SPACE-REACTOR ELECTRIC SYSTEMS

SUBSYSTEM TECHNOLOGY ASSESSMENT

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Rockwell International
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SPACE-REACTOR ELECTRIC SYSTEMS: SUBSYSTEM TECHNOLOGY ASSESSMENT

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PREFACE

Planners in the cognizant United States government agencies are again evaluating the potential of reactor electric systems to provide power in space. This evaluation must be performed at the system level.

As part of the evaluation, the Department of Energy contracted with Rockwell International in September 1982 to conduct a system-level assessment of space reactor electric power. The end product of this assessment will be a plan to guide out-year development of these systems.

The first step in effectively understanding the system is to thoroughly understand the building blocks that make up the system — the subsystems. Rockwell has completed this step, and the results are presented in this Subsystem Technology Assessment Report.

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I. INTRODUCTION

A. BACKGROUND

Throughout the late 1950s, the 1960s, and the early 1970s, substantial development was performed on compact space reactor electric systems. A variety of reactor, power conversion, and heat rejection options were investigated and developed to varying degrees of operational capability. The first operational space reactor electric system — the SNAP-10A — was launched into earth orbit and operated in space in 1965 as part of this intensive national effort.

Due to the lack of an identified mission for these types of systems, the space reactor power effort dwindled substantially in the 1970s. In the past few years, however, interest has been rekindled as a result of higher power needs foreseen by both the military services and the civilian sector.

Although there is not yet a firmly defined mission for such systems, planners in the cognizant government agencies foresee the need to restart work on the space reactor electric system option. Deployment of such systems could be needed as early as the late 1980s. Due to the inherently long lead time required to develop, qualify, accept, and fly such systems, activities are now getting underway in DOE, NASA, and DOD to provide the management and technical groundwork necessary to the reestablishment of this option.

As part of these activities, DOE contracted with Rockwell International's Energy System Group (ESG) to conduct a system-level assessment of space reactor electric power early in FY 1983. The end product of this assessment will be a plan to guide out-year development of space reactor electric systems. The subsystem technology assessment presented in this report is the first major step of Rockwell's system-level assessment.

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B. PURPOSE AND SCOPE

This report documents the subsystem technology assessment. For the purpose of this report, five subsystems were defined for a space reactor electric system, and the report is organized around these subsystems:

- Reactor
- Shielding
- Primary heat transport
- Power conversion and processing
- Heat rejection

The major components and equipment used in these subsystems are shown in Figure I-1.

The purpose of the assessment was to determine the current technology status and the technology potentials for different types of the five subsystems. The cost and schedule needed to develop these potentials were estimated, and sets of "development-compatible" subsystems were identified. Development compatibility is an important quality. It means that specific subsystems (e.g., a reactor with a power converter and a heat rejector) are compatible with each other: (1) technically (e.g., temperatures, powers, materials, fluids) and (2) schedularly (the compatibility occurs at the same point in time). This first "from the bottom up" development compatibility step provides the framework for a technology roadmap to be further refined "from the top down" during the planned system-level assessment.

The scope of the assessment was oriented to identify these sets of development-compatible subsystems. The assessment encompassed four primary activities:

- Technology review and survey
- Analysis and technology projection

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Figure I-1. The Five Subsystems of a Reactor Electric System

- Cost and schedule estimation
- Development compatibility determination

Approaches taken in these activities are discussed in Section I.C.

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C. METHOD OF APPROACH

The approach that was used to perform the subsystem technology assessment is shown in Figure I-2.

The technology review and survey activity was performed by Rockwell personnel with expertise and past experience in the subsystem of interest. These personnel reviewed available literature to enhance their background knowledge. A list of industrial firms and national laboratories that had been or were currently active in space power was formulated. Thirty-two firms and laboratories were identified, and formally contacted by Rockwell with a letter that explained the purpose of the subsystem technology assessment. The letter requested assistance in providing more data on technologies and more insight into the potentials of technologies. Of the 32 organizations contacted, 23 responded favorably and were selected for surveys, as shown in Table I-1. One- to four-man teams from ESG performed the 23 surveys in November and December 1982.

The analysis and technology projection activity was performed by the survey team members. Subsystem technologies were reviewed and analyzed to determine when they had been worked on and how far they had progressed in the development cycle. The current status of generic subsystem technologies was determined, considering such important factors as past development status, vintage of the technology, and status of the technology infrastructure. Starting from the current status, the team members identified the logical steps in development for each generic subsystem technology.

