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SUMMARY OF SPACE NUCLEAR REACTOR POWER SYSTEMS (1983-1992)

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Abstract

Major developments in the last ten years have greatly expanded the space nuclear reactor power systems technology base. In the SP-100 program, after a competition between between liquid-metal, gas-cooled, thermionic, and heat pipe reactors integrated with various combinations of thermoelectric, thermionic, Brayton, Rankine, and Stirling energy conversion systems, three concepts were selected for further evaluation. In 1985, the high-temperature (1,350 K), lithium-cooled reactor with thermoelectric conversion was selected for full scale development. Since then, significant progress has been achieved, including the demonstration of a 7-y-life uranium nitride fuel pin. Progress on the lithium-cooled reactor with thermoelectrics has progressed from a concept, through a generic flight system design, to the design, development, and testing of specific components. Meanwhile, the U.S.S.R. in 1987-88 orbited a new generation of nuclear power systems beyond the thermoelectric plants on the RORSAT satellites. Two satellites using multicell thermionic Topaz I power plants operated six months and eleven months respectively in space at 5 kWe. An alternate single cell thermionic power plant design, called Topaz II, has been operated in a ground test for 14,000 h in one unit. The U.S. has continued to advance its own thermionic fuel element development, concentrating on a multicell fuel element configuration. Experimental work has demonstrated a single cell operating time of about 1 1/2-y. Technology advances have also been made in the Stirling engine; an advanced engine that operates at 1,050 K is ready for testing. Additional concepts have been studied and experiments have been performed on a variety of systems to meet changing needs, such as powers of tens-to-hundreds of megawatts and highly survivable systems of tens-of-kilowatts power.

INTRODUCTION—TEN YEAR TREND (1983-1992)

In 1983, the National Aeronautics and Space Administration (NASA), the Department of Energy (DOE), and the Department of Defense (DoD) entered into an agreement to fund a space nuclear reactor power program. This program, called SP-100, superseded the heat pipe reactor development program (also called SP-100) that had started in 1979. The goals of this program are to develop the technology to provide tens-to-hundereds of kWe of electric power with the specific initial design concentrating on 100 kWe at an operational time of 7 y and lifetime of 10 y. Starting with a broad range of candidates (Hylin and Moriarty 1985, Chiu 1985, Harty et al. 1985, Yoder and Graves 1985 and Terrill and Putnam 1985) including liquid-metal, gas-cooled, thermionic, and heat pipe reactors with various combinations of thermoelectric, thermionic, Brayton, Rankine, and Stirling energy conversion systems, three concepts were selected for further evaluation. These were: (1) a high-temperature, pin-fuel element reactor with thermoelectric conversion, (2) an in-core thermionic power system, and (3) a low-temperature pin-fuel element reactor with

Stirling cycle conversion. In 1985, the high-temperature pin-fuel element reactor with thermoelectric conversion was selected for development to flight readiness. The thermionic powerplant offered a more compact heat rejection subsystem and lower temperature structural materials, but issues of lifetime excluded its selection. The Stirling system offered higher energy conversion efficiency and lower temperature materials, but the technology risk was considered greater at that time because of the preliminary status of its development. Because sufficient merit was recognized in the other options, a technology program was started on in-core thermionic fuel elements, called the Thermionic Fuel Element Verification Program, and another program on developing a high-temperature Stirling engine that can be mated with the high-temperature pin-fuel reactor. Since 1985, the SP-100 power system has progressed from a concept, through a generic flight system design, to the design, development, and testing of specific components.

In the meantime, the U.S.S.R. launched a new generation of space reactors with flights of Cosmos 1818 and 1867 in 1987 and 1988. Details of these systems, known to the U.S. as Topaz I, were first introduced to the West at the Sixth Space Nuclear Power Systems Symposium in 1989.

During the past ten years, a number of activities occurred in developing multimegawatt-level power systems to support power needs for directed energy weapons and electric propulsion. The Multimegawatt Program evaluated systems for: (1) open loop, power levels of tens-of-megawatts for hundreds of seconds, (2) closed loop, power levels of tens-of-megawatts for one year, and (3) open loop, power levels of hundreds-of-megawatts for hundreds of seconds. Six concepts were considered during the Phase I preconceptual activities. The purpose of Phase I was to identify key technology feasibility issues. Phase II was to resolve the issues prior to technology selections. This program was terminated in 1990, because of funding constraints, before Phase II design contracts were awarded.

There is some interest in multimegawatt systems for use in electric propulsion (Doherty and Gilland 1992 and Gilland and Oleson 1992). The major candidate for this mission is a version of the SP-100 nuclear subsystems integrated with a Rankine cycle power conversion system.

In this report, emphasis is being given to the major system level developments. In the U.S., development concentrated on SP-100; in the U.S.S.R., development was on in-core thermionic Topaz power systems. In addition, all areas of activities where technology developments have occurred are summarized. The major U.S. technology efforts besides SP-100 have been in reactors with in-core thermionic power converters. The ten-year period ends with uncertainty as a result of changing directions and budget constraints.

HIGH-TEMPERATURE, LIQUID-METAL-COOLED POWER PLANT DEVELOPMENT (SP-100)

Requirements

SP-100 is being designed to provide tens-to-hundreds of kWe power for 7 y at full power and 10 y overall operation. Power plant components are to provide this wide range of power without significant requalification of the basic building blocks. These requirements were originally derived from projected DoD needs for robust surveillance pumps, (5) solid-state thermoelectric power conversion units using silicon germanium/gallium phosphide (SiGe/GaP) to convert thermal power to electricity, and (6) the heat rejection system consisting of a carbon/carbon matrix structure and armor with titanium/potassium heat pipes (Mondt 1989 and Josloff 1988).

The building blocks that form SP-100 can be configured in a number of arrangements depending on mission needs and desired power levels. The baseline configuration arrangement for a 100 kWe power plant has the reactor at the forward end of the power plant away from the payload. The largest segment of the power plant is the heat rejection area. The total length of the deployed power plant without the boom is 12 m. The boom, used to minimize the amount of shielding needed, makes the overall length 25 m. This can be stowed for launch.

The heart of the reactor design is the fuel pin, shown in Figure 3. The UN fuel is fabricated in the form of pellets, having a density of 94.5 percent theoretical and uranium enrichment of 97 percent. Approximately 50,000 pellets are needed for a 100 kWe thermoelectric system core. Cladding provides structural strength for the fuel pin, while a layer of rhenium acts as a barrier between the fuel and lithium coolant, prevents loss of nitrogen from the UN fuel, allows thinner cladding, and acts as thermal poison if the reactor is immersed in water. The fuel pin cladding is the niobium alloy PWC-11 refractory material tubing with rhenium (Re) tubing bonded to the internal surface. The UN fuel operates at a peak surface temperature at the beginning of life of 1,400 K and end of life of 1,450 K and a peak burnup of 6 atom percent. Surrounding the fuel pin is a wire wrap spacer that provides space for the lithium coolant to flow. The fuel pins have an upper plenum region to contain fission products.

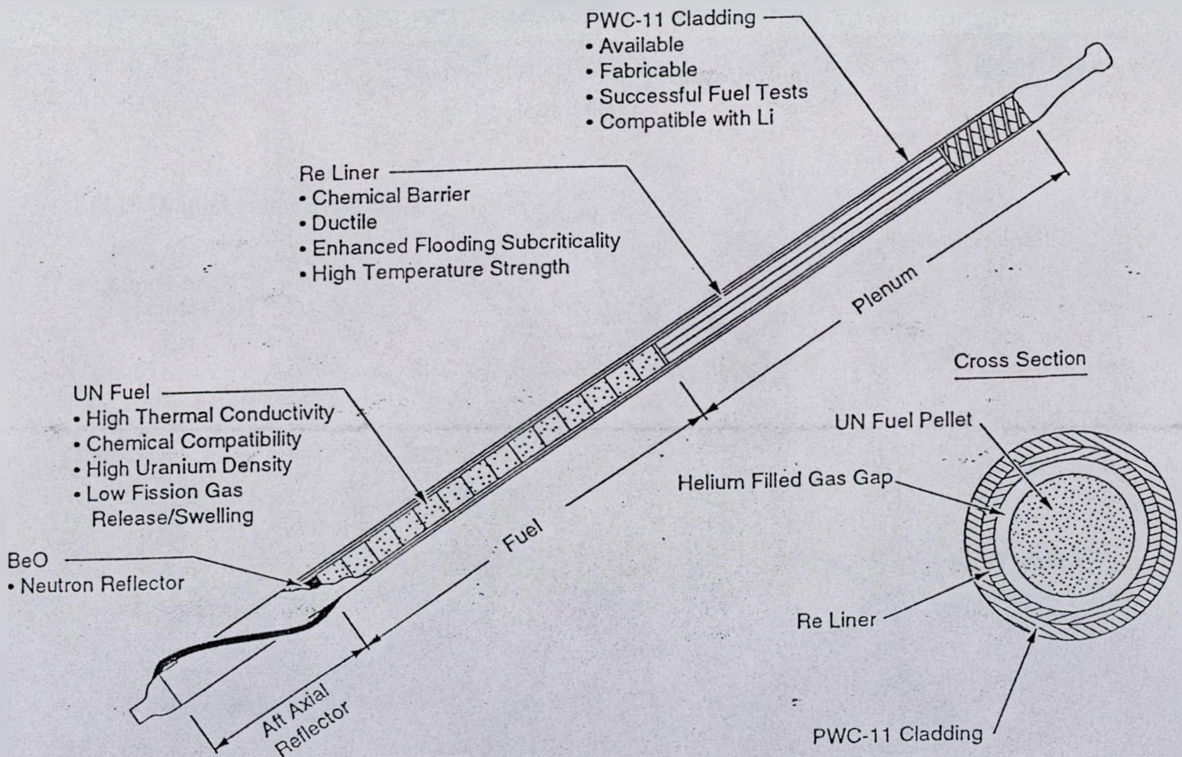


FIGURE 3. SP-100 Fuel Pin (Truscello and Rutger 1992).

The reactor core is separated into 10 hexagonal-shaped assemblies with approximately 61 fuel pins per assembly and six partial assemblies, each with approximately 50 pins, around the outer edge (similar to the configuration shown in Figure 4). The partial assemblies result in the hexagonal shape being closer to a circular form. The core also contains three safety rods for accidental criticality mitigation and, in case of loss-of-coolant flow, an auxiliary cooling system to prevent core melt down. The in-core safety rods provide a redundant shutdown system and help maintain the core subcritical in case of fire, explosion, water submersion, or compacting accidents. The spacing between the fuel pins is used as coolant paths for the lithium which transfers heat to the converters.

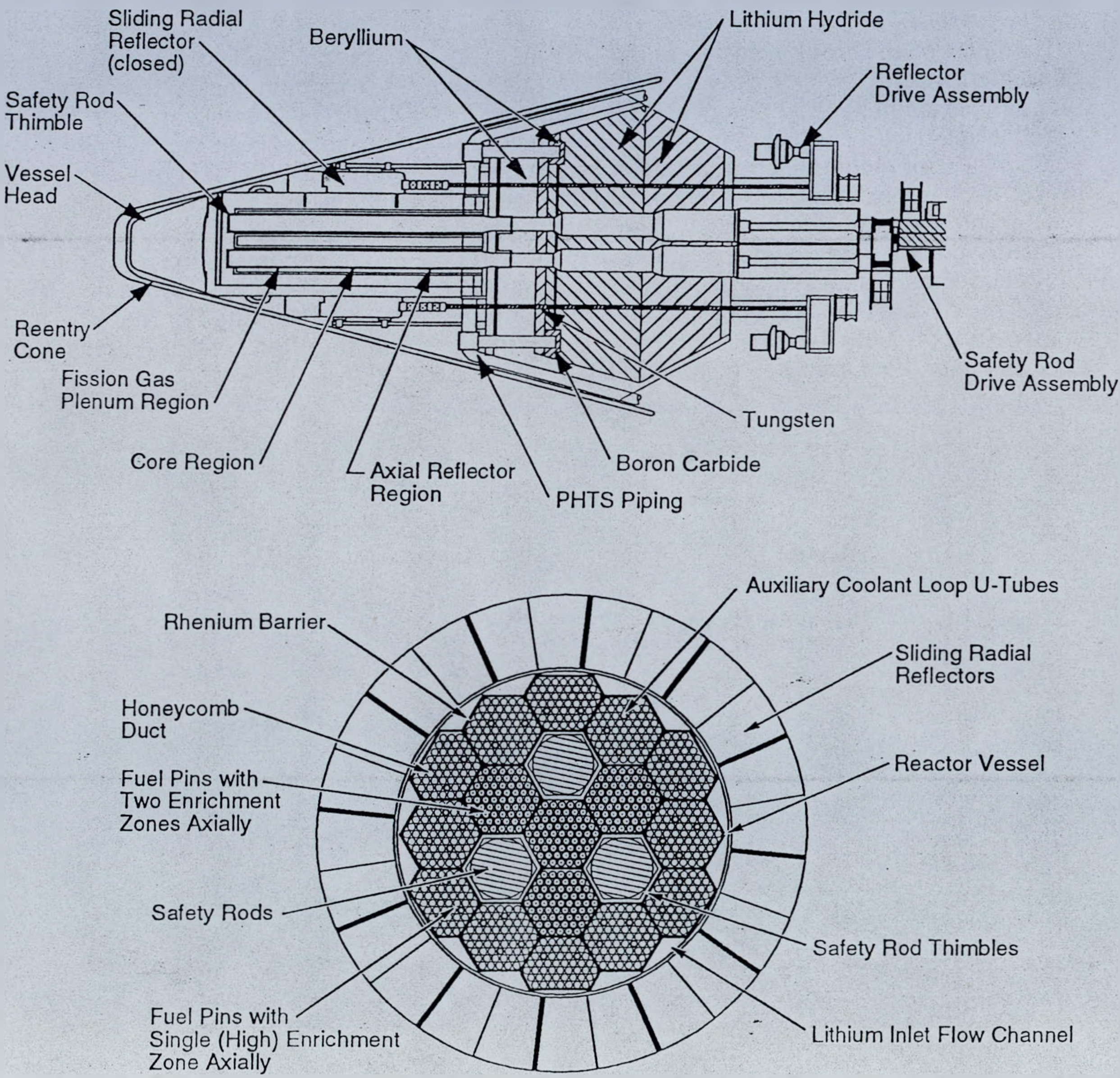


FIGURE 4. SP-100 Reactor Subsystem Components (Truscello and Rutger 1992).

The auxiliary coolant function option is designed to limit reactor fuel temperatures to less than 2,000 K in case of a loss-of-coolant accident. It provides coolant loops totally independent of the normal heat removal system. Heat removal is by 42 u-tubes located throughout the core pin bundles (Mondt 1992). The auxiliary lithium coolant passes through the core into a collection manifold, from which it is pumped to an independent radiator for rejection of heat to space.

The in-core safety rods are of boron carbide (B4C) neutron absorbing material inside structural thimbles that can be located in or out of the core. The rods have a follower segment of beryllium oxide (BeO) that acts as moderator during operation.

Normal control is by means of twelve sliding tapered reflector segments. The segments control neutron leakage from the core. The positioning of the reflector segments is used to bring the reactor critical once the in-core safety rods are removed and to compensate for fuel burnup and swelling. The reflector segments consist of BeO contained within a Nb-1% Zr shell.

The radiation shielding approach for minimizing system mass resulted in a conical shadow-shield with a cone half-angle of seventeen degrees (Disney et al. 1990). Within the shield, the design must thermally isolate high performance/low mass shielding material with limited temperature capabilities from the adjacent high temperature components. The shield is fabricated from lithium hydride (LiH) pressed into a stainless-steel honeycomb to attenuate neutrons. Depleted uranium plates are added primarily to attenuate gamma radiation. Beryllium is used to transfer heat to the outer surface of the shield, where it is radiated to space. Stainless steel is used as a structural element and to contain the shield.

Surrounding the reactor and butting against the radiation shield is a carbon-carbon reentry heat shield. The purpose of the reentry shield is to ensure confinement of the core fission products if the reactor reenters the atmosphere. It is conical in shape, and the geometry protects the reactor vessel from overheating during reentry, keeping the temperature to 300 K even through the reentry shield might reach 3,200 K (Deane et al. 1989).

Heat is transferred to the converter by six interrelated pumped lithium loops (Atwell et al. 1989). Thermoelectric-electromagnetic (TEM) pumps are used to circulate liquid lithium in each loop (Figure 5). Each pump also circulates lithium coolant on the cold side of two heat rejection loops. These are self-energized, DC conduction, electromagnetic TEM pumps. Hot and cold ducts run parallel to each other along the active length of the pump. Sandwiched between them is a series of thermoelectric elements connected on each side by compliant pads. The temperature differential imposed across the thermoelectric converters generates an electrical current that travels at right angles to the lithium flow in the ducts and in closed paths around a magnetic center iron; as the current circulates, an induced magnetic flux is generated in the center iron that is directed through the lithium ducts perpendicular to the current. The interaction of the magnetic flux and the current produces a force on the molten lithium that drives it along the ducts. Hence, flow responds to the temperature in a self-regulated way.

During reactor operation, helium gas is generated in the coolant loop by neutron reactions with lithium. Enriched lithium in ^{7}Li (99.9% ^{7}Li, 0.1% ^{6}Li in contrast to natural lithium, which has an isotope ratio of 92.6/7.4) is used to minimize production

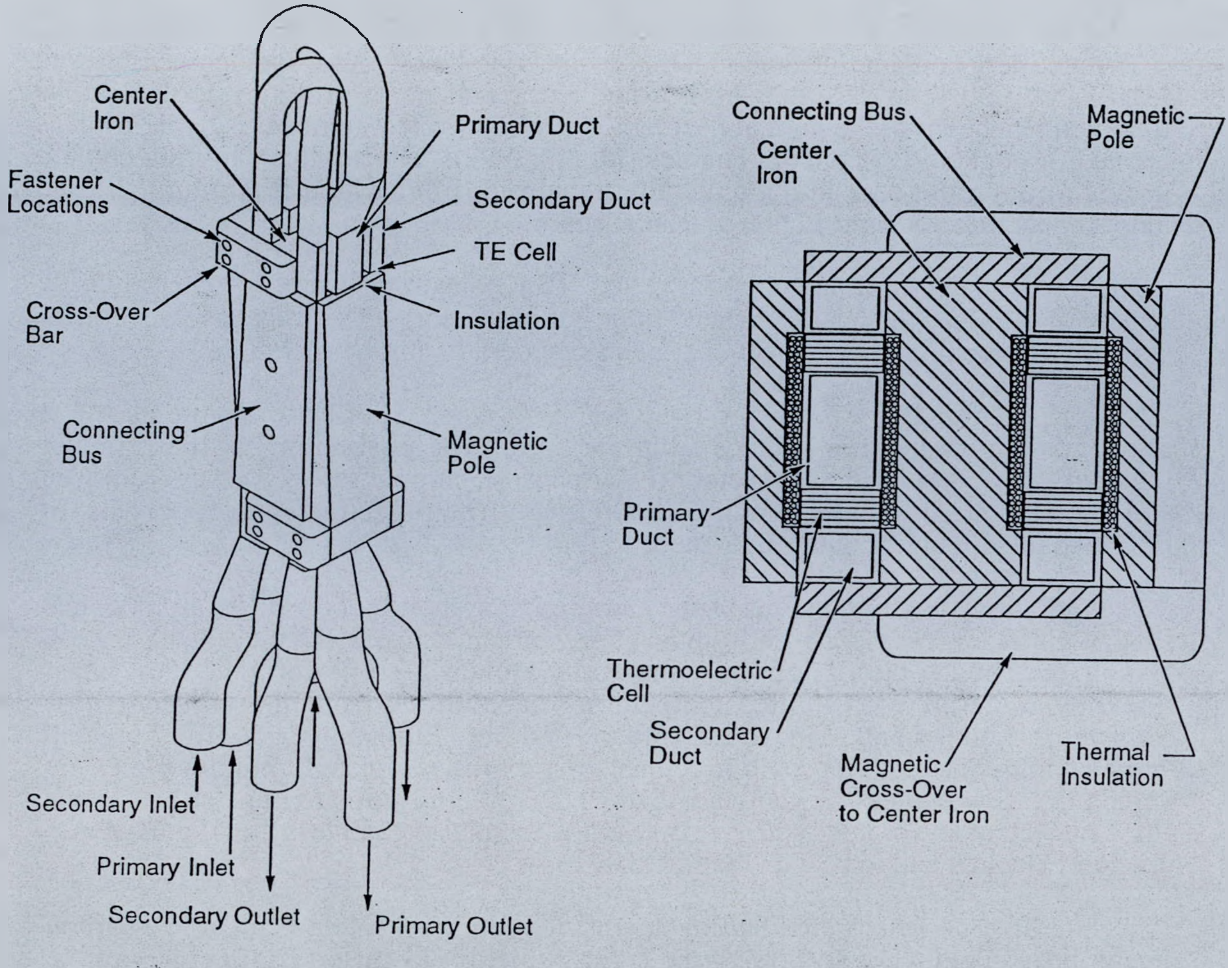


FIGURE 5. SP-100 TEM Pump Development (Truscello and Rutger 1992).

of helium. A gas separator/accumulator (Figure 6), used to separate the helium from circulating lithium, utilizes the principles of surface tension and centrifugal force.

The reactor is controlled based on temperature and neutronic measurements. Flow is inherently maintained by the TEM pumps (for example, higher reactor temperatures increase TEM pumping). Sensors are multiplexed in an analog multiplexer/amplifier located behind the reactor shield. These are designed to operate in a radiation environment of 4×10^{15} n/cm^2 and 2.4×10^8 rad (gamma) in 10 y of operation. N-type junction field effect transistors (JFET) semiconductor devices are being used.

