TITLE: A REVIEW OF THE LOS ALAMOS EFFORT IN THE DEVELOPMENT OF NUCLEAR ROCKET PROPULSION

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A REVIEW OF THE LOS ALAMOS EFFORT IN THE DEVELOPMENT OF NUCLEAR ROCKET PROPULSION

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ABSTRACT

This paper reviews the achievements of the Los Alamos nuclear rocket propulsion program and describes some specific reactor design and testing problems encountered during the development program along with the progress made in solving these problems. The relevance of these problems to a renewed nuclear thermal rocket development program for the Space Exploration Initiative (SEI) is discussed.

INTRODUCTION

Renewed interest in nuclear rocket space propulsion for application in the SEI has lead to numerous review papers on the broad experience gained and performance potential of the prismatic reactor fuel elements developed in the Rover/NERVA program and tested in full-scale propulsion reactor tests conducted at the Nuclear Rocket Development Station in Nevada from 1959 through 1972. A renewed nuclear thermal rocket program will require a major effort on reactor development and testing, in addition to fuel-element development, in order to achieve the full performance potential of the propulsion system. The purpose of this paper is to describe some specific reactor design and testing problems encountered by Los Alamos during the development program and the progress made in solving these problems, along with a brief review of program achievements. The areas to be discussed are

- Matching flow and power in the reactor fuel,
- High temperature measurement and control,
- The fuel element and core support structure,
- Design of the reactor core periphery, and
- An emergency cooldown system for reactor tests.

Historical Background and Performance Achieved

The nuclear rocket program, which began at Los Alamos in 1955, continued for 17 years and resulted in the development of solid-core nuclear propulsion reactor systems that far exceeded the performance goals outlined at the beginning of the program. Initial program sponsors were the Air Force and the Atomic Energy Commission. After the establishment of NASA in 1959, a joint NASA/AEC office was formed to manage the program and

in 1961 the industrial team of Aerojet and Westinghouse was chosen to develop the basic Los Alamos design into a flyable nuclear rocket engine, called NERVA. Meanwhile Los Alamos continued with its development and test program to improve the basic technology and explore advanced reactor designs. The Los Alamos goals included higher temperature, longer life, higher power, and higher power density. Los Alamos, with major industrial support from Rocketdyne, ACF Industries, and EG&G, went on to build and test 13 reactors before the termination of the program in January 1973, the Aerojet-Westinghouse team built and tested 7 reactors, two of which were part of engine tests. A chronology of these tests is shown in Fig. 1.

The Los Alamos program was a test-oriented experimental effort. Not all of the reactors achieved their experimental objectives; in fact there were some rather spectacular failures. However, there was rapid test feed-back and design modification between the early tests, and a reliable basic design configuration (Kiwi B4E) was successfully demonstrated only five years after the initial, rudimentary, Kiwi A test. A summary of the major performance records achieved in the Los Alamos program following Kiwi B4E is listed below:

- Power (Phoebus 2A)   4100 MW
- Thrust (Phoebus 2A)   930 kN
- Hydrogen flow rate (Phoebus 2A)   120 kg/s
- Equivalent specific impulse   845 s
  (Pewee 1)
- Minimum reactor specific mass   2.3 kg/MW
  (Phoebus 2A)
- Average coolant exit temperature   2550 K
  (Pewee 1)
- Peak fuel temperature   2750 K
  (Pewee 1)
- Core average power density   2340 MW/m^3
  (Pewee 1)
- Peak fuel power density   5200 MW/m^3
  (Pewee 1)
- Accumulated time at full power   109 min.
  (Nuclear Furnace)

Matching Flow to Power

The neutron flux and the consequent number of fissions per unit volume tend to vary considerably within

