards would require similar tests to be conducted in a closed cycle mode (ref. 7, 8). For interplanetary missions, after chemical systems such as SSME or RL-10, NERVA-derivative nuclear rockets are next in their technology maturation. The NERVA and ROVER programs were terminated short of actual flight test on January 5, 1973, due to the indefinite postponement of manned Mars missions; following the Apollo program national priorities changed drastically (ref. 9). For a comparison, an overview of the USSR's nuclear rocket design philosophy is presented in Reference 10.

ROVER Program

The initial nuclear rocket program (ROVER) commenced in 1953 as a backup for the chemical ICBM rocket propulsion development efforts. The ROVER program initially consisted of two exploratory studies, KIWI and TORY, at Los Almos Scientific Laboratory (LASL) and Lawrence Livermore Laboratory (LLL), respectively. After review of these studies, it was decided that LASL should proceed with a nuclear rocket development program (ROVER) and that the efforts of LLL should be redirected towards a nuclear ramjet development program (PLUTO). Under the PLUTO program, several successful tests of air-cooled reactors, TORY II-A and TORY II-C, were conducted to demonstrate the feasibility of nuclear powered ramjet engines, for use at low altitude, Mach 3 flight up 10 hours in duration (ref. 11, 12, 13).

Under the ROVER program, extensive research was completed on solid core nuclear rocket engines. The main phase began in 1955 at Los Almos Scientific Laboratory under the auspices of the

Atomic Energy Commission (AEC) and the United States Air Force (USAF) (ref. 14). Under the ROVER program, several H_2-cooled, graphite-moderated, beryllium-reflected, U_{235}-fueled reactors were built and tested. The basic concept was to heat a high pressure propellant to temperatures of 4500 °R and to expand the high temperature propellant in a high expansion ratio nozzle. The program consisted of several research reactor series including KIWI, Phoebus, Peewee-1, and Nuclear Furnace-1.

KIWI Reactors. The 8 ground reactors under the KIWI series were tested from mid-1959 to mid-1964 (Table 2). These reactors ranged in power from 70 Mwt to 1000 Mwt.

The KIWI-A reactor (ref. 15) featured an 18 inch diameter core moderator of D_2O, surrounded by four layers of UO_2 loaded, uncoated graphite fuel plates and one unloaded layer of plates. The resulting core size was roughly 33 inches in diameter and 54 inches in length. The annular graphite reflector surrounding the core was approximately 17 inches in thickness. Under testing in 1959, the hydrogen working fluid flow rate was 7 pounds per second during the 5 minute run time; a power level of 70 Mwt was achieved. Post-test core examination revealed that some core elements reached 5200 °R.

KIWI-A' and KIWI-A3 reactors were similar to KIWI-A, except the graphite plates were replaced by long cylindrical, 4-hole, graphite fuel elements with coolant holes to reduce the observed element temperature. The KIWI-A' reactor (ref. 16) was tested in 1960 at 88 Mwt for six minutes. The existence of a major structural weakness within the KIWI-A' core was rather dramatically illustrated during the full-power portion of the run by three separate bursts of glowing fuel element fragments ejected from the nozzle. Post-test examination show some fuel element blistering, corrosion, and transverse fracturing. The KIWI-A3 reactor (ref. 17) was tested in 1960 at an average power of 112.5 Mwt for 259 seconds. As with the previous test, the core experienced structural damage indicating that tensile loads on graphite structures should be avoided. The KIWI-A series of tests (ref. 18) demonstrated the following technologies: instrumentation and control, fuel element design and fabrication, structural design, and testing techniques.

The KIWI-B series (ref. 19, 20, 21) was designed to achieve a 10-fold increase in power (1000 Mwt) over the KIWI-A series while holding the size constant, thus demonstrating the basic reactor concept for the Westinghouse/Aerojet General team to develop. This was achieved by eliminating the 18 inch core moderator, increasing the number of fuel elements and coolant holes, and by increasing working fluid density (liquid versus gaseous hydrogen). Neutronic control was achieved by 12 rotating drums, containing boron carbide, within the beryllium reflector. Like the KIWI-A reactors, the KIWI-B reactors used pyro-coated UO_2 fuel beads in a graphite matrix; except the last reactor, KIWI-B4E (ref. 22), which used 50-150 µm diameter UC_2 particles coated with 25 µm pyrocarbon. Throughout this series, six hexagonal fuel elements were clustered around a single tie-rod support element, cooled by a single pass of H_2 that exhausted into the nozzle plenum chamber.

Beginning in the fall of 1961, the early KIWI-B reactors were slowly increased in power from 300 to 1000 Mwt. Post-test examinations revealed a core fuel element instability problem which resulted in broken and missing core elements; this result was evident from bright flashes in the nozzle exhaust during the tests.

It was concluded that a dynamic flow instability, in the gap between adjacent fuel element clusters, had caused strong vibration in the core. The KIWI-B4 series incorporated design changes to constrain element movement. The positive results from these reactor tests at full power cleared the way for design and fabrication of flight type reactors, such as the NRX series of the NERVA program.

At the end of the KIWI reactor test series, nuclear rocket engine clustering was investigated. In September 1964, two KIWI reactors were positioned adjacent to one another in a cluster. The results of this zero-power experiment verified there is little nuclear cross-talk between reactors and that they could be operated in clusters, much like chemical engines (ref. 23).

The final reactor to carry the KIWI name was used in a transient nuclear test, KIWI-TNT. This reactor test was a special flight safety experiment to study the behavior and effluent of a KIWI reactor undergoing a sudden excursion and explosion. The modified KIWI-B4E reactor was intentionally destroyed at the NRDS by placing it on a fast excursion through rapid rotation of the modified control drums, followed by mechanical explosion (non-nuclear). Test results showed 1) a maximum core temperature of 3900 °R, 2) only 50% of the core material could be located within 25,000 feet, and 3) most likely only 5-15% of the core vaporized.

Phoebus Reactors. After the Apollo program's Saturn-booster chemical rocket had developed to an advanced state, it was clear that the nuclear thermal rocket would not be needed for the lunar mission. Advanced interplanetary missions were targeted for use of NTP systems. A project was undertaken to design a nuclear rocket for a manned Mars mission, the Phoebus reactor series. The design requirements were a thrust of 250,000 pounds and an Isp of 840 seconds; this requires a reactor power level of 5000 Mwt.

The Phoebus I series reactor tests were designed to investigate the level of power density achievable. Phoebus-IA (ref. 24) was tested in 1965 to a power level of 1090 Mwt for more than 10 minutes before exhausting the hydrogen supply and damaging the core; the hydrogen supply gauges were affected by the intense radiation environment. Phoebus-IB (ref. 25) was tested in 1967 at 1460 Mwt for the planned 30 minutes. The post-test examination showed excellent core condition and the test overall demonstrated an average power density of 1 Mwt per element. Exhaust gas analysis indicated a release of 1.5% of the core fission product inventory, with 0.5% from fission product bearing uranium fuel and 1.0% from thermally diffused fission products.

The ability to achieve high power density, as shown by the Phoebus I series, proved that the goals of the Phoebus program could be achieved.

The next reactor in the program, Phoebus-2A (ref. 26), was the largest propulsion reactor ever tested. The core measured 52 inches in length and 55 inches in diameter; it contained 4068 fuel elements along with 721 tie-rod support elements. Phoebus-2A was about 2.5 times larger than KIWI-B and incorporated two-pass tie-rods to maintain high exit temperature, and hence Isp, by exhausting the relatively low temperature coolant back into the core inlet plenum instead of the nozzle chamber. This feature should add 25 seconds to the Isp.

During four tests in 1968, Phoebus 2A achieved a maximum power of 4082 Mwt, limited only by inadequate cooling of its aluminum pressure vessel, for 12 minutes. Post-test examination showed that the core was in excellent condition and could have operated at the designed 5000 Mwt. Moreover, examination of the fuel elements showed an average mass loss of 10-13 grams per element. Phoebus 2A resulted in a thrust level of over 200,000 pounds and a power density of under 6 pounds per Mwt.

The Phoebus test series firmly established or demonstrated the following: 1) the basic core and fuel element configuration was very satisfactory for NTP systems, 2) methods were available to control reactor parameters safely over a wide range of operating conditions, 3) NbC-Mo coatings will protect the UC_2-graphite matrix fuel elements from H_2 corrosion, 4) a two-pass regeneratively cooled support element was demonstrated which allows for full core performance, and 5) large rocket nozzles, capable of high heat flux and nuclear heating, were shown feasible.

Peewee-1 Reactor. Peewee-1 was a smaller reactor than those investigated under the KIWI series and was designed to evaluate advanced fuel elements for the Phoebus and NRX reactors (ref.27). To provide sufficient neutron moderation in a 20 inch core diameter, zirconium hydride sleeves were placed around the tie rods to increase neutron moderation & reduce the uranium load. Peewee-1 was tested with NbC and ZrC coated fuel elements in late 1968 for a total of 40 minutes at 514 Mwt. A record exit temperature of 4590 °R was achieved along with a power density of 1.3 Mwt per fuel element (5200 Mwt per cubic meter). Post-test inspection showed a structurally sound core, although core material was ejected and there were numerous areas of damage. Moreover, examination of the fuel elements showed the ZrC coated elements out performed the NbC coated elements in reducing corrosion.

Nuclear Furnace. Much like the Peewee-1 reactor, the Nuclear Furnace (NF-1) was built for testing H_2 cooled fuel elements and other components of high temperature, long life nuclear rocket reactors (ref. 28). The NF-1 was a heterogeneous water-moderated, beryllium-reflected thermal nuclear reactor, designed to operate at 44 Mwt. The advanced hexagonal elements tested were designed for increased corrosion resistance and strength for reaching higher temperatures.

Forty-seven of the forty-nine fuel element cells contained (U,Zr)C-graphite composite fuel elements with a carbide content of either 30% or 35% by volume; these ZrC coated elements had a coefficient of thermal expansion of 3.4 or 3.7 microinch per inch per °R, respectively. Testing of these elements confirmed the belief that minimizing the thermal expansion mismatch between the coating and the fuel matrix reduces coating cracks and carbon mass loss. These elements withstood peak power densities of 4500 to 5000 Mwt/m^3 at fuel element temperatures of 4410 °K without difficulty, except for ZrC coatings susceptibility to radiation damage.

Two of the forty-nine fuel element cells contained (U,Zr)C solid-solution fuel elements with a carbide content of 100%; the elements were impregnated with 0%, 3% or 8% Zr. The primary purpose for testing these elements was to determine their fracture mode at high power densities; a major concern was possible crumbling of the carbide, due to poor thermal stress resistance, that could block the flow passage. Testing of these elements showed many transverse and longitudinal fractures, but no fragmentation into small particles. The 8% Zr elements showed the least amount of fracturing.

The NF-1 assembly was tested in 1972 for a total duration of 108 minutes at a peak exit gas temperature of 4590 °R along with a power density of 4500 Mwt/m^3. The NF-1 operated with a closed-cycle effluent cleanup system for fission fragment scrubbing instead of the traditional open-cycle atmospheric exhaust.

NERVA Program

After the announcement of the Apollo program, a joint NASA/AEC program was initiated to develop a flight rated Nuclear Engine for Rocket Vehicle Application (NERVA). Based on technology developed in the ROVER program, NERVA demonstrated a reusable NTP system with high specific impulse and thrust. The program began in 1960 and was conducted by the joint Space Nuclear Propulsion Office (SNPO). The engine was designed to produce a thrust level of 75 klbf and a thrust specific impulse of 825 seconds, twice the Isp of chemical systems, for a run time of 600 minutes. The Isp of nuclear rockets is limited by the melting temperature and high temperature strength of the fuel, moderator, and core structure. The operation time is limited by structural integrity and by the exhaustion of the critical mass due to U_{235} burnup and carbon corrosion. The success of the NRX reactor and XE engine tests has amply demonstrated that the technology is ready for development of the flight version of the NERVA engine and reactor.

Requirements and Objectives. The design requirements of a flight-rated NERVA rocket engine are summarized by the following.

- Multi-mission capability
- Man-rated (i.e. high reliability)
- Based on full-flow topping cycle
- Minimum chamber temperature and pressure of 4250 °R and 450 psia, respectively
- Minimum 75,000 pounds thrust
- Endurance of 600 minutes with up to 60 cycles
- Capable of 150 °R/s and 50 psia/s transients
- Incorporates adequate shielding for manned operations
- Storable for 5 years on ground, 6 months on pad, and 3 years in space
- Transportable by land, air, or sea

To reach these flight requirements, the following objectives were set forth early in the NERVA program (1964):

- Develop a nuclear reactor capable of operating at full power and temperature for 60 minutes
- Evaluate the performance capabilities and to demonstrate the stable operation of the hot-bleed cycle
- Provide the necessary information for the design of future reactor and engine systems
- Develop a simple, reliable reactor control system
- Experimentally determine the extremes of the steady-state operating map of the test reactors
- Demonstrate the multiple full power restart capability

To satisfy the objectives, NERVA consisted of 2 projects, the Nuclear Reactor eXperiment (NRX) and the eXperimental flight Engine Prototype (XE-Prime); components of both projects were built by the Westinghouse-Aerojet General team. The NRX project was a test of five 1100 Mwt reactors while the XE project tested integrated nuclear and non-nuclear flight components.