Conversion of thermal energy from the reactor to electrical power is accomplished through the Power Conversion Assembly (PCA) (Figure 7) (Bond et al. 1993). Heat is conducted from hot lithium in a central heat exchanger, through TE cells on either side (which convert some of the heat to electricity), and then to a pair of heat exchangers where cooler lithium carries the waste heat to the heat rejection subsystem. The PCA building blocks can be packaged in any combination to generate the desired output. These building blocks are the thermoelectric cell and thermoelectric converter assembly (TCA). Each assembly has two arrays of 60 TE cells. The TCA consists of two cell arrays, one hot side heat exchanger, and two cold side heat exchangers. A

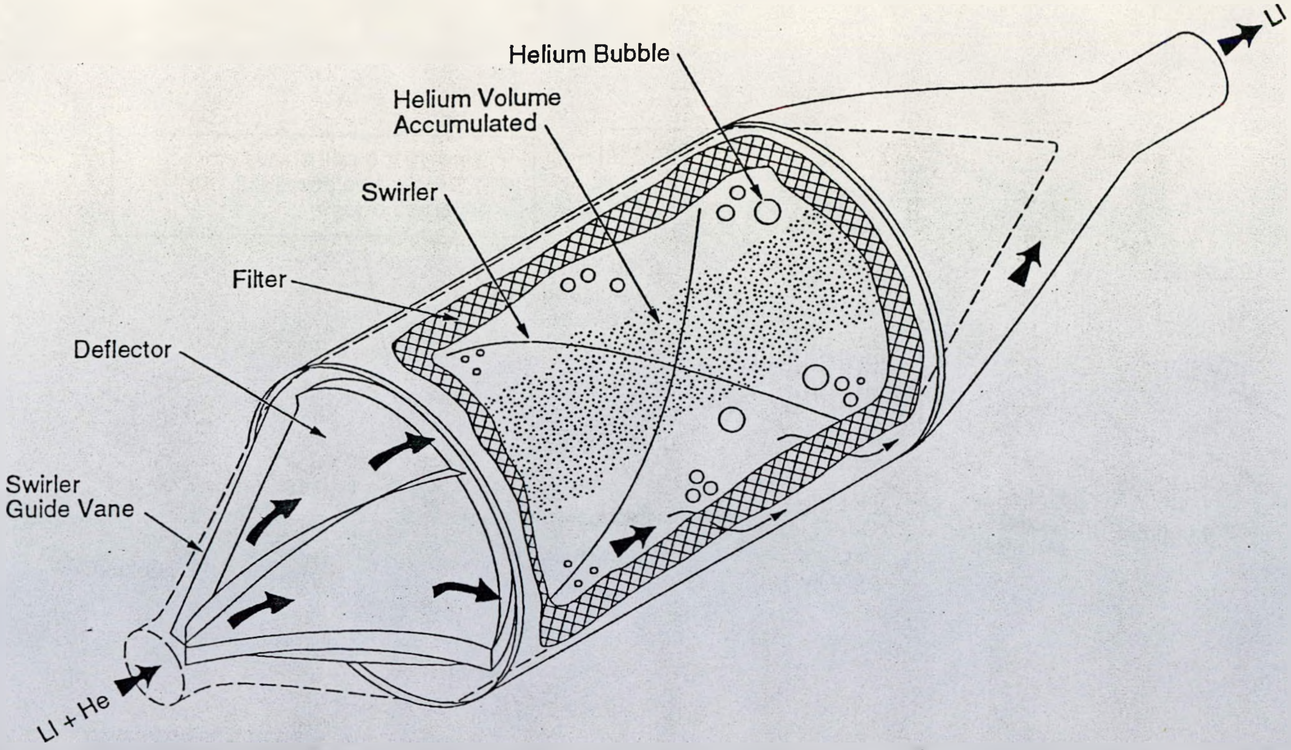


FIGURE 6. SP-100 Lithium/Helium Gas Separator-Accumulator (Truscello and Rutger 1992).

typical TCA configuration have 6 x 10 cell arrays. The thermoelectric (TE) converter assemblies are in a stack of six plate-and-frame configuration, each of which produces 1.5 kWe at 34.8 VDC. A total of 8,640 cells is required for a 100 kWe system.

The thermoelectric cells (see Figure 8) use silicon germanium/gallium phosphide (SiGe/GaP) materials. Each cell consists of a thermoelectric module, compliant pads to accommodate thermal stresses, electrical insulators to isolate the electrical power from the spacecraft, and conductive coupling to the heat exchangers. The cells are arranged in a parallel/series electrical network to provide the desired 200 V output. The thermoelectric material figure-of-merit is being increased to 0.85×10^{-3} K^{-1} by the addition of GaP to 80:20 SiGe from 0.67×10^{-3} K^{-1} for SiGe. A graphite-electrode-SiGe bond is used to make the electrical contact resistivity of that joint less than 25 $\mu\Omega$/cm. The compliant pad is used to prevent cracking of the TE elements from thermal expansion. It consists of niobium fibers bonded to niobium face sheets on both sides. The niobium face sheet matches the thermal expansion of both the heat source material (PWC-11) and the heat sink material (Nb-1% Zr). The insulators are single-crystal alumina, with 4,000 V/cm voltage gradient at 1,375 K.

The radiator will be tailored to the particular application. In the GFS, the heat rejection subsystem includes twelve radiator assemblies, each constructed of an array of varying lengths of potassium heat pipes brazed to a lithium duct-strongback structure at the center (see Figure 2). The lithium duct-strongback structure consists of the lithium supply and return ducts. Each of the twelve radiator panels have flexible joints for the supply and return ducts so that the radiator can be folded for launch and deployed for reactor operation. Accumulators are included within the heat rejection subsystem to accommodate the variable volume necessary to compensate for expansion

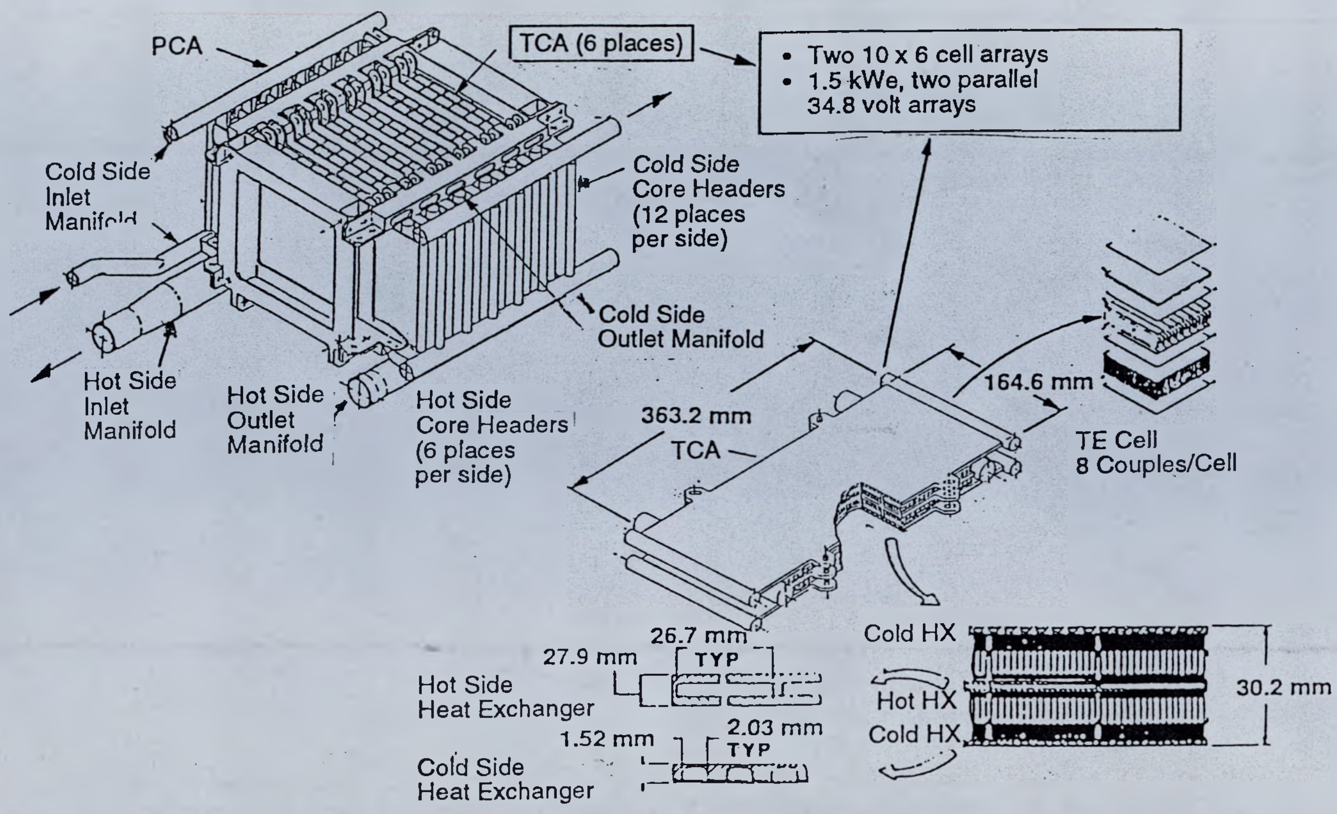


FIGURE 7. SP-100 Power Converter Subsystem Components (Truscello and Rutger 1992).

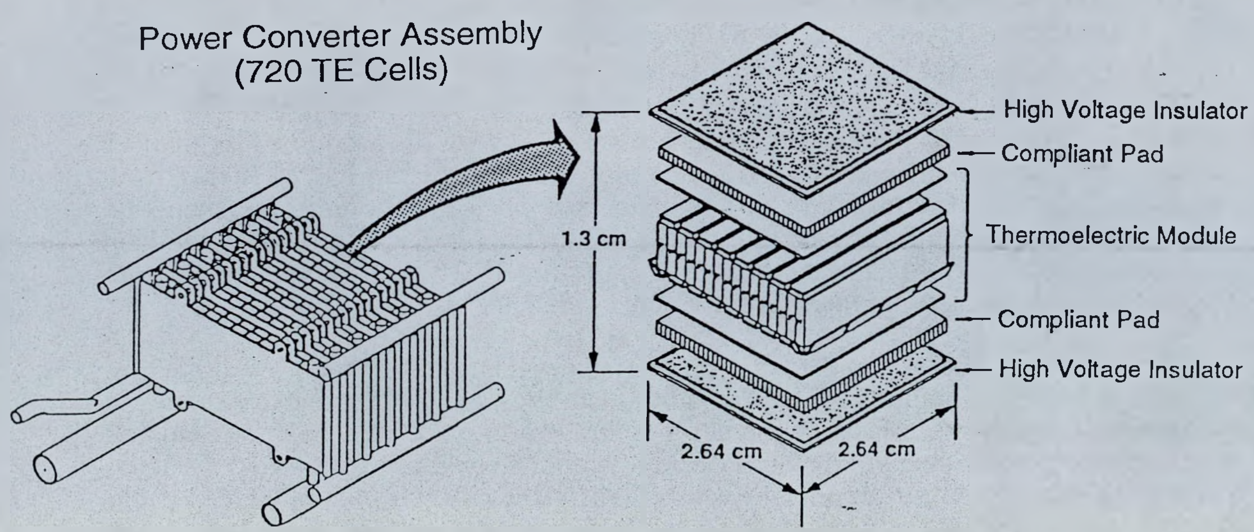


FIGURE 8. SP-100 Thermoelectric Cell (Truscello and Rutger 1992).

and contraction of the lithium coolant. The accumulators are fabricated out of titanium. Heat rejection area is 104 m^2.

For restarting the power plant in space, the reactor is used to heat an auxilliary liquid metal loop to melt the lithium throughout the system (Hwang et al 1993).

Table 2 summarizes some key power plant performance parameters.

TABLE 2. SP-100 GFS Design Performance Parameters.

Parameters	Values
Reactor power (MWt)	2.5
Peak reactor outlet temperature EOL (K)	1,375
Heat loop ΔT (K)	92
Heat loop mass flow (kg/s)	5.9
Reactor rejection loop ΔT (K)	48
Reactor rejection loop mass flow (kg/s)	10.4
Radiator inlet temperature (K)	840
Average radiator surface temperature (K)	790
Radiation black body area (m^2)	94
Radiation physical area (m^2)	104
Thermopile area (m^2)	5.5
Thermoelectric leg length (mm)	5.5
Gross power generated (kWe)	105.3
Subsystem Mass (kg)	
Reactor	650
Shield	890
Primary heat transport	540
Reactor instrumentation and controls	380
Power conversion	530
Heat rejection	960
Power conditioning, control and distribution	400
Mechanical/structural	250
Total	4,600

Performance and Scaleability

SP-100 is designed as a set of building blocks. Thus, the reactor power level can be varied by changing the number of fuel pins, changing the number of thermoelectric converters, or using different power conversion units. The radiator area can be adjusted according to the power level. This flexibility was one of the major reasons for selecting the concept in 1985. The mass of the system is a function of the various combinations selected. Concepts of 8, 10, 20, 30 40, 50 100, 200, 300, 1,000,

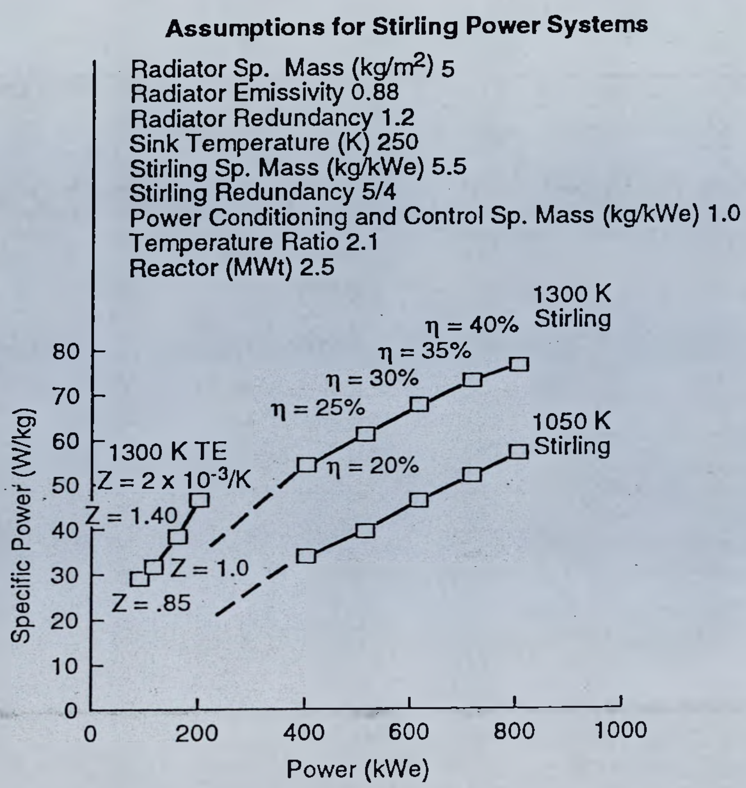


FIGURE 9. Extending SP-100 Reactor Power Systems Capability

5,000, 10,000 and 15,000 kWe have been configured. Figure 9 shows the specific power difference as a function of thermoelectric figure-of-merit and using Stirling engines at various efficiencies and peak operating temperatures. The same 2.5-MWt reactor, used in conjunction with thermoelectric conversion to generate 100 kWe, can be coupled to the Stirling conversion system to generate nearly 600 kWe.

Nearer term options are given in Figures 10 and 11 (Schmitz et al. 1992). The mass is shown to be a trade-off at the design thermoelectric conditions with Brayton or Stirling conversion systems. The Brayton conversion work was performed in the 1960s where 46,000 h of test data were accumulated on the rotating machinery. More details on the status of Brayton technology is given in Dudenhoefer et al. in this volume. The Brayton and Stirling conversion systems require much larger radiators than the thermoelectric conversion systems.

The SP-100 reactor can be reconfigured for higher thermal power levels and mated with a high-temperature Rankine cycle to provide even higher powers, such as several megawatts for nuclear electric propulsion. Also, SP-100 can be configured in a number of arrangements for lunar and Mars surface applications.

Development Status

Since 1985, substantial progress has been made in all key technology areas, and early feasibility issues have been resolved (Josloff et al. 1992a and Truscello and Rutger 1992). Fuel development includes the fabrication of the fuel pins and testing in an environment sufficient to develop high confidence of meeting the 7-y full-power operational lifetime requirement. The fabrication processes necessary to produce high

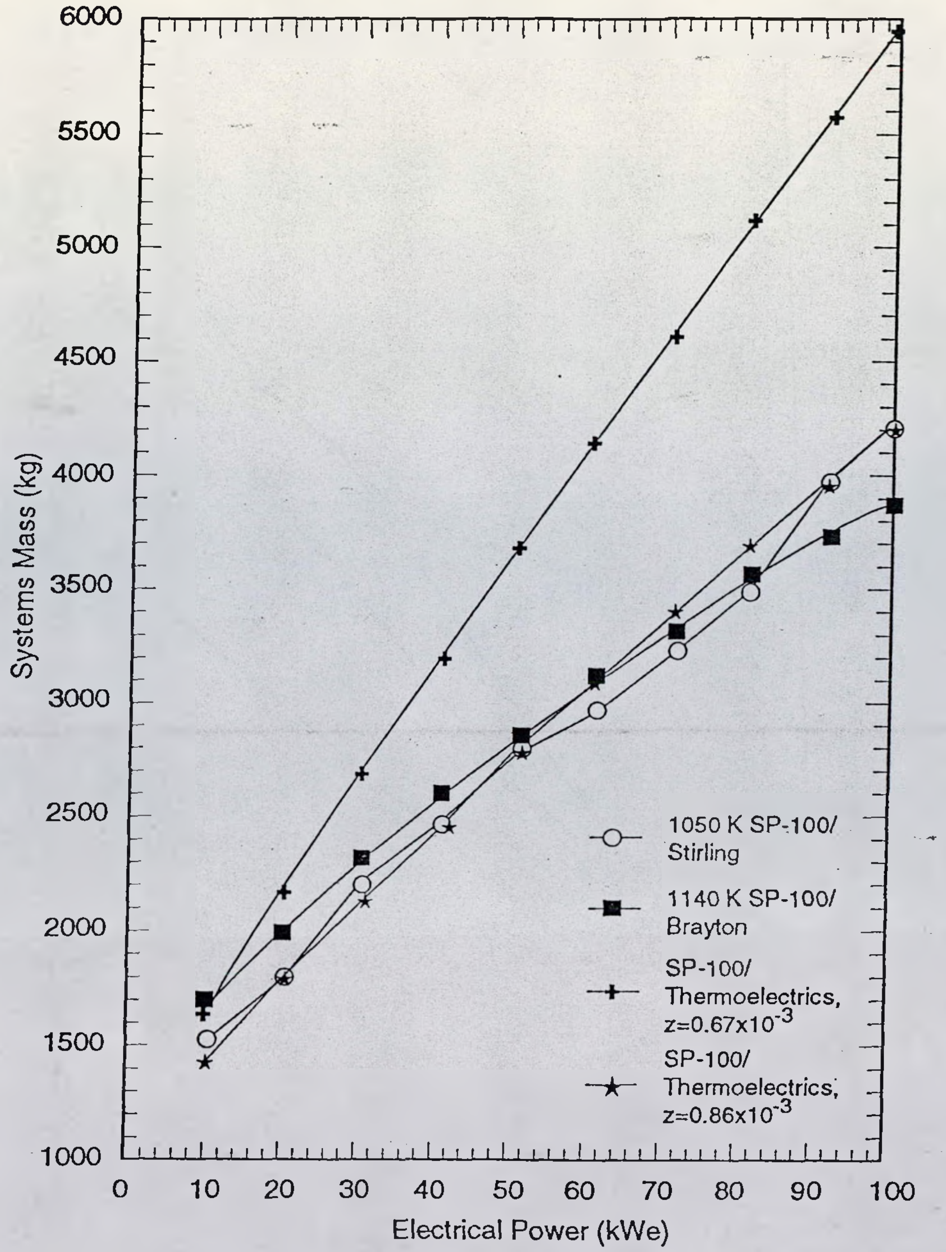


FIGURE 10. SP-100 System Mass Comparisons For Near Term Options (Schmitz et al. 1992).

quality UN fuel pellets have been successfully developed and the fuel irradiated in the Experimental Breeder Reactor II (EBR-II) and Fast Flux Test Facility (FFTF) test reactors. Approximately 75 fuel pins have been tested with some tests performed at three times the nominal power and at fuel pin surface temperatures as high as 1,500 K. All tests have met or exceeded expectations, and the goal of 6 atom-percent fuel burnup has been achieved. The pins tested used the Nb-1% Zr alloy cladding because PWC-11 was not yet qualified. This alloy is predicted to provide similar design margins, but at less mass. The tests show the fuel pins can meet the requirements of a 7-y system. Fuel pins testing with PWC-11 is planned to be completed in 1998 (Truscello and Rutger 1992).

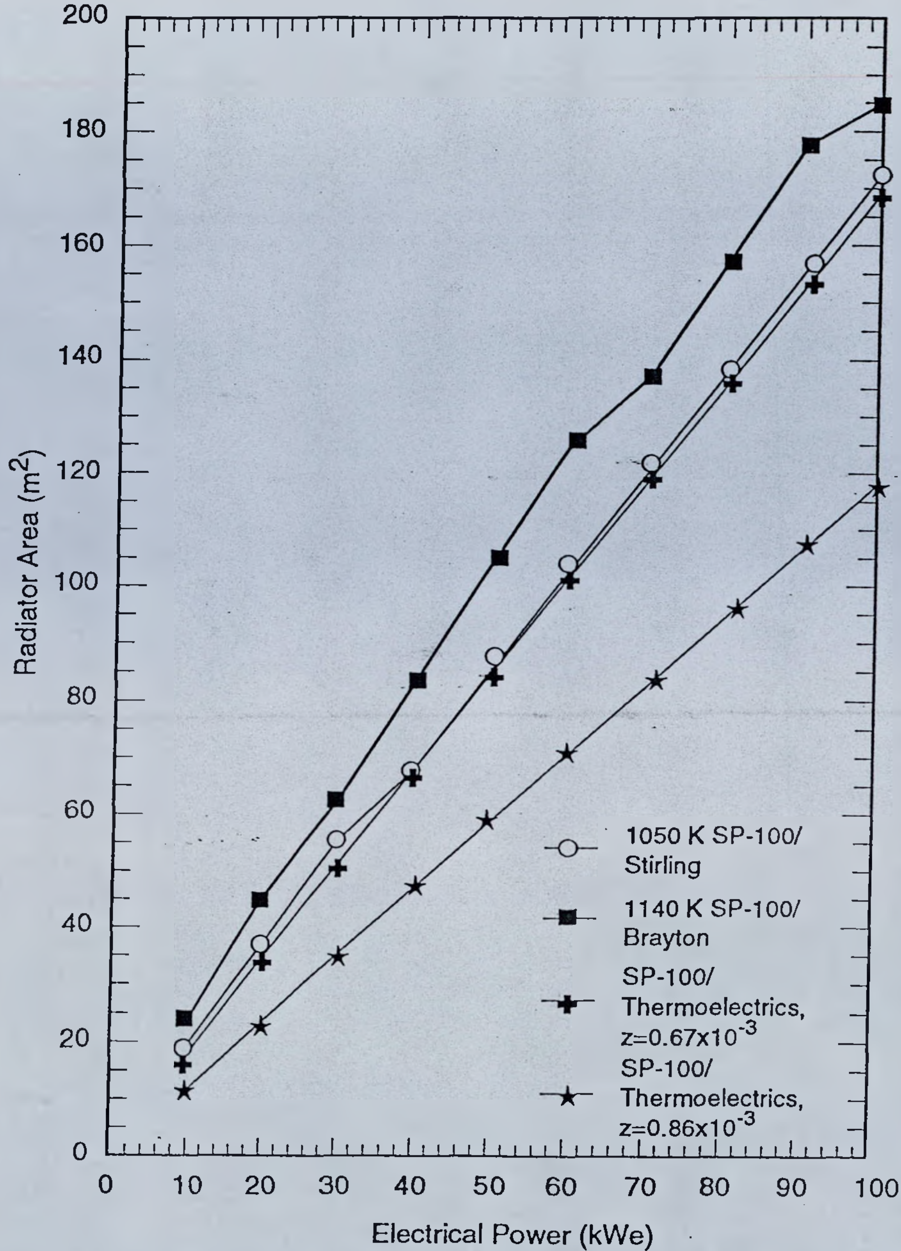


FIGURE 11. SP-100 Radiator Area Comparisons For Near Term Options (Schmitz et al. 1992).