The cost and schedule estimating activity was based on both historical data from the Systems for Nuclear Auxiliary Power (SNAP) program and current data for subsystem technologies. Time-dependent cost trends were established from this data base. A methodology for estimating the cost and schedule needs

Figure I-2. Approach to the Subsystem Technology Assessment

TABLE I-1
SUBSYSTEM TECHNOLOGY ASSESSMENT — SURVEY SUMMARY

Contacted	Surveyed	Subsystems Surveyed				
		Reactor	Shielding	Primary Heat Transport	Power Conversion and Processing	Heat Rejection
Aerojet Energy Conversion Company	X	X			X	
Argonne National Laboratory						
Battelle Memorial Institute	X	X		X		X
Battelle Pacific Northwest Laboratories	X	X		X		
Boeing Aerospace Company	X				X	
Brookhaven National Laboratory	X	X				
GA Technologies Incorporated	X	X			X	
Garrett Turbine Engine Company	X			X	X	X
General Electric Company - King of Prussia	X				X	
General Electric Company - Sunnyvale	X	X				
Grunman Aerospace						
Hughes Aircraft						
Jet Propulsion Laboratory	X				X	
Lawrence Livermore National Laboratory	X	X				
Los Alamos National Laboratory	X	X		X	X	X
Mechanical Technology Incorporated	X				X	
NASA-Lewis Research Center	X			X	X	X
Oak Ridge National Laboratory	X	X	X	X	X	X
Pratt and Whitney						
Rasor Associates	X	X			X	
Rockwell International - Energy Systems Group	X	X	X	X	X	X
Rockwell International - Rocketdyne Division	X				X	
Sundstrand Energy Systems	X				X	
Teledyne Energy Systems	X				X	
Thermacore, Incorporated	X			X		X
Thermo Electron Corporation	X			X	X	X
TRW Aircraft Components Group						
TRW Space and Technology Group						
Vought Corporation						
University of Washington	X					X
Westinghouse Advanced Reactors Division						
Westinghouse Hanford Company	X	X		X		

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of projected technologies was developed. Using the step-wise technology projections identified for generic subsystems, development costs and schedules were estimated for subsystem technologies applicable through the turn of the century.

The development compatibility determination activities were then performed to complete the subsystem technology assessment. Groups of development-compatible subsystem technologies were identified by comparing the technical compatibility and the schedular availability of the various types of subsystems assessed. This led to a determination on where to focus the upcoming system-level assessment. The results from this determination, and the overall cost and schedule requirements for the subsystem technologies, are summarized in Section II.

D. REPORT CONTENTS

This report contains six sections:

- I. Introduction
- II. Summary of Findings
- III. Study Basis
- IV. Subsystem Technology Assessment
- V. Subsystem Development Projections
- VI. Development Compatible Subsystem Technologies

Sections I and II act as an "Executive Summary" for the report. They are bound separately and also included in the total report. They summarize the methods and findings of the assessment.

Section III presents the techniques used in the assessment.

Section IV presents the technology assessment by subsystem. First, background information on each subsystem is presented. Special design or other considerations concerning the subsystem are then presented. The various types of technologies that have been used or studied for this subsystem are described. Lastly, generic types of subsystems are defined and assessed in terms of the present state of their technology and how that technology can progress through time. The key technical issues that pace this progression are discussed in detail.

Section V presents the schedule and cost estimates for developing each generic type of subsystem technology.

Section VI presents the groups of development-compatible subsystem technologies that were identified: current technology groups for near-term needs, improved technology groups for mid-term needs, and advanced technology groups for far-term needs.

II. SUMMARY OF FINDINGS

A. SUBSYSTEM TECHNOLOGY STATUS

The purpose of this section is to summarize the technology status for the five major subsystems that comprise a space reactor electric system:

- 1) Reactor subsystem
- 2) Shielding subsystem
- 3) Primary heat transport subsystem
- 4) Power conversion and processing subsystem
- 5) Heat rejection subsystem.

Detailed information on the status of the technology is presented in Section IV for each of these subsystems.

In defining the status of a technology, the descriptive ratings given in Table II-1 were used to define where the subsystem stood when work was terminated.