NRX Reactors. This reactor series was developed to prove that the KIWI-B4 series reactor structure could be adapted to withstand booster-type vibration and shock environments, and that reactor controls could handle rapid exhaust property variations, such as temperature changes of 100 °R per second. In all, testing time and power levels exceeded NERVA design goals.

The objectives of the NRX-A2 (ref. 29) test were to provide significant information for verifying the steady-state design analysis of power operation and for assessing the suitability of reactor operation at power and temperature levels required within the experimental engine system. The NRX-A2, which closely resembled the KIWI-B4E, was tested in September 1964 and operated for 6 minutes with 40 seconds at 1096 MW; the test duration was limited by the supply of hydrogen available. Post-mortem inspection revealed no broken elements but showed incipient corrosion, especially around the core periphery.

The NRX-A3 (ref. 30) was tested in spring 1965 for a total of 6.7 minutes at 1093 MW. The main objectives of this reactor test were to operate for 15 minutes at full power and to shut and cool down using only hydrogen. Post-test disassembly did not show any damage to the core or corrosion to the periphery; NRX-A3 was the first reactor to use externally coated fuel elements along the periphery.

The NRX-A4 (ref. 31) was the first reactor to be coupled with the major engine components, including the turbopump and a nozzle with an expansion ratio of 10:1, in their functional relationship; the test of this setup, in March 1966, was known as NRX/EST. The goal of the test was to demonstrate the bootstrap start up capability and to evaluate the engine system under transient and steady-state conditions. In all, the engine system was tested under eleven startups, with a total test time of 29 minutes at full-power. The fuel elements showed significant mid-band corrosion and a total of 528 broken elements. The NRX/EST test series was a significant milestone in the development of a nuclear rocket engine. The hot bleed bootstrap principle of nuclear rocket engine operation was demonstrated for the first time, and system stability under a number of control modes and over a wide operating range of pressure and temperature was also proven. Moreover, the multiple restart capability of the engine system was shown, and significant reactor engine operating endurance at rated conditions was demonstrated.

Of prime interest in the NRX-A5 (ref. 32) test series, was the extent of the in-core corrosion and reactivity variation following extended full-power operation. NRX-A5 was tested twice at 1120 Mwt for 29.6 minutes in June 1966. Over the total run duration of 40 minutes, $2.2 of reactivity was lost due to corrosion.

The NRX-A6 was successfully tested in December 1967 at 1125 Mwt for 62 minutes (ref. 33). The primary objective was to accomplish a full power run to a predetermined loss of reactivity or for a time of 60 minutes. Post-test examination revealed severe cracks in the reflector assembly, which was of a new design. This was attributed to a large temperature spike 2 minutes before the end of the test. The NRX-A6 run more than doubled the full power and temperature endurance of previous reactors, with a reduction of 75-80% in the fuel element time rate of corrosion compared to NRX-A4 and NRX-A5. The reduction of fuel element corrosion is attributed to the improved channel coating techniques, dimensional control across element flats, better regard for the coefficients of thermal expansion, and a flattened core power distribution.

XE-PRIME Engine. This engine was the main focus of the NERVA program with a vertical downward firing into a simulated space vacuum of 1.6 psia. The XE engine test (ref. 34) was a full prototype nuclear engine system experiment involving the integration of the reactor (similar to NRX-A6), pressure vessel, nozzle, turbopump assembly and valves in a close-coupled fashion. The XE engine test objectives were to investigate start-up characteristics under different operating modes, determination of engine and component performance parameters, and investigate engine shutdown and pulse-cooling characteristics. This 40,000 pound test engine was designed to produce a thrust of 55,430 pounds at a power level of 1140 Mwt, within its 272" length and 102" diameter. These tests commenced in early 1969 and accomplished full power tests at 1140 Mwt and exit temperatures of 4090 °R. The first run duration was 11 minutes with 3.5 minutes at maximum conditions. Also, bootstrap startups without external power were demonstrated.

NERVA Results. The significant feasibility questions to be answered by the NERVA program, as stated in the objectives, were regarding system structural integrity, restart capability, predictability, controllability, and reliability. It was mainly the NRX reactor tests which answered these questions (ref. 35).

Structural Integrity. In the KIWI tests, severe vibrations induced by a destructive flow pattern were observed. These noted vibrations had to be prevented in the NERVA program, and the reactor integrity must be maintained under the operating temperature and pressure drop conditions. This had to be achieved with acceptably low fuel element weight loss. NRX-A1, NRX-A3 and NRX-A6 reactors demonstrated that the desired endurance capability could be achieve without structural integrity problems, and the latter reactor inparticular showed over the full 60 minute endurance at nominal power.

Restart Capability. The capability to restart the reactor multiple times throughout its design life was a necessity. This capability was proven by all of the NRX tests except NRX-A6. Ten high power start-ups were conducted on NRX/EST alone. Overall 34 restarts were conducted. The XE tests showed multiple restarts and shutdowns; a total of 23 engine starts to power were conducted. A significant result of the NERVA program was the recognition of the difficult start-up and shutdown process. A typical operating map, a chamber pressure versus temperature curve, is shown in Figure 8. The initial start-up bootstrap is complex but the interrelated phenomena involve transfer of heat from the engine to the H_2, the engine flow resistance, and the driving force feeding the engine, such as the tank pressure and the turbopump.

Predictability. To certify a reactor for flight, its performance must be highly predictable within tight constraints. Throughout the NRX tests, predictability has been enhanced by the obtained data. Prior to the NRX-A6 tests, reactor operation predictions were generated and latter compared to the actual operation. The results of the comparison showed excellent agreement when the differences between the planned and actual test profiles are considered. In further support, post examination of the fuel elements revealed their condition to be excellent; the elements could have endured significantly longer operation.

Controllability. Also for flight certification, the system requires close controlling of start-up, steady-state operation, and shutdown (ref. 36). This was well demonstrated by each NRX test. Moreover, NRX-A2, NRX-A3, NRX/EST AND NRX-A5 tests each incorporated advanced control concepts which had been developed during the program.

Reliability. Flight certification places extreme reliability requirements on nuclear thermal systems, especially on reactor components. Particular emphasis is on the reliability of the fuel elements with full endurance and restart capability (ref. 37). The ten successful NRX, KIWI and Phoebus test series have contributed to the demonstration of reactor reliability.

NERVA Flight Engine Development. The actual NERVA flight engine design was based on the success of the NRX reactor and XE engine tests, but incorporated the topping-cycle instead of the hot-bleed-cycle. The flight engine (Figure 9) was designed to use hydrogen at a tank pressure of 30 psia. Dual turbopump assemblies were incorporated, for redundancy, to deliver the hydrogen at 1400 psia to the nozzle and structure as coolant. After cooling the peripheral shield, the warm hydrogen was used to drive the turbine. Turbine exit flow was routed to cool the reactor central shield and the core support plate. Next, the hydrogen entered the reactor core.

The NERVA flight reactor (Figure 10), a complete subassembly, is a hydrogen cooled, graphite-moderated, intermediate neutron energy unit designed to operate at a nominal 1575 Mwt and supply 92 pounds per second of hydrogen propellant to the nozzle entrance at 4250 °R and 450 psia. Reactor components include the following: core, reflector, structural support, shield and reactivity control devices which consist of neutron absorbing vanes assembled in the control drums. The core assembly consists of clustered graphite-uranium fuel elements 54 inches long, including end caps. The reflector assembly consists of a right circular cylinder of beryllium housing eighteen control drums and providing longitudinal cooling holes and lateral support spring pockets. The reactor support structure consists of the core support plate, dome end support cone, nozzle end support ring, and the locating cone. The light weight reactor shield is made of neutron and gamma attenuating material, such as B_4CAl-TiH (BATH). The prime purpose of this shield is to reduce heating of propellant in the flight tank and protect sensitive components.

ANALYZED NTP SYSTEMS

The following two sections present a review of NTP systems which have been designed and analyzed but not tested to the extent of the ROVER and NERVA reactors. The first section covers NTP systems with reactors containing axial flow, prismatic (hexagonal) fuel elements; the latter section covers systems with alternate element forms. Table 3 summarizes pertinent data on selected systems tested under the ROVER and NERVA programs for comparison to the following. Note, the data in parenthesis was inferred from published material.

Prismatic Element NTP Systems

During the ROVER and NERVA programs, the benefits of the hexagonal fuel element form were demonstrated. Using the success of this form, two classes of reactor systems have been further studied, NERVA-derivative reactors and cermet-matrix fast reactors. Table 4 presents the systems based on prismatic elements along with the reference NERVA-1 engine design.

NERVA-Derivative Reactors. Generally, NERVA-derivative reactors are based on the "mixed" core type, with graphite moderator in the carbon-based matrix fuel elements (as in homogeneous cores) and with ZrH moderator sleeves in the support elements (as in heterogeneous cores). The mixed NERVA-derivative core type results in a lower uranium fuel load and lower overall weight.

The Enabler (Figure 11), as conceived by Rocketdyne, is a reactor designed around the graphite/carbide composite fuel elements tested in the Nuclear Furnace. It is of similar scale to the NERVA-1 baseline but was designed to operate at a higher nozzle chamber pressure and temperature with a lower core pressure drop.

The Small Engine (Figure 12), designed by Los Alamos, was scaled to operate at a lower thrust rating for Earth orbit missions. The design chamber pressure was comparable to NERVA-1 however the chamber temperature was higher due to the usage of the composite fuel elements.

The Small Nuclear Rocket Engine (SNRE) is shown in Figure 13 as designed by Aerojet in the mid-1960's. The engine was scaled to operate at a lower thrust rating and the reactor incorporated the graphite matrix fuel elements proven in the NRX reactor tests; a higher chamber temperature is achieved by this design through the usage of the topping-cycle.

Cermet-Matrix Fast Reactors. In the late 1950's, a nuclear rocket engine design effort commenced at General Electric. This effort was focused around a fast neutron fission reactor and was known as the 710 Program. Under this program, engines from 30,000 to 250,000 lbf were designed and analyzed. The fast reactor design incorporated hexagonal fuel elements made of UO_2 dispersed in a refractory-metal matrix (cermet); several elements were manufactured and tested under this program. Excellent thermal and mechanical performance was demonstrated over the thousands of hours of testing. The advantage of cermet elements is positive fuel retention due to metal lined coolant channels.

The 710 engine (Figure 14) in Table 4 is an example of the 710 point design results near a NERVA-1 thrust level. The tungsten cermet fuel elements allow for a high chamber temperature to be achieved. Note, the fuel elements were designed with 91 coolant channels to maximize the surface area per unit volume; however, this resulted in a higher core pressure drop.

The Cermet engine (Figure 15) is a 710 engine derivative developed subsequent to cancellation of the 710 Program's rocket engine design phase in 1963. This design incorporated a larger width fuel element with only 19 coolant channels.

Alternative Element NTP Systems

Although the prismatic element forms dominated testing throughout the 1960's, engines designed around reactors with other element forms deserve considerations. These other element forms include particle (500 μm diameter), bead (1,000 μm diameter), pellet (10,000 μm diameter), and wire (900 μm diameter), and the reactors designed around these are shown in Table 5.

Particle Fuel Elements. Particle fuel elements are designed with annular beds of 500 μm diameter fuel particles contained between two coaxial porous cylinders (frits); essentially, the frits replace the matrix for containment of the fuel. The propellant flows radially through the element from the cool outer to hot inner frit; the propellant exits axially out the inner frit. A particle bed reactor (PBR) core is composed of particle bed elements arranged in a hexagonal pattern, surrounded by moderating material; the resulting core is encased in a reflector. The advantage of this design is a high surface area per unit volume. A potential problem could be clogging of the frits by the particles; experiments have been conducted at Brookhaven National Laboratories (BNL) to investigate this area.

Table 5 presents data for a 75,000 lbf and a 7,400 lbf thrust particle bed reactor system designed by BNL/Babcock-Wilcox/Grumman (Figure 16). Note, the design chamber temperature in both designs is significantly higher than in the prismatic element reactors.

Bead Fuel Elements. Bead fuel elements are designed with beds of 1000 μm diameter fuel beads contained between two porous frits. Table 5 presents design date for two thrust levels of the Low Pressure Nuclear Thermal Rocket (LPNTR) as conceived by Idaho National Engineering Laboratories (INEL). The LPNTR system (Figure 17) is designed incorporate bead (or wafer) elements and to operate at extremely low chamber pressures, comparatively. By operating at low pressure, the system takes advantage of the increase in the specific heat of hydrogen with decreasing pressure; this results in a significantly higher specific impulse at a given chamber temperature. Moreover, tank pressure alone is sufficient to achieve the design chamber pressure; therefore, no engine turbopump assembly is required.

Pellet Fuel Elements. Pellet fuel elements are designed with 0.394" diameter fuel pellets contained between two porous frits. The Pellet Bed Reactor (PeBR) concept, shown in Figure 18, is a fast neutron reactor. The pellets consist of a UC-TaC core with layers of PyC/TaC and ZrC.

Wire Fuel Elements. A study was conducted in the mid-1960's by General Atomics of a compact, high performance nuclear rocket engine which employs tungsten wire fuel elements (Figure 19). The fueled wire was formed by filling a braided tungsten wire tube with 100 μm UN particles, vapor-depositing tungsten on the tube, and then swaging the filled tube to 900 μm. The core is constructed of layers of wire wound over alternate layers of spacer wires, which form a rugged annular lattice. The wire core fast reactor is compact in size due to its high surface area per unit volume.