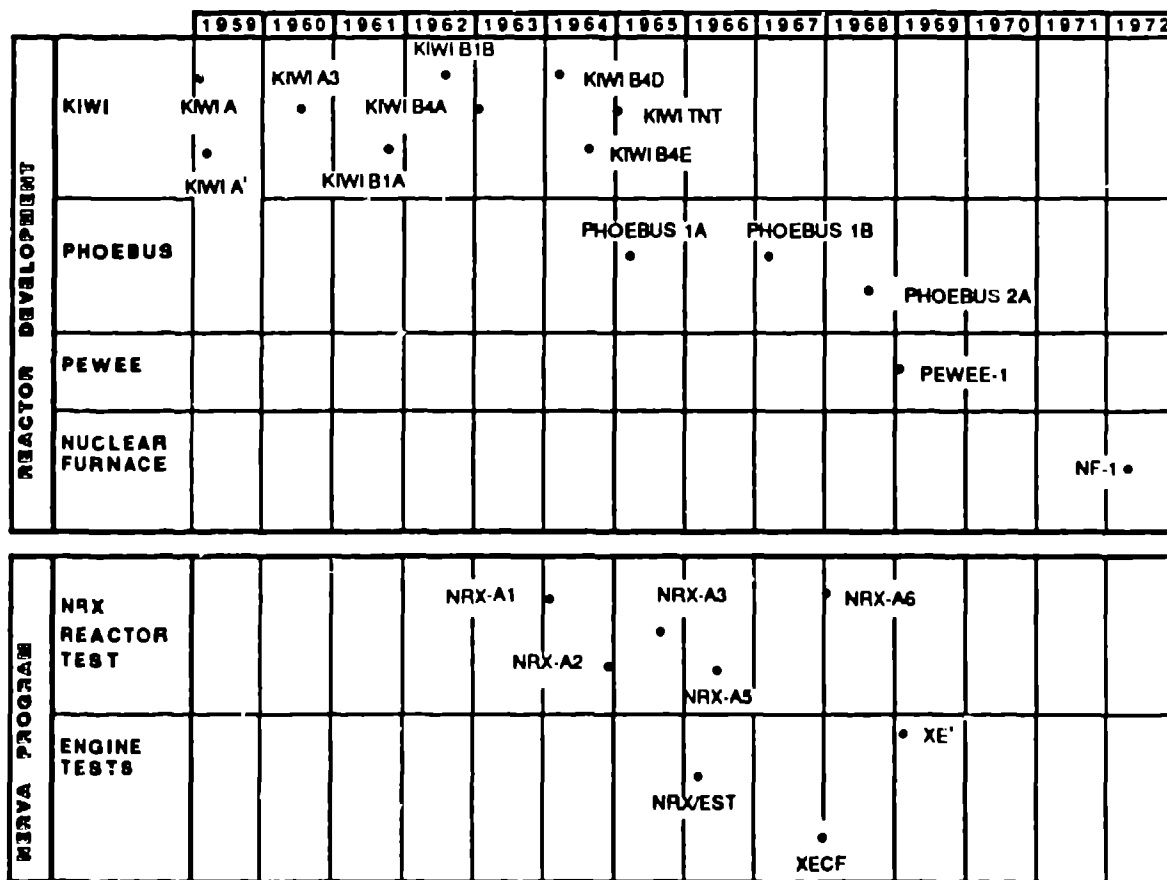|		1959	1960	1961	1962	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972
REACTOR DEVELOPMENT	KIWI		KIWI A / KIWI A'	KIWI A3	KIWI B1B / KIWI B4A / KIWI B1A		KIWI B4D	KIWI TNT / KIWI B4E							
	PHOEBUS							PHOEBUS 1A		PHOEBUS 1B	PHOEBUS 2A				
	PEWEE										PEWEE-1				
	NUCLEAR FURNACE														NF-1
NERVA PROGRAM	NRX REACTOR TEST					NRX-A1	NRX-A2	NRX-A3	NRX-A5	NRX-A6					
	ENGINE TESTS							NRX/EST	XECF		XE'				

Fig. 1. Chronology of the major nuclear rocket reactor tests.

a reactor core. For simple geometries, there are classic analytical solutions for the flux and power shape; for instance, the shape of the flux in a slab geometry is a cosine function. Except for some minor perturbations due to hydrogen density, temperature, and inlet end reflection, the axial power shape in the Rover reactors approximated a cosine distribution. The radial power distribution is not only more complex, but far more important to reactor/engine performance. A radial reflector was used for a number of reasons; perhaps the most important one being that the neutronic control elements could then be located in the reflector, a low-temperature region, and thus avoid numerous materials and safety problems. With the reflector and with a uniform fuel loading in the core, the radial power shape would be like that shown in Fig. 2. As can be seen in the figure, there is a factor of four or more variation in power density. However, in order to maximize the average fuel exit-gas temperature for a given fuel material temperature limit, the flow through each channel should be matched to the power generated in the fuel adjacent to that channel. The first step in accomplishing this goal was to flatten the radial power distribution as much as possible with fuel loading variations; that is, tailoring

the uranium content in each element to counteract the spatial variation of the fissioning flux. That was done, using twenty or more different fuel loadings to reduce the ratio of peak to average power. At the same time, of course, the system reactivity had to be kept under control. With loading variation, the power distribution was like that shown in Fig. 3. Because the neutron flux varies so rapidly near the edge of the core, there are still appreciable variations in power across a fuel element. The range of this variation depends on several aspects of the reactor design, particularly the ratio of the fuel-element thickness to core radius. To accommodate this remaining variation, individually calibrated flow orifices were used at the entrance to each coolant passage. Hole-by-hole measurements of power density were also made with uranium wires in a final cold, critical assembly of each reactor at Los Alamos before the Nevada test. Of course it was necessary to correct this measured power distribution with calculations to account for reactor operating conditions of temperature and hydrogen density in the core and reflector. The goal was to produce a uniform fuel element exit-gas temperature near the mid-point of the planned test. This goal takes into account the fact that material loss
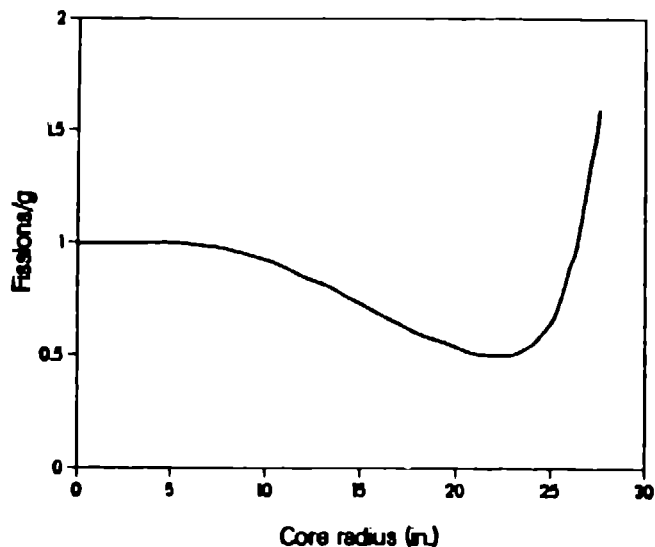
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Fig. 2. Radial power profile with uniform uranium loading.

from the fuel due to corrosion must be compensated by control drum motion which, in turn, changes the radial power distribution.