One point should be kept in mind when reviewing the status of those subsystem technologies that might be used for future space reactor electric systems. With few exceptions, subsystem development work has been at a standstill since 1973, when the U.S. program was terminated. This hiatus has significantly depleted the government and industry infrastructure of space reactor electric system expertise. In the meantime, however, a substantial government/industry infrastructure has developed in liquid metal fast breeder reactor (LMFBR), high-temperature gas-cooled reactor (HTGR), and power conversion technologies for terrestrial use. This expertise has had a substantial influence on the space reactor electric systems that are being considered. For example, today's proposed reactor subsystems are fast reactors, and a substantial data base for fuels and materials exists for LMFBRs and HTGRs. Material compatibilities for these LMFBRs and HTGRs are understood, and this understanding is backed by long-term test and operating data.

TABLE II-1
DESCRIPTIVE RATINGS

Subsystem Technology Descriptive Rating ^a	Description	Comments
0	Idea	No study
1	Early concept	Simple studies
2	Concept	Improved studies
3	Conceptual design	Component development needs identified
4	Early preliminary design	Scoping component development tests performed
5	Preliminary design	Component development tests well underway
6	Final design	Critical technology development complete and design approved for fabrication
7	Fabricated	Assembled and checked out
8	Ground tested	Significant testing done at power and temperature
9	Flight qualified	Components and subsystem qualified in system environment
10	Flown and operated	Acceptance tested and produced power in space

^aNote that the rating refers to the status of the technology as of the termination point, not to the status of the technology today. For much of the technology, "picking up the pieces" to reestablish the technology infrastructure will probably be a time-consuming process.

The following sections summarize the technologies for the five subsystems.

1. Reactor Subsystem

Table II-2 summarizes the reactor subsystems for space electric power production that existed at the termination of the U.S. program in 1973, plus a