NUCLEAR THERMAL PROPULSION SAFETY

The usage, or mere mention, of nuclear power tends to cause great public concern, especially when connected to space flight. Prior to public acceptance of nuclear rockets, a comprehensive and publicly credible safety plan must be established (ref.38). In anticipation of flight, extensive safety plans were developed during the ROVER and NERVA programs (ref. 39, 40). Generally, the objectives of nuclear safety are as follows: 1) to protect workers and the public against "unreasonable" exposure to radiation and toxic materials, 2) to protect the Earth and local space environment against risk of "significant" alteration, and 3) to protect the mission against nuclear system failure (ref. 41). Specifically, the hazards are inadvertent criticality, toxic material release, failure to "poison" reactor on final shutdown, radiation after shutdown, and diversion of special nuclear materials.

As a guideline, space nuclear reactors and rockets should be able to withstand the following launch hazards: 1) the worst-case pressure gradient associated with the most credible scenario for detonation of the liquid and/or solid rocket propellant, 2) the worst-case temperature due to flame from the detonation, 3)the LEO reentry and earth impact in sea or on land, 4) the worst-case credible combination of pressure gradients, temperature, and vibration due to range safety destruct of launch vehicle during ascent. Also, the reactor must have a positive and permanent shutdown system, along with a redundant, automatic shutdown control for all contingencies.

As part of the SNAP-10A space power reactor flight test, a comprehensive nuclear safety program was conducted. This safety program presents a good model for future efforts and is well documented in Reference 42. Also, the SP-100 system development effort is integrated around the SP-100 Surety Program, where surety is defined as an integration of safety, safeguards, environmental protection, reliability, and quality assurance (ref. 43). The SP-100 Surety Program has been developed to provide confidence that the surety issues will be adequately integrated in the designs and that flight approval can be obtained while minimizing undue design penalties. The subject of Environmental Impact Statements is well discussed in Reference 44.

CONCLUSIONS

The exploration of space is one of mankind's greatest adventures. Engineers and technicians over the next 20 years will be making a permanent mark on history by extending technology to allow for human contact with our neighboring planets. The exploration of the distant planets with unmanned vehicles is a phenomenal achievement; however, manned exploration will require extraordinary technological advances. The task of resurrecting NTP technology and developing it to a man-rated level is one of those extraordinary advances.

Table 1

Melting Points of Reactor Core Material (Ref. 45)

Type of Material	Material	Temperature (°R)
Fuel	Uranium	2530
Fuel Compound	Uranium Nitride	5690
	Uranium Dioxide	5535
	Uranium Carbide	4810
Refractory Metal	Tungsten	6580
	Rhenium	6200
	Tantalum	5890
	Molybdenum	5170
Refractory Non-Metal	Hafnium Carbide	7490
	Tantalum Carbide	7480
	Carbon (sublimation)	7190
	Niobium Carbide	6790
	Zirconium Carbide	6210

Table 2

Chronology of ROVER and NERVA Tests (ref. 6)

Project	Date	Max.Power (Mwt)	Time (@ max.)
KIWI-A	1 July 1959	70	300 sec
KIWI-A'	8 July 1960	88	307 sec
KIWI-A3	19 Oct. 1960	112.5	259 sec
KIWI-B1A	7 Dec. 1961	225	36 sec
KIWI-B1B	1 Sept. 1962	880	several sec
KIWI-B4A	30 Nov. 1962	450	several sec
KIWI-B4D	13 May 1964	990	40 sec
KIWI-B4E	28 Aug. 1964	937	480 sec
KIWI-B4E	10 Sept. 1964	882	150 sec
KIWI-TNT	12 Jan. 1965	---	---
NRX-A2	24 Sept. 1964	1096	40 sec
NRX-A3	23 April 1965	1093	210 sec
NRX-A3	20 May 1965	1072	792 sec
PHOEBUS-1A	25 June 1965	1090	630 sec
NRX-A4/EST	March 1966	1055	1740 sec
NRX-A5	June 1966	1120	1776 sec
PHOEBUS-1B	23 Feb 1967	1450	1800 sec
NRX-A6	15 Dec. 1967	1125	3720 sec
PHOEBUS-2A	26 June 1968	4082	750 sec
PEEWEE-1	Fall 1968	514	2400 sec
XE-PRIME	11 June 1969	1140	210 sec
NF-1	July 1972	44	6528 sec

Table 3 - Comparison of Tested NTP Systems.

	PHOEBUS-1B	PHOEBUS-2A	PEWEE-1	NF-1	NRX/EST	NRX-A6	XE-PRIME
Net I_{sp} (s)	835	805	845^{v}	830^{v}	820	847	710
Thrust (klbf)	68	209	N/A	N/A	55	55	55.4
Nozzle Area Ratio	12		N/A	N/A	10	10	10
Endurance (h)	.5	.21	.72	1.82	.48	1.05	0.06
TPA Cycle	COLD-BLEED	HOT-BLEED	N/A	N/A	HOT-BLEED	N/A	HOT-BLEED
Propellant	H_2	H_2	H_2	H_2	H_2	H_2	H_2
Reactor Type	THERMAL	THERMAL	THERMAL	THERMAL	THERMAL	THERMAL	THERMAL
Max. Core Power	1455	4082	514	44	1170	1125	1140
Core Flow	AXIAL	AXIAL	AXIAL	AXIAL	AXIAL	AXIAL	AXIAL
Core Type	HOMO.	HOMO.	MIXED	MIXED	HOMO.	HOMO.	HOMO.
Fuel Type	U^{235}	U^{235}	U^{235}	U^{235}	U^{235}	U^{235}	U^{235}
Compound	UC_2	UC_2	UC_2	UC_2	UC_2	UC_2	UC_2
Particle Coating	PyC	PyC	PyC	PyC	PyC	PyC	PyC
$D_{PARTICLE}$ (μm)	50-150	50-150	50-150	50-150	50-150	50-150	50-150
Matrix Mat'l	GRAPHITE	GRAPHITE	GRAPHITE	COMP/CARB	GRAPHITE	GRAPHITE	GRAPHITE
Element Form	HEXAGONAL	HEXAGONAL	HEXAGONAL	HEXAGONAL	HEXAGONAL	HEXAGONAL	HEXAGONAL
$D_{ELEMENT}$ (in)	.752	.753	.752	.752	.754	.753	.752
$N_{COOLANT\ CHANNELS}$	19	19	19	19	19	19	19
$D_{COOLANT\ CHANNELS}$	.10	.11	.11	.11	.009	.10	.10
Channel Liner Mat'l	NbC/Mo	NbC/Mo	NbC	NbC/ZrC	NbC	NbC	NbC
Liner Thickness (μm)					35-70		
$N_{FUEL\ ELEMENTS}$	1498	4068	402	49	1584	1584	1584
$N_{SUPPORT\ ELEMENTS}$		721	132	N/A	(289)		
Support Element Type	1-PASS	2-PASS	1-PASS	1-PASS	1-PASS	1-PASS	1-PASS
Moderator Mat'l	GRAPHITE	GRAPHITE	GRAPH/ZrH_x	GRAPH/H_2O	GRAPHITE	GRAPHITE	GRAPHITE
D_{CORE} (in)		54.7	21	13.4	35	35	35
L_{CORE} (in)	52	52	52	57.5	52	52	52
Reflector Mat'l	Be	Be	Be	BeO	Be	Be	Be
Reflt. Thickness (in)				10.75			
$T_{FUEL,MELT}$ (°R)							
$T_{FUEL,PEAK}$ (°R)			4950	4950	(5048)		
$T_{FUEL,EXIT}$ (°R)	4401	4158	4570	4590	>4320	4600	>4320
T_c (°R)	4125	4068	3303	3263	4125	4330	4105
P_c (psia)	735	624	620	464	587	593	565.8
ΔP_{CORE} (psia)	203		175	217.5	(124)		146.7
w_{CORE} (lb/s)	94.4	262	41.5	2.4	86.6	72	70.2
$T_{TURB,INLET}$ (°R)	600		N/A	N/A	(1140)		
$N_{TURBINES}$	2	2	N/A	N/A	1	N/A	1
Turbine Type	Mk25	Mk25	N/A	N/A	MkIII-4	N/A	Mk25
TPR	10.4						
N_T (RPM)					20,343		
η_T (%)					58.4		
η_P (%)					71.2		
NPSP (psi)					30.23		
Nozzle Mat'l			N/A	N/A			
$W_{REACTOR}$ (lb)		20419	5665				6964
W_{FUEL} (lb)		661.4	79.4	<11	401.2	401.2	
W_{ENGINE} (lb)			N/A	N/A			6613
W_{TOTAL} (lb)			5665				13578
$W_{INT.SHIELD}$ (lb)			N/A	N/A			2901
$W_{EXT.SHIELD}$ (lb)			N/A	N/A			
REFERENCE(S)	46	47,14	47,14,27	48,28	14	47,14	14,49

Table 4 - Comparison of Prismatic Element Based NTP Systems.

	NERVA-1	ENABLER	SMALL ENGINE	SNRE	710	CERMET
Net I_{SP} (s)	825	916	875	833	873	930
Thrust (klbf)	75.1	75[100]	16.4	10	100	100.1
Nozzle Area Ratio	100	500	100	100	100	120
Endurance (h)	1 - 10	2	1	1	10	<10
TPA Cycle	TOPPING	TOPPING	TOPPING	TOPPING	TOPPING	TOPPING
Propellant	H_2	H_2	SH_2	H_2	H_2	H_2
Reactor Type	THERMAL	THERMAL	THERMAL	THERMAL	FAST	FAST
Max. Core Power	1570	1586[1856]	367	210	2010	2000
Core Flow	AXIAL	AXIAL	AXIAL	AXIAL	AXIAL	AXIAL
Core Type	MIXED	MIXED	MIXED	MIXED	N/A	N/A
Fuel Type	U^{235}	U^{235}	U^{235}	U^{235}	U^{235}	U^{235}
Compound	UC_2	UC-ZrC	UC-ZrC	UC_2	UO_2	60V/°UO_2
Particle Coating	PyC	UNCOATED	UNCOATED	PyC	TUNGSTEN	TUNGSTEN
$D_{PARTICLE}$ (μm)	50-150	(~3.4)	(~3.4)	50-150		
Matrix Mat'l	GRAPHITE	COMPOSITE	COMPOSITE	GRAPHITE	W-45UO_2	40V/°W
Element Form	HEXAGONAL	HEXAGONAL	HEXAGONAL	HEXAGONAL	HEXAGONAL	HEXAGONAL
$D_{ELEMENT}$ (in)	.7519	.75	.75	.753	.996	1.87
$N_{COOLANT\ CHANNELS}$	19	19	19	12	91	19
$D_{COOLANT\ CHANNELS}$	.11	(.11)	0.09		0.0584	
Channel Liner Mat'l	ZrC	ZrC	ZrC	NbC		
Liner Thickness (μm)			50-150		(0.008)	
$N_{FUEL\ ELEMENTS}$	(1878)	1277[1870]	564	(~280)	661	163
$N_{SUPPORT\ ELEMENTS}$		249[325]	241	55		
Support Element Type	2-PASS	2-PASS	2-PASS	2-PASS		
Moderator Mat'l	(ZrH/Graph)	ZrH/Graph	ZrH_2/Graph	(ZrH/Graph)	N/A	N/A
D_{CORE} (in)	54.7	32.56[35]	25.4	17	27.9	
L_{CORE} (in)	52	52	35	36	25.9	34.25
Reflector Mat'l	Be	Be	Be	Be	Be	(Be)
Reflt. Thickness (in)		(9.6)[7]	5.6	(8.0)		
$T_{FUEL,MELT}$ (°R)		(~5200)	(5547)		5559	(5200)
$T_{FUEL,PEAK}$ (°R)	(4860)		5187		5352	4912
$T_{FUEL,EXIT}$ (°R)			4802			
T_c (°R)	4250	4860	4741.2	4500	4759	4512
P_c (psia)	450	1000	450	400	484	600
ΔP_{CORE} (psia)	171	106.2	124.7	125	416	322
W_{CORE} (lb/s)	92.37	81.86[109]	18.74	12	112.3	120
$T_{TURB,INLET}$ (°R)	277.2	(973)	772.2	500		
$N_{TURBINES}$	2	2	1	1	1	1
Turbine Type	Mk25	[Mk25]	Mk25	(see ref.)	(see ref.)	
TPR		[1.25]	1.17			(1.42)
N_T (RPM)		[37,500]	46,951			
η_T (%)			80			75.4
η_P (%)			65			77.
NPSP (psi)		(0)				
Nozzle Mat'l			Inc'l 625		Incnl.X/TZM	
$W_{REACTOR}$ (lb)	12985	13008.7	3499	1450	8875	
W_{FUEL} (lb)	(606.3)		115.5			
W_{ENGINE} (lb)	9319	5687.8	1486	1500	2097	
W_{TOTAL} (lb)	22304	18696.6	4985	2950	10972	(20000)
$W_{INT.SHIELD}$ (lb)	3499	3344	527	550		
$W_{EXT.SHIELD}$ (lb)		10304				
REFERENCE(S)	27	50,51	52	53	54,55	50

Table 5 - Comparison of Alternative Element Based NTP Systems.