A dual material tube has been qualified with an outer tube of niobium alloy and an inner tube of rhenium for strength as well as a barrier between the fuel and niobium alloy. Tubes 0.6 m long have been tested that show the bond is sufficiently rugged for 10 y of operation. Approximately 50 tubes have been fabricated with Nb-1% Zr cladding and another 10 were produced with PWC-11 cladding material.

The nearly 50,000 fuel pellets needed for a reactor core have been fabricated and are in storage awaiting the fabrication of the cladding. Production of the fuel pins for a reactor is being delayed until they are needed.

Materials are fundamental to the success of the development of a reactor that must operate for 7 y with a maximum coolant outlet temperature of 1,375 K. Sufficient materials data did not exist on the alloys of choice at the start of the program. Now, suppliers have been qualified for refractory alloys fabrication including Nb-1% Zr, PWC-11, and rhenium. Fabrication procedures include electron beam and gas tungsten arc welding, cold forging, drawing, hot isostatic pressing, diffusion bonding, vacuum sputtering, chemical vapor deposition, and high-temperature heat treatment. Compatibility testing with lithium has been performed in Nb-1% Zr and PWC-11 test loops at 1,350 K for thousands of hours without failures (Josloff et al. 1992b).

Reactor physics behavior has been experimentally verified in critical assembly tests. Measurements performed on the assembly showed close agreement with predictions under both normal design and postulated accident conditions. These tests provide assurance that the reactor will meet safety criteria imposed to protect the environment. From Mondt (1992), the experimental results verify that: (1) the internal shutdown rod reactivity met the shutdown requirement with margin, (2) reflector control worth versus position confirmed necessary control margin, (3) flooding reactivity worth was confirmed and subcriticality assured, and (4) buried reactor reactivity worth was confirmed and subcriticality was assured.

Control and safety drives are the only mechanical moving parts in the SP-100 power plant design. Low-temperature development tests have been completed to confirm the key mechanical design features. Environmental tests of control drive motor, clutch, and brake assemblies have demonstrated predicted performance of these components at 700 K. The self-aligning bearings of the reflector control drives have been successfully tested for 30,000 cycles at temperatures up to 1,170 K, and tests of safety rod bearings are underway at 1,570 K. Tribological coatings necessary to protect against self-welding, friction, and wear have been tested using refractory borides, carbides, and nitrides. These must operate at temperatures >1,600 K. Test data at 1,700 K indicate refractory carbides are the best material. Accelerated testing has been completed demonstrating that the equivalent life at SP-100 conditions is 50 y.

Long-life temperature and pressure sensors are needed to measure lithium coolant conditions. Temperature sensor concepts using a Johnson Noise Thermometer and W/Re thermocouples have been developed and fabricated units found to be mechanically robust. Sensor lifetime, accuracy, and stability are presently being established in a series of tests. In addition, the key features of a pressure transducer hydraulically coupled to the lithium coolant have been tested.

Multiplexers are located near the reactor where the nuclear radiation and temperature environments are severe (1.2×10^8 rad gamma, 1.6×10^{15} n/cm^2, and 800 K over 10 y). The temperature will be reduced to 375 K by the use of insulating blankets and radiators. Gamma testing indicates that radiation damage annealing will prevent unacceptable levels of drift in the circuit. Neutron testing is still underway.

Shadow shielding is used to attenuate the neutron and gamma radiation to acceptable levels for the spacecraft. Characterization of the LiH shield material for thermal conductivity and expansion, material compatibility experiments to confirm the long term behavior of the materials, and irradiation of LiH to establish swelling rates indicate no long-term issues with the current design. This includes irradiation testing at temperature. Material compatibility testing showed that LiH is not compatible with beryllium (Be). Therefore, the shield now has a stainless steel barrier between the Li and Be (Mondt 1992).

Work has been done on the major issues associated with the heat transport subsystem. For the gas separator, a feasibility experiment using air and water indicates that the design is sound. Experiments of a prototypical gas separator with helium and lithium are planned. A magnetic bench test has been performed on the TEM pump to demonstrate the ability to accurately predict the three-dimensional magnetic field. An electromagnetic integrated pump test has been performed to verify the calculated pumping forces.

Power conversion major issues included: (1) thermoelectric material figure-of-merit, (2) bonding of the cell between the heat exchangers to accommodate critical problems of thermal stress and electrical isolation, and (3) electrical insulation at 1,350 K while sustaining a voltage gradient of 8,000 V/cm for a period of 7 y in a deoxidizing environment created by the proximity of molten lithium. The SiGe material used in radioisotope thermoelectric generators (RTGs) has been improved so that the figure-of-merit has increased from $0.65 \times 10^{-3}\ K^{-1}$ to $0.72 \times 10^{-3}\ K^{-1}$. The design goal of $0.85 \times 10^{-3}\ K^{-1}$ must still be achieved.

A major technical achievement has been the development of a compliant pad to connect the thermoelectric cell to the heat source and sink. This was accomplished by use of a brush-like design that carries heat conductively while absorbing mechanical stresses due to the large temperature difference across the cell. The pad fibers are coated with a thin film of yttria to prevent self welding. Pads have performed under prototypic conditions for thousands of hours satisfactorily. Tests are continuing to demonstrate lifetime performance.

High voltage insulators are positioned at the top and bottom of the thermoelectric cell. Single crystal Al_2O_3 insulators, equipped with oxygen permeation barriers made of molybdenum, have been developed that will maintain the necessary electrical isolation for more than twice the lifetime required.

Very high electrical conductivity is needed to interconnect the TE couples. A multilayer electrode consisting of tungsten or niobium sandwiched between graphite layers has been developed. The tungsten provides good intercouple conductivity and strength, while the graphite isolates the tungsten from the TE material with which it reacts. Initial experiments have been performed, indicating that low resistivity bonds can be achieved, but long-term stability has yet to be demonstrated.

The process for fabricating the thermoelectric converter heat exchanger with the integral headers has been successfully demonstrated.

Cells can now be routinely fabricated using validated processes. The development has progressed through three phases. These all use SiGe thermoelectric materials. Progress in each phase includes (Don Matteo, May 25, 1993):

- PD-1 first demonstrated the basic concept of conduction coupling of thermoelectric cells to their heat source and heat sink heat exchangers. The PD-1 cell contained certain features (such as low temperature braze) which prevented driving the cell to full prototypic temperature levels, and the test fixture limitations resulted in thermal inefficiencies which prevented the cell from reaching maximum potential power output. The PD-1 cell delivered 4.0 We at 500 K temperature change for prototypic conditions and 4.8 We for 545 K ΔT at peak power (Matteo et al. 1992).

- The PD-2 cell improvements allowed operation of the cell at prototypic temperatures (1335 K hot side), but still was constrained by certain thermal inefficiences. The PD-2 cell produced 4.0 We at 500 K ΔT prototypic conditions and 8.7 We peak power at 730 K ΔT.

- New analytical techniques (Bond et al. 1993) were developed and applied to design a "fracture safe" configuration. The TA cells are near prototypic. These were subjected to prototypic and higher thermal conditions. The measured cells were 8.8 We (verus 8.9 We predicted) at the 500 K ΔT prototypic temperature conditions and 13.7 We under 660 K ΔT peak power temperature conditions. The two TA cells tested had efficiencies of conduction coupling of 75% and 79%, respectively, almost identical to pretest predictions. Lifetime testing has now reached 3,670 h on one cell, and is continuing.

Only limited work has been performed on the heat rejection system. A half-dozen titanium heat pipes, with potassium as the working fluid, were fabricated and have been life tested. A 0.9-m section of radiator duct was fabricated and tested using a low melting point liquid-metal (Cerrobend) that substituted for the lithium to demonstrate the ability of the lithium in the radiator to thaw during startup. Test results show that actual thaw rate was twice as fast as predicted.

The early SP-100 development issues are summarized in Table 3 (Mondt 1991). These challenges have been mainly resolved. To meet the safety challenges, an anuxiliary coolant loop has been designed to maintain the fuel pin clad below 2,000 K. This maintains the fuel structural integrity for disposal. The disposal location will be in space either at high Earth orbit or at its planetary destination. Thermoelectric cell design issues were resolved with development of the compliance pad and validation of the electrical insulator and electrical contact resistance. The fuel pin design and performance has been confirmed in reactor radiation testing. Heat transport hermiticity has been demonstrated in a lithium loop test. The major elements of the pump, including the magnetics, were demonstrated in element testing. The nitrogen loss that could limit lifetime has been resolved in testing that verified the success of the rhemium liner in the fuel pin to contain the nitrogen. System mass continues to be a challenge. Currently, the mass is at 4,600 kg for the GFS design and 3,900 kg for outer planet design; the specification is 4,000 kg. The gas separator to remove helium from the lithium loop has been demonstrated in air/water tests in Earth's gravity with further tests planned using lithium/helium. Enriched ^7Li is being used to minimize helium generation. To meet the low cost, low mass radiator heat pipe challenge, a design has been develped and successful operated for limited periods of time. For the shield temperature control, the foreward LiH material was replaced with Be and B_4C to reduce the radiation dose by a factor of 200 and the temperature of the LiH to 700 K in the LiH. This resolved the shield temperature control challenge. Remaining challenges for flight readiness will be discussed under the Flight Readiness section.

Flight Safety

The key in the design of SP-100 has been safety; the nuclear safety requirements are given in Table 4 (General Electric Co. 1989a). Figures 12 and 13 summarize key safety features built into the SP-100 design (General Electric 1989b). The reactor is

TABLE 3. SP-100 Top Ten Challenges in FY 1987.

1. Safety
   - Core coolability with loss of coolant
   - Reactor control and safety drives
   - End of mission disposal
2. Thermoelectric cell technology
   - Electrical insulator development and performance
   - Electrical contact resistance
3. Fuel pin design and performance validation
   - Fuel pellet development and scheduled
   - Fuel clad liner development
   - Fuel pin clad creep strength
4. Thawing coolants
   - Startup from frozen lithium
5. Highly reliable heat transport loop
   - Hermetic
   - TEM pump development and performance
6. System lifetime
   - N_2 loss from fuel elements
7. System mass
   - Compaliance with specification
8. Gas accumulator/separator
   - Li^7 versus natural lithium
9. Heat pipe design and manufacture
   - Transient performance/rethaw
10. Radiation shield temperature control

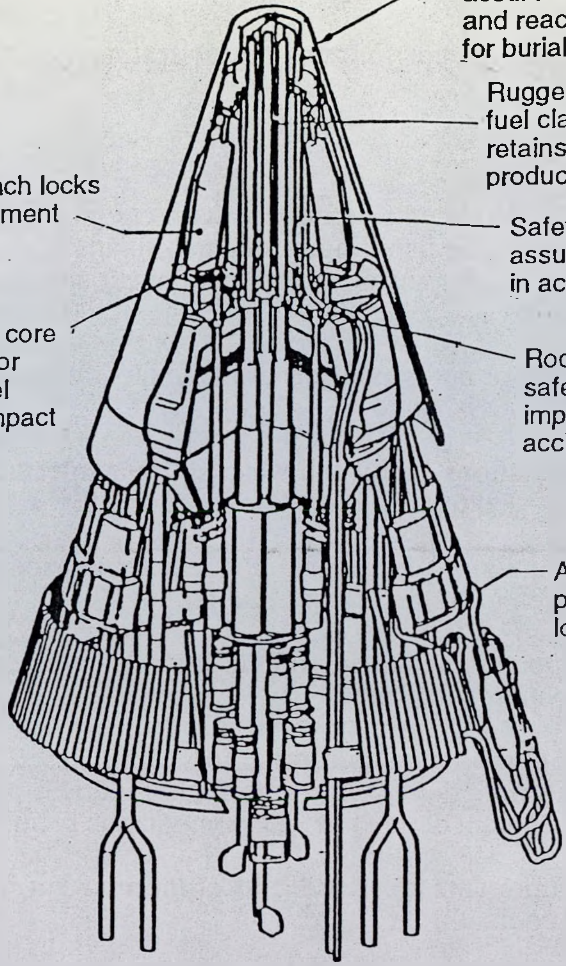


FIGURE 12. SP-100 Key Safety Features (General Electric 1989b).

designed to prevent inadvertent criticality during handling or in accident situations. This is accomplished by including two independent control elements that are physically locked in their shut down positions during ground transport, handling, launch, ascent, and final orbit acquisition (assuming Shuttle launch). These cannot be released until two independent signals are given. A large reactivity shutdown margin is provided by the safety rods to ensure that the reactor remains shutdown should any accident occur. This includes accidents involving severe fires, core compacting, projectile impacts, overpressure, and immersion/flooding environments. At the beginning of the mission, opening of any 7 of 12 radial reflectors or the insertion of any 1 of 3 safety rods will shut the reactor down. As the mission progresses, the number of reflectors required for shutdown reduces, becoming 3 after 7 y of full power operation. The reactor is designed with a prompt negative reactivity coefficient to ensure stable reactor control and enhance shutdown if a loss-of-coolant should occur.

TABLE 4. SP-100 Key Safety Requirements (General Electric 1989a).

---

1. Reactor Operation

Not started and operated (except for zero power testing) until operational orbit achieved.

2. Subcriticality

Remain subcritical to environments associated with credible failures or accidents during assembly, transportation, handling, prelaunch, launch, ascent, deployment, orbit acquisition, shutdown, and transfer to high permanent storage orbit. Minimum situations:

A. Core internal structure and vessel generally intact, all exterior components removed for:

a. All possible combinations of soil and water surrounding core.
b. Reactor vessel exposed to solid propellant fire for 1,000 s.

B. Core internal structure and vessel generally intact, compaction along the pitch line of the pins to produce pin-to-pin contact, and for:

a. All exterior components removed and all possible combinations of soil and water filling and surrounding the core.

b. Normal exterior components and reflectors compressed around the core; exterior absorber material, if any, in its normal shutdown position; and core containing its original coolant or any possible combination of soil and water.

c. All exterior components removed and aluminum surrounding the core containing its original coolant.

C. Reactor vessel as designed, core fuel pins spread radially apart to the maximum distance allowed within the fuel channel lattice design. All exterior components removed:

a. Water filing and surrounding the core.
b. Saturated soil filling and surrounding the core.
c. Dry soil filling and surrounding the core.

Calculated effective reactivities for these conditions, with margin for modeling and calculational uncertainties, shall be <0.98.

3. Response to Fires

Reactor, without reflector elements, neutron shield and reentry heat shield, remain subcritical in the liquid and solid propellant fire environments. Limited melting and creep deformation allowed, however, the as-built geometry essentially maintained.

TABLE 4. SP-100 Key Safety Requirements (General Electric 1989a) (continued).

4. Structural Response to Explosions

Reactor remain subcritical in launch vehicle explosion environments.

5. Reentry

For inadvertent reentry following reactor operation, reactor designed to reenter through the earth's atmosphere sufficiently intact to prevent the dispersion of fuel and fission products. Essentially intact defined as:

A. Reentry structural and thermal loads not cause loss of effective fuel/safety rod alignment.

B. If after reactor operation, the reactor structure remains sufficiently intact to allow effective burial on impact.

C. Reentry not breach the reactor vessel nor impair the predictability of its structural response on impact.

6. Burial

Intact reentry through the earth's atmosphere, the reactor capable of producing effective burial as it impacts on water, soil, or pavement-grade concrete. Effective burial defined that the fuel, reactor vessel, and internal components are within the formed impact crater and below normal grade level.

7. Transfer to Permanent Storage Orbit

Designed that reactor can be transferred to high permanent storage orbit at end-of-mission. Reactor designed to prevent such core disruption and structural degradation during and following operation that compromise structural integrity and predictability of desired reactor behavior during final shutdown and transfer to high permanent storage orbit.

8. Final Shutdown

Reactor designed to ensure high-confidence permanent subcriticality at the final shutdown to preclude further production of fission products and activated material and ensure subsequent reduction of radioactive inventory. Final shutdown activated automatically, irreversible, and not initiated or rendered inoperable by any credible single failure or initiating event.

9. Final Shutdown Clock

Final shutdown clock irreversibly interrupt the supply of power to all in-core safety rods and control reflector clutches when preset final shutdown time is reached.

TABLE 4. SP-100 Key Safety Requirements (General Electric 1989a) (continued).

10. Loss of Primary Coolant

Reactor designed to accommodate an instantaneous complete loss of main loop primary coolant followed by scram during operational phases without,

A. Exceeding fuel design cladding temperature limits.
B. Impairment of capability to achieve final shutdown.
C. Loss of structural integrity sufficient to impair the capability to boost to permanent storage orbit.

11. Core Heat Removal Capability

Coolability assured with high confidence for all credible accident conditions to maintain the structural integrity and thereby the predictability of desired reactor behavior during final shutdown and transfer to high permanent storage orbit.

12. Nuclear Feedback Calculations

Reactor core and associated coolant systems designed over entire power operating range, net effect of the prompt inherent nuclear feedback characteristics tends to compensate for the rapid increases in reactivity.

13. Power Oscillations

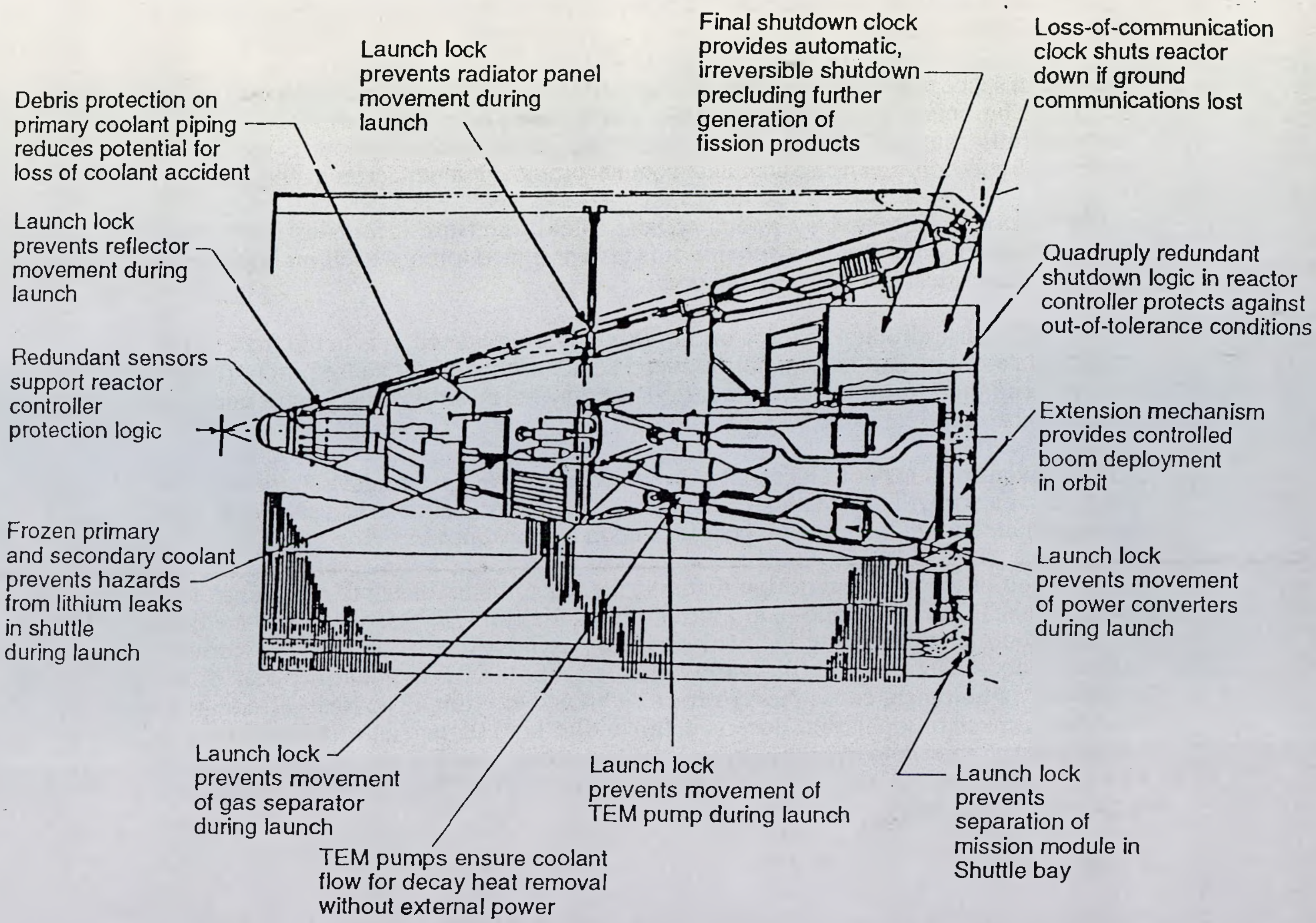
Nuclear subsystems designed to assure that thermal power oscillations which can result in conditions exceeding acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

14. Reactivity Control Redundancy

Two means of reactivity control provided. Suitable independence and diversity provided to assure adequate protection against common cause failures. Each of these means capable of performing its nuclear safety function with a single active failure.

15. Inhibits

Until operational orbit achieved, startup shall be precluded by three independent inhibits, one of which precludes startup by radio frequency energy.



Other features

- Encryption and decryption devices (in mission module) and specific command sequence guard against unauthorized startup and operation
- Inhibits on safety critical functions
- Monitoring of inhibits while in Shuttle

FIGURE 13. Additional SP-100 Safety Features (General Electric 1989b).

The reactor fuel pins include a rhenium poison that acts as thermal neutron absorption in case of water flooding.

While in space, the reactor is protected against impacts of micrometeorites and orbital debris by bumpers. If the debris would cause a loss-of-coolant, the reactor would automatically shut itself down. A loss-of-coolant auxiliary cooling loop is included in some configurations to ensure adequate core heat removal under accident conditions.

The reactor is protected by a reentry heat shield to ensure it remains intact if the reactor should reenter the atmosphere. It is also designed to bury itself on impact, and to remain subcritical if it falls into the ocean.

The SP-100 uses lithium as the working fluid. During launch, it is in the frozen state for added safety so that any launch-induced accident that might cause piping rupture will not endanger crew or equipment. Reactor energy is used for thaw once the operational altitude is reached.

In the event of a loss of electrical power to the control system during operation in space, the reactor will automatically shut down. Reflector elements and safety rods are spring actuated to their shutdown positions upon a loss of power.