By and large, a uniform fuel exit temperature was successfully attained with these techniques, particularly when one basic reactor geometry was used repeatedly, as Los Alamos did with Kiwi B4A, B4D, and B4E and Phoebus 1A and B and Westinghouse did with the entire series of NRX reactors. Predictions were less successful for first-of-a-kind reactors such as Phoebus 2A, Pewee 1, and Nuclear Furnace in which both control drum position and power distribution were somewhat different than expected. Most of the discrepancy could be explained after the fact and it was believed that subsequent similar reactors could have been correctly adjusted.

To return briefly to the axial power distribution, it is obvious that its shape affects the heat exchanger effectiveness of the reactor as well as thermal stress levels in the fuel. Consideration was given to changing the axial power shape by axial variations in loading or fuel material, but the concept was not implemented because of lack of confidence in the endurance capability of fuel-joining techniques. Nor was it completely clear what the optimum axial power shape should be, considering thermal stress, corrosion, and heat exchanger effectiveness. With today's large and fast computers, it may be possible to progress to an optimization for specific fuel materials. Today's much more elaborate analytical tools, such as 3-dimensional neutron transport and Monte Carlo codes, should also substantially improve predictions of cold-to-hot reactivity and power distribution changes.
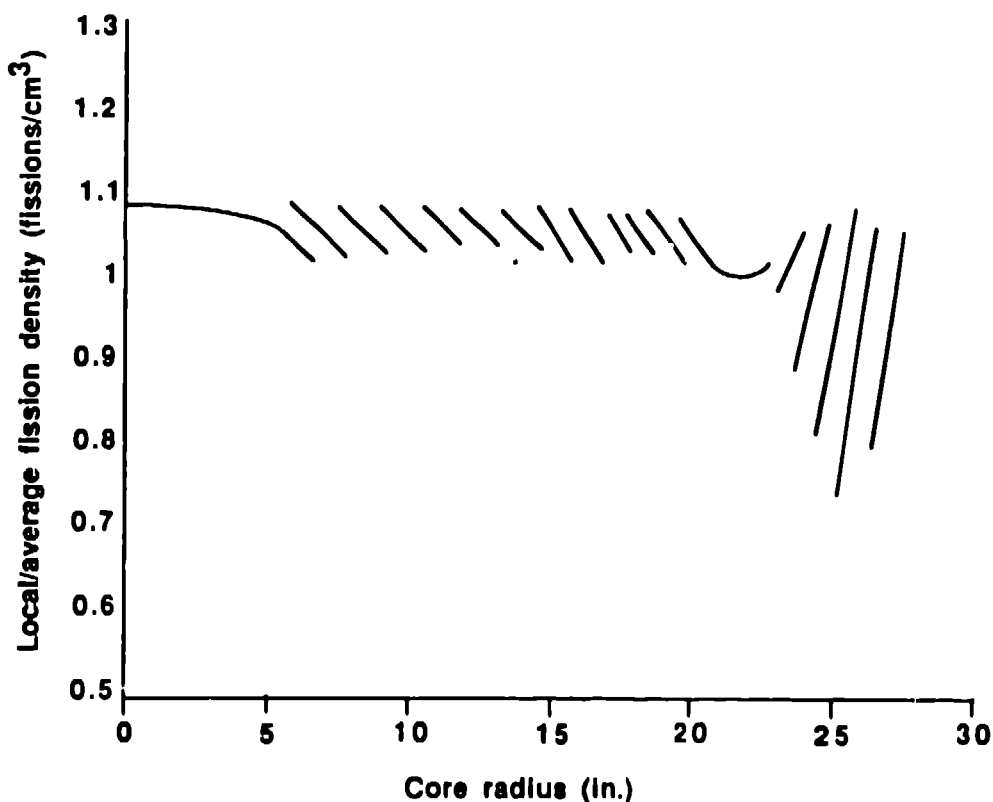

Fig. 3. Radial power profile with variable uranium loading.

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With more powerful analytical techniques, it may even prove to be feasible to add a variation that was considered but discarded as impractical; that is, varying hole size for different elements or even within an element to adjust flow distribution and thermal stress. Obviously, analysis of the thermal-hydraulic and corrosion behavior of the core becomes much more complex in this case.

High Temperature Measurement and Control

Large quantities of reactor (and facility) instrumentation were used for the Nevada reactor tests. In fact, there was usually a spirited battle between those who hoped to acquire more and more test data to assist in reactor design and development and those who were concerned that an instrumentation overload would delay or misfocus the test or even lead to dangerous perturbation of the quantities to be measured. The measurement of core material and exit gas temperatures was of particular interest and difficulty, both because of the high temperature level and the nuclear/chemical environment. An intense and eventually notably successful thermocouple development program paralleled the reactor development program. Of course, as reactor endurance and temperature capability improved, the demands on the instruments continued to increase as well.