	PBR #1	PBR #2	PeBR	LPNTR #1	LPNTR #2	MARRS WIRE CORE
Net I_{SP} (s)	971	780	1000	1075	1050	930
Thrust (klbf)	75	7.4	70.8	25	10.7	205.5
Nozzle Area Ratio		125		60	40	92.5
Endurance (h)		.55		.55		5-10
TPA Cycle	HOT-BLEED	HOT-BLEED	TOPPING	N/A	N/A	HOT-BLEED
Propellant	H_2	$.4H_2/.6D$	H_2	H_2	H_2	H_2
Reactor Type	THERMAL	THERMAL	FAST	THERMAL	THERMAL	FAST
Max. Core Power	(~1945)	150	1500	525	260	4400
Core Flow	RADIAL	RADIAL	RADIAL-IN	RADIAL-OUT	RADIAL-OUT	RADIAL-OUT
Core Type	MIXED	MIXED	N/A	HOMO	HOMO	N/A
Fuel Type	U^{235}	U^{235}	U^{235}	U^{235}	U^{235}	U^{235}
Compound	UC_2	UC-ZrC	UC-TaC	UC-ZrC	UC-ZrC	UN
Particle Coating	PyC/ZrC	PyC/ZrC	PyC/TaC			TUNGSTEN
$D_{PARTICLE}$ (μm)	500	500	500	1000	1000	100
Matrix Mat'l	NONE	NONE	NONE	NONE	NONE	TUNGSTEN
Element Form	PARTICLE	PARTICLE	PELLET	BEADS	BEADS	WIRE
$D_{ELEMENT}$ (in)			0.394			0.035
$N_{COOLANT\ CHANNELS}$	N/A	N/A	N/A	N/A	N/A	N/A
$D_{COOLANT\ CHANNELS}$	N/A	N/A	N/A	N/A	N/A	N/A
Channel Liner Mat'l	N/A	N/A	ZrC	N/A	N/A	N/A
Liner Thickness (μm)	N/A	N/A	0.5	N/A	N/A	N/A
$N_{FUEL\ ELEMENTS}$						N/A
$N_{SUPPORT\ ELEMENTS}$						N/A
Support Element Type			N/A	N/A	N/A	N/A
Moderator Mat'l		Be		Be+ZrH	Be+ZrH	N/A
D_{CORE} (in)			31.5	19.68	27.6	24
L_{CORE} (in)			51.28	N/A		32
Reflector Mat'l		Be	BeO	Be	Be	Be
Reflt. Thickness (in)		10.2			3.94	4.0
$T_{FUEL,MELT}$ (°R)			<6606	6660	6660	6606
$T_{FUEL,PEAK}$ (°R)			5580		6545	5580
$T_{FUEL,EXIT}$ (°R)			5400	5760	5760	
T_C (°R)	5760	4950	5400	5760	5760	5400
p_C (psia)	1000	896		10	15	
Δp_{CORE} (psia)		(45)		20	20	
w_{CORE} (lb/s)	77.24	8.4	70.54	23.2	10.5	70.54
$T_{TURB,INLET}$ (°R)	2800	2500		N/A	N/A	
$N_{TURBINES}$	1	1		N/A	N/A	
Turbine Type				N/A	N/A	
TPR						
N_T (RPM)						
η_T (%)		80.				
η_P (%)		70.				
NPSP (psi)						
Nozzle Mat'l	C/C					
$W_{REACTOR}$ (lb)	2250	1323	(2204)		1527	(2204)
W_{FUEL} (lb)				154.3	(88.2)	
W_{ENGINE} (lb)	1500		(1929)		2528	
W_{TOTAL} (lb)	3750		4133	4360	4056	
$W_{INT.SHIELD}$ (lb)			8818			
$W_{EXT.SHIELD}$ (lb)			N/A			
REFERENCE(S)	50	56	57,50	58	50	59

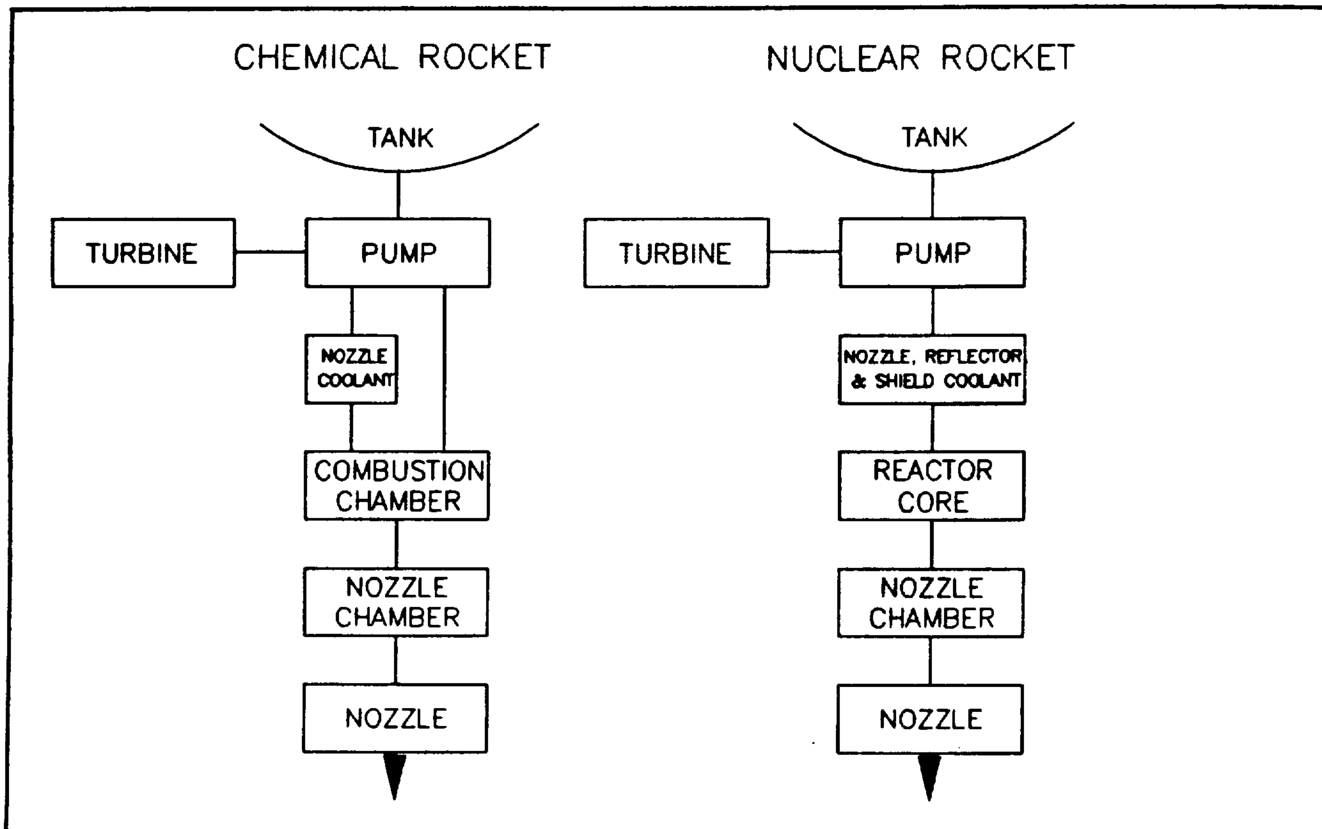

Figure 1 - Chemical & Nuclear Rocket Engine Schematics.

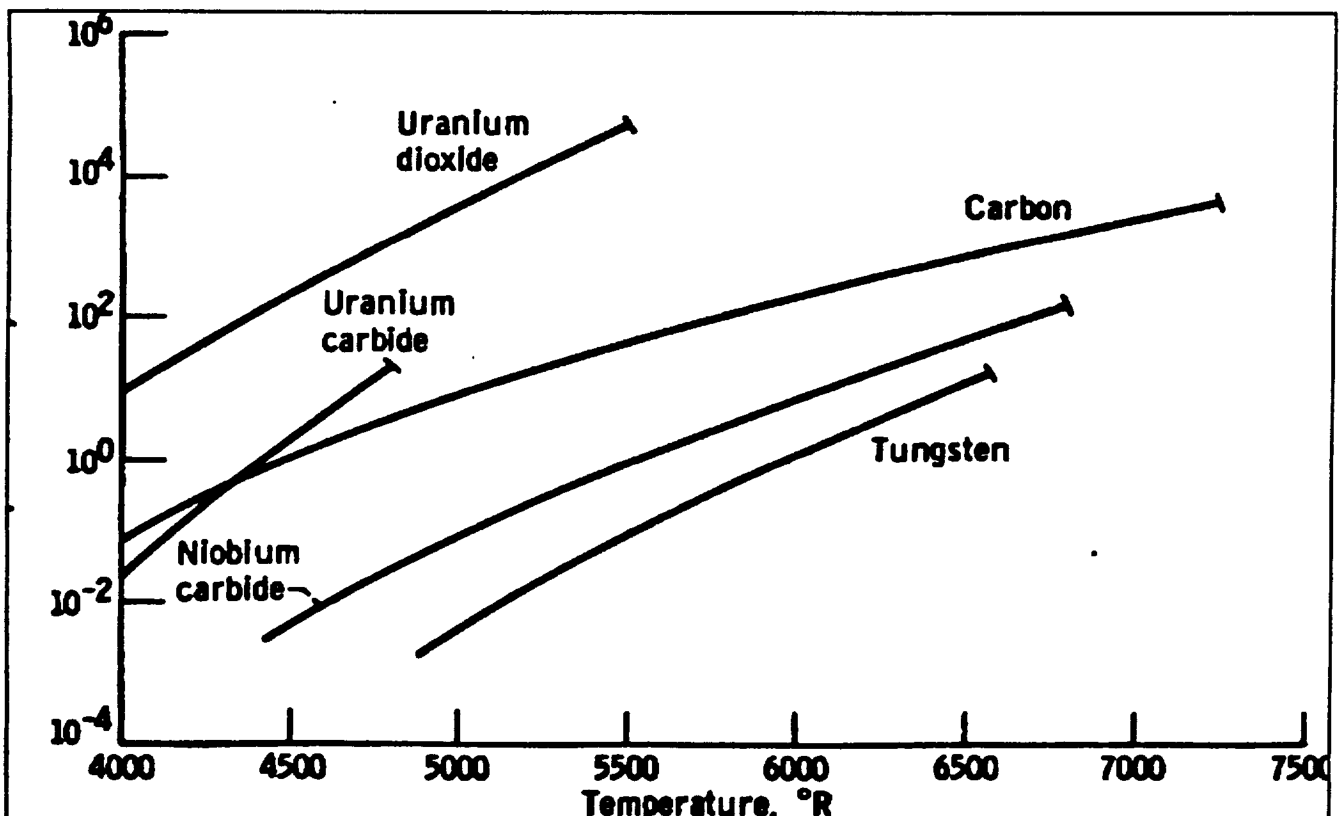

Figure 2 - Vaporization Rate of Several Reactor Core Materials.

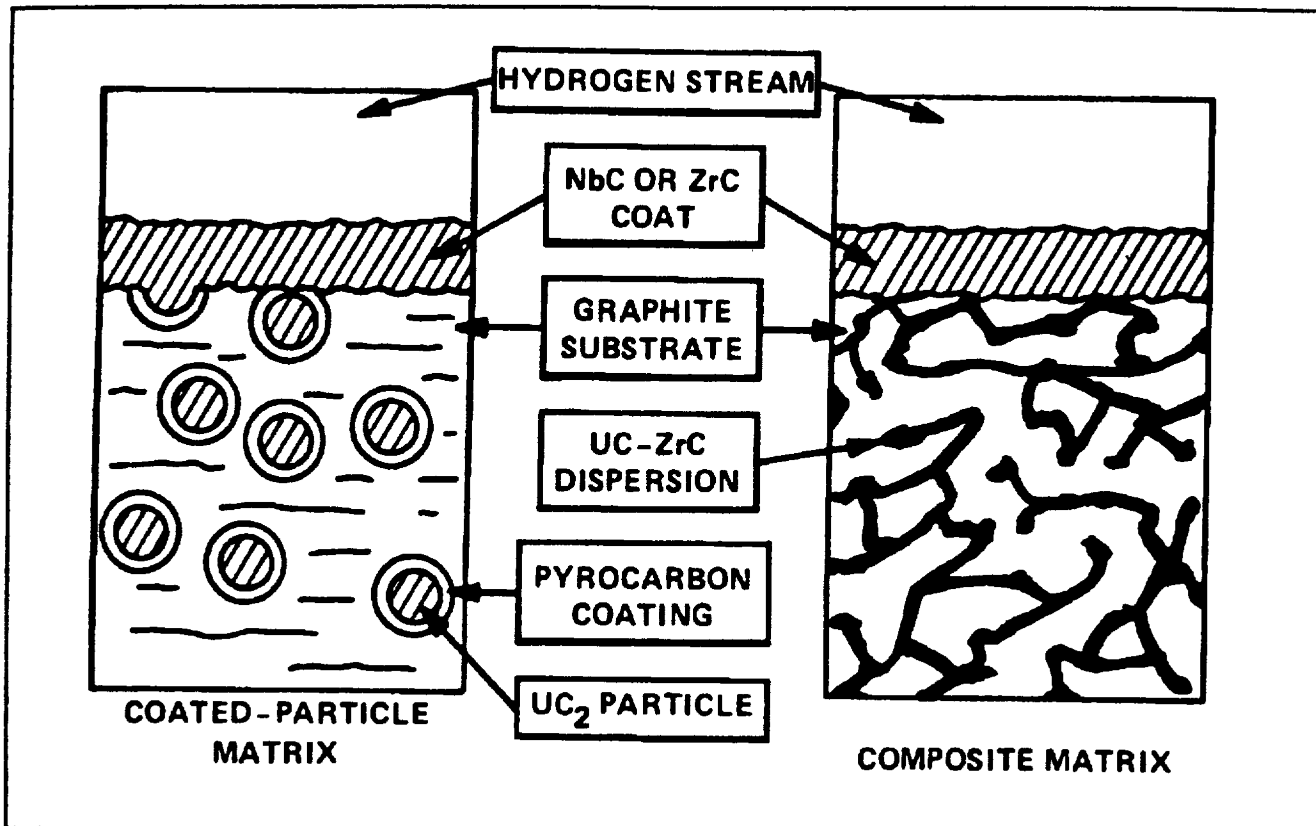

Figure 3 - Comparison of Graphite-Coated Particle Matrix Structure to Graphite-Carbide Composite.