As a result of the stringent design features, the reactor is predicted to offer much less radiation hazard to the public than transcontinental airline flights, diagnostic medical examinations, or therapeutic medical services. A mission risk analysis performed indicates no outstanding public safety issues. The analysis quantifies risk from accidental radiological consequences for a reference mission. The total mission risk based on expected population dose is estimated to be 0.05 person-rem based on a 1 mrem/y as the threshold for radiological consequences; this is a negligible amount in absolute terms and relative to the 1.5 billion person-rem/y that the world population experiences from natural radiation sources.

Flight Readiness

A summary of the technical status and challenges as seen in 1991 is given in Table 5 (Mondt 1991). The progress to date on each of the FY 91 technical challenges provides an assessment of flight readiness. Much of this information was prepared by J. Mondt (1993).

System Level

The power-verus-lifetime prediction codes are available and the verification of these codes is dependent on the completion of all component lifetime performance tests being conducted under the subsystems. With regards to verification of a 10-y system, all of the critical components, their failure modes, and the failure mechanisms have been identified. The failure mechanisms have been well defined and analyzed. The failures have been placed in three categories: i.e. 1) design margins determined to be adequate based on existing data, 2) addditional test data needed, which is included in the SP-100 planned effort, and 3) additional test data need which will be obtained during the flight development phase. System start-up and restart from frozen lithium in zero G has been conceptually designed based on component thaw tests and will be verified with system level ground tests. Flight system acceptance tests are still being defined.

TABLE 5. FY-91 Technical Challenges (Mondt 1991).

System Level

- Verified power versus lifetime prediction codes
- Verified reliable 10-y system-design margin codes
- Startup from frozen lithium in zero-gravity
- Flight system acceptance tests

Subsystems

1. Reactor

   - Verified prediction of fuel pin behavior
   - Verified 10-y creep strength of PWC-11
   - Verified transient behavior

2. Reactor Instrumentation and Controls

   - Reflector control drive actuator insulators and electromagnetic coil lifetime
   - Temperature sensors lifetime
   - Radiation hardened multiplexer amplifiers lifetime

3. Shield

   - Verified LiH swelling properties

4. Heat Transport Subsystem

   - Gas separator performance and plugging lifetime
   - TEM pump (TE/Busbar) bond performance and lifetime
   - TEM pump (Cu/Graphite Bus Duct) bond performance and lifetime

5. Converter Subsystem

   - Electrodes and bonds to TE legs lifetime
   - TE cell assembly low cost fabrication
   - High figure-of-merit TE material performance and lifetime ($Z = 0.85 \times 10^{-3}\ K^{-1}$)
   - Cell to heat exchanger bond performance and lifetime

6. Heat Rejection Subsystem

   - Carbon-carbon to titanium bond performance and lifetime
   - Low cost and low mass heat pipe lifetime

Reactor Subsystem

The fuel pin swelling and fission gas release predictions have been verified up to the 10-y mission lifetime (6% burnup) based on inpile accelerated fuel tests. The 10-y creep strength of Nb-1Zr has been verified by uniaxial and biaxial long term creep test. The very high creep strength of PWC-11 is dependent on the material processing and requires the results of existing long term creep tests, which are now scheduled to be complete in FY 98. The reactor transient behavior will be verified by the first operating SP-100 reactor. The SP-100 reactor component development is complete and technology ready for a flight system.

Reactor Instrumentation And Controls Subsystem

The actuator insulator and EM coil lifetime tests and analyses verify that these components will operate for the 10-y mission lifetime. The temperature sensors are verified for 5-y missions. The lifetime of the multiplexers in a very high radiation field still needs to be verified.

Shield Subsystem

The LiH swelling properties as a function of temperature, radiation dose, and radiation dose rate are verified. The SP-100 shield development is complete and technology ready for a flight system.

Heat Transport Subsystem

The gas separator performance and plugging is scheduled to be verified in FY 94 in a flowing lithium test. The TEM pump bond performance has been verified and the pump lifetime is primarily dependent on the TE cell lifetime. The pump TE cell performance is scheduled to be verified in FY93 and the lifetime by the end of FY98.

Converter Subsystem

The TE cell electrodes and bonds have been developed and incorporated into a prototype TE cell. Three of these prototype TE cells are now on test at design conditions, with one operated for 3,670 h, another for 3,100 h and one just started (from a telephone converstion with D. Matteo of Martin Marietta on 25 May 1993). The low cost fabrication of TE Cells has been factored into the design and will be verified in FY 98 with the manufacture of a large number of cells. The present converter manufacturing process uses hot isostatic pressure bonding of niobium-to-niobium for the cell-to-heat-exchangers bond. Small scale fabrication has shown this bonding to be very successful The high figure-of-merit TE material is progressing and should be available by the end of FY 98.

Heat Rejection Subsystem

An intregral carbon-carbon tube and fin has been developed and successfully bonded to a thin wall (0.085 mm) Nb-1Zr potassium heat pipe, 25 mm diameter by 376 mm long and operated as a radiator heat pipe. Since the Nb-1Zr tubes can be manufactured and are leak tight with such a thin wall it may not be necessary to go to a titanium liner. A high conductivity integral carbon-carbon tube and fin 25 mm diameter by 1 m long has also been fabricated. Nb-1Zr potassium screen-wick heat pipes have been tested for long times (>10,000 hr) with no failures or degration. The heat rejection FY 91 technical challenges are nearly resolved.

Design Growth—Stirling Engine Development

As shown in Figure 9, Stirling engines offer an attractive power conversion system in the hundreds of kWe when mated with an SP-100 reactor. For instance, a 600 kWe power plant using Stirling engines that operate at 1,050 K has a specific power over 45 W/kg. Therefore, in 1985, it was decided to develop Stirling engine technology along with thermoelectric power conversion for SP-100. The Stirling engine development program (Slaby 1987) is based on a free-piston design that features only two moving parts (displacer and power piston), close clearance, noncontacting seals (no wear of mating parts), hydrostatic gas bearings for dynamic members (no surface contact of dynamic components and no oil lubrication necessary), dynamic balancing, and the potential for a hermetically sealed power module. The free-piston Stirling concept utilizes gas springs, which have hysteresis losses.

The program started with a 650-K Space Power Demonstrator Engine (SPDE) technology development and is proceeding with the development of common designs for 1,050-K and 1,300-K (Dudenhoefer 1990). The 1,050-K engines provide the means to demonstrate that the engine technology issues have been solved using easier-to-work-with superalloy materials before demonstration of the refractory or ceramic materials version. The plan is shown in Figure 14 with the goals and specifications for the 1,050-K design in Table 6. A schematic of the 1,050-K engine is shown in Figure 15.

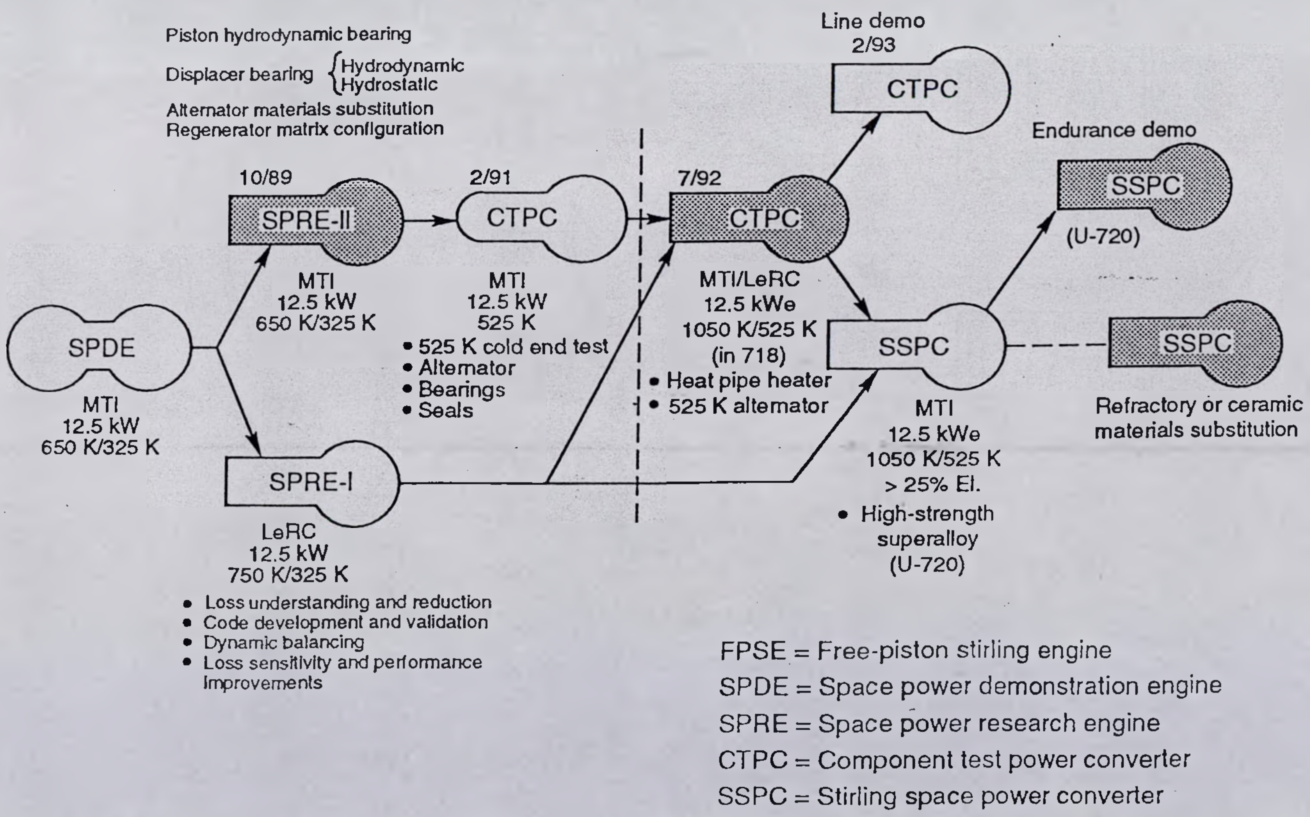


FIGURE 14. Evolution of a High Temperature Stirling Engine (Dudenhoefer 1990 and Dudenhoefer and Winter 1991).

TABLE 6. 1,050 K Stirling Space Engine Goals and Specifications.

Balanced opposed configuration total power output (kWe)	50
End of Lift power (kWe/piston)	25
Efficiency (percent)	>25
Life (h)	60,000
Hot side interface	Heat Pipe
Heater temperature (K)	1,050
Cooler temperature (K)	525
Vibration - casing peak-peak (mm)	<0.04
Bearings	Gas
Specific mass (kg/kWe)	<6.0
Frequency (Hz)	70
Pressure (MPa)	15

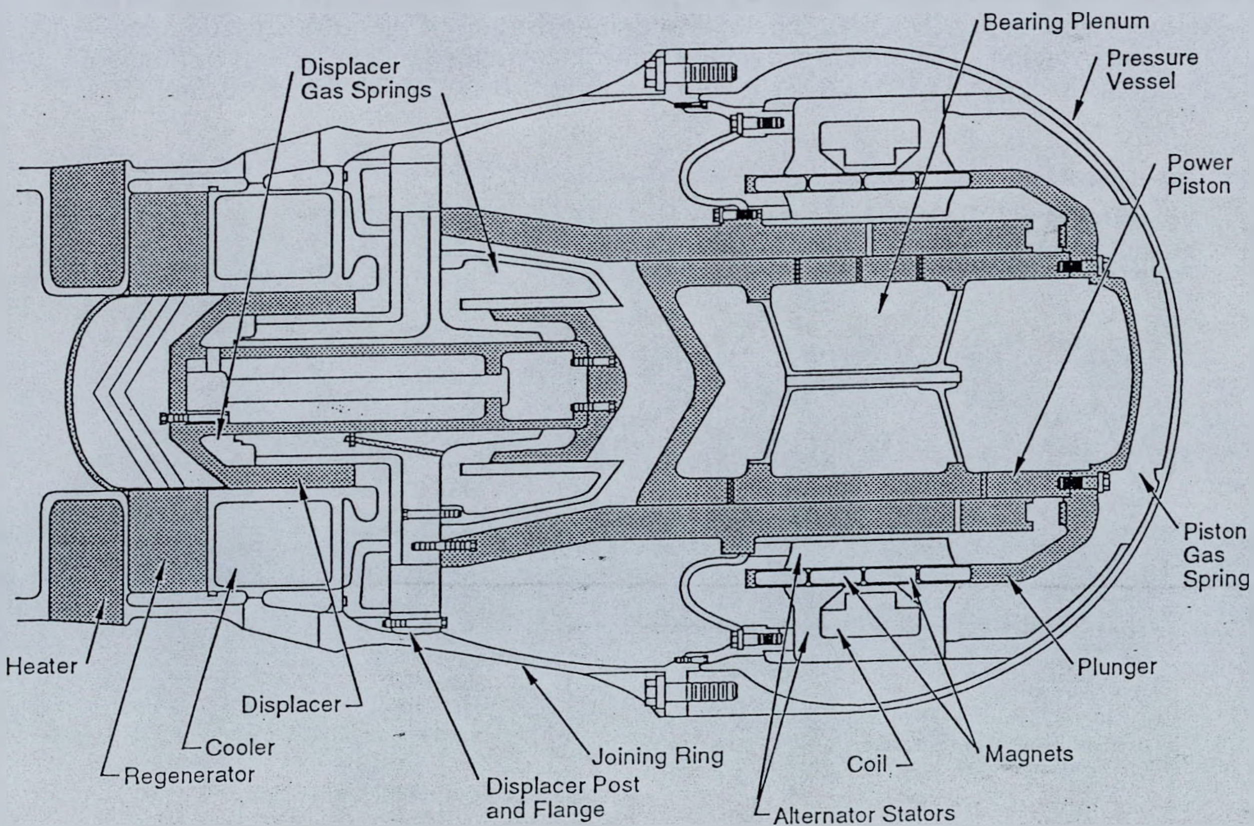


FIGURE 15. Preliminary Design of 1,050 K Stirling Space Power Converter (Dudenhoefer 1990).

In October 1986, the 650-K SPDE demonstrated 25 kWe (Dudenhoefer and Winter 1991). The SPDE was a dual-opposed configuration consisting of two 12.5-kWe converters. After this successful demonstration, the engine was cut in half to serve as test beds for evaluating key technology areas and components, now called Space Power Research Engines (SPRE). The electrical output has been measured as 11.2 kWe at overall efficiency of about 19%. The goal is 12.5 kWe and efficiency greater than 20%. Sensitivity of the engine performance to the displacer seal clearance and the effects of varying the piston centering port area are under study.

The Component Test Power Converter (CTPC) is a 12.5 kWe cylinder technology engine for the Stirling Space Power Converter (SSPC). Inconel 718 is being used as the heater head material to permit early testing for short terms (100-1,000 h at 1,050 K). This testing has demonstrated 12.5 kWe at a 1,050 K operation temperature and greater than the 20% goal. A heater for the SSPC of Udimet 40 L1 superalloy that will have a design life of 60,000 h still needs to be fabricated. The CTPC is being used to evaluate critical technologies identified as: bearings, materials, coatings, linear alternators, mechanical and structural issues, and heat pipes. The impact of temperature on close-clearance seal and bearing surfaces in the cold end of the power converter has been tested, with no problems observed. The CTPC linear alternator uses an alternator that can reach a peak temperature of 575 K, close to the upper operating limits of samarium cobalt magnets. Test results indicate sufficient design margins (Dudenhoefer et al. 1992).

Endurance testing is underway on a 2-kWe free-piston Stirling engine called EM-2 that operates at a heater temperature of 1,033 K. At the end of 5,385 h, only minor scratches were discovered due to the 262 dry starts/stops, and no debris was generated. The heater head of Stirling power conversion systems is the major design challenge because heater head creep is predicted to be the life-limiting mechanism for Stirling engines. The difficulty in creep analysis stems from inadequate knowledge of elevated temperature material behavior and inadequate knowledge of inelastic analysis techniques. Typically, the high operating temperatures and long operating periods of Stirling engines are taxing the ultimate capabilities of even the strongest superalloys.

THERMIONIC DEVELOPMENT PROGRAMS

U.S.S.R. Space Reactors

Since 1967, the U.S.S.R. has orbited approximately 33 thermoelectric reactor power systems as a power source for ocean surveillance radars in satellites called RORSAT. This includes nine in the decade starting in 1983, with the last one on March 14, 1988. Power levels ranged from several hundred watts to a few kilowatts. Limited information is available on the details of the RORSAT power system. We know that the RORSAT power systems are fast reactors using SiGe thermoelectric conversion system. The general characteristics (Bennett 1989) are summarized in Table 7.

In 1987–1988, the U.S.S.R. tested a different type of reactor power system using thermionic power conversion. Two space tests were performed, with one operating six months (Cosmos 1818) and the other operating 346 days (Cosmos 1867). These power plants are designated in the U.S. as Topaz I. Topaz I design output is 10 kWe. The flight-tested units used a multicell thermionic fuel element with an output power of approximately 5 kWe, one with a molybenum emitter and the other with a tungsten emitter. The power system with the tungsten emitter operated for the longer period of time; degradation of performance occurred, with the thermal power increased to compensate for this degradation.

TABLE 7. RORSAT Power System (Bennett 1989).

Thermal Power (kWt)	≤100
Conversion System	Thermoelectric
Electrical Power Output (kWe)	≤5 (~1.3 to 2)
Fuel Material	U-Mo (≥3wt% Mo)
Uranium-235 Enrichment (%)	90
Uranium-235 Mass (kg)	≤ 31 (~20 to 25)
Burnup (fissions/g of U)	$\leq 2 \times 10^{18}$
Specific Fuel Thermal Power (W/g of U)	~5
Core Arrangement	37 cylindrical elements (probably 20 mm dia)
Cladding	Possibly Nb or stainless steel
Coolant	NaK
Coolant Temperature Outlet (K)	≥970
Core Structural Material	Steel
Reflector Material	Be (6 cylindrical rods)
Reflector Thickness (m)	0.1
Neutron Spectrum	Fast (~ 1 MeV)
Shield	LiH (W and depleted U)
Core Diameter (m)	≤0.24
Core Length (m)	≤0.64
Control Elements	6 in/out control rods composed of B_4C with LiH inserts to prevent neutron streaming and Be followers to serve as the radial reflector
Overall Reactor Mass (kg)	<390

A schematic of the fuel element (Bennett 1989), shown in Figure 16, portrays the fuel element as divided into five fuel cells. Urania fuel is used, the cathodes are made from a tungsten or molybdenum alloy, anodes are made from the niobium alloy, insulators are beryllia, outer casings are stainless steel, and cesium vapor is used in the interelectrode gap. A cutaway of the Topaz I reactor showing the principal subsystems and design features is shown in Figure 17. During operation, some hydrogen leaks from the moderator; this is continuously removed by the cesium. During a year's operation about a kilogram of cesium is used.

A second form of thermionic reactor, called Topaz II, has been purchased by the U.S. for testing and evaluation. Topaz II also has a 6 kWe design output, but a single thermionic cell fuel element has replaced the multicell fuel element. A significant advantage of Topaz II is the ability to test the entire system at full temperature in an electrically heated configuration (Nicitin et al. 1992). The single-cell fuel element makes this possible.

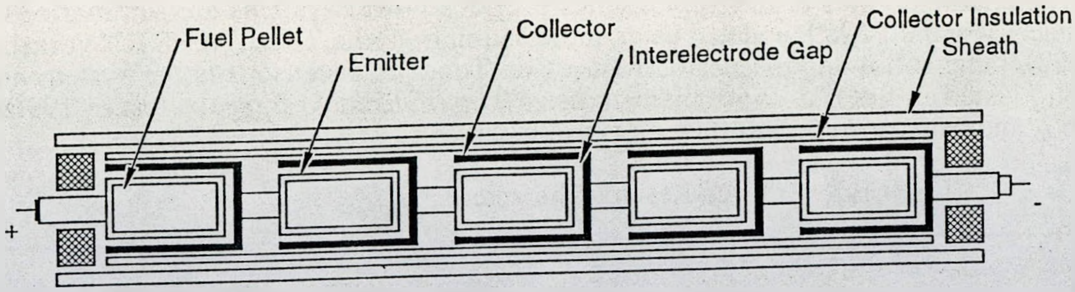


FIGURE 16. Basic Arrangement of the Multicell TOPAZ Thermionic Fuel Element (TFE) (Bennett 1989).

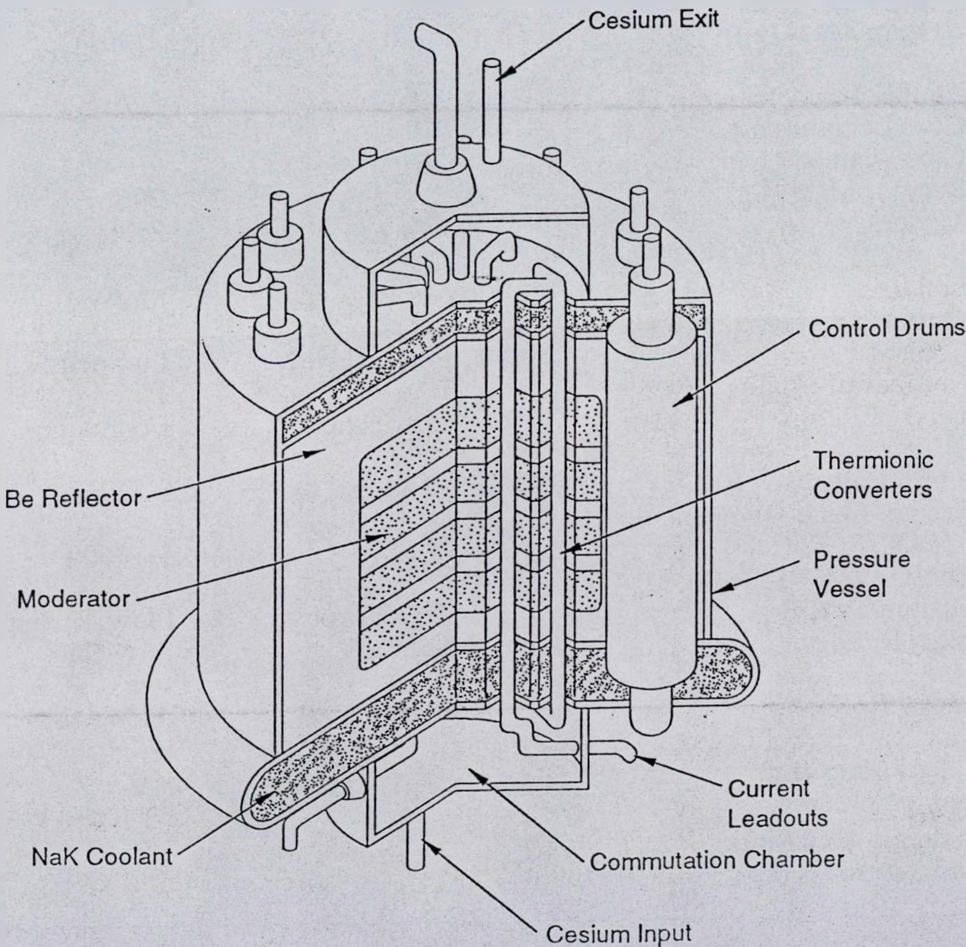


FIGURE 17. Configuration of the TOPAZ I Reactor (Bennett 1989).