In the larger reactors, measuring temperature for each fuel element was out of the question; therefore, thermocouple arrays were devised that would sample the fuel environment and performance in useful patterns.

Also, an average of the readings of a selected set of thermocouples was used as a control variable during reactor tests. The earliest reactor tests used exit-gas thermocouples for control with somewhat unsatisfactory results because of thermocouple failures or questionable readings. A change was then made to control based on fuel material temperature measurements near the axial mid-point of the core, using a predicted ratio of exit-gas temperature to the measured material temperature to set the control temperature. This technique, too, was not completely satisfactory even though thermocouples utilizing many new design and fabrication features had demonstrated excellent performance in furnace tests. The basic problem encountered in several reactor tests, was a time-dependent drift between the thermocouple temperature indications and apparent chamber conditions, both as observed during the test and verified by post-run analysis. When the mid-reactor "temperature" was held constant, chamber temperature gradually drifted down. There are several plausible explanations, including a radiation-induced degradation of thermocouple components, or a real temperature variation at the measurement point due to radiation or corrosion effects on the fuel. Eventually control temperatures were routinely trimmed to keep chamber conditions near the desired value. Also, more accurate and reliable exit gas thermocouples were developed. The exit gas thermocouples used to measure exit gas temperatures from 46 of the 49 cells in the Nuclear Furnace, shown in Fig. 4, performed
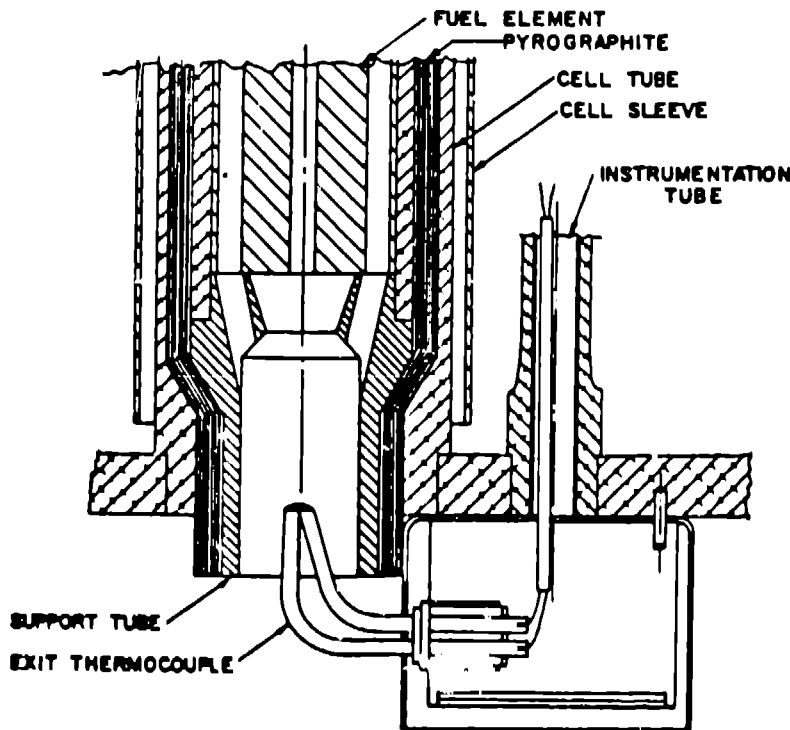
Fig. 4. Fuel-element exit-gas thermocouple installation in the Nuclear Furnace.

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almost flawlessly, with only one channel producing an apparently erroneous reading. It is worth noting however, that increasing reactor exit temperatures beyond those demonstrated by the Rover reactors will require a corollary increase in temperature measurement capability. In view of the importance of accurate temperature measurement to the entire fuel and reactor development process, high-temperature sensors should receive high priority in a renewed nuclear thermal rocket program.

Fuel Element and Core Structural Support

The fuel elements in a solid-core nuclear rocket engine must be supported against launch acceleration and vibration and the core pressure-drop loads. Acceleration loads were not simulated in the Rover tests, but a significant effort was made to develop an axial support system that would have a minimum impact on the nozzle chamber temperature. The failure of support concepts employed early in the program that made use of graphite modules loaded in tension led to the development of cooled, metallic (stainless steel or Inconel) support rods that transmitted the fuel element loads from the hot end of the core to an aluminum support plate at the core inlet. Typically, as shown in Fig. 5, the load from a cluster of six fuel elements was transmitted to a NbC-coated graphite support block and from the support block through a pyrolytic graphite insulator cup to a perforated molybdenum cone and then to a support rod that was enclosed in a central graphite element containing a pyrolytic graphite insulating sleeve which also served as the coolant flow channel for the support rod. Larger clusters and support

blocks, some with two support rods, were required at the core periphery for the transition from hexagonal to circular geometry.

The support rod hydrogen coolant mixed with the high-temperature hydrogen from the fuel elements and resulted in an undesirable reduction in chamber temperature (typically about 8%). This type of support cooling performed well and was used on Kiwi B4A, Kiwi B4D, Kiwi B4E, Phoebus 1A, Phoebus 1B, Pewee 1, and all the NERVA reactors.