TABLE II-2
SUMMARY OF REACTOR SUBSYSTEMS TECHNOLOGY

DESIGNATION	POWER LEVEL (kWt)	MAXIMUM OPERATING TEMPERATURE [K (°F)]	FUEL/CLADDING	COOLANT	CONTROL METHOD	FULL POWER TEST OPERATING TIME (h)	TECHNOLOGY DESCRIPTIVE RATING										
							0	1	2	3	4	5	6	7	8	9	10
SPACE REACTOR CONCEPTS																	
FIXED BED REACTOR (FBR)	300,000	1500 (2240)	HTGR PARTICULATE FUEL	He (or H ₂)	PERIPHERAL DRUMS	0											
GULF GENERAL ATOMIC THERMIONIC REACTOR	120 kW _e	THERMIONIC SYSTEM	90 UC - 10 ZrC Nb - 1 Zr	NaK (FOR TFE WASTE HEAT)	18 PERIPHERAL BeO SHUTTERS	0											
MEDIUM POWER REACTOR EXPERIMENT (MPRE)	1000	1111 (1540)	UO ₂ 316 SS	BOILING K	4 MOVABLE BOTTOM BeO PLUGS	0											
NASA LEWIS FAST REACTOR	2170	1279 (1843)	UN, T-111	Li	6 PERIPHERAL FUEL/T-111 POISON DRUMS	0											
SNAP 50/SPUR (PWAR-20)	2200	1367 (2000)	UN, Cb 1 Zr - 0.1 C	Li	6 PERIPHERAL BeO SEGMENTS	0											
SPACE ARENA BASELINE REACTOR (SABRE)	1470	1500 (2240)	UO ₂ TILES, Mo-Re	HEAT PIPES	12 PERIPHERAL B ₄ C DRUMS & CENTRAL B ₄ C SAFETY PLUG	0											
MINIPIPE	1470	1500 (2240)	UO ₂ TILES, Mo Re	MINIPIPES	12 PERIPHERAL B ₄ C DRUMS & CENTRAL B ₄ C SAFETY PLUG	0											
SPR-4 & SPR-5	2000	1500 (2240) (HEAT PIPE EVAP)	UN, W 25% Re	HEAT PIPES	Mo SIDE REFLECTORS +12 Li - 6 IN-CORE RODS	0											
SPR-6	57000	1650 (2510)	UN, W - 25% Re	Li	Mo SIDE REFLECTOR	0											
SNAP 5 kW _e	110	923 (1200)	U - ZrH HASTELLOY X	NaK	2 PERIPHERAL Be SLIDING SEGMENTS	0											
OPERATIONAL SPACE REACTORS																	
SNAP 2	65	923 (1200)	U - ZrH, HASTELLOY N	NaK	4 PERIPHERAL Be DRUMS	11,000											
SNAP 8	600	978 (1300)	U - ZrH, HASTELLOY N	NaK	6 PERIPHERAL Be DRUMS	15,000											
SNAP 10A	43	833 (1040)	U - ZrH HASTELLOY N	NaK	4 PERIPHERAL Be DRUMS	11,000											
APPLICABLE GROUND-BASED REACTORS																	
AVR	46000	1223 (1741)	U Th, GRAPHITE COATED	He	IN-CORE RODS	>70,000 IN OPERATION											
ML 1	3000	920 (1200)	UO ₂ /BeO, HASTELLOY X	NITROGEN	IN CORE BLADES	2,500											
ANP REACTORS																	
HEAT TRANSFER REACTOR EXPERIMENT 1 (HTRE 1)	20000	1000 (1350)	UO ₂ /Ni/Cr CERMET 80 Ni/20 Cr + Ni	AIR	IN CORE RODS	151											
HTRE 2		MODIFIED HTR-1 FOR USE AS A TEST REACTOR		AIR	Eu ₂ O ₃	TEST REACTOR											
HTRE 3	30,000	1000 (1350)	UO ₂ /Ni/Cr CERMET 80 Ni/20 Cr + Nb	AIR	30 SHIMS, 3 REGS, 15 SAFETIES (IN CORE)	126											
PROPULSION REACTORS																	
710 REACTOR	220,000	2480 (4000)	UO ₂ /W, TANTALUM	HYDROGEN	9 PERIPHERAL BeO DRUMS	0											
KIWI	1,000,000	1900 (3000)	UC ₂ CARBON COATED SPHERES EMBEDDED IN GRAPHITE	HYDROGEN	PERIPHERAL DRUMS	0.6											
LIGHT BULB REACTOR	4,600,000	6700 (11,600)	UF ₆ PLASMA	HYDROGEN	Ne PRESSURE/FLOW	0											
METALLIC ANNULAR ROCKET REACTOR SYSTEM (MARRS)	4,400,000	3033 (5000)	UN - W CERMET, W 3% Re	HYDROGEN	CENTRAL CONTROL ROD	0											
NRX	1,000,000	2220 (3540)	SAME AS KIWI	HYDROGEN	PERIPHERAL DRUMS	4.5											
NuERA II	275,000	1255 (1800)	UN, ASTAR B11C	Li	12 PERIPHERAL B ₄ C DRUMS AND 7 IN CORE SHIMS	0											
PEWEE	500,000	2555 (4140)	SAME AS KIWI	HYDROGEN	PERIPHERAL DRUMS	1.3											
PHOEBUS	4,100,000	2200 (3500)	SAME AS KIWI	HYDROGEN	PERIPHERAL DRUMS	2.5											
PLASMA	NOT REPORTED	45,000 (80,000)	U 233/H ₂ PLASMA	HYDROGEN	NOT REPORTED	0											
PLUTOTORY II C	500,000	1450 (2150)	UO ₂ /BeO CERAMIC TUBES	AIR	14 IN CORE HI RODS + 6 SAFETY RODS	0.1											
ROTATING BED REACTOR (RBH)	5,000,000	3000 (4940)	ADVANCED PARTICULATE FUEL	H ₂ (OPEN CYCLE)	PERIPHERAL DRUMS	0											
SMALL NUCLEAR ROCKET ENGINE (SNRE)	500,000	2700 (4397)	UC - ZrC	HYDROGEN	12 BeO PERIPHERAL DRUMS	0											

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few other pertinent current reactors and reactor concepts. Table II-1 shows that the technology fell into two broad categories: (1) conceptual reactor subsystems that had advanced to either the conceptual or early preliminary design phase (descriptive ratings up to 4) and (2) reactor subsystems that had actually been built and tested (descriptive ratings of 7 or higher). The single exception to this broad categorization is the in-core thermionic reactor: it progressed to the point of single thermionic fuel element (TFE) development testing but did not progress to actual reactor fabrication or testing.