The characteristics of the Topaz series of space power systems are summarized in Table 8 (Bennett 1989, updated using information from the TSET/NEPSTP Workshop 1992). The following discussion focuses on Topaz II because it is the system now being tested in the U.S., with plans to use a Topaz II in a U.S. space test in 1996 or 1997, and because more information is available.

TABLE 8. TOPAZ Reactor Characteristics.

	TOPAZ I	TOPAZ II
Electrical Power (kWe)	5 to 6	6
Voltage at Lead (V)	5 to 30	28 to 30
Thermal Power (BOL/EOL)(kWt)		115/135
Number of TFEs	79	37
Cells/TFE	5	1
System Mass (kg)		1,061
Emitter Diameter (cm)	1.0	1.73
Core Length (cm)	30	37.5
Core Diameter (cm)	26	26
Reactor Mass (kg)		290
Moderator	$ZrH_{1.8}$	$ZrH_{1.85}$
Emitter	Mo/W	Mo/W
Emitter Temperature (K)	1,773	1,800 to 2,100
Coolant	Pumped NaK	Pumped NaK
Number of Pumps	1	1
Type of Pump	Conduction	Conduction
Pump Power	19 TFEs	3 TFEs
Reactor Outlet Temperature (BOL/EOL)(K)	773/873	560/600
Coolant Flow Rate (kg/s)		1.5
Cesium Supply	Flow Through	Flow Through
Axial Reflector	Be Metal	Be Metal
Radial Reflector	Be Metal	Be Metal
Number Control Drums	12	12
Shield Mass (kg)		390
Radiator		Tube and Fin
Radiator Area (m^2)		7.2
Radiator mass (kg)		50

Topaz II Description

A schematic of Topaz II is shown in Figure 18. At the beginning of life, the reactor produces approximately 115 kWt with a conversion efficiency of 5.2%; and at the end of life, the reactor produces 135 kWt with a conversion efficiency of 4.4%. The Topaz

II is cooled by a liquid metal of eutectic sodium-79% potassium-21% (NaK) that remains liquid during all phases of the Topaz II lifetime, excluding the end-of-mission shutdown. The NaK coolant removes the waste heat from the reactor and transports it to the radiator, where it is rejected to space.

The Topaz II core is a right circular cylinder 260 mm in diameter and 375 mm high (Space Power Inc. 1990). Thirty-seven cylindrical thermionic fuel elements (TFEs) are arranged in an approximately triangular pitch within a block of moderator. The

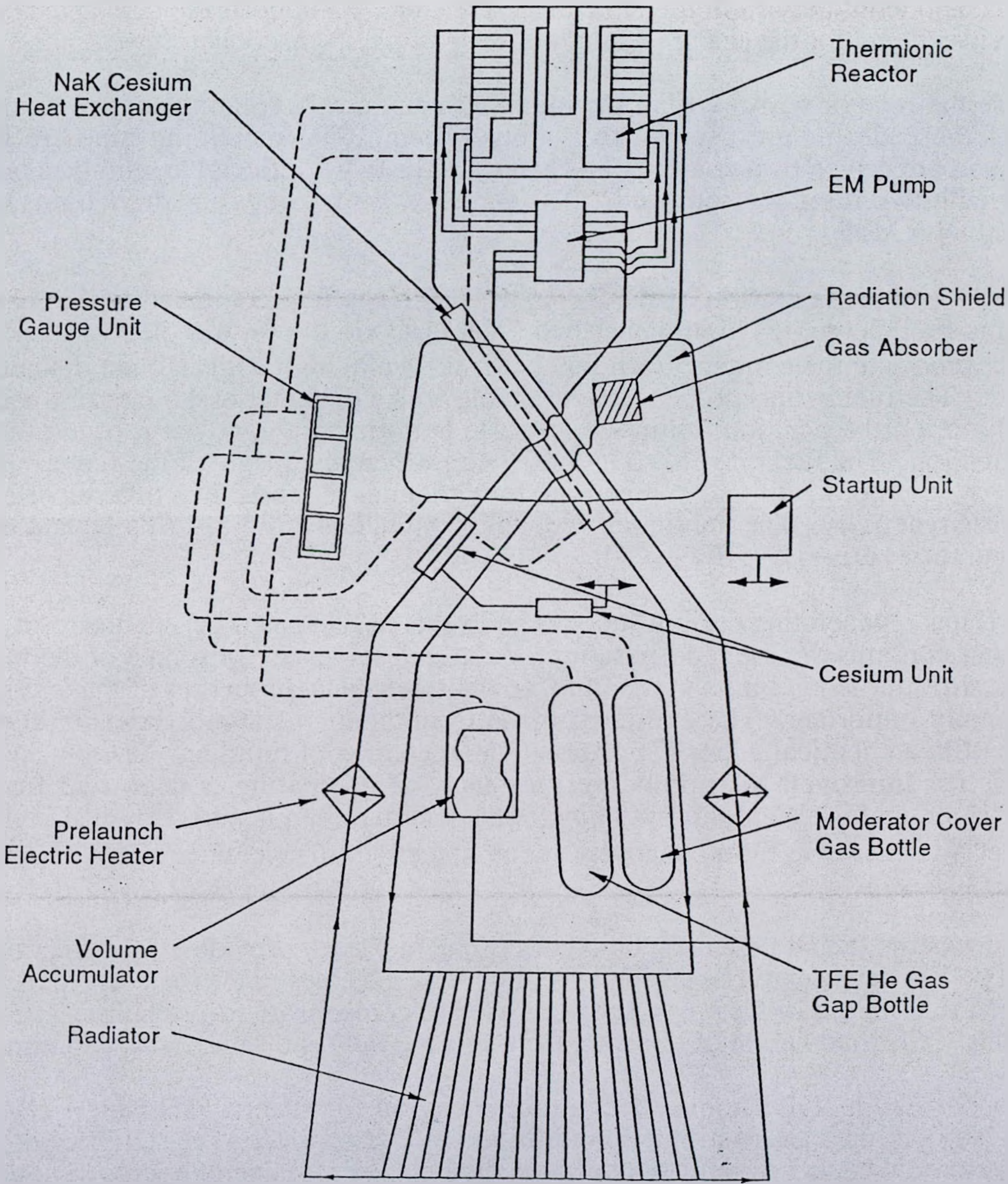


FIGURE 18. TOPAZ II Schematic (Topaz CoDR 1992).

moderator is epsilon phase zirconium hydride (ZrH_{2-x}) with hydrogen stoichiometry in excess of 1.8. The moderator is canned in a stainless steel calandria that has 37 circular channels in it to accommodate the TFEs and NaK coolant. The coolant channel gap between the TFE outer sheath and the calandria wall is a grooved surface.

A gas mixture of approximately 50% CO_2, 50% helium, and other trace gases is maintained within the moderator/axial reflector region to help inhibit the release of hydrogen from the $ZrH_{1.85}$, and to increase the heat transfer from the $ZrH_{1.85}$ to the outer surface of the vessel and to the coolant channels.

The thin-walled, stainless steel cylinder reactor vessel encloses the TFEs, moderator calandria, and axial beryllium metal neutron reflectors. It supports the core and TFEs and provides plena for the cesium vapor, helium gas, and NaK coolant.

Outside the reactor vessel within the radial reflector are 12 rotating control drums. Three of these drums are used in the safety system. The remaining nine are for control, and are driven by a common mechanism. The radial reflector assembly is held together with two fused tension bands. The control drum bearings and drive trains are lubricated with MoS_2.

Figure 19 is a cross section of the Topaz II TFE. The fuel contained within each thermionic fuel element is highly enriched (96%) UO_2 in the form of pellets stacked within the cavity of the emitter. Each fuel pellet is ~8 mm high with an outer diameter of 17 mm. The fuel pellets possess a central hole with a diameter of 4.5 mm or 8 mm, depending upon the position within the core, to help flatten the power profile in the radial direction. The fuel height is 355 to 375 mm, where the height of the fuel can be varied at the time of loading to compensate for variations in fabrication that can affect core excess reactivity. The maximum fuel temperature is ~1,775 to 1,925 K, and the end temperatures are ~1,575 K.

Topaz II has regenerating cesium supplies and vents fission gases outside the reactor. The emitter contains the fuel and fission products and serves as the source of thermal electrons. Emitter strain limits system life, so the mechanical properties of the emitter are extremely important. The emitter is made from monocrystal molybdenum alloy substrate with a chemical vapor deposition (CVD) coating of tungsten. The tungsten coating is for improved thermionic performance. This coating is deposited from chloride vapor, and is also monocrystalline. The outer tungsten layer is enriched in the isotope ^{184}W in order to limit the adverse neutronic effect. The emitter temperature is 1,873 K.

Beryllium oxide (BeO) pellets on both ends of the fuel stack provide axial reflection. The BeO pellets have central holes that match up with the holes in the fuel. The pellets are stacked to a height of 55 mm. They are used to compensate for variations in the fuel loading. The total height of the core, including the BeO end reflector, is 485 mm.

The monocrystal molybdenum collector tube is coaxial with the fueled emitter. High collector temperatures are desirable to reduce radiator size, while lower temperatures reduce thermionic back-emission and keep the dissociation pressure of hydrogen in the moderator within bounds. A temperature of 925 K is used at the outlet of the reactor; the temperature is about 100 K lower at the inlet.

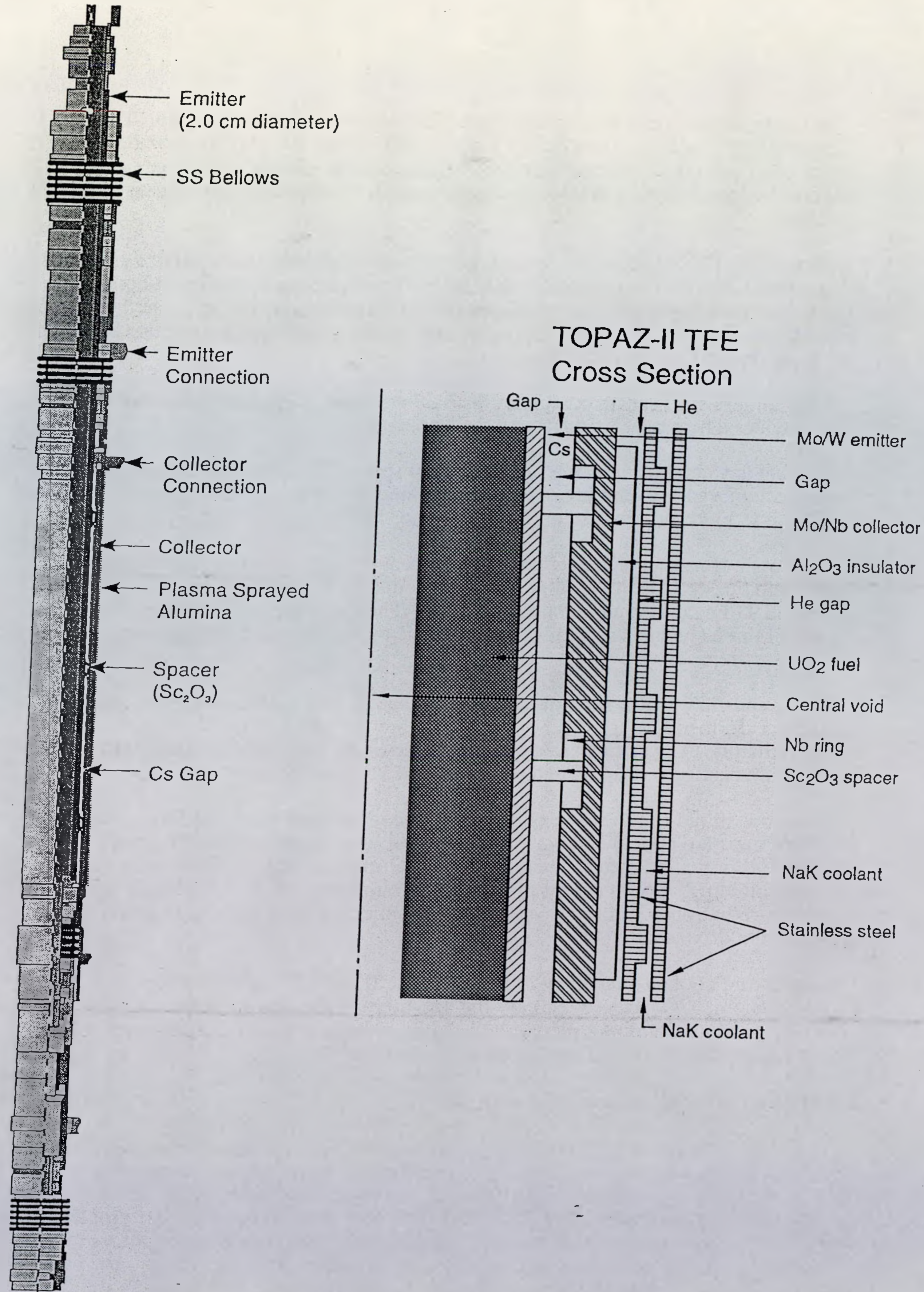


FIGURE 19. TOPAZ II Thermionic Fuel Element Cross Section.

Between the collector and emitter is the interelectrode thermionic gap. This gap is 0.5 mm (U.S. Topaz II Flight Safety Team 1992). There are scandium-oxide (Sc_2O_3) spacers between the emitter and collector surfaces in the interelectrode gap to prevent the emitter from shorting to the collector as a result of emitter distortion caused by fuel swelling.

External to the collector is a helium gap between the collector insulator and inside diameter of the inner coolant tube. The helium provides a good thermal bond for the transfer of heat to the coolant, while maintaining electric insulation of the thermionic fuel element. A helium bottle is located in the radiator region that maintains helium in the gap over the lifetime of the system.

The stainless steel sheath completes the TFE structure. The sheath supports the TFE and provides a heat transfer surface to the NaK coolant.

The fuel is vented from its interior directly to space. The cesium supply is a still that feeds cesium vapor to the interelectrode gap and vents any gases to space. Cesium venting is 0.5 g/d for this system (Marshall et al. 1993).

The Topaz II uses eutectic NaK to remove heat from the reactor. The NaK must be kept liquid during launch. A single DC conduction pump powered by a group of three dedicated TFEs connected in parallel is used. The coolant piping is divided into two groups of three channels as it passes through the pump. An on-board current source is used for startup.

A tube-and-fin radiator is employed in the shape of a truncated cone. The surface of this cone is formed from steel tubes welded to circular manifolds at the top and bottom of the radiator. Copper fins are welded to these tubes. To improve emissivity of the fins, a glass coating is used that adheres well and has good thermal resistance.

A shadow shield in the shape of a truncated cone is used for radiation attenuation. Both end caps are concave downward, spherical, and thick walled. The sides of the shield are thin-walled steel. The space between the end caps is filled with LiH for neutron shielding. Four coolant pipes pass through the shield at angles designed to minimize radiation streaming. A stepped channel through the shield contains the control drum drive shaft.

The gas and coolant systems within the Topaz II include the NaK coolant, cesium supply system, CO_2/He cover gas, the He thermionic fuel element gas gap, the argon/He gas in the volume accumulator, and the helium gas in the radiation shield. The last four systems are fed from pressurized bottles.

Technology Status of Topaz I and II

Topaz I demonstrated 1 y operation in space, while Topaz II has demonstrated 1.5 y of nuclear ground testing. A claim of 3-y lifetime is based on component life data. Experience with the multicell TFE indicates that swelling and intercell leakage are significant life-limiting problems. The single cell TFE tends to correct these problems, partially by using a high void fraction. A primary life limiting element appears to be the loss of hydrogen from the ZrH moderator. The rate is about one percent per year. Also, with only 65 cents of excess reactivity at design (Topaz CoDR 1992) the fuel burnup can be life limiting, especially if the reactor cools down before a restart is achieved Another issue is the oxygen getter. Modifications may be needed in the cesium supply, but these do not appear to be life limiting.

Topaz I

For the thermionic fuel elements in Topaz I, the chloride-deposited tungsten layers on Mo alloy monocrystalline substrates have been subjected to in-pile tests of up to 17,340 h duration and high temperature (1,900 K) tests of over 13,000 h. For the collector, after 1.5 y of in-pile testing opposite a tungsten emitter, a condensed mass transfer layer has reached a thickness of 300 nm. The layer consists primarily of tungsten, but also contains some oxygen, carbon, and cesium. The UO_2 fuel behavior under irradiation is very complex. The high temperature at the outer edge (~1,900 K), combined with the relatively large diameter of the fuel pin (~10 to 15 mm), implies high temperatures at the center of the fuel pin. Within a matter of hours after startup, the fuel column restructures so that the original stack of pellets has become a single fused structure. The interior of this structure is an isothermal void, which extends the length of the TFE. The 25 to 40 mm gap between the fuel pellets and the interior of the emitter closes due to sublimation and redistribution of the fuel. This void and the micropores originally present in the UO_2 fuel pellet are swept to the interior void. As burnup proceeds, xenon, krypton, and a few volatile fission products are similarly swept from the fuel into the interior. This central void is vented by a passage in a screw hold-down plug fitting in the end of the TFE.

Extensive testing on components has been performed by the U.S.S.R. Coated zirconium hydride material to reduce hydrogen loss has been tested in a reactor with losses of hydrogen being less than 1% of the initial inventory. Zirconium hydride swelling data from neutron irradiation show a volume change of 2 % at 823 K for a fluence of 1.5×10^{21} n/cm^2. The temperature in the system is actually 50 to 75 K less during design operation. Cladding the hydride improves performance by six orders of magnitude, and the use of a cover gas was found to provide another factor of five improvement.

Topaz II

For Topaz II, extensive component and systems testing has been performed. Table 9 provides summary information for some of the major components including the number of components tested, the type of testing, and the time at test. During the development phase, component testing was done in two stages. The first preliminary assurance of functional sufficiency and identified potential problems in an informal manner. Later, more formal tests were performed to ensure that the components met the defined acceptance criteria.

Table 10 (prepared by Susan Voss, Los Alamos National Laboratory) summarizes the power plant testing data on Topaz II. The longest test was 14,000 h. Transient behavior, shown in Figure 20, shows stable operation.

A new automatic control system must be designed that meets U.S. qualification standards for space applications and that is compatible with U.S. launch systems. Also, it is uncertain whether the nuclear fuel will be procured from Russia or fabricated in the U.S. If fabricated in the U.S., the fuel will need to meet both U.S. and Russian quality standards.

TABLE 9. TOPAZ II Component Testing.

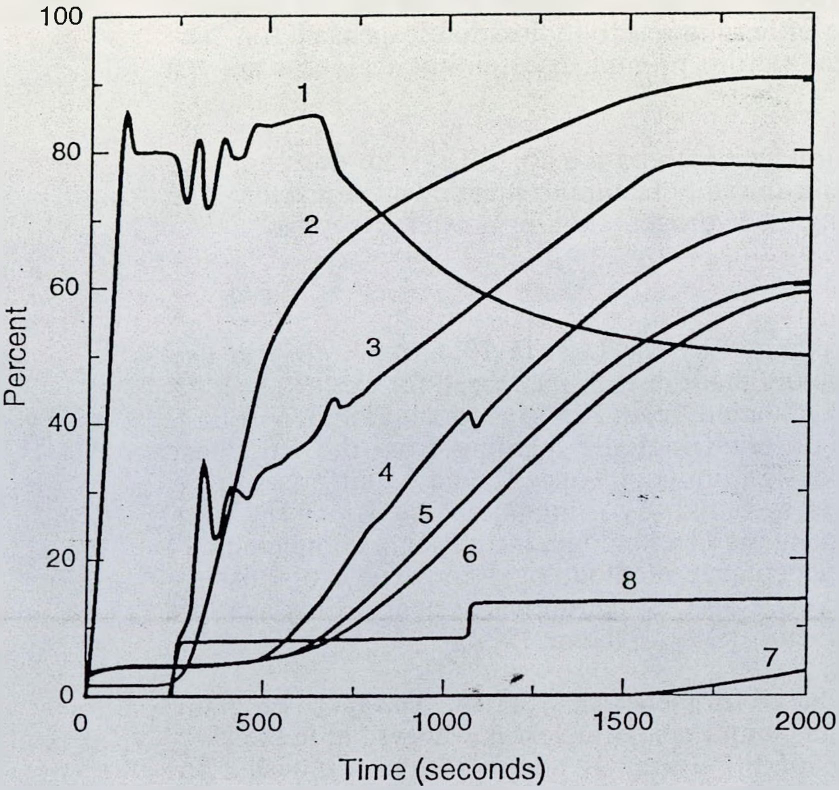
Single Component Tests				System Test		
Component	Test Description	Number of Tests	Time on Test (h)	Test Description	Number of Test	Time on Test (h)
1. Reactor	Thermo-Physical	3	12,000	Electrical Tests	7	12,500
	Mechanical (Dynamic)	2		Mechanical		
				• Ground Transport	4	
				• Static	2	
				• Dynamic	4	
				Cold Testing	4 & 1	
				Nuclear Tests	6	
2. Control Drum Drive	Operational	5	13,000	Electrical Tests	6	14,000
	Lifetime	6		Mechanical		
				• Ground Transport	1	
				• Static	1	
				• Dynamic	3	
				Cold Testing	3	
				Nuclear Tests	4	
3. Safety Drive	Mechanical	3		Mechanical		
	Thermo-Physical	3		• Ground Transport	12	
	Climatic	3		• Static	6	
	Operational	3		• Dynamic	12	
				Cold Testing	12 & 3	
4. Radiation Shield	Mechanical (Static)	2		Electrical Tests	7	14,000
	Lifetime	2		Mechanical		
	Characteristics and Material Changes			• Ground Transport	4	
				• Static	2	
				• Dynamic	4	
				Cold Testing	4 & 1	
				Nuclear Tests	6	
5. Cesium Unit	Thermal Lifetime	7	26,400	Electrical Tests	7	14,000
	Operational Lifetime	3		Mechanical		
				• Ground Transport	4	
				• Static	2	
				• Dynamic	4	
				Cold Testing	4 & 1	
				Nuclear Tests	6	
6. Radiator	Thermal Lifetime	1	10,000	Electrical Tests	7	14,000
				Mechanical		
				• Ground Transport	4	
				• Static	2	

TABLE 9. TOPAZ II Component Testing (continued).