To eliminate this loss in specific impulse, a regeneratively-cooled tie-tube support system was developed and successfully tested as an experiment in Phoebus 1B and throughout the Phoebus 2A cores. In this support system, shown in Fig. 6, a part of the flow was taken directly from the turbopump outlet to the tie-tube inlet manifold where it cooled the tie tubes in parallel with the primary nozzle wall/reflector coolant flow. The tie-tube coolant was then returned to the core inlet plenum where it mixed with the coolant stream from the reflector before entering the fuel elements. Thus, there was no reduction in chamber temperature from the fuel-element support cooling system. The separate coolant circuit provided by the tie tubes can also be used as a turbine energy source in a topping (expander) cycle to further increase engine specific impulse, as compared with that for a bleed cycle, in which the turbine drive gas is discharged at a much lower temperature than that of the nozzle chamber. The topping cycle feature was not demonstrated in any of the Rover/NERVA tests, but appears to be feasible based on a preliminary design study of a small nuclear rocket engine conducted at Los Alamos in 1972. A schematic of the
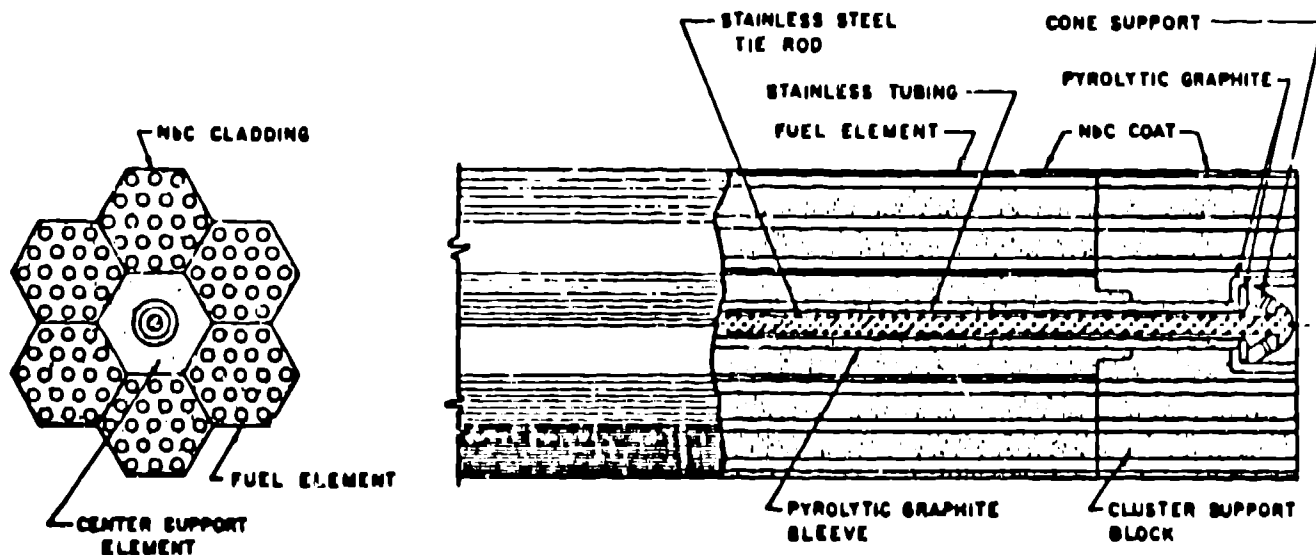

Fig. 5. Details of fuel-element cluster assembly showing cooled support rod and graphite support block.

OUTER TUBE
WASHER
SPACER
END CAP
RIBBED
SUPPORT

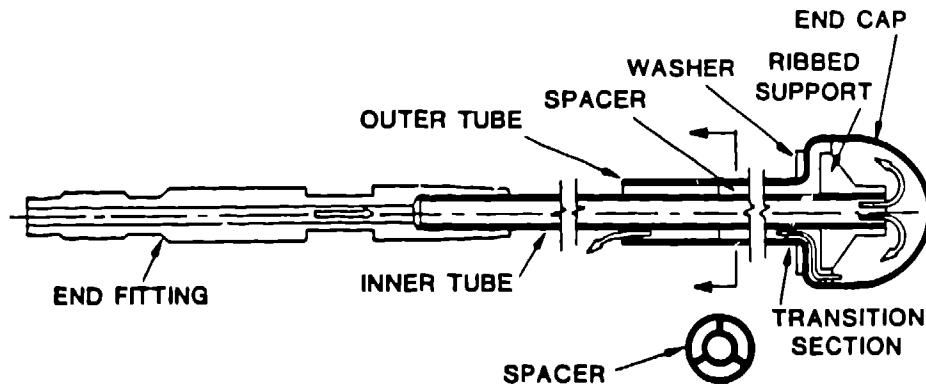END FITTING
INNER TUBE
SPACER
TRANSITION
SECTION

Fig. 6. Details of the regeneratively-cooled tie tube employed to support the Phoebus-2A fuel clusters.

small engine topping cycle is shown in Fig. 7. The tie-tube coolant loop has also been considered for the generation of 10 to 25 kilowatts of electrical power for dual-mode operation.