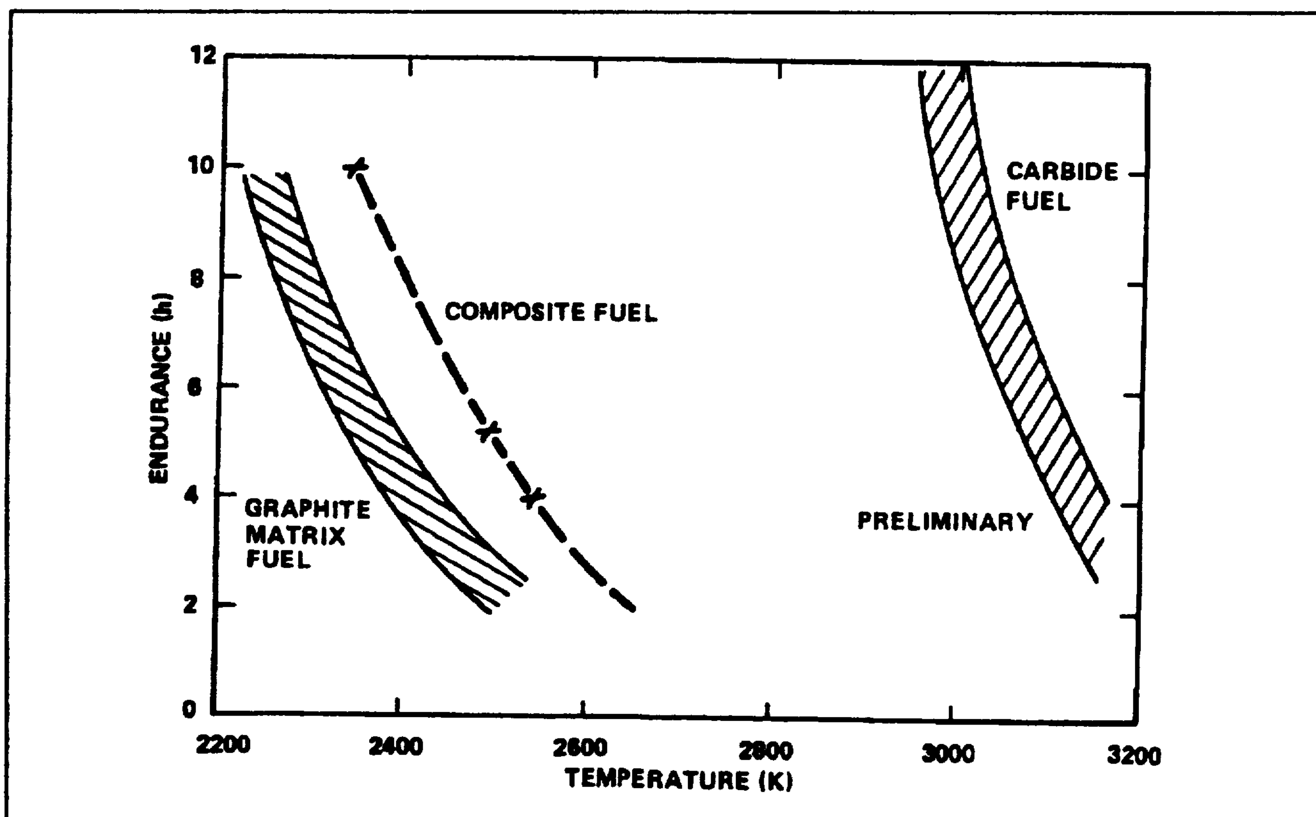

Figure 4 - Comparison of Projected Endurance Level.

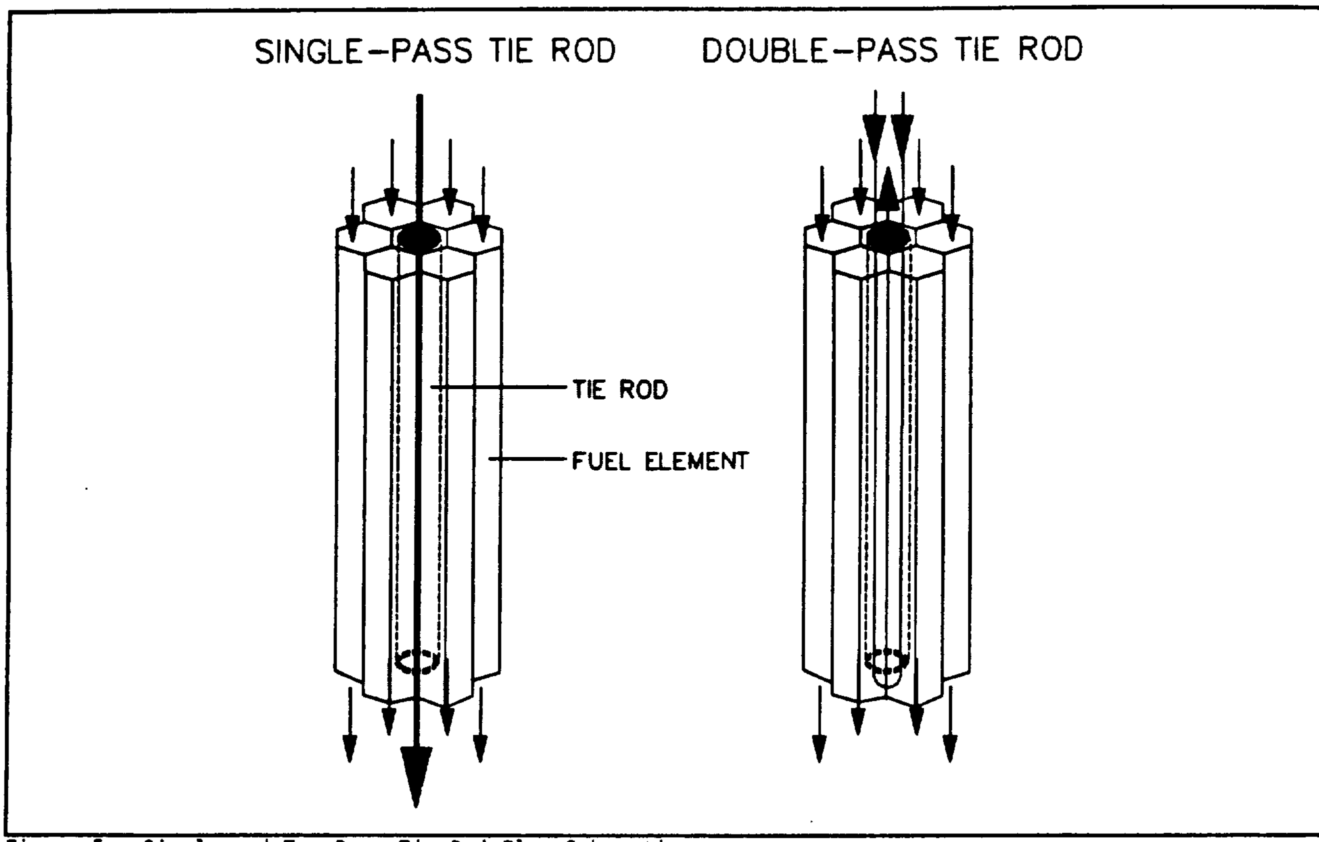

Figure 5 - Single and Two Pass Tie-Rod Flow Schematics.

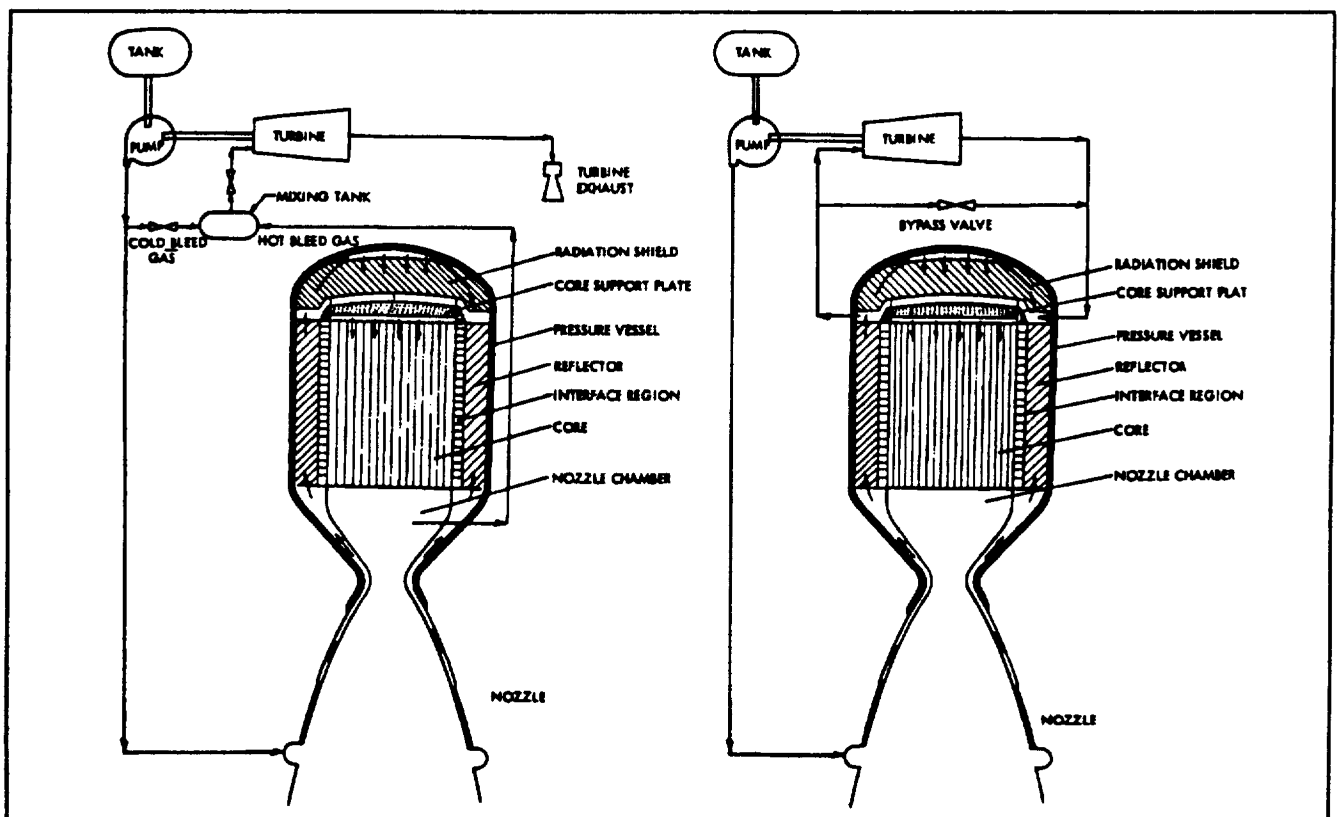

Figure 6 - Hot-Bleed (Left) & Topping (Right) Cycle.

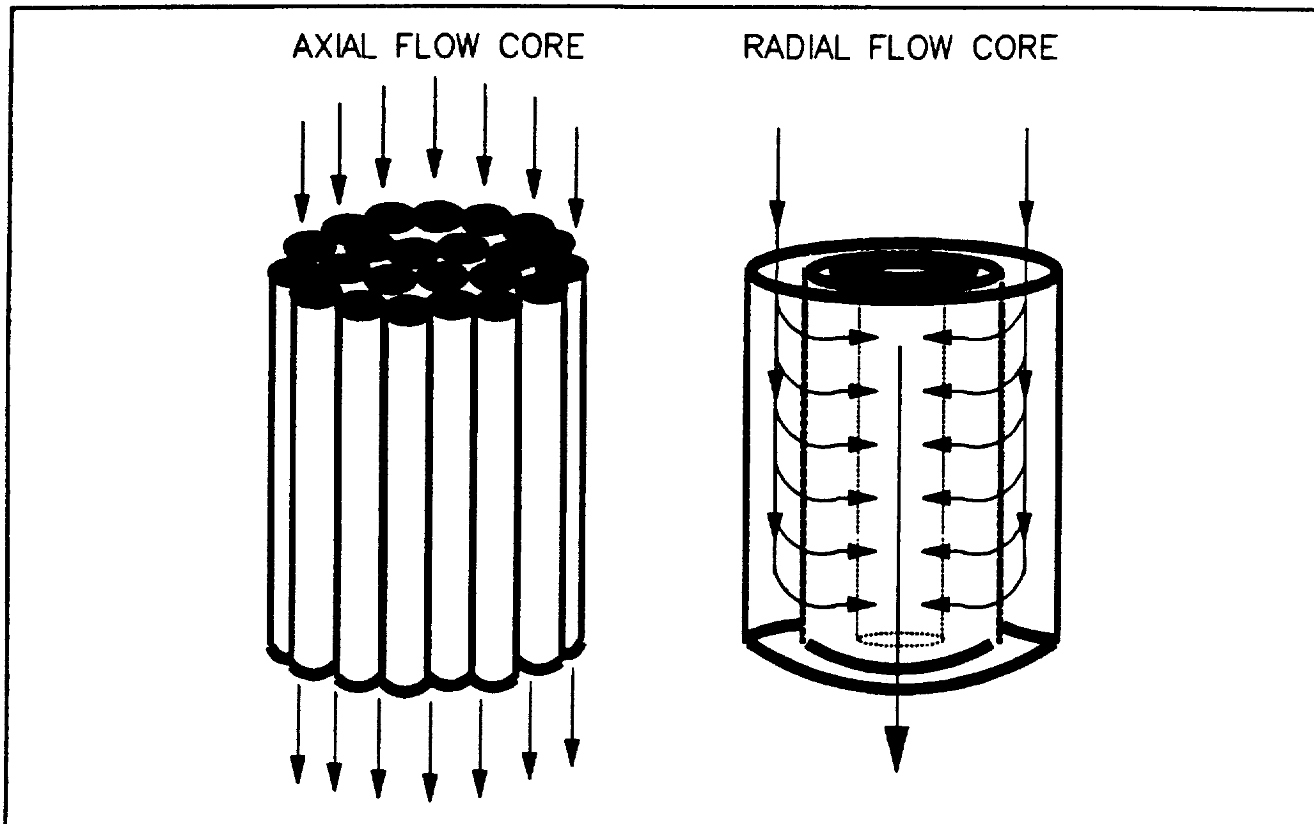

Figure 7 - Axial & Radial Core Flow Schematics.