	Single Component Tests			System Test		
Component	Test Description	Number of Test	Time on Test (h)	Test Description	Number of Tests	Time on Test (h)
				• Dynamic	4	
				Cold Testing	4 & 1	
				Nuclear Tests	6	
7. Volume Compensator	Thermal Lifetime	7	40,000	Electrical Tests	7	14,000
	Operational	5		Mechanical		
				• Ground Transport	4	
				• Static	2	
				• Dynamic	4	
				Cold Testing	4 & 1	
				Nuclear Tests	6	
8. Ionization Chamber	Thermal Lifetime	3	26,300	Electrical Tests	7	14,000
				Mechanical		
				• Ground Transport	4	
				• Static	2	
				• Dynamic	4	
				Cold Testing	4 & 1	
				Nuclear Tests	6	
9. Pressure Sensor Unit (4)	Lifetime	1	13,500	Electrical Tests	6	2,000
	Operational	1		Mechanical		
				• Ground Transport	3	
				• Static	2	
				• Dynamic	3	
				Cold Testing	3 & 1	
				Nuclear Tests	6	
10. Start-up Unit	Operational	3		Electrical Tests	3	
	Mechanical			Mechanical		
	• Ground Transport	6		• Ground Transport	3	
	• Dynamic Launch	2		• Static	1	
	Thermal Lifetime	6		• Dynamic	3	
				Cold Testing	1 & 1	
				Nuclear Tests	1	
11. Thermal Cover	Mechanical (Static)	2		Mechanical		
	Mechanical (Dynamic)	1		• Ground Transport	4	
	Operational	2		• Static	2	
				• Dynamic	4	
				Cold Testing	4 & 1	

TABLE 10. TOPAZ II Systems Test Data.

Test	Date	Time at Test	Findings
Plant 23	1975-76	2,500 h at 6 KWe 5,000 h total test	Ground Demonstration Unit. Significant degradation of power due to TFE.
Plant 31	1977-78	4,600 h	Flight Demonstration Unit. Same as TFEs as Plant 23 Startup with Automatic Control System (ACS). 4,600 h was planned test time.
Plant 24	1980	14,000 h	Ground Demonstration Unit. Same TFEs as Plant 23 Startup with ACS Provided life testing of many components.
Plant 81	1980	12,500 h at 4.5 to 5.5 kWe	Ground Demonstration Unit. New TFE design. Did not complete 1000-h electric testing prior to nuclear testing. NaK leaked 150 h into nuclear test. Fixed NaK leak and continued testing.
Plant 82	1983	8,300 h at 4.5 to 5.5 kWe	Ground Demonstration Unit. Loss of flow progressing to a loss-of-coolant accident. Final shutdown of the reactor due to loss of H from the $ZrH_{1.85}$. No major structural damage to the reactor.
Plant 38	1986	4,700 h at 4.5 to 5.5 kWe	Flight Demonstration Unit. Test ended due to a NaK leak in upper radiator collector First ground test with the temperature regulator as part of the ACS.



Parameter	Range
1. Drive position	0 to 180 deg
2. Temperature at the fuel element center	0 to 2275 K
3. Reactor power	0 to 150 kW
4. Coolant temperature at the reactor outlet	0 to 1075 K
5. Average moderator temperature	0 to 1075 K
6. Vessel temperature	0 to 1075 K
7. Load current	0 to 400 A
8. Coolant flow rate	0 to 6 kg/s

FIGURE 20. TOPAZ II Regular Power Plant Startup (Topaz CoDR 1992).

Safety

Topaz II was originally designed for operation in geosynchronous orbit. Therefore, the Russians were not concerned with dispersal after operation. If reentry occurs, it is doubtful that complete break up at sufficiently high altitudes will achieve full reactor dispersal (Topaz CoDR 1992).

A number of design modifications are being considered to meet U.S. safety philosophy. These include the inclusion of a reentry thermal shield to avoid breakup if reentry occurs and the addition of removable poison in the annulus of a number of TFEs or removal of fuel during launch to ensure against reactor supercriticality when immersed and flooded with water. For the planned U.S. flight in 1996 -1997, the initial operational altitude is planned to be well above a sufficiently high orbit for fission product decay.

Another safety concern is the delayed positive temperature coefficient in the reactor core. This coefficient has been confirmed experimentally, and the Topaz reactors have been proven to be experimentally controllable. The delayed positive temperature coefficient effect is due primarily to moderator spectrum hardening when temperature is increased. This leads to fewer moderator captures and more fuel captures. The positive feedback time constant is very long (~330 s) relative to the control system. The long time constant is due to a large heat capacity and high thermal resistance. An effect of having a delayed positive feedback coefficient is to reduce the amount of excess reactivity required to very low levels (65 cents). This results from no negative temperature defect and a minimum of burnup reactivity loss to compensate for. The reactor has the unique feature that startup prompt disassembly accidents are not probable.

If the reactor has a loss of coolant accident and the control system does not shut down the reactor, Topaz II has a built-in safety feature to shut down the reactor. The ZrH moderator will heat up, releasing the hydrogen, shutting down the reactor.

Flight Readiness

A launch of Topaz II is being planned by the U.S. in 1996-1997. It will use information from two unfueled reactors that the U.S. purchased for electric heating testing in the Thermionic System Evaluation Test (TSET) program (TSET/NEPSET Workshop 1993). Goals of this program include: learning from the Russians, evaluating performance within the design limits of Topaz II, and training a cadre of U.S. experts on space nuclear power systems. By using purchased Topaz II power plants, the U.S. obtains insights into Russian technology, insights into complete non-nuclear satellite qualification and acceptance methodology, knowledge of Russian safety methods, and reduced cost to develop U.S. thermionic systems. First heat-up testing of TSET got underway in November 1992 (Thome 1992).

Additional reactors are being purchased for the space program. The space program will demonstrate the feasibility of launching a space nuclear power system in the U.S. and evaluate in-orbit performance of the Topaz II power system. It will also demonstrate electric propulsion options and measure the nuclear electric propulsion self-induced environments. Modifications to the Topaz II power plant will include a new automatic control system, acquiring the nuclear fuel, possibly adding the thermal reentry shield, and adding removable poison in the TFEs or removeable fuel for launch.

Thermionic Fuel Element (TFE) Verification Program

In 1986, the Thermionic Fuel Element Verification Program was initiated to resolve the technical issues identified with thermionic reactors during the SP-100 Phase I concept selection. The program's objectives are to resolve technical issues for a multicell TFE suitable for use in thermionic space reactors with electric power output in the 0.5 to 5 MWe range and a full-power lifetime of 7 to 10 y. The main concerns are fuel/clad swelling, insulator integrity for long irradiation times, and demonstration of performance and lifetime of TFEs and TFE components (Brown and Mulder 1992). The program has been restructured to better address the performance and lifetime requirements currently of interest to the DoD (i.e., 5 to 40 kWe, 1.5 to 5 y lifetime).

System Concept

Thermionic power systems are an attractive option for providing electric power in space because their radiators tend to be smaller than for other concepts (higher heat rejection temperatures and/or efficiencies); outside of the fuel and emitter, the temperatures are sufficiently low that refractory metals are not needed; and scaleability is from a few kilowatts to megawatts (Homeyer et al. 1984, Snyder and Mason 1985 and Holland et al. 1985). The heart of these systems is the thermionic fuel element

(TFE). A single thermionic fuel cell is shown schematically in Figure 21 (Strohmayer and Van Hagan 1985). The TFE is a building block, as seen in Figure 22.

One of the objectives of the TFE Verification Program is to size the cell for a 2 MWe space nuclear power system with a 7-y operating life. Another objective is to demonstrate scaleability from 500 kWe to 5 MWe. Table 11 summarizes systems parameters for the 2 MWe system; Table 12 defines the TFE design parameters (General Atomics 1988).

TABLE 11. 2 MWe System Parameters (General Atomics 1988).

Net electrical power (MWe)	2
System efficiency (percent)	8.9
Lifetime at full power (y)	7
Shield cone half-angle (deg)	12
Reactor thermal power (MWt)	22.5
Coolant	Lithium
Heat rejection (MWt)	20.3
Radiator temperature (K)	1,020
Radiator area (m^2)	426
System Mass (kg)	
Reactor	8,720
Shield	2,803
Coolant loop	1,788
Main radiator	6,390
Power conditioning and shielding	1,230
Power conditioning radiator	262
Cables	506
Structure and miscellaneous	791
Total	22,490

Technology Developments

TFE cells were extensively developed during the 1960s and early seventies with the operation of elements out-of-pile of over 47,000 h and over 12,000 h in-pile (Samulelson and Dahlberg 1990). The major issues with developing a 7-y TFE are fuel/clad dimensional stability, seal insulator integrity, and sheath insulator integrity under temperature, irradiation, and applied voltage. Since 1986, component and TFE cell verification testing has been underway through accelerated and real-time irradiation testing and analytical modeling. The results of these activities are summarized in Tables 13 and 14 (Begg et al. 1992).

Significant progress in the TFE Verification Program include:

- Higher burn up for fuel emitters to 3% at 1800 K and 4% at 1700 K equivalent to over 5 y operation
- Sheath insulators tested under 10-12 V for 8 mo (the test were stopped by a water leak with the test samples still in good condition)

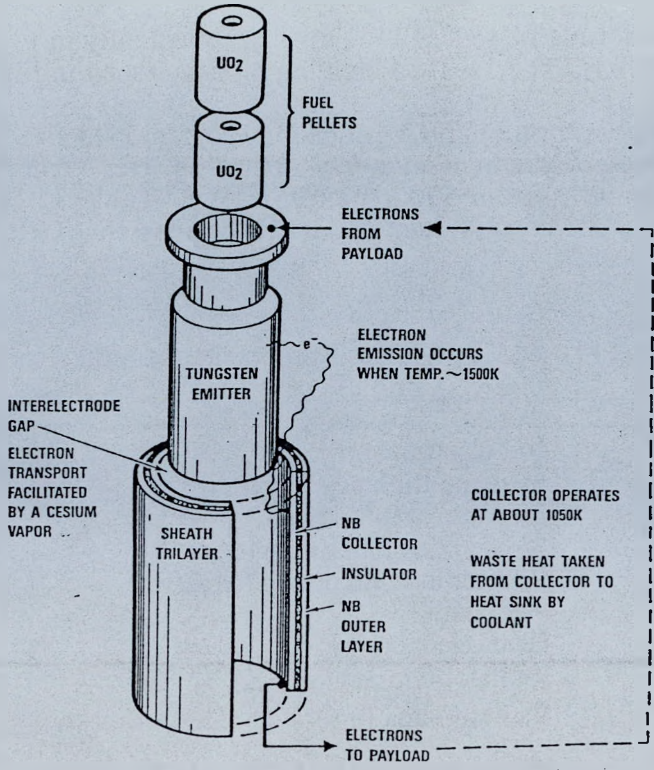


FIGURE 21. Schematic of a Thermionic Fuel Cell (Strohmayer and Van Hagan 1985).

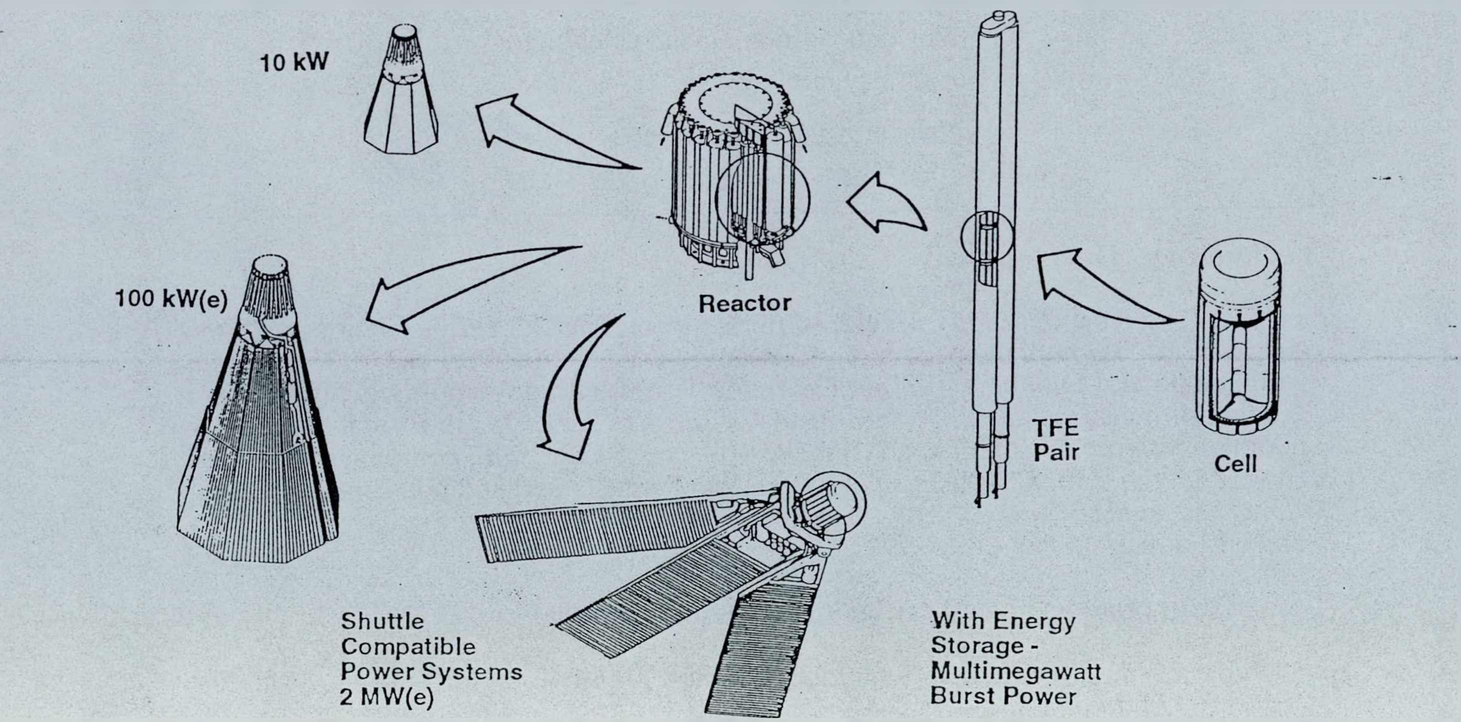


FIGURE 22. Thermionic Cell is the Building Block for Space Nuclear Power Systems to All Levels (Samuelson and Dahlberg 1990).

TABLE 12. TFE Design Definition (General Atomics 1988).

Performance	Values
Overall TFE:	
Output electrical power (We)	705
Efficiency (%)	8.9
Maximum voltage (v)	5.9
U-235 burnup (a/o)	4.1 average, 5.3 peak
Fluence (n/cm^2)	2.7 x 10^{22} average, 3.5 x 10^{22} peak
Converter:	
Converter power (Wt/We)	658/58.8
Thermal power/length (Wt/cm)	137.8
Emitter power flux (We/cm^2)	2.9
Diode current density (A/cm^2)	7.0
Thermionic work function (eV)	4.9
Emitter temperature (K)	1,800
Collector temperature (K)	1,070
Cesium pressure (Pa)	2.7
Converter output voltage (V)	0.49
Converter current (A)	140
Configuration	
Overall TFE:	
TFE length (active core) (cm)	100.6
TFE length (overall)	TBD
Sheath tube O.D. (cm)	1.8
Lead O.D. (cm)	2.2
Lead length (cm)	10.2
Converters per TFE	12
Converter	
Emitter O.D. x L x t (cm)	1.3 x 5.1 x 0.1
Emitter stem L x t (cm)	1.1 x 0.05
Diode gap (cm)	0.025
Trilayer thickness:	
collector(cm)	0.07
insulator (cm)	0.04
outer cylinder(cm)	0.07
Fuel specification	93% enriched UO_2; variable volume fraction
Intercell axial space (cm)	1.88

TABLE 13. TFE Verification Program Key Components Demonstrated Lifetimes (Begg 1992).

Key Components	Tested To	Life in 40 kWe Design (y)
Sheath insulators	4×10^{20} n/cm^2 (in-core with voltage)	0.75
Insulator seals	2.3×10^{21} n/cm^2	4.3
Graphite reservoir	3×10^{22} n/cm^2	>>10
Fueled emitters	>3% at 1800 K	>5
TFEs	13,500 h	1 to 1.5

TABLE 14. TFE Test Results (Begg 1992).

TFE Designation	Time at Power (h)	Cause of Performance Degradation	Comments
1H1	12,000	In PIE	First TFE test in 16 y
1H3	13,500	Fission gas related interelectrode space (planned)	Mixed fission gasses into
1H2	8,800	Internal short caused by over voltage (external)	First graphite reservoir
3H1	13,000	Under evaluation	Still at power, output performance started declining at 8,000 h)
3H5	3000	N/A	Good output, end of February 1993
6H1	N/A	N/A	Planned start in March 1993

- Insulator seals in converters still functional after testing for 39,000 h
- Graphite reservoirs radiation tested to equivalent of over 10 y operation
- TFEs tested to 13,500 h.

Forty Kilowatt Thermionic Power Systems Program

In 1992, a 40-kWe thermionic power plant program was initiated. The design life goal is 10 y; however, the initial lifetime requirement is 1.5 y. Currently, there are two concepts selected. One, called S-Prime Thermionic Nuclear Power System, builds on the multicell TFE Verification Program as well as Topaz I technology. The second concept, called the SPACE-R Thermionic System, builds on the single cell Topaz II technology. Initial mass calculations at 40 kWe indicate both systems have a specific power of 18 W/kg and growth capabilities above 100 kWe. This program is just getting underway with a plan to complete preliminary designs and demonstrate key

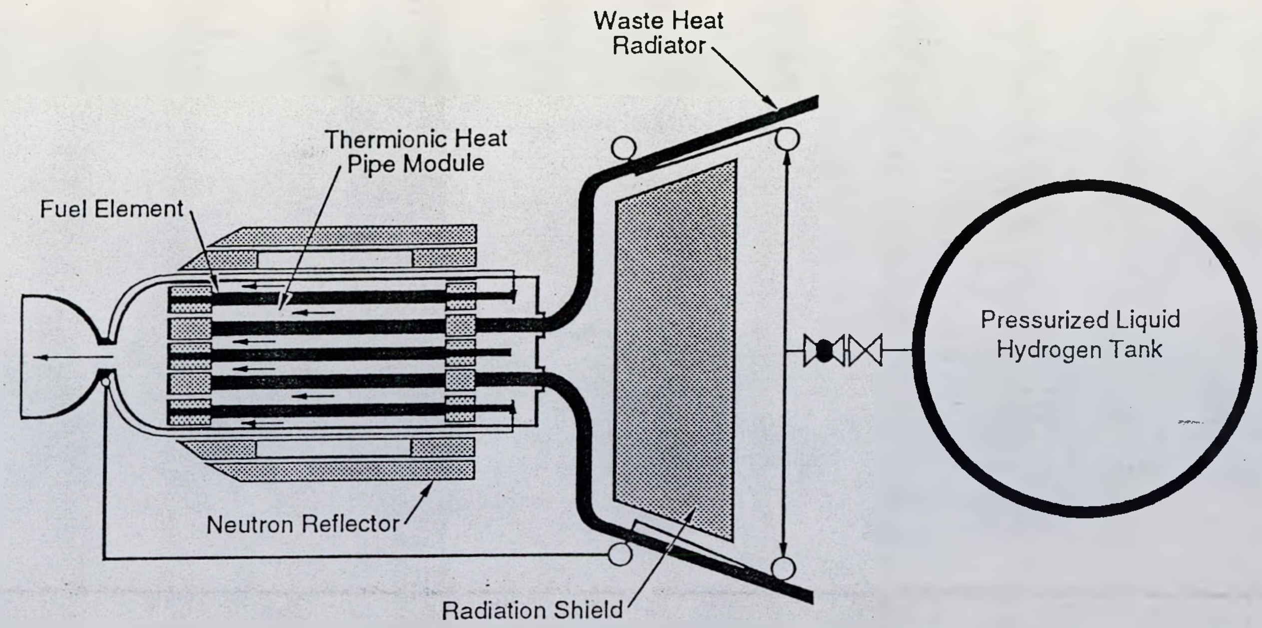


FIGURE 23. Schematic of Propellant Flow for Bimodal Power and Propulsion SEHPTR Operation (Zubrin et al. 1992).

the tungsten cladding must provide enough ductility at 2,100 K to allow for about 8% creep. While creep rupture values for tungsten are much higher than this, data are needed to indicate what effect irradiation will have on ductility. The tungsten clad for the SEHPTR fuel is designed to operate 300 degrees higher than the TFE emitters; data from the 710 Program indicates that chemical compatilility should not be a problem if the UO_2 is stablized. A heat pipe of W-Li has operated for 1.5y; if the heat pipes all fail, there is a 20% degradation in power output. In the tested heat pipe, the mass flow was ten times that needed in the SEHPTR design. This indicates that the 7 to 10-y life is possible. Data on emissivity coatings is now being developed.

A technology demonstration program is underway on the SEHPTR thermionic converter, called Thermionic Heat Pipe Module (THPM) (Horner-Richardson et al. 1992). A schematic is shown in Figure 24. Emitter and collector heat pipes of a preprototype (THPM 1127A) were successfully designed and built in 12 weeks. The converter produced a maximum output of 33.4 W with an emitter temperature of 1,810 K, collector temperature of 1,000 K, and cesium reservoir temperature of 550 K. The emitter heat pipe developed a leak after 50 h due to corrosion of the molybdenum by the lithium working fluid. It is hypothesized that this corrosion was driven by residual nitrogen and/or oxygen in the lithium. Another THPM (1127B) has also been fabricated with a thick emitter sleeve of tungsten. This device has been tested up to temperatures of 2,050 K and has produced 180 W of power. Full length, full power THPMs are planned for later in 1993.