The coated graphite fuel-element cluster support blocks performed adequately for the Kiwi and NERVA tests, but were a source of concern because of bore corrosion and thermal stress cracks, which would be more severe at the higher fuel element temperatures and longer operating times planned in the program. Along with the development of ZrC.UC-graphite composite fuel elements, a NbC.TaC-C composite material was developed for the hot end supports to replace the graphite support blocks. These hot end supports, called pedestals, were much smaller than the support blocks and had superior resistance to corrosion and thermal stress. The pedestal not end support, shown in Fig. 8, was successfully tested in Pewee 1. A small number of thermal stress cracks were observed in the pedestals, but there was no observable corrosion. While the pedestal supports behaved as expected, it is of interest to note that there was extensive corrosion damage to the hexagonal support elements, which was later determined to be caused by improper seal tolerances at the inlet end of the core that permitted hydrogen to flow between the pyrolytic graphite insulating sleeve and the inner wall of the graphite support element.

A further reduction in the size of the hot end support pieces to increase thermal stress resistance was planned for the small engine. These supports, called mini-arches, were to be essentially one-sixth of a pedestal, as shown in Fig. 9.

Core Periphery Design

The design of the interface region between the fuel elements, which typically operated at temperatures of 2200 to 2500 K, and the beryllium reflector, which typically operated at temperatures of 150 to 300 K, was among the most challenging in the Rover program. In

addition to the very large temperature gradients, an expansion gap was required to accommodate the thermal growth of the reactor core. This expansion gap was a potential channel for flow to bypass the core; it was also found to be necessary to provide a method of radially bundling the core within this gap to minimize interstitial flow among the fuel elements and prevent flow-induced vibrations, as learned from the Kiwi B4A test. A further complicating factor was the required transition from the characteristic
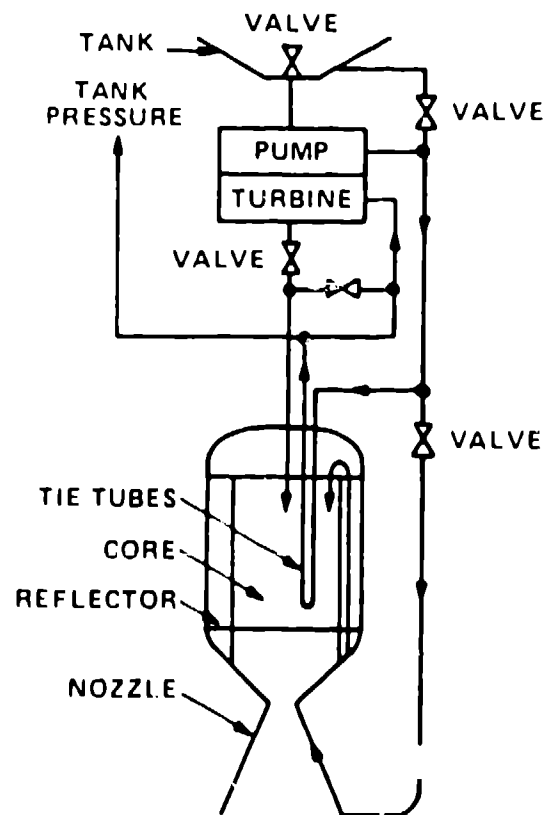

TANK
VALVE
TANK
PRESSURE
VALVE
PUMP
TURBINE
VALVE
VALVE
TIE TUBES
CORE
REFLECTOR
NOZZLE

Fig. 7. Schematic flow diagram of engine topping cycle.