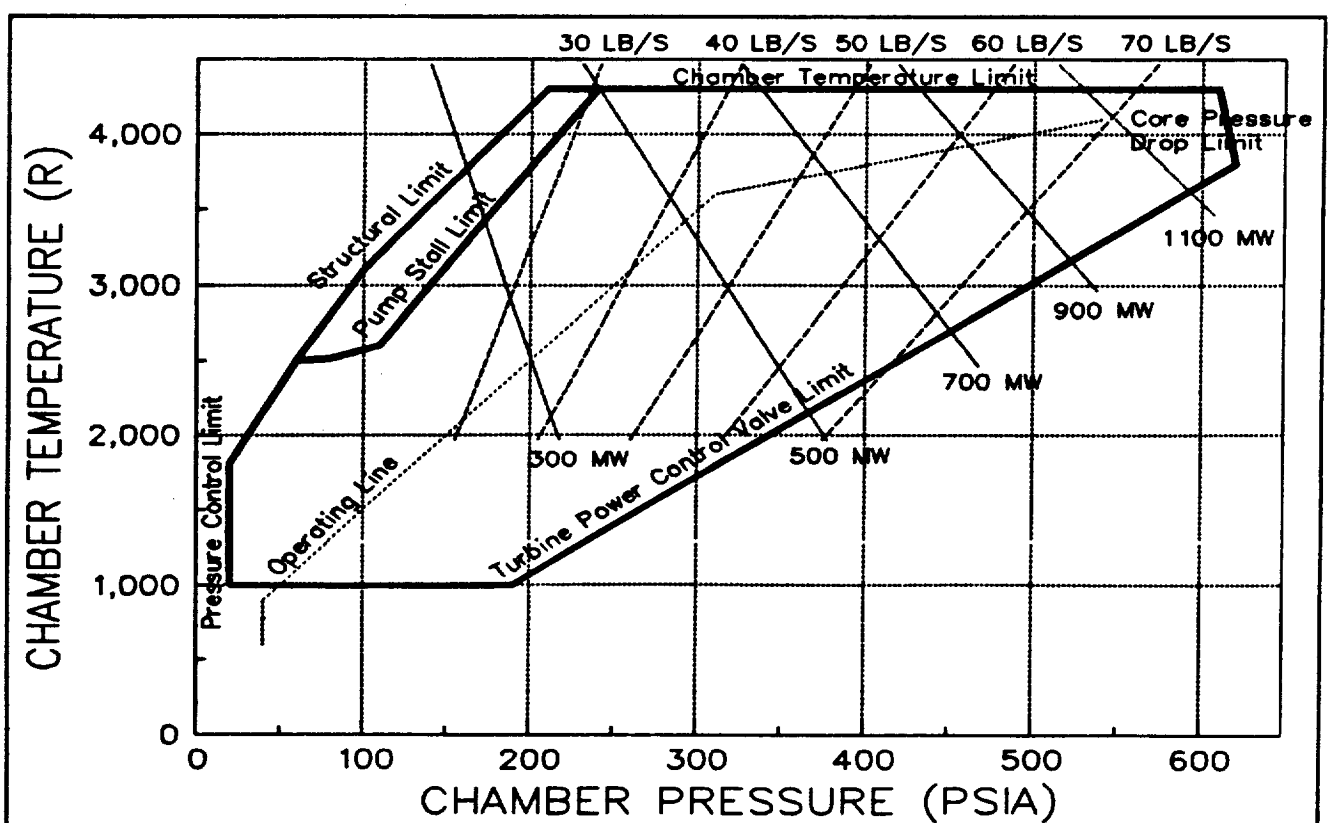

Figure 8 - XE-PRIME Operating Map.

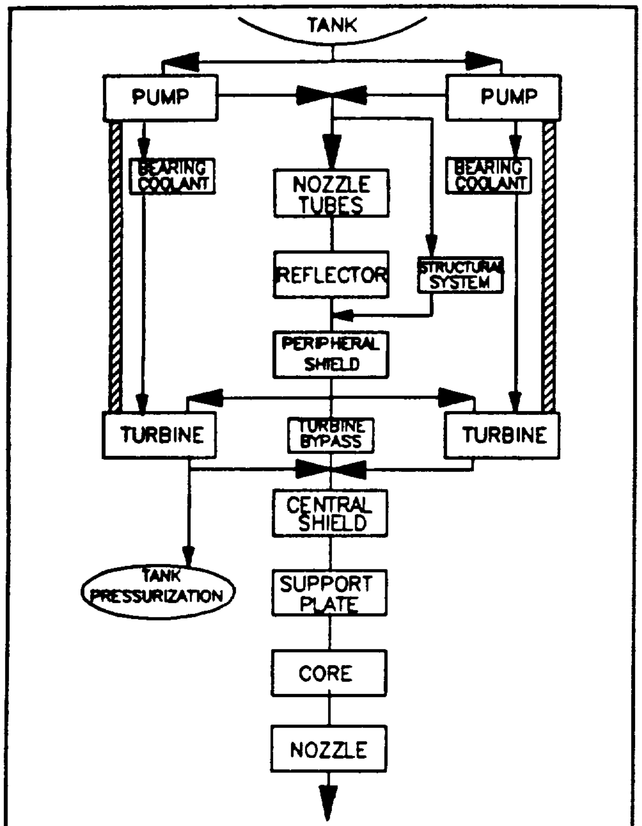

Figure 9 - NERVA Flight Engine Block Diagram.

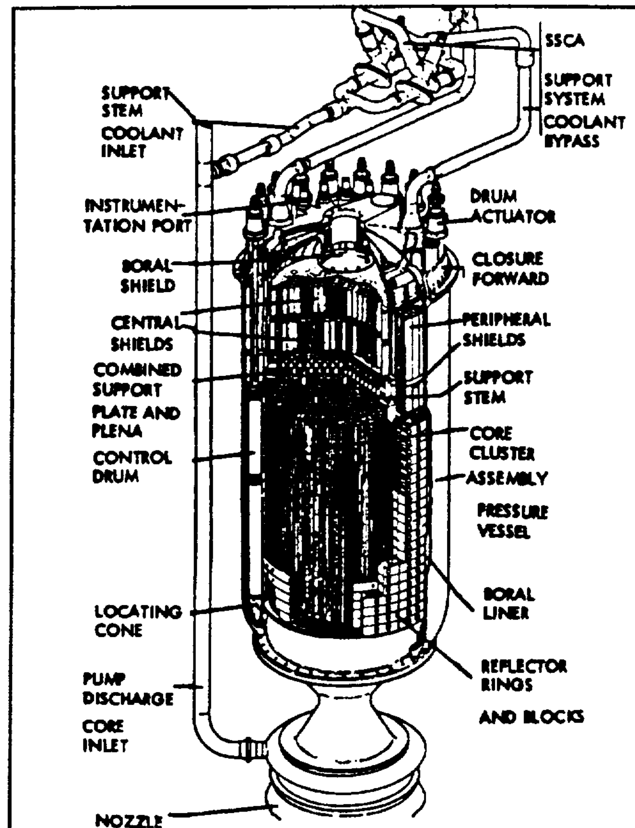

Figure 10 - NERVA Flight Engine Nuclear Subsystem

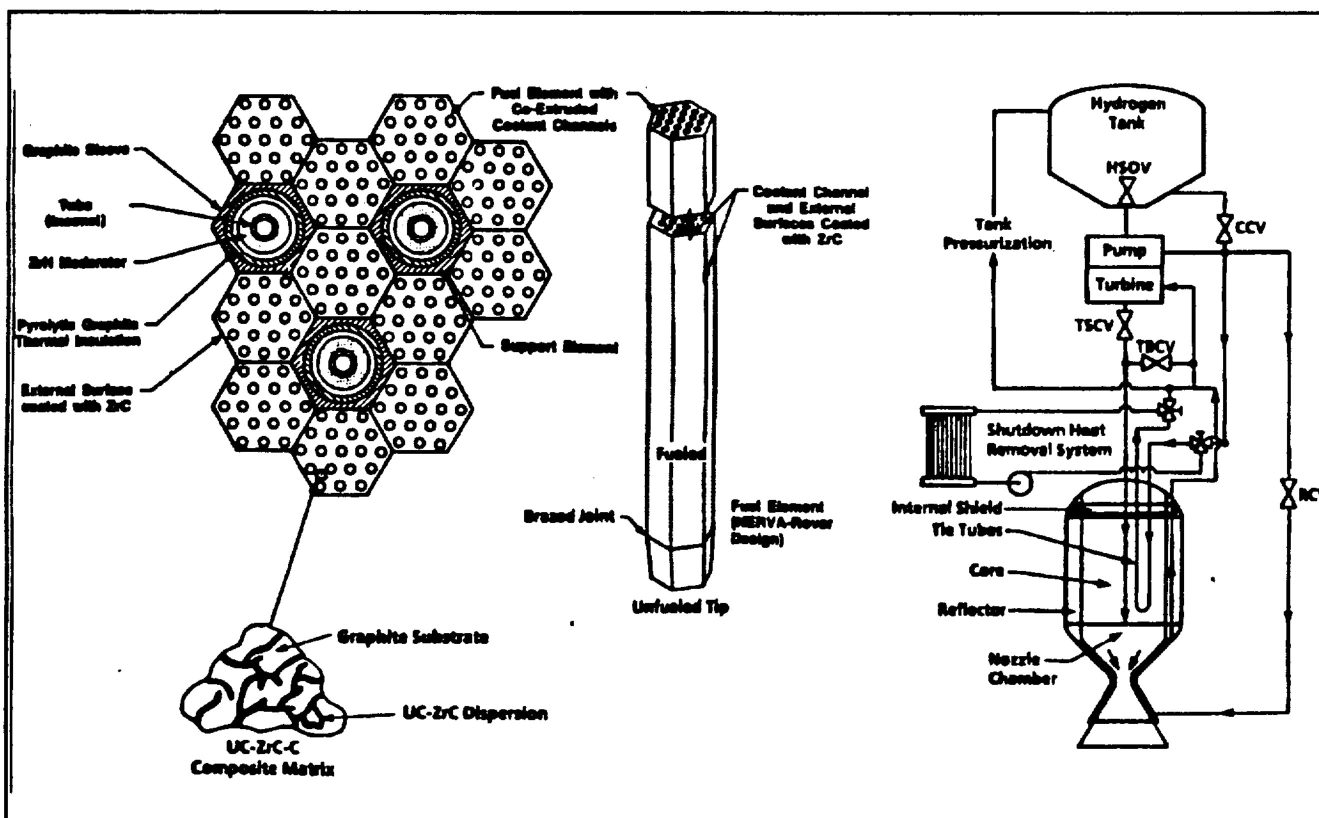

Figure 11 - Schematic of Enabler Engine and Fuel Structure.