Of those reactor subsystems falling in the second category (rating of 7 or greater), a further subdivision into propulsion reactors and space electric power production reactors is evident. The propulsion reactors (ROVER/NERVA, PLUTO, ANP) were relatively high-power, high-temperature, short-duration reactor subsystems, whereas the space electric power reactors (specifically SNAP UZrH) were lower power, lower temperature, and longer duration reactors.

2. Shielding Subsystem

Current technology for the shielding subsystem is a lithium hydride neutron shield encased in stainless steel. This type of shield received substantial development as part of the UZrH programs in the 1960s and 1970s and has been developed to a high degree of readiness. The major operating limit associated with the use of lithium hydride for a neutron shield is associated with its ~960 K (1266°F) melting temperature. Future space reactor electric systems will operate at temperatures above the lithium hydride melting point. Such systems could either incorporate active cooling of a solid shield or make provisions in the design to allow operation of the shield in the molten condition.

Gamma shielding developed in the past used heavy metal shadow shields. The major penalty associated with the use of such materials for the gamma shield is, of course, the mass burden. Nevertheless, judicious engineering of the gamma shield can at least minimize this mass penalty.

In summary, the shielding subsystem materials and fabrication technology are well developed. For reactor subsystems with high operating temperature, active cooling methods for the neutron shield or the use of a molten shield will require development effort.

3. Primary Heat Transport Subsystem

The primary heat transport subsystems that have received the majority of past development are those associated with convective heat transport using either liquid metals or gasses as the heat transfer fluid. Recently, work has begun at LANL on the potential use of heat pipes for transporting thermal energy from the reactor subsystem to the power conversion subsystem (PCS).

Regarding liquid metal heat transport subsystems, both dc electromagnetic and ac linear induction pumps were developed to a high degree, at temperatures up to 922 K (1200°F), during the previous space reactor electric programs and, more recently, during development of the fast breeder reactor. Some development has been performed on pumps for higher temperatures. Similarly, volume accumulators and piping for liquid metal heat transport subsystems were proven in 978 K (1300°F) environments as part of the SNAP ZrH program. SNAP-50 and other programs investigated refractory metal piping at higher temperatures. Use of liquid metals in higher temperature space reactor electric systems will no doubt require certain development, particularly in the area of materials (component fabrication and interaction/mass transfer effects).

For gas heat transport subsystems, turbomachinery (specifically compressors) has been developed, as part of the ongoing effort for commercial jet engines, to temperatures in the range of approximately 1250 K (1800°F). Bearings, seals, and fan technology appear to be in place to allow final proof-of-principle testing of prime movers for gas heat transport subsystems in the range of approximately 1250 K (1800°F). The technology for using gas heat transport subsystems up to 1250 K (1800°F) seems to be well developed.

Heat pipes are attractive for heat transport subsystems because they contain no moving parts and provide an internal fluid transport mechanism. However, it appears that considerable development work remains before heat pipes can be considered a proven technology for use in primary heat transport subsystems.

4. Power Conversion and Processing Subsystem

The power conversion technology summarized in Table II-3 relates to five generic types of devices:

- 1) Thermoelectric
- 2) Thermionic
- 3) Brayton cycle
- 4) Rankine cycle
- 5) Stirling cycle.

Most of the work performed since the U.S. space reactor electric system program was terminated in 1973 has focused on the thermoelectric subsystem (used extensively in radioisotope thermoelectric generators), the closed-cycle Brayton (for terrestrial applications and the Brayton Isotope Power System), the open-cycle Brayton (used in commercial jet engines and for central power station load-peaking units), the organic Rankine (for the Dynamic Isotope Power System), and thermionics for terrestrial applications.

PCS efficiencies can range from 1.8% (as for the SNAP-10A thermoelectric converters) to over 40% (in certain turboelectric converters). An insight to efficiency can be gained by considering it in two parts: (1) the Carnot efficiency (for an ideal "heat engine") and (2) the fraction of Carnot efficiency that is actually achieved. This distinguishes the purely thermodynamic efficiency from the subsystem or "device" efficiency.