STAINLESS LINER TUBE

PYROGRAPHITE SLEEVE

ZIRCONIUM HYDRIDE MODERATOR

SUPPORT ELEMENT

FUEL ELEMENT

SPIDER

SUPPORT ROD

PYROGRAPHITE

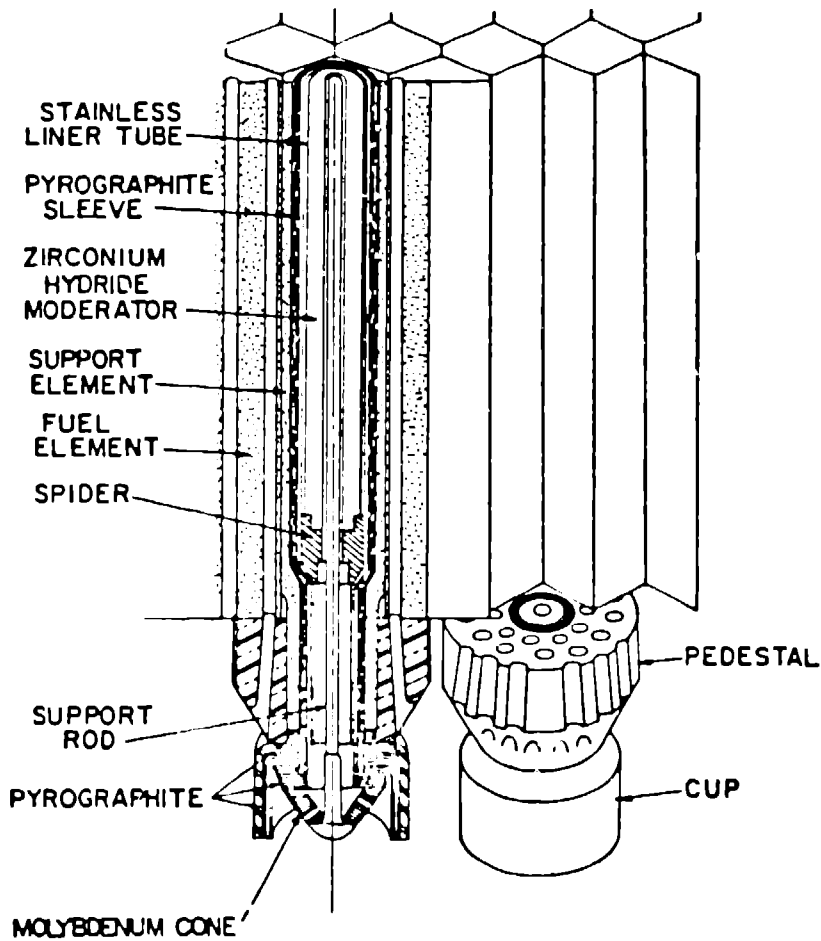MOLYBDENUM CONE

PEDESTAL

CUP

Fig. 8. Details of the hot end of the Pewee 1 fuel-element support system showing NbC.TaC-C composite pedestal and protective cup.
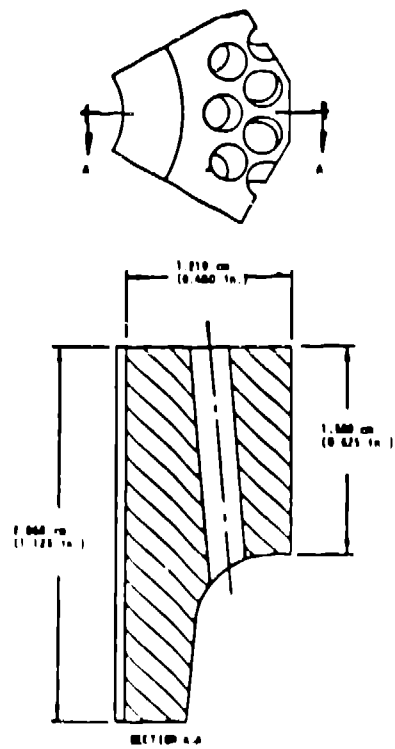
Fig. 9. Proposed mini-arch fuel-element support.

hexagonal shape of the core resulting from the assembly of fuel-element clusters to the circular shape of the reflector.

The core periphery design went through considerable development and evolution during the course of the program. While most of the periphery designs were adequate for accomplishing the primary test objectives, each design tested left room for improvement. In particular, dependence on pyrolytic graphite, with its inherent susceptibility to hydrogen corrosion, as the insulating material between the core and reflector presented a potential limitation on reliability and operating time.

A cross section of the Pewee 1 core showing details of the periphery region, including both cooled and un-cooled filler slats, an Invar wrapper, and pyrolytic graphite insulating strips, is shown in Fig. 10. An external view of the the Pewee 1 core showing details of the core-bundling differential pressure system, including an inlet centering ring, flow impedance rings to reduce the bundling pressure, hot end seal, and garter springs, is shown in Fig. 11. Not shown in Fig. 11 is the beryllium reflector surrounding the core. Pewee 1 was cycled from low temperature to design power and temperature three times for a total, full-power operating time of 40 minutes and achieved the highest fuel-element temperature and power den y in the Rover/NERVA program, as noted above. An additional 20-minute full-power cycle had been

planned for the final test, but this cycle was terminated just before reaching full-power because of corrosion-induced damage to the periphery that resulted in the ejection of several small pieces of slat material from the hot end of the core. Post-test disassembly and exami-nation of the periphery region showed that extensive corrosion in the pyrolytic graphite insulating strips had caused the Invar wrapper to buckle inward with resulting transverse fractures of the filler strips.

The Pewee 1 periphery damage reinforced the need for an insulator material that is not subject to hydrogen corrosion. A promising candidate is low-density ZrC. Development of this material was underway at Los Alamos for several years prior to the termination of the program, and a ZrC insulator experiment was included in the Nuclear Furnace test with encouraging results. Devel-opment of an improved high-temperature insulator mate-rial may be of equal or higher priority than the further development of high-temperature fuel elements in a re-newed nuclear thermal rocket program. Much more work needs to be done in this area.

Emergency Cooldown System

The shutdown of a reactor must be followed by a cooldown period to remove the residual heat generated in
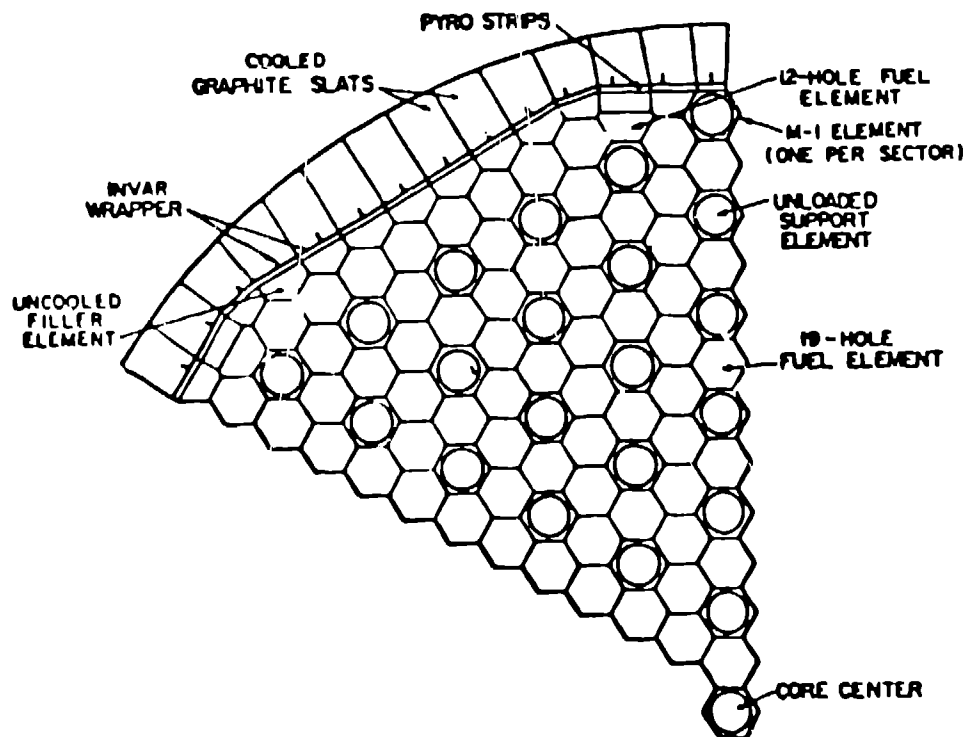