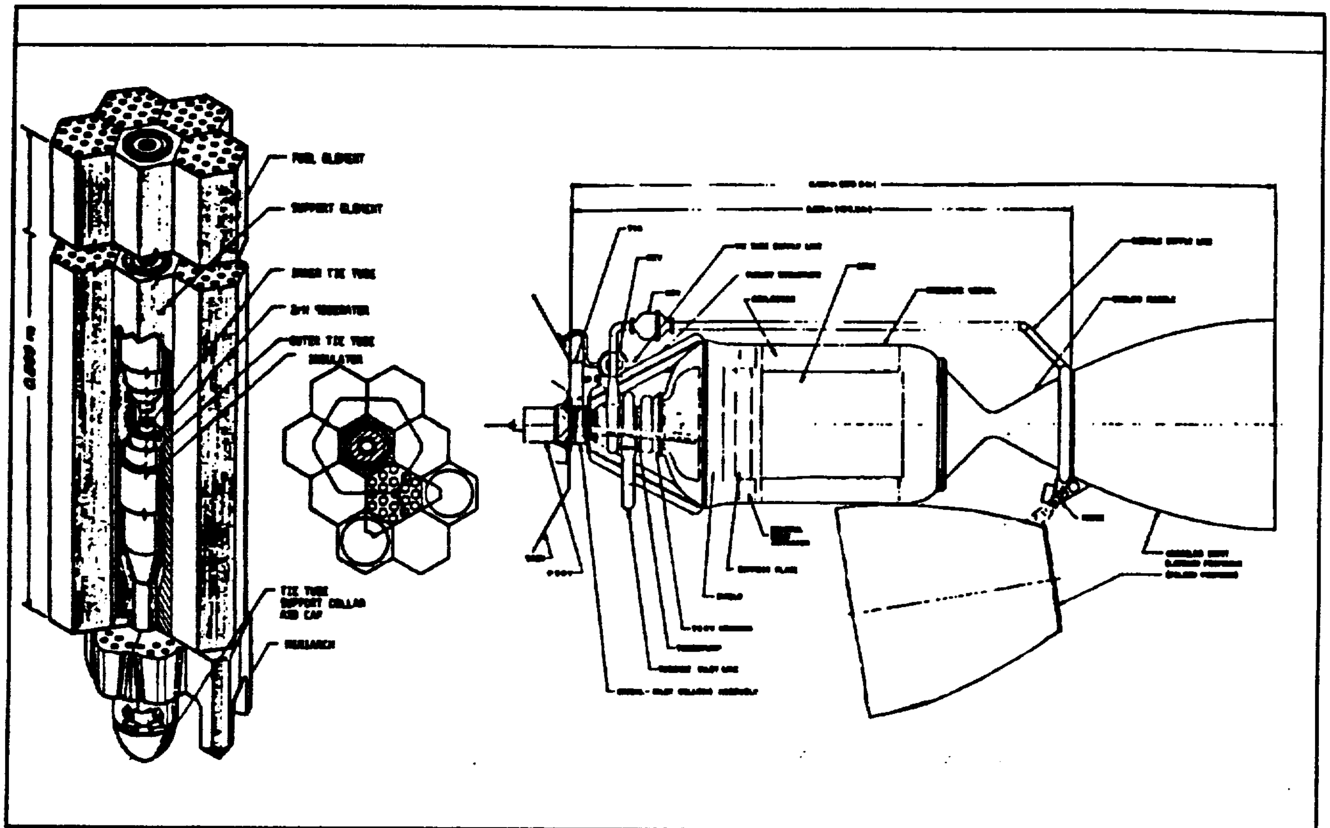

Figure 12 - Schmatic of Small Engine and Fuel Element Sturcture.

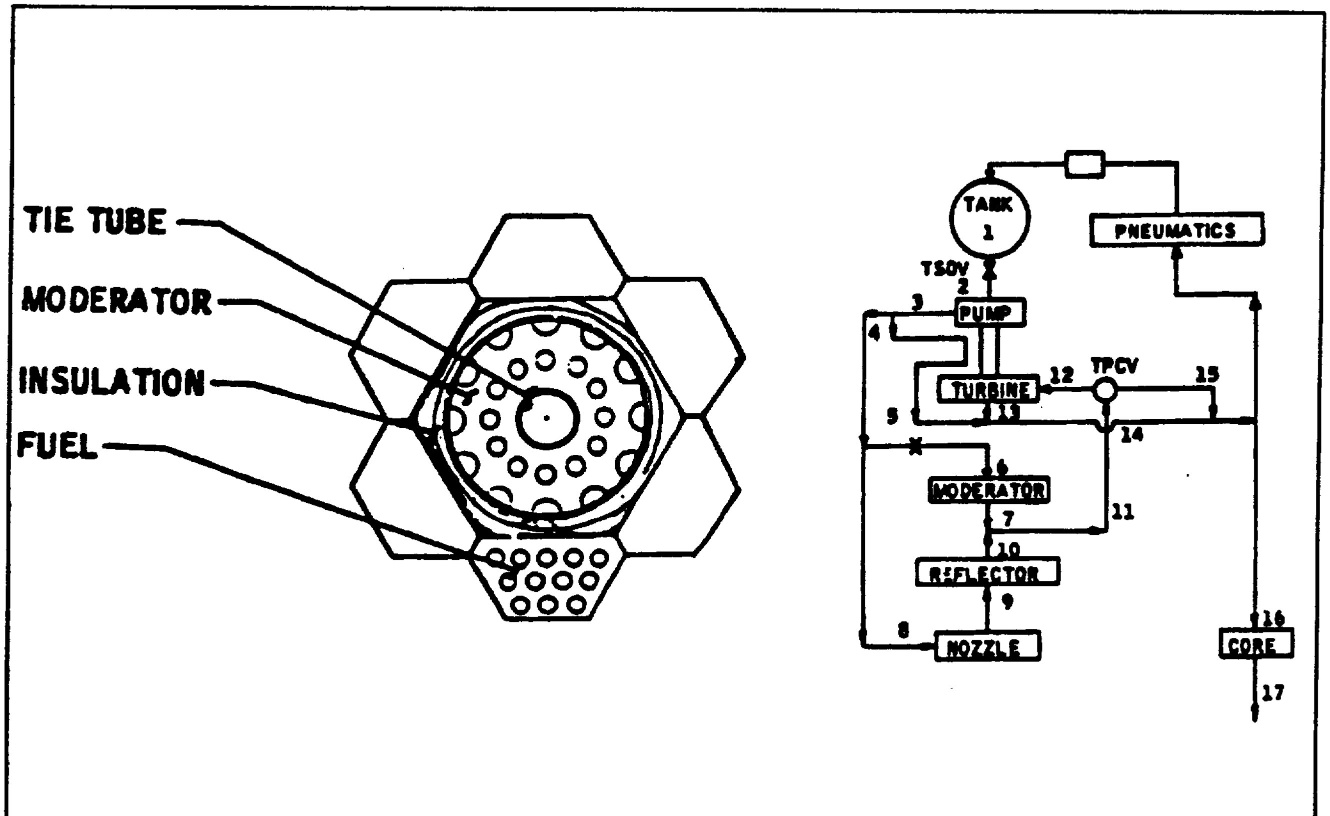

Figure 13 - Schematic of SNRE and Fuel Cross-Section.

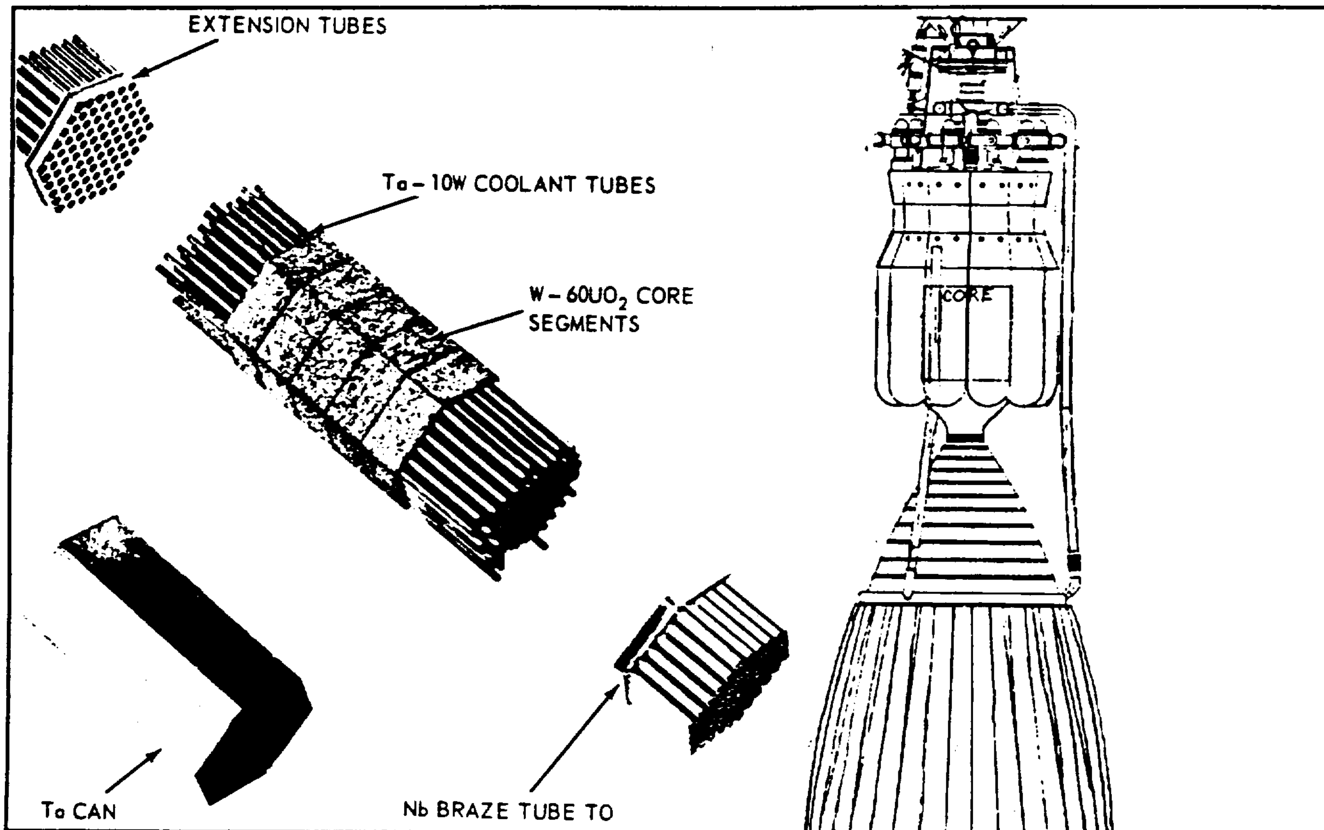

Figure 14 - Schmatic of 710 Engine and Fuel Element Structure.

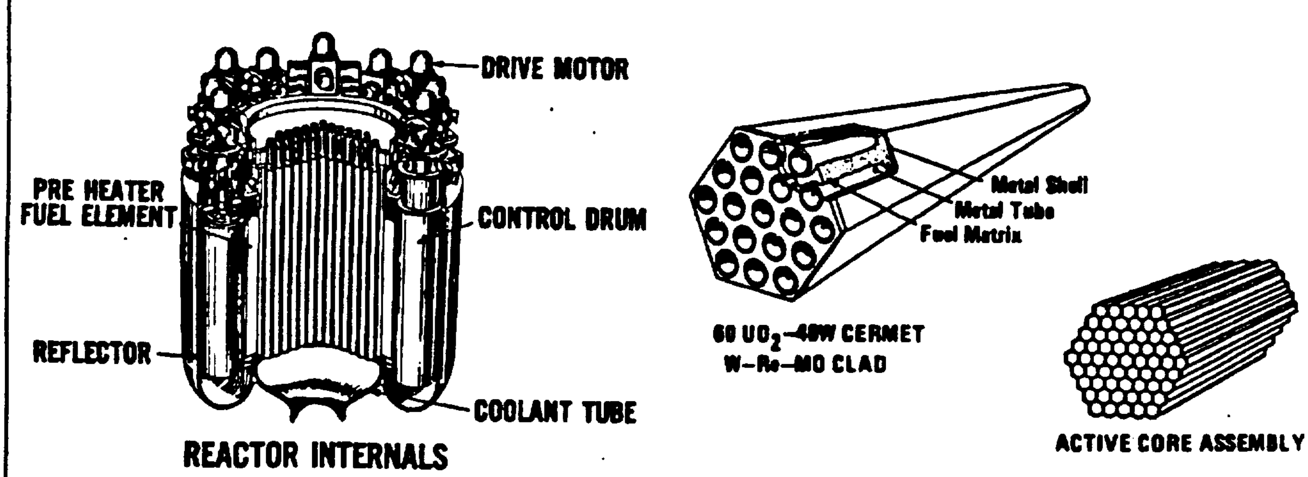

Figure 15 - Schmatic of Cermet Engine and Fuel Element.