The Carnot efficiency (η_c) is a function of the hot and cold operating temperatures for the power conversion cycle. The device efficiency (η_D) is

TABLE II-3

SUMMARY OF POWER CONVERSION SUBSYSTEMS TECHNOLOGY

TYPE	DESIGNATION	MATERIAL OR WORKING FLUID	MAXIMUM OPERATING TEMPERATURE [K (°F)]	RANGE OF DEVICE EFFICIENCY ^a (%)	SPECIFIC MASS (kg/kWe)	TOTAL TEST HOURS		TECHNOLOGY DESCRIPTIVE RATING										
						ONE UNIT	ALL COMPONENTS	0	1	2	3	4	5	6	7	8	9	10
STATIC	<u>THERMOELECTRICS</u>																	
	SNAP-10A Si-Ge	BINARY-DOPED Si-Ge	772 (930)	10-11 (EARLY 60's TECH)	120 (INCLUDES RADIATOR)	~40,000	1,000,000											
	RTG Si-Ge	BINARY-DOPED Si-Ge	1273 (1830)	16-17	20	~80,000	1,300,000											
	RTG TERNARY-DOPED Si-Ge	GaP-DOPED Si-Ge	1273 (1830)	18-19	16	—	200											
	HIGH-PERFORMANCE	NEW	1570 (2370)	20-25	—	—	—											
	SABRE CONVERTER	NEW	1355 (1980)	20-22	2	—	—											
	SNAP LEAD TELLURIDE COMPACT	PbTe	858 (1090)	15-16	20	>40,000	800,000											
	RTG TAGS	TAGS - PbTe	805 (990)	17-18	37	~80,000	1,000,000											
	<u>THERMIONICS</u>																	
	G A IN-CORE	Cs	1820 (2810)	18-20	5	12,000	65,000											
DYNAMIC	GE IN-CORE	Cs	1820 (2810) est.	18-20 est.	—	10,000	—											
	USSR TOPAZ IN-CORE	Cs	1873 (2910)	12-14	11	6,000	7,000,000											
	OUT-OF-CORE	Cs	1800 (2780)	25-34	3	2,200	6,600											
	<u>TURBOELECTRIC</u>																	
	SNAP-2 MERCURY RANKINE	Hg	895 (1150)	18-20	38	21,000	35,000											
	SNAP-8 MERCURY RANKINE	Hg	980 (1306)	21-23	33	12,000	420,000											
	6-kWe ORGANIC RANKINE	DOWTHERM A	617 (650)	52-55	—	3,000	28,000											
	DYNAMIC ISOTOPE POWER SYSTEM ORGANIC RANKINE	DOWTHERM A	645 (700)	40-42	52	11,000	20,000											
	NASA POTASSIUM RANKINE	K	1420 (2100)	54-56	10	—	19,000											
	MEDIUM POWER REACTOR EXPERIMENT POTASSIUM RANKINE	K	1110 (1540)	53-56	5	—	160,000											
	ADVANCED MPRE POTASSIUM RANKINE	K	1365 (2000)	62-65	5	—	0											
	NASA B ENGINE BRAYTON	He - Xe	1145 (1600)	40-43	27	41,000	41,000											
	TURBOALTERNATOR COMPRESSOR BRAYTON (TAC)	He - Xe	1145 (1600)	45-48	4	—	—											
	MINI-BRU BRAYTON (BIPS)	He - Xe	1025 (1385)	35-38	59	1,100	1,100											
	<u>RECIPROCATING</u>																	
	10-50 kWe SPACE FREE PISTON STIRLING	He	978 (1300)	65-70	8	—	—											
	TERRESTIAL FREE PISTON STIRLING	He	980 (1300)	55-60	—	2,000	16,000											

^aDEVICE EFFICIENCY MUST BE MULTIPLIED BY THE CARNOT EFFICIENCY TO GIVE OVERALL SUBSYSTEM EFFICIENCY.

a function of irreversible losses in the power converter itself, such as mechanical/frictional losses, pressure losses, and electrical losses.

The overall PCS efficiency is the product of the Carnot efficiency and the device efficiency, $\eta_C \cdot \eta_D$. The heat rejection temperature of a PCS is usually selected on the basis of system mass and radiator area. At low rejection temperatures, the Carnot efficiency is high and the benefits of high efficiency can be obtained at the price of larger and therefore more massive radiators. High radiator temperatures, however, lead to smaller and less massive radiators because of the T^4 effect but pay the price of lower overall efficiencies. There is usually some intermediate heat rejection temperature that leads to an optimum overall power system. This temperature depends mainly on the power system requirements and the type of PCS selected.