Fig. 10. Cross section of the Pewee 1 core showing details at the periphery.
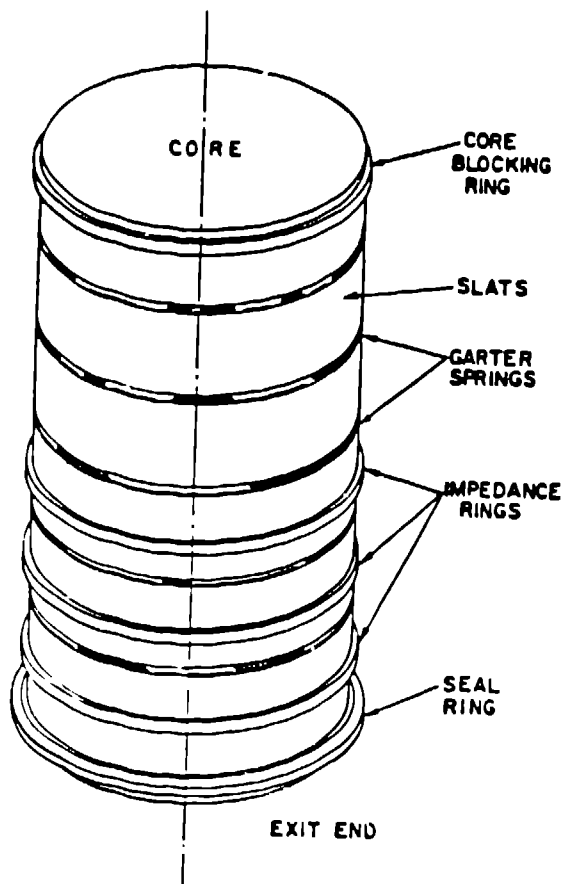
Fig. 11. External view of the Pewee 1 periphery.

the reactor core as a result of the radioactive decay of fission products formed during the powered operation. This heat is generated in three types of decay: (1) fission chains triggered by delayed-neutron precursor decay, (2) beta decay, and (3) gamma decay. The fission heat drops off quickly and is virtually zero within about ten minutes, but the beta and gamma decay tail off m e slowly and require some cooling for a day or longer, depending on the run duration.

During the delayed-neutron heating period, programmed cooldown of Kiwi B4D and Kiwi B4E required continued use of the turbopump, which supplied liquid hydrogen from the main dewars. This high temperature phase was followed by a switchover to gaseous hydrogen

at a pre-determined decay power and core temperature and a later switchover to gaseous nitrogen flow and pulsed cooling. The same cooldown sequence was planned for Phoebus 1A. The objective of the Phoebus 1A test was to operate at full power and temperature for as long as the limited (100 000 gal.) liquid hydrogen supply at Test Cell C would permit. (The two 500 000 gal. dewars were under construction at that time and were not available.) Because of erroneous liquid level capacitance gauge readings caused by the intense radiation field, the reactor was operated at full power until the remaining liquid hydrogen was not sufficient to prevent the core from overheating immediately after shutdown with resulting severe damage to the core and the loss of post test fuel-element examination and performance analysis.

Following this test an emergency cooldown system was designed and installed that, when activated at full power, would automatically come on line when the turbopump outlet pressure dropped below 5.2 MPa (750 psi) and supply high-pressure liquid hydrogen to the reactor until the time in the cooldown that the reactor could be safely cooled with gaseous hydrogen. This system, which was in place for the Phoebus 1B test, consisted of a 30 280-L (8 000-gal.), high-pressure dewar connected to the Test Cell C flow system through a check valve downstream of the turbopump outlet. It was successfully demonstrated and routinely operated in the normal shutdown sequence during both the Phoebus 1B and Phoebus 2A experimental plans. A modified emergency cooldown system having a much smaller high-pressure coolant volume than that required for Phoebus 2A was successfully demonstrated in the relatively low-powered Pewee 1 tests (500 MW vs 4 000 MW).

CONCLUSIONS

A renewed nuclear thermal rocket program will require a major effort on reactor development and testing, in addition to fuel-element development, in order to achieve the full performance potential of the propulsion system. Significant progress was made in the problem areas discussed above that were encountered in the Rover program, but much work remained to be done. These or similar problems can be expected to require further development for the higher temperatures and higher power densities envisioned for nuclear thermal rocket application in the SEI.

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