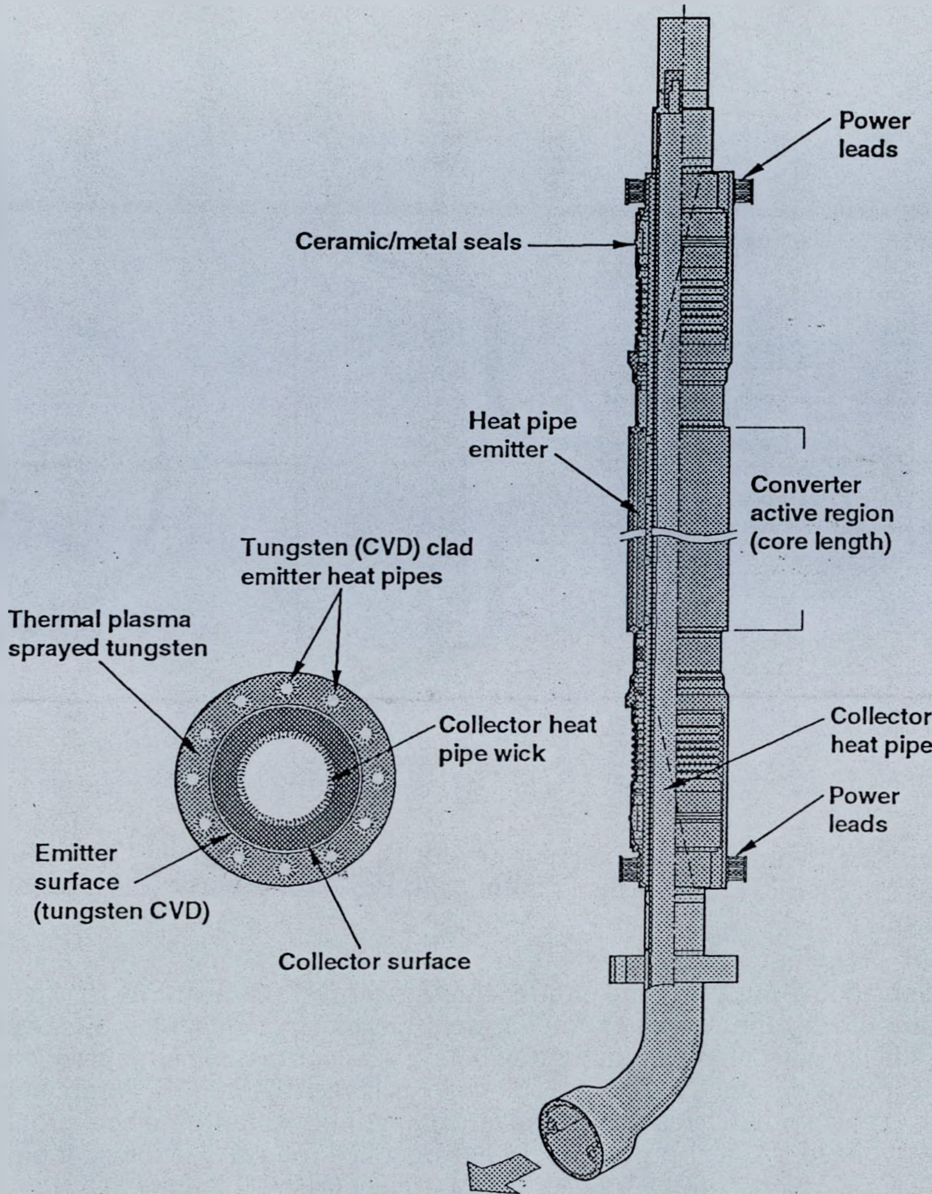


FIGURE 24. Schematic of SEHPTR Thermionic Heat Pipe Module (Horner-Richardson et al. 1992).

Space Thermionic Advanced Reactor-Compact (STAR-C)

In STAR-C (Begg et al. 1992 and Allen et al. 1991), illustrated in Figure 25, the solid core is composed of annular plates of UC_2 fuel supported in graphite trays. Heat is radiated from the radial core surface to a surrounding array of thermionic converters where electric output is generated. Planar thermionic converters operate in the ignited, high-pressure cesium mode with an interelectrode gap of 0.1 mm. The emitter is tungsten, collector niobium, and insulator-seal alumina tri-layer. Reject heat is taken from the thermionic converter collectors through the radial core reflector by heat pipes to an extended surface immediately on the outside of the reactor, from which heat is radiated directly to space. Key parameters are given in Table 15. Design studies are the extent of the funded activities on this concecpt.

STAR-C has the advantages of relatively simple thermionic converters located outside the core, a simple fuel design, no pumped loops, and being a compact power plant overall. The major concerns relate to the quantity and containment of fuel swelling, sublimation of graphite, venting of fission gases without loss of fuel, and limited power system growth.

TABLE 15. STAR-C Key Performance Features.

Thermal Power (kWt)	340
Reactor Output (kWe)	42.8
Net Electrical Power (kWe)	40.9
Net System Efficiency (%)	12.0
Peak Fuel Temperature (K)	2,150
Core Surface Temperature (K)	2,000
Emitter Temperature (K)	1,854
Collector Temperature (K)	1,031
Main Radiator Area (m^2)	5.9
Mass (kg)	2,502

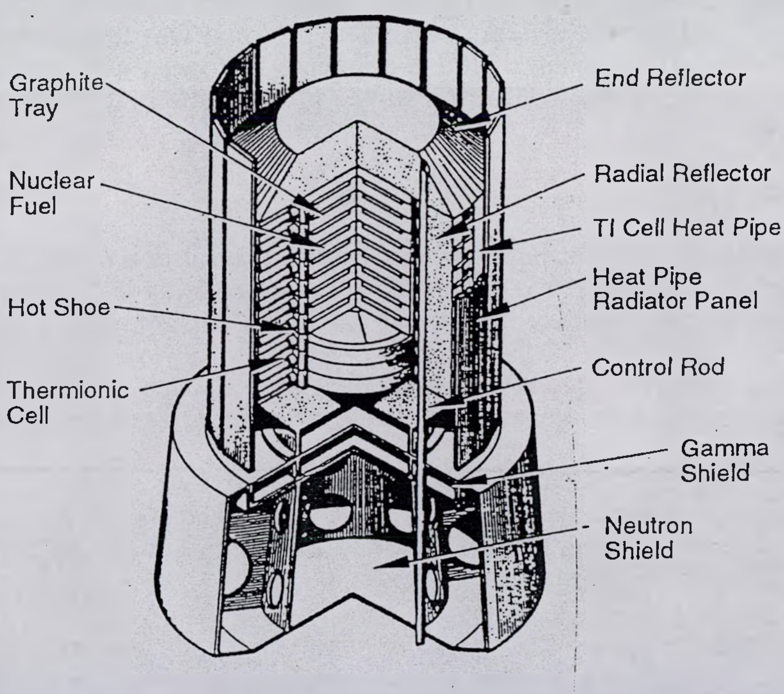


FIGURE 25. STAR-C Configuration (Allen et al. 1991).

MULTIMEGAWATT PROGRAM

Beginning in 1985 and until 1990, a program was underway to develop electrical power in the multimegawatt range for neutral particle beams, free electron lasers, electromagnetic launchers, and orbital transfer vehicles. The program was discontinued because of a shift in emphasis within the Strategic Defense Initiative. The power requirements were grouped into three categories, as seen in Table 16.

TABLE 16. Multimegawatt Space Power System Requirements

	Category I	Category II	Category III
Power requirements (MWe)	10s	10s	100s
Operating time (s)	100s	100s + 1 y of total life	100s
Effluents allowed	Yes	No	Yes
1 Orbit Recharge	No	Yes	No

System Concepts

Six concepts were selected for Phase I studies: three for Category I, two for Category II, and one for Category III. The program was terminated prior to Phase II awards. A brief summary of these concepts follows (much of the information is curtesy of Richard Shutters, Multimegawatt Project Office, Idaho National Engineering Laboratory).

For Category 1:

(1) A fast-spectrum, cermet-fuel, gas-cooled reactor derived from the 710 program (Angelo and Buden 1985) drives twin counter-rotating open Brayton cycle turbines coupled to super-conducting generators. Table 17 summarizes key parameters.

(2) A fast-spectrum, gas-cooled reactor with a two-pass core heats hydrogen and drives twin gas turbo-generators. Table 18 summarizes key parameters.

(3) A gas-cooled, nuclear rocket derivative based on the Nuclear Engine Rocket Vehicle Application reactor (NERVA) (Angelo and Buden 1985) drives two open-cycle, counter-rotating turbine generators (Schmidt et al. 1988 and Chi and Pierce 1990). Table 19 summarizes key parameters.

For Category 2:

(1) A space thermionic advanced reactor with energy storage system (STARS) consisting of a liquid-metal-cooled, in-core thermionic reactor coupled to alkaline fuel cells for burst power. Table 20 summarizes key parameters.

(2) A lithium-cooled, cermet-fuel, fast reactor drives a potassium-vapor Rankine cycle with a Na/S battery storage system. Table 21 summarizes key parameters.

TABLE 17. Category I, MMW, Fast-Spectrum, Cermet-Fuel Reactor (710 Program Derivative) with Twin Counter-Rotating Open Brayton Cycle Turbines Coupled to Super-conducting Generators.

Concept Details	
Reactor Inlet Temperature (K)	164
Reactor Outlet Temperature (K)	800
Turbine Inlet/Exit Temperature (K)	1,400/800
Reactor	
Type (derivative of 710 reactor)	Gas-cooled fast reactor
Fuel (86 to 97% enriched U-235)	Cermet (UO_2-Mo),
Materials	Reactor structure, W-Re; pressure vessel, Inconel-750X
Maximum Fuel Temperature (K)	1,083
Coolant	Hydrogen
Burnup	Negligible
Power Conversion	
Turbine/Compressor Type	Conventional axial turbine
Generator Type	Superconducting
Recuperation	No
Materials	Single-crystal Rene N5 for first blades; second stages A286; shaft and rotor astrology; casing HS188; bearings M50 99.999% aluminum stator
Heat Rejection Method	
Main	Effluent
Decay Heat Removal	Hydrogen flow
System Mass (kg)	8,136

For Category 3:

(1) A gas-cooled, particle-bed reactor drives a turbine generator (Powell and Horn 1985). Table 22 summarizes key parameters.

Technology Developments

The major Multimegawatt Program development activities were concerned with fuels. Scoping tests were performed to evaluate the compatibility of UN fuels with W-Re and Mo-Re alloys. The test results showed some problems at high temperatures, but these could be mitigated through control of the UN stoichiometry. Thermodynamic analyses were performed to estimate the chemical compatibility of UC fuels with these alloys. Also, a testing program was performed on two particle bed fuel elements. The elements did not perform as expected. Post-irradiation examinations indicated power/flow matching problems exhibited by nonuniform flow distributions, particle-frit chemical and mechanical interactions, and cycling problems.

TABLE 18. Category I, MMW, Fast-Spectrum Reactor with a Two-Pass Core and with Gas Turbo-generators.

Concept Details	
Reactor Inlet Temperature (K)	150
Reactor Outlet Temperature (K)	1,200
Turbine Inlet/Exit Temperature (K)	1,200/800
Reactor	
Type	Gas-cooled fast reactor with two-pass core
Fuel	UC pins, 316 S.S. clad first pass, Mo-41 Re for second pass
Materials	Vessel is 316 stainless steel
Maximum Fuel Temperature (K)	1,650
Coolant	Hydrogen
Burnup	Insignificant
Power Conversion	
Turbine/Compressor Type	Twin axial turbine counter-rotating; 15,000 rpm
Generator Type	Wound field, non-salient pole, hydrogen cooled
Recuperation	Yes
Heat Rejection Method	
Main	Effluent
Decay Heat Removal	Hydrogen flow
System Mass (kg)	12,887

Some materials development activities were undertaken on Ta-, Mo-, and W-based alloys. Several material lots were produced, but the efforts were terminated before conclusive data were obtained on the alloys.

FOREIGN CONCEPTS

A number of international participants have presented concepts during the Space Nuclear Power Systems Symposiums in Albuquerque, NM in addition to the U.S.S.R. concepts described above. Rolls-Royce of the United Kingdom participated in the Multimegawatt Program. The most active foreign participation at the Symposia has been by France. In addition, Japan presented a concept of a reactor that could be launched with a fuel cartridge separate from the reactor (Yasuda et al. 1990). The fuel is coated particle and the reactor is cooled using heat pipes.

The French space nuclear power program has entered a period of lower activity. They have concentrated on power levels in the 10 to 30 kWe range. The power conversion system selected is the Brayton cycle. Several reactor concepts have been reported, including gas-cooled, liquid-metal-cooled, and water moderated reactors (Tilliette et al. 1990, Proust et al. 1990, Carre et al. 1990 and Tilliette and Carre 1990). The selected reference system is a 930 K, NaK-cooled, fast spectrum reactor (Figure 26). This selection was based on available or near term technologies.

TABLE 19. Cateorgy I, MMW, Gas-Cooled, NERVA-Derivative Reactor (NDR) with Two Open-Cycle, Counter-Rotating Turbine-Generators.

Concept Details	
Reactor Inlet Temperature (K)	35
Reactor Outlet Temperature (K)	1,150
Turbine Inlet Temperature (K)	1,150
Reactor	
Type	NERVA derivative, graphite-moderated near thermal
Fuel	Pyrolitic carbon and SiC-coated $UC_{1.7}$ particles in graphite matrix
Materials	Titanium vessel, Zr-C coated graphite internals
Maximum Fuel Temperature (K)	1,200
Coolant	Hydrogen
Burnup	Negligible
Power Conversion	
Turbine/Compressor Type	Two counter-rotating, high pressure axial turbines exhaust to a pair of counter-rotating, low pressure axial turbines. Pressure provided by LH_2 turbopumps.
Generator Type	Hyper-conducting
Recuperation	None except by jacketing the reactor outlet line
Materials	99.999% Aluminum conductors
Heat Rejection Method	
Main	Effluent
Decay Heat Removal	Hydrogen flow through tie tubes to a radiator
System Mass (kg)	10,513

The initial version of the liquid-metal-cooled reactor consisted of a tight lattice of 780 UO_2 fuel pins arranged with a pitch to diameter ratio of 1.07 (9.1/8.5 mm). This leads to a fuel inventory of 75 kg of 93% enriched uranium with active core height of 270 mm and diameter of 290 mm. Alternative core designs, including an SP-100 derivative architecture using a honeycomb fuel supporting structure, are being considered, and the mass penalty associated with the lesser fuel volume fraction within the core is being evaluated. The reduction of core size and reactor mass afforded by the use of UN fuel is also being assessed. The reactor control system has 12 rotating drums, and the core contains 7 safety rods. A shadow shield consists of B_4C and LiH elements fitted in a stainless steel honeycomb structure for neutron attenuation and tungsten for gamma ray attenuation are included in the design. The Brayton systems are designed with a single recuperated turbogenerator directly coupled to an armored gas cooled radiator as a heat

TABLE 20. Category II, MMW, Space Thermionic Advanced Reactor with Energy Storage System (STARS) and with a Liquid-Metal-Cooled, In-Core Thermionic Reactor Coupled to Alkaline Fuel Cells for Burst Power.

Concept Details	
Reactor Inlet Temperature (K)	230
Reactor Outlet Temperature (K)	1,130
Main Radiator Inlet/Exit (K)	1,073/1,023
Reactor	
Type	Liquid-metal-cooled, in-core thermionic
Fuel	UO_2 pellets
Materials	W-HfC
Maximum Fuel Temperature (K)	2,520
Coolant	Lithium
Burnup	Negligible
Power Conversion	
Type	Thermionic
Emitter	W-HfC, 2,200 K
Collector	Nb, 1,200 K
Energy Storage	
Type	Fuel cell
Heat Rejection Method	
Main	Heat-pipe panel radiators for steady-state system, expandable radiator for fuel cell system with water.
Decay Heat Removal	Heat pipe radiators
System Mass (kg)	58,152

sink. Alternative design options under study include using redundant dual Brayton engines and a heat pipe radiator. The mass of a 20 kWe system is calculated to be 2319 kg.

ADDITIONAL CONCEPTS

A number of other concepts deserve discussion. Space limits mentioning all of the concepts that have appeared in the Space Nuclear Power Systems Symposium over the last ten years.

TABLE 21. Catgegory II, MMW, Li-cooled, Cermet-Fuel, Fast Reactor with a Potassium-Vapor Rankine Cycle and Na/S Battery Storage System.

Concept Details	
Reactor Inlet Temperature (K)	1,500
Reactor Outlet Temperature (K)	1,660
Turbine Inlet/Exit Temperature (K)	1,500/1,255
Radiator Inlet/Exit (K)	1,040 average
Reactor	
Type	Liquid-metal-cooled fast reactor Cermet (UN in W-Re)
Materials	Pressure vessel and piping, ASTAR 811-C
Maximum Fuel Temperature (K)	1,870
Coolant	Lithium
Burnup	~7%
Power Conversion	
Working Fluid	Potassium
Turbine/Compressor Type	Axial turbine
Generator Type	Wound rotor
Energy Storage	
Type	Sodium-sulfur batteries
Heat Rejection Method	
Main	Carbon-carbon heat pipes
Decay Heat Removal	Heat pipe radiators
System Mass (kg)	54,544

Heat Pipe Reactors

Heat-pipe-cooled reactors were being pursued before the SP-100 program started in 1983. These used heat pipes to cool the reactor and transfer the heat to the electric power conversion equipment. Significant features of such a system are that all heat transport is by passive means and there is multiple redundancy throughout the system. The disadvantages of such a system are that no heat pipe reactors have actually been built, there are technical questions as to whether the redundancy is actually achieved, and the technology base for 1,400 K, long-life heat pipes is limited (El-Genk et al. 1985 and Koenig 1985). More recent studies have suggested a heat pipe reactor with operating temperature of 1,125 K to reduce development risks (Ranken 1990).

TABLE 22. Category III, MMW, Gas-Cooled, Particle-Bed Reactor with a Turbine Generator.

Concept Details	
Reactor Inlet Temperature (K)	220
Reactor Outlet Temperature (K)	1,050
Turbine Inlet Temperature (K)	1,050
Reactor	
Type	Gas-cooled, particle-bed reactor
Fuel	UC_2 coated with ZrC and pyrolytic C particles
Materials	Al vessel, Mo-Re hot frit
Maximum Fuel Temperature (K)	1,240
Coolant	Hydrogen
Burnup	Negligible
Power Conversion	
Turbine/Compressor Type	Axial turbines on counter-rotating shafts, direct coupled generator
Generator Type	Cryo-cooled, wound rotor alternator, 3-phase output at 20 kV
Recuperation	None
Heat Rejection Method	
Main	Effluent
Decay Heat Removal	H_2 flow via multiple independent channels
System Mass (kg)	42,000

Gas Vapor Cores

Gas vapor cores have been studied for hundreds of MWe of power. The concepts are futuristic and could find applications in large power plants for beaming power back to Earth. The Ultra-High Temperature Vapor Reactor (UTVR) is an externally-moderated (with BeO), circulating fuel reactor with highly enriched (>85%) UF_4 fuel (Kahook and Dugan 1991). The working fluid is in the form of a metal fluoride. A side view schematic is shown in Figure 27. The reactor has two fissioning core regions: (1) the central ultrahigh temperature region contains a vapor mixture of highly-enriched UF_4 and a metal fluoride working fluid at an average temperature of ~3,000 K and a pressure of ~5 MPa., and (2) the boiler columns which contain highly enriched UF_4 fuel. This reactor has symmetry about the midplane, with identical tandem vapor cores and boiler columns separated by the midplane BeO slab region and the magnetohydrodynamic ducts where power is extracted. The walls are maintained at about 2,000 K by tangential injection of the metal fluoride. A schematic is shown in Figure 28 with representative values of some parameters (Diaz et al. 1991). This is a research program with recent results: UF_4 is the fuel of choice above 1,800 K; UF_4 is compatible with tungsten, molybdenum, and glassy carbon to 2,200 K for up to 2 h; specialized gaseous core neutron data were developed; and nuclear-induced ionization enhances electrical conductivity by factors greater than 10.

SUMMARY

Table 23 summarizes major potential applications of space nuclear power. The table divides applications into near-Earth, solar system exploration (Yen and Sauer 1991), Lunar-Mars exploration, and Near-Earth resources. The requirements presented are representative values, there are many possible optimizations for each specific application.

In the near term, most applications would involve missions to support planet Earth, and most of those would be defense missions that provide unique capabilities. There is a need for rapidly deployable satellites from launch site storage in times of conflict and for redeployment of satellites already in space for better coverage of key geographic areas. Improved surveillance, communications, battlefield illumination, and electronic jammers are some of the unique systems that would be enabled by nuclear power.

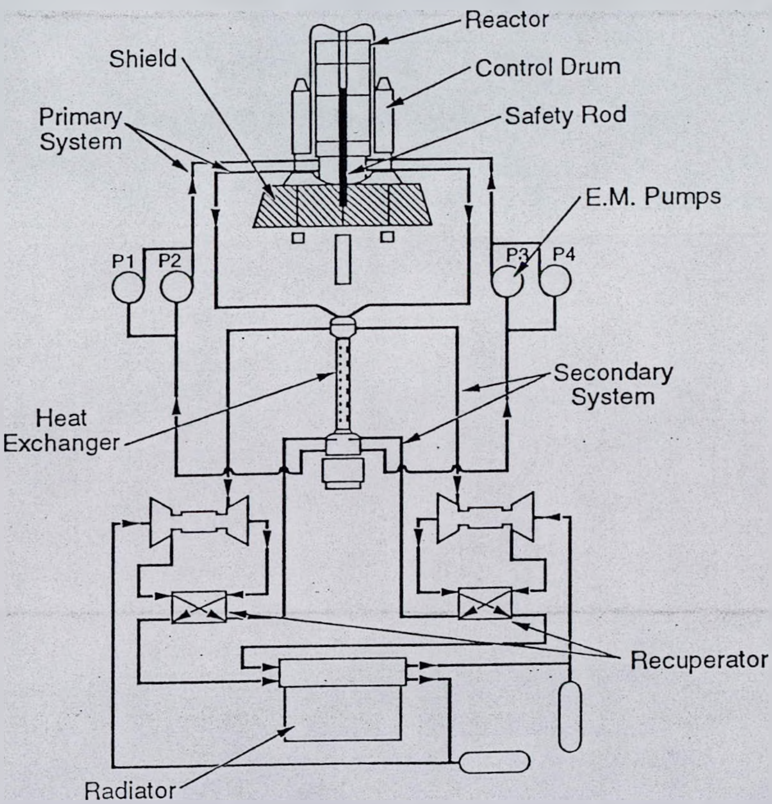


FIGURE 26. Double Loop 20 kWe Nuclear Brayton Power System Concept proposed by French (Tilliette et al. 1990).