To further emphasize the point, a review of historical space reactor electric power system designs reveals that even for the wide device efficiency ranges for various PCSs, the overall specific power (We/kg) of system designs fell into a fairly narrow band. This is a very important observation. It illustrates the fact that many design variables interact at the system level. Judgments about the "goodness" or "desirability" of one subsystem relative to another based on a single, isolated variable (such as device efficiency, for example) often lead to erroneous conclusions in the absence of system-level assessments.

Thermoelectrics are a proven static technology for RTGs. But they suffer from the disadvantages of being costly in multikilowatt quantities and of having a relatively low device efficiency. In addition, specific converter designs must be evolved for reactor electric system applications.

Thermionics, which were developed in the 1960s and early 1970s, are high-temperature devices for the direct conversion of reactor thermal power to electric power. Thermionics, like thermoelectrics, are static converters, but in-core converters can suffer from the disadvantage of limited life (due to

fuel swelling and radiation effects). They operate at a higher device efficiency than thermoelectrics, but are in a less well developed state.

Closed-cycle inert-gas Brayton subsystems were being developed for space applications in the 1960s and 1970s. This development was well underway in the 10-kWe power range for temperatures up to 1090 K (1500°F) at program termination. The development has continued in the 1970s and 1980s for isotope and terrestrial heat sources. The closed-cycle Brayton operates at a moderate device efficiency for a turboelectric machine but requires a relatively low heat rejection temperature. Brayton subsystems would require specific design and development for space reactor electric system use.

Open-cycle gas Brayton machines would be used for high-level semicontinuous electric power in space. The commercial jet engine industry uses these up to temperatures of 1250 K (1800°F), but they would require significant development for semicontinuous space use.

Rankine-cycle PCSs were developed using both two-phase liquid metal and two-phase organic working fluids. The Rankine subsystem, of course, has millions of hours of operating experience in central power stations. The organic cycle was well developed in a 3-kWe size for isotope power systems; however, it is temperature limited to 670 K (750°F) by its working fluid. The liquid metal (mercury and potassium) cycles were in development. The use of a two-phase organic or liquid metal PCS in the zero-gravity space environment introduces questions that still need to be resolved. The Rankine turboelectric equipment has moderate device efficiencies and relatively high heat rejection temperatures.

The reciprocating Stirling engine has yet to be fully developed. Its advantages are a high device efficiency and operation with a single-phase gas. It has the disadvantage of a relatively low heat rejection temperature. The Stirling engine is regarded as requiring additional development before it can be used in a space reactor electric system.

5. Heat Rejection Subsystem

Waste heat from the PCS must be rejected by radiation to space. Heat rejection subsystems received extensive development in the 1960s, especially in the areas of emissivity coatings and special low-mass geometries to enhance micrometeoroid protection. Direct-conducting (radiating), pumped-fluid, and heat-pipe subsystems have been designed and tested. Deployable radiator subsystems that would have significantly increased the available areas were on the drawing boards in the late 1960s and early 1970s. Currently, advanced concepts, such as the liquid droplet radiator, are receiving much attention, but are still in the feasibility study phase. In the near- and mid-term, traditional radiators will be used in most systems until the advanced radiators are through the development cycle.

6. Conclusions – Subsystem Technology Status

If it were required to launch an operational space reactor electric system by the late 1980s, the recommended system (based on the results of this subsystem technology assessment) would necessarily have an established technology base and extensive test and operating experience. Selection of subsystems for this specific system could only be performed at the system level, based on a firm set of system functional requirements. However, few subsystems are at the necessary stage of development.

The development and testing performed under the ROVER program has demonstrated a workable reactor subsystem with potential for driving a semi-continuous multimewatt space reactor electric system. Development of the open-cycle Brayton PCS for this application is not as far advanced as the reactor subsystem, but a flight-qualified system might be possible by the early 1990s.

B. SUBSYSTEM TECHNOLOGY PROJECTIONS

This section projects the improvements and advancements that might, with renewed development activity, be possible for various subsystem technologies. The methodology for making these projections is presented in Section V. These projections refer only to what might be accomplished and not to the advisability of attempting to do so. Those judgments must await the results of system-level assessments to be performed during the remainder of FY 1983.

Briefly, the subsystem technology projections are based on (1) the current status of the technology and (2) assessments of the time required to develop potential improvements in the technology (based on a review of key technology issues and on the views and opinions of knowledgeable people in the various subsystem disciplines both from government and industry). Cost projections are based on extrapolations of historical cost data from the SNAP UZrH reactor and