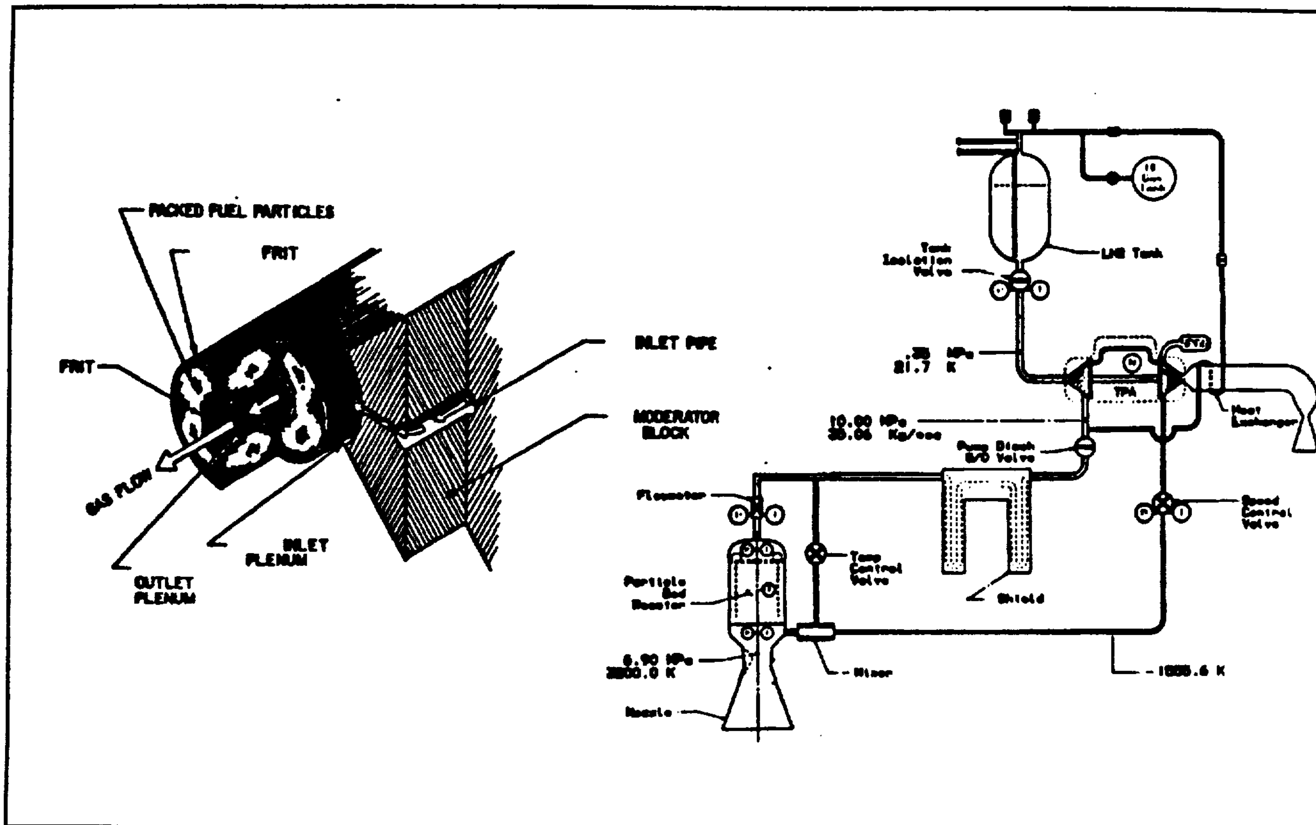

Figure 16 - Schmatic of Particle Bed Reactor and Fuel Element.

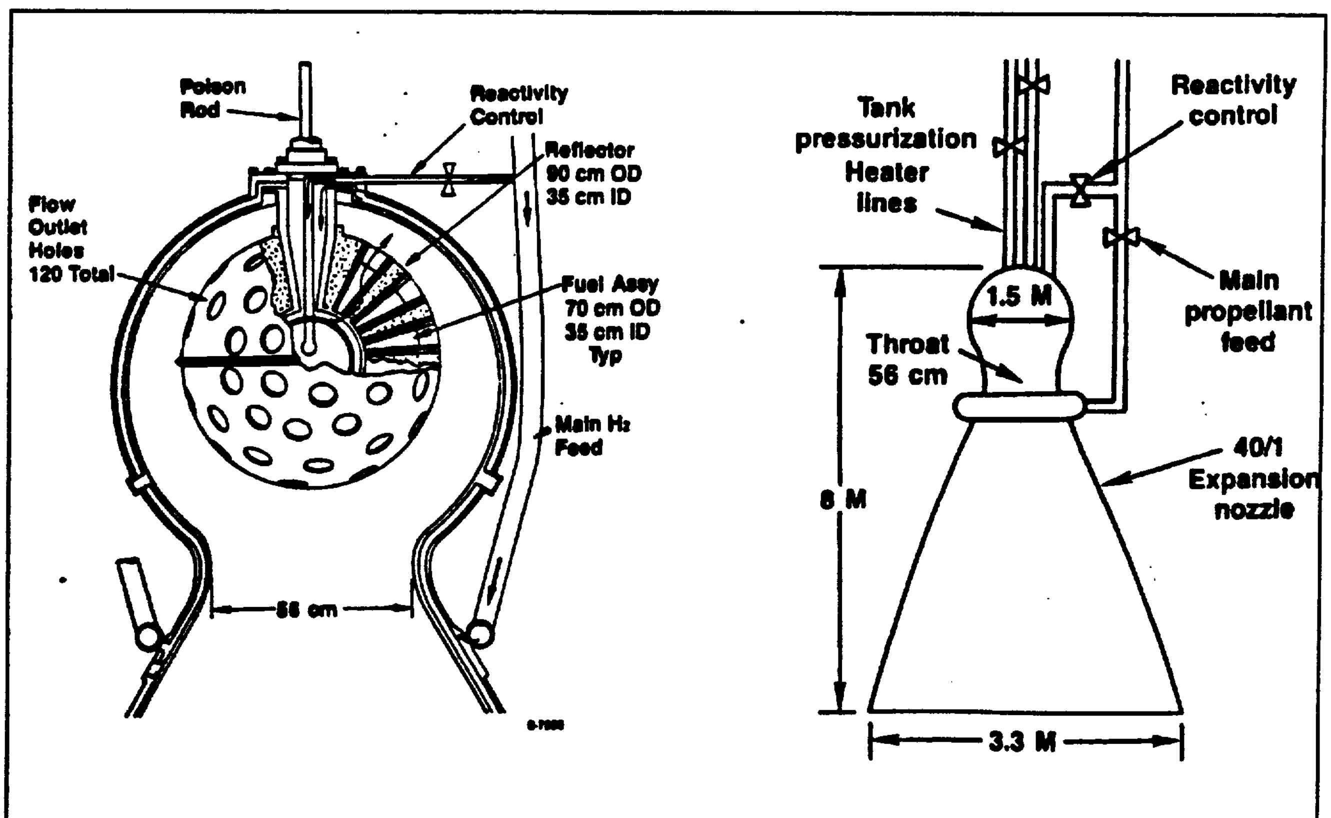

Figure 17 - Schmatic of Low Pressure Nuclear Thermal Rocket.

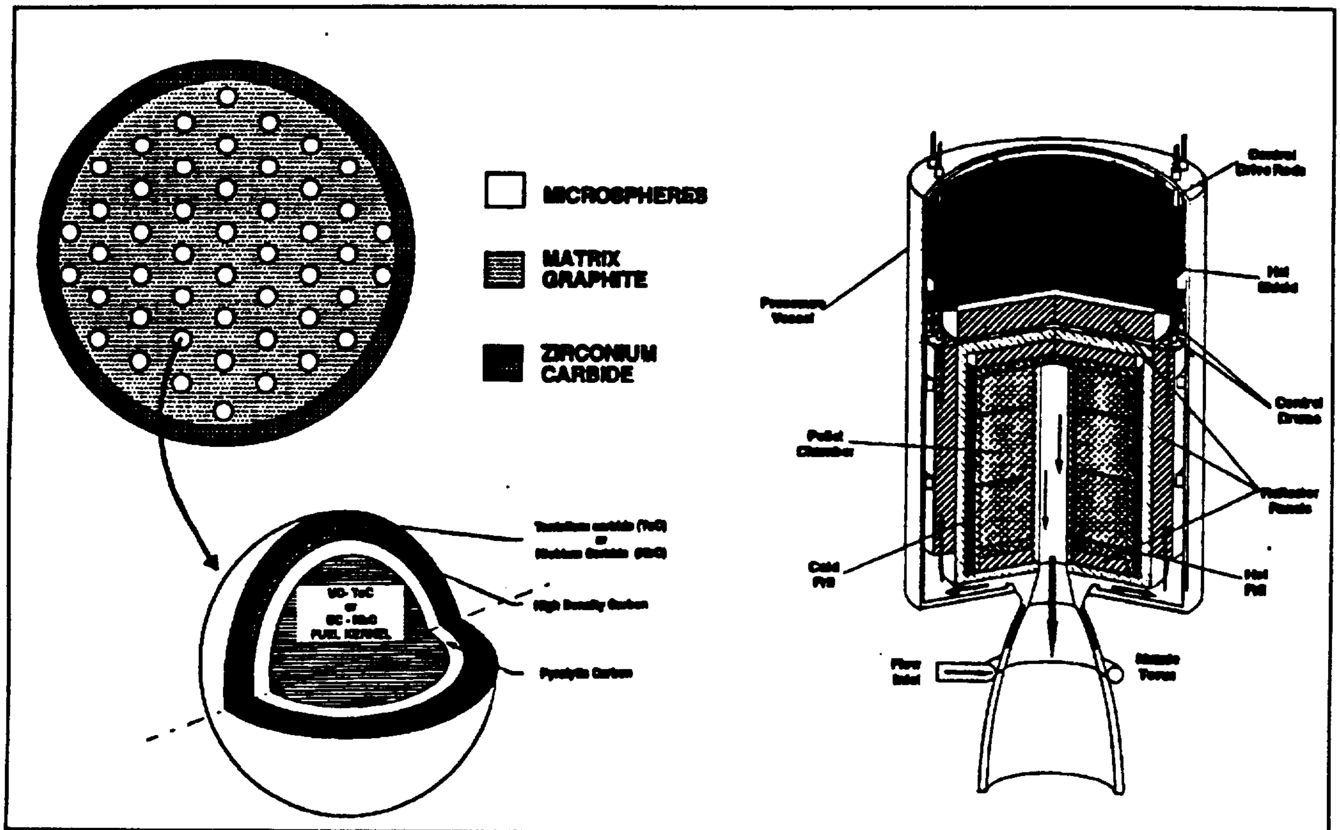

Figure 18 - Schmatic of Pellet Bed Reactor and Fuel Pellet.

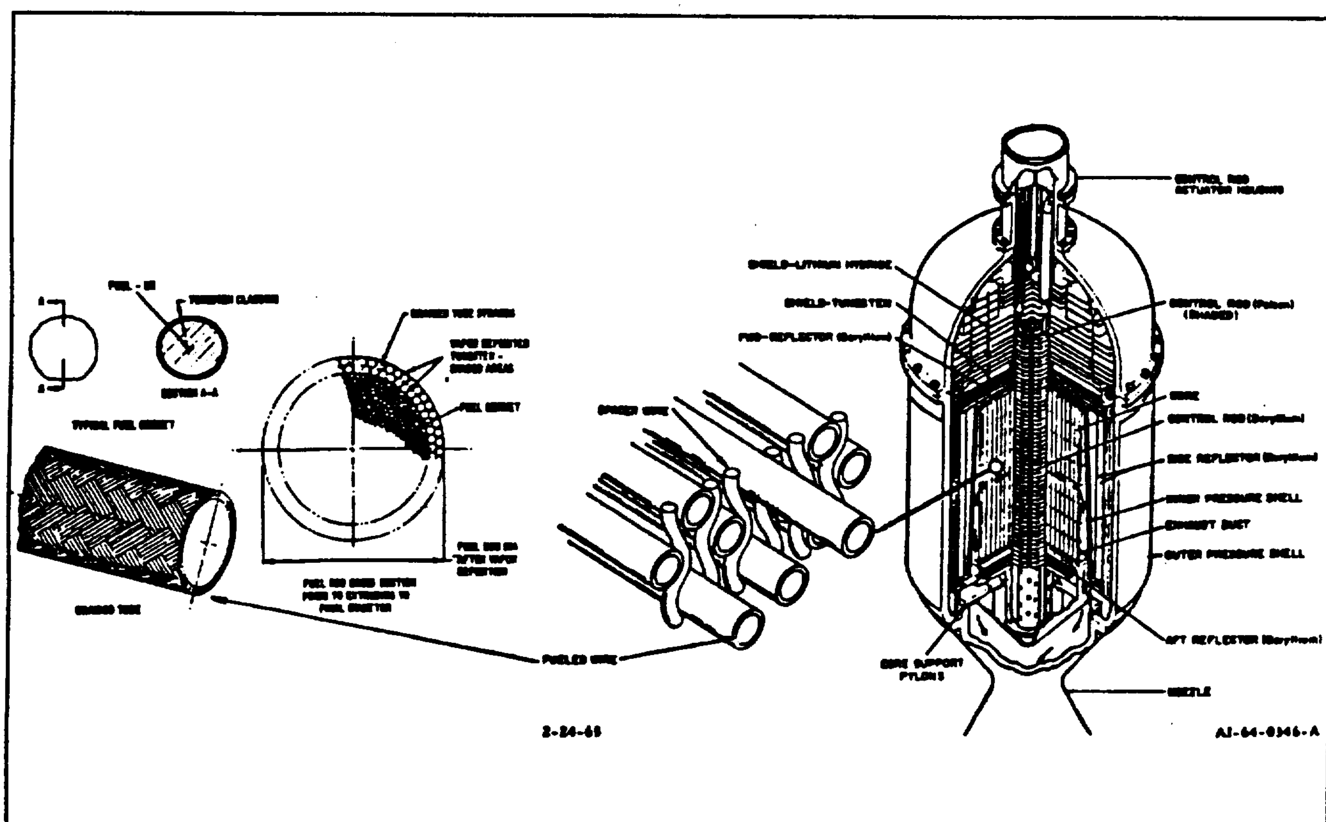
Figure 19 - Schematic of Wire Core Engine.

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13. ABSTRACT ***(Maximum 200 words)***

If we buy into the goals of the Space Exploration Initiative (SEI) and accept that they are worthy of the hefty investment of our tax dollars, then we must begin to evaluate the technologies which enable their attainment. The main driving technology is the propulsion system; for interplanetary missions, the safest and most affordable is a Nuclear Thermal Propulsion (NTP) system. This paper presents an overview of the NTP systems which have received detailed conceptual design and, for several, testing.

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