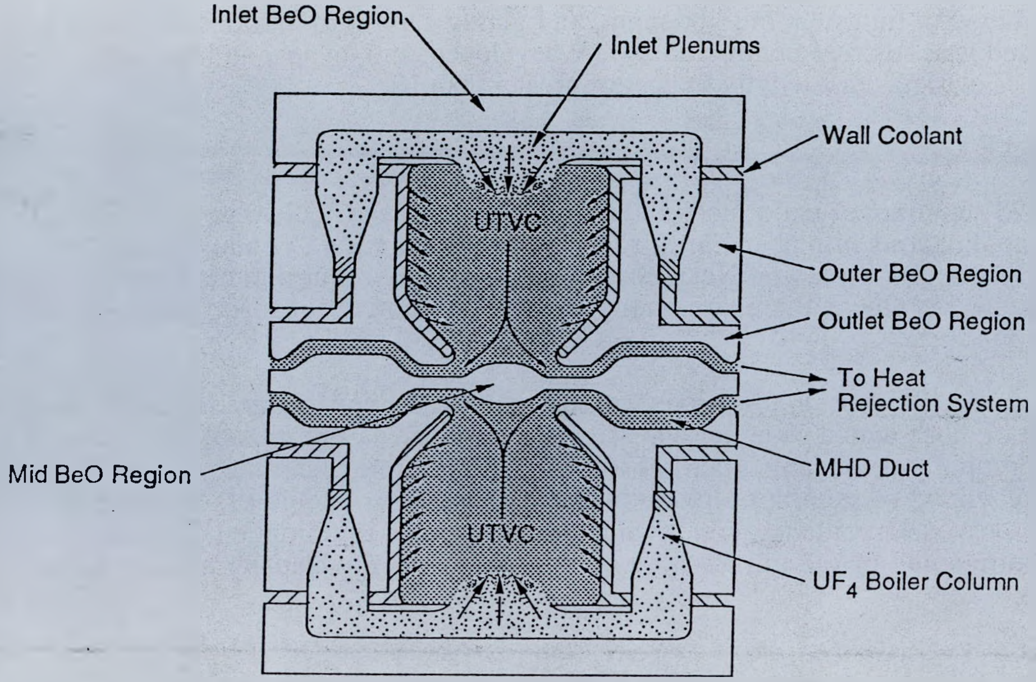
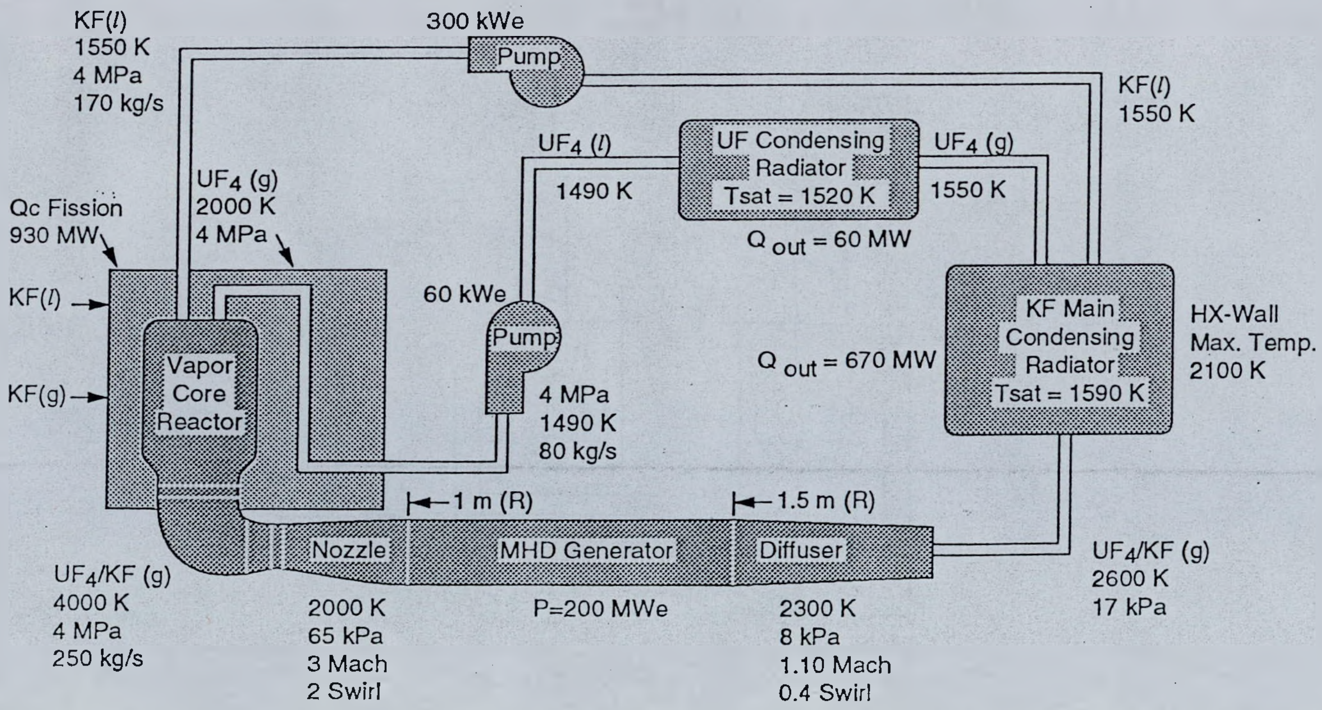


FIGURE 27. Side View Schematic of the UTVR (Diaz et al. 1991).



UF_4/KF UTVR-MHD

Net Cycle Efficiency (%) 22
Gas Core Reactor Volume (m^3) 3
Neutron Flux (n/cm^2s) 10^{16}
Electric Conductivity (mho/m) 60
Magnetic Field (Tesla) 4

FIGURE 28. Schematic Diagram of a 200 MWe UTVR-MHD Conceptual Design with Specific Mass of ~ 1 kg/kWe (Diaz et al. 1991).

These require power levels of 10 to 40 kWe with lifetimes of 7 to 10 y. The orbits are sufficiently high to satisfy concerns about safety and possible interference with gamma ray observatories. Other possible missions include performing underground measurements using ground penetrating radar or microwaves and chemical, nuclear and biological effluent monitoring using laser spectroscopy for non-proliferation and treaty verification. The Federal Aeronautics Administration has a need for oceanic anti-collision aircraft radar. Environmental monitoring needs world-wide measurements of ozone and pollution and also upper air turbulence. Commercial use could include a satellite component of the data information superhighway for remote and mobile sites and high definition television satellites.

Robotic exploration of the solar system and piloted exploration of the Moon and Mars are given less weight in the next decade. However, if nuclear systems enable low cost orbiters to the outer planets, these might be planned at the beginning of the next century. Near-Earth resource recovery from comets and asteroids is another possible application that might become a driver, but only at the beginning of the next century. Precursor missions are needed to establish the viability of this approach.

During the last ten years, many challenges have been overcome in the development of space nuclear power systems. For instance, SP-100 has demonstrated a 7-y fuel pin that can operate at 1,400 K. Based on the work already performed, there is high confidence that the liquid metal cooled, fuel pin SP-100 reactor, with thermoelectric or some other converter, can be successfully developed. The SP-100 program has progressed to the hardware demonstration of most of the key components. NASA has tended to favor the development of the SP-100 to meet their mission needs.

Thermionic power systems have received a major boost with the change in relations with the former Soviet Union and the accessibility to its technology. Defense missions have tended to favor thermionic systems because of being more compact, with smaller radiators. Themionic power plants have been demonstrated by the Russians in flight tests. There are plans for the U.S. to space test a Topaz II power system, modified to meet U.S. safety standards, in 1996 or 1997. Follow-on thermionic developments in both the U.S. and Russia are planned to develop a flight-ready 40 kWe system. Thermionic fuel element lifetime is still the key issue.

Changing mission emphasis is leading to emerging interest in other forms of nuclear space systems. This interest includes a possible combination of electric power and thermal propulsion in a single power plant.

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TABLE 23. Representative Potential Space Nuclear Power Missions.

Mission	Key Requirements
Near-Earth	
Defense	
Wide-area surveillance	20 to 40 kW, lifetime 7 to 10 y, rapid deployment, high elliptical Earth orbit (HEEO)/medium Earth orbit (MEO)
Battlefield communications	10 to 20 kW, lifetime 7 to 10 y, rapid deployment, geosynchronous Earth orbit (GEO) orbit
Battlefield illumination	10 to 40 kW, lifetime 7 to 10 y, rapid deployment, HEEO/MEO orbits
Electronic jammers	>10 kW, lifetime 7 to 10 y, rapid deployment, HEEO/MEO orbits
Submarine communications	10 to 40 kW, lifetime 7 to 10 y, rapid deployment, HEEO orbit
Non-Proliferation and Treaty Verification	
Under ground measurements	to 40 kW, lifetime 7 to 10 y
Moveable surface sensors	to 40 kW, lifetime 7 to 10 y
CBN effluent monitoring	to 40 kW, lifetime 7 to 10 y
Federal Aeronautics Administration	
Anti-collision aircraft radar	20 to 40 kW, lifetime 7 to 10 y, HEEO/MEO orbits
Commercial	
Electronic information highway	25 to 100 kWe
Direct broadcast television	25 to 100 kWe
Environmental Monitoring	
Earth observations	>10 kW, lifetime 7 to 10 y
Upper air turbulence	>10 kW, lifetime 7 to 10 y
Solar System Exploration	
Neptune orbiter/probe	Payload 1.8 Mg, 100 kW, power system mass 3.7 Mg
Pluto orbiter/probe	Payload 1.4 Mg, 56 kW, power system mass 2.8 Mg
Uranus orbiter	Payload 1.4 Mg, 100 kW, power system mass 3.7 Mg
Jupiter grand tour	Payload 1.4 Mg, 58 kW, power system mass 2.9 Mg
Rendezvous	Payload 1.4 Mg, 40 kW, power system mass 2.35 Mg
Comet Sample/Return	Payload 1.8 Mg, 100 kW, power system mass 3.7 Mg
Lunar-Mars Exploration	
First lunar outpost	>12 kW
Enhanced outpost (ISRU)	>200 KW
Mars transportation	Flight time <180 d, payload 52 MT
Mars stationary (600 d)	75-150 kW
Mars in situ resources	> 200 kW
Mars comsats	20 KW
Near-Earth Resources	
In situ probes	30 kW, payload 1.5 Mg, lifetime 3 y
Transportation	20 kN

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References

Allen, D. T., G. O. Fitzpatrick, and D. M. Ernst (1991) "The STAR-C Nuclear Power With High-Efficiency Transparent Thermionic Converters," in *Proceedings Eighth Symposium on Space Nuclear Power Systems*, Conf-910116, M. S. El-Genk and M. D. Hoover, eds., American Institue of Physics, New York, AIP Conf. Proc. No. 217, 3:1093–1099.

Ambrus, J. H. and R. G. Beatty (1985) "Potential Civil Mission Applications for Space Nuclear Power Systems," in *Space Nuclear Power Systems 1984*, Proceedings of the First Symposium on Space Nuclear Power Systems, held in Albuquerque, NM, Jan. 1984, Chapter 10, 43–51.

Angelo, J. A., Jr. and D. Buden (1985) *Space Nuclear Power*, Orbit Book Company, Malabar, FL.

Atwell, J. C., et al. (1989) "Thermoelectric Electromagnetic Pump Design For The SP-100 Reference Flight System," in *Transactions Sixth Symposium On Space Nuclear Power Systems,* held in Albuquerque, NM, Jan. 1989, 280-283.

Begg, L. L., T. J. Wuchte, and W. D. Otting (1992) "STAR-C Thermionic Space Nuclear Power System," in *Proceedings Ninth Symposium on Space Nuclear Power Systems* , held in Albuquerque, NM, Jan. 1992, 1:114–119.

Bennett, G. (1989) "A Look At The Soviet Space Nuclear Power Program," in *Proceedings 24th Intersociety Energy Conversion Engineering Conference*, held in Washington, D.C., August 1989, 2:1187-1194.

Bond, J. A., D. N. Matteo, and R. J. Rosko (1993) "Evolution of the SP-100 Conductivity Coupled Thermoelectric Cell," in *Proceedings Tenth Symposium Space Nuclear Power and Propulsion*, held in Albuquerque, NM, January 1993, 2:753-758.

Brown, C. E. and D. Mulder (1992) "An Overview of the Thermionic Space Nuclear Power Program," presented at the *1992 Space Nuclear Power Systems Symposium*, held in Albuquerque, NM, Jan. 1992.

Carre, F., et al. (1990) "Update of the ERATO Program and Conceptual Studies on LMFBR Derivative Space Power Systems," in *Proceedings Seventh Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1990, 1:381–386.

Chi, J. W. H. and W. L. Pierce (1990) "NERVA Derivative Reactors and Space Electric Propulsion Systems," in *Proceedings Seventh Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1990, 1:208-213.

Chiu, W. S. (1985) "Impact of Stirling Engines to a Space Nuclear Power System," in *Proceedings Second Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1985, EC-2.

Deane, N. A. et al. (1989) "SP-100 Reactor Design and Performance," in *Proceedings 24th Intersociety Energy Conversion Engineering Conference*, held in Washington, DC, Aug. 1989, 2:1225–1226.

Diaz, N. J. et al. (1991) *Ultrahigh Temperature Reactor and Energy Conversion Research Program, Annual Report For Period July 1990-Sept. 1991*, Innovative Nuclear Space Power & Propulsion Institute Report INSPI-91-110, Oct. 1991.

Disney, R. K., J. E. Sharbaugh, and J. C. Reese (1990) "Shielding Approach And Design For SP-100 Space Power Applications," in *Proceedings Seventh Symposium On Space Nuclear Power Systems*, held in Albuquerque, NM, 1:121–124.

Doherty, M. P. and J. H. Gilland (1992) "NEP Systems Engineering Efforts In Fy92 Plans and Status," in *Proceedings Nuclear Technologies For Space Exploration NTSE-92*, American Nuclear Society Meeting held in Jackson, WY, August 1992, 1:42-49.

Dudenhoefer, J. E. (1990) "Programmatic Status of NASA's CSTI High Capacity Power Stirling Space Power Converter Program," in *Proceedings 25th Intersociety Energy Conversion Engineering Conference*, held in Reno, Nevada, Aug. 1990, 1:40.

Dudenhoefer, J. E. and J. M. Winter (1991) "Status of NASA's Stirling Space Power Converter Program," in *Proceedings 26th Intersociety Energy Conversion Engineering Conference*, held in Boston, MA, Aug. 1991, 2:38-43.

Dudenhoefer, J. E., D. Alger, and J. M. Winter (1992) "Progress Update of NASA's Free-Piston Stirling Space Power Converter Technology Project," in *Proceedings Nuclear Technologies For Space Exploration, NTSE-92*, American Nuclear Society Meeting held in Jackson, WY, August 1992, 1:294–303.

El-Genk, M. S. et al. (1985) "Review of the Design Status of the SP-100 Space Nuclear Power System," in *Space Nuclear Power Systems 1984*, Chapter 23, Orbit Book Co., Malabar, FL, 1985, 177–188.

General Atomics (1988) *TFE Verification Program Semiannual Report for the Period Ending September 30, 1987*, General Atomics Documents GA-A19115, LaJolla, CA, March 1988.

General Electric Co. (1989a) "SP-100 Mission Risk Analysis," in *Report No. GESR-00849*, San Jose, CA, Aug. 1989.

General Electric Co. (1989b) "SP-100 Safety Feature," presented at the *Sixth Symposium On Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1989.

Gilland, J. H. and S. R. Oleson (1992) "Combined High and Low Thrust Propulsion For Fast Piloted Mars Missions," *in Proceedings Nuclear Technologies For Space*

Exploration NTSE-92, American Nuclear Society Meeting held in Jackson, WY, August 1992, 1:22-31.

Harty , R. B., M. P. Moriarty, and T. A. Moss (1985) "Capabilities of Brayton Cycle Space Power Systems," in *Proceedings of the Second Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1985, EC-3.

Holland, J.W. et al. (1985) "SP-100 Thermionic Technology Program Status," in *Space Nuclear Power Systems 1985*, Chapter 24, Orbit Book Co., Melabar, FL, 179–188.

Homeyer, W. G., et al. (1984) "Thermionic Reactors for Space Nuclear Power," in *Space Nuclear Power Systems 1984*, Chapter 25, Orbit Book Co., Melabar, FL, 197-205.

Horner-Richardson, D., J. R. Hartenstine, D. Ernst, and M. G. Jacox (1992) "Fabrication and Testing of Thermionic Heat Pipe Modules for Space Nuclear Power Systems," in *Proceedings 27th Intersociety Energy Conversion Engineering Conference*, Paper No. 929075, San Diego, CA, August 1992, 6:6.35-6.41.

Hwang, Choe et al. (1993) "SP-100 Lithium Thaw Design, Analysis, and Testing," in *Proceedings Tenth Symposium on Space Nuclear Power and Propulsion Systems*, held in Albuquerque, NM, Jan. 1993, 2:697-702.

Hylin, E. C. and M. P. Moriarty (1985) "Impact of Stirling Engine Characteristics upon Space Nuclear Power Systems Design," in *Proceedings Second Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1985, EC-1.

Josloff, A. T. (1988) "SP-100 Space Reactor Power System Scaleability," in *Proceedings 23th Intersociety Energy Conversion Engineering Conference*, held in Denver, CO, Aug. 1988, 1:263–265.

Josloff, A. T., H. S. Bailey, and D. N. Matteo (1992a) "SP-100 System Design and Technology Progress," in *Proceedings Ninth Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1992, 1:363–371.

Josloff, A. T., D. N. Matteo, and H. S. Bailey (1992b) "SP-100 Technology Development Status," in *Proceedings Nuclear Technologies For Space Exploration, NTSE-92*, American Nuclear Society Meeting held in Jackson, WY, August 1992, 1:238–246.

Kahook, S. D. and E. T. Dugan (1991) "Nuclear Design Of The Burst Power Ultrahigh Temperature UF4 Vapor Core Reactor System," in *Proceedings Eighth Symposium on Space Nuclear Power Systems*, CONF-910116, American Institute of Physics, 1:205–210.

Koenig, D. R. (1985) "Heat Pipe Reactor Designs for Space Power," in *Space Nuclear Power Systems 1984*, Chapter 27, Orbit Book Co., Malabar, FL, 1985, 217–228.

Marshall, A., et al (1993) "Topaz II Preliminary Safety Assessment," in *Proceedings Tenth Space Nuclear Power and Propulsion Systems Symposium*, held in Albuquerque, NM, Jan. 1993, 1:439-445.

Matteo, D. N., J. A. Bond and R. J. Rosko (1992) "SP-100 Thermoelectric Converter Technology Development," in *Proceedings of the 27th IECEC*, held in San Diego, CA, August 1992, 6:73-77.

Matteo, D. N. (1993) Personal Communication, Martin Marietta, Valley Forge, PA, 25 May, 1993.

Mondt, J. F. (1991) "Overview of the SP-100 Program," in *AIAA/NASA/OAI Conference on Advanced SEI Technologies*, AIAA 91-3585, held in Cleveland, OH, September 1991.

Mondt, J. (1992) *SP-100 Accomplishments*, in SP-100 Project Review held June 15, 1992, JPL D-9650.

Mondt, J. F. (1989) "Development Status Of The SP-100 Power System," in *Proceedings AIAA/ASME/SAE/ASEE 25th Joint Propulsion Conference,* Paper No. AIAA-89-2591, held in Monterey, CA, July 1989.

Mondt, J. F. (1993) *SP-100 Project Memorandum No. 8501-93-004*, 25 March 1993.

Nicitin, V. P., et al. (1992) "Special Features And Results Of The "Topaz II" Nuclear Power System Tests With Electric Heating," in *Proceedings Ninth Symposium on Space Nuclear Power Systems*, Conference 920104, American Institute of Physics, 1992, 1:41–46.

Phillips Laboratory (1992) *Thermionic Space Nuclear Power Technical Program Review,* held in Albuquerque, NM, 9-10 November 1992.

Powell, J. R. and F. L. Horn (1985) "High Power Density Reactors Based on Direct Cooled Particle Beds," in *Space Nuclear Power Systems 1985*, Chapter 39, Orbit Book Co., Melabar, FL, 319–329.

Proust, E., S. Chaudourne, and J. Tournier (1990) "Performance Modeling and Optimization of Design Parameters for the Space Nuclear Brayton Power Systems Considered in the ERATO Program," in *Proceedings Seventh Symposium on Space Nuclear Power Systems* , held in Albuquerque, NM, Jan. 1990, 1:426–431.

Ranken, W. A. (1990) "Low Risk Low Power Heat Pipe/Thermoelectric Space Power Supply," in *Proceedings Seventh Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1990, 1:488–496.

Redd, F. J. and E. V. Fornoles (1985) "Emerging Space Nuclear Power Needs," in *Space Nuclear Power Systems 1984*, Proceedings of the First Symposium on Space Nuclear Power Systems, held in Albuquerque, NM, Jan. 1984, Chapter 9, 41–42.

Samulelson, S., L. and R. C. Dahlberg (1990) "Thermionic Fluid Element Verification Program - Summary of Results," in *Proceedings Seventh Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1990, 1:77–84.

Schmidt, J. E., J. F. Wett, and J. W. H. Chi (1988) "The NERVA Derivative Reactor and a Systematic Approach to Multiple Space Power Requirements," in *Space Nuclear Power Systems 1988*, Chapter 19, Orbit Book Co., Melabar, FL, 95–100.

Schmitz, P., H. Bloomfield, and J. Winter (1992) "Near Term Options For Space Reactor Power," in *Proceedings of Nuclear Technologies For Space Exploration*, American Nuclear Society Meeting held in Jackson, WY, August 1992, 2:313-321.

Slaby, J. G. (1987) "Overview of NASA Lewis Research Center Free-Piston Stirling Engine Technology Activities Applicable to Space Power Systems," in *Space Nuclear Power Systems 1987,* Chapter 5, 35–39.

Snyder, H., J. and J. H. Mason (1985) "Thermionic Space Power System Concept," in *Space Nuclear Power Systems 1985*, Chapter 29, Orbit Book Co., Melabar, FL, 223–229.

Space Power Inc. (1990) "*Characteristics Of The Soviet Topaz II Space Power System*," Report SPI-52-1, San Jose, CA, October 1990.

Strohmayer, W.H. and T. H. Van Hagan (1985) "Parametric Analysis of a Thermionic Space Nuclear Power System," in *Space Nuclear Power Systems 1985*, Orbit Book Co., Melabar, FL, Chapter 20, 141–149.

Terrill, W. and L. R. Putnam (1985) "SP-100 Thermocouple Assembly, Design and Technology Development," in *Proceedings of the Second Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1985, EC-5.

Thome, F. (1992) "TSET," presented at the *Thermionic Space Nuclear Power Technical Program Review*, Phillips Laboratory, Albuquerque, NM, Nov. 9-10, 1992.

Tilliette, Z. P. and F. O. Carre (1990) "French Investigations on an Alternative, Longer Term, 20-kWe, Direct Cycle, Gas-Cooled Reactor, Space Power System," in *Proceedings Seventh Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1990, 1:508–513.

Topaz CoDR (1992) *Flight Topaz Conceptual Design Review, U.S. Air Force Phillips Laboratory*, review hold in Albuquerque, NM, Sept. 16-17, 1992.

Truscello, V. C. and L. L. Rutger (1992) "The SP-100 Power System," in *Proceedings Ninth Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1992, 1:1–23.

TSET/NEPSTP Workshop (1993) held in the Albuquerque Convention Center, sponsored by Phillips Laboratory (AFSC), 15 January 1993.

U.S. Topaz II Flight Safety Team (1992), *Topaz II Flight Program Safety Assessment,* Draft report Aug. 24, 1992.

Yasuda, H. et al. (1990) "Conceptual Study of a Very Small Reactor With Coated Particle Fuel," in *Proceedings Seventh Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1990, 1:103–107.

Yen, C. L. and C. G. Sauer (1991) "Nuclear Electric Propulsion for Future NASA Space Science Missions," in *AIDAA, AIAA/DGLR/ JSASS 22nd International Electric Propulsion Conference,* Paper No. IEPC-91-035, Viareggio, Italy, October 1991.

Yoder, G. L. and R. R. Graves (1985) "Analysis of Alkali Liquid Metal Rankine Space Power Systems," in *Proceedings of the Second Symposium on Space Nuclear Power Systems*, held in Albuquerque, NM, Jan. 1985, EC-4.

Zubrin, R. M. et al. (1992) "The Integrated Power and Propulsion Stage: A Mission Driven Solution Utilizing Thermionic Technology," in *Proceedings Ninth Space Nuclear Power Systems Symposium*, held in Albuquerque, NM, Jan. 1992, 3:1259–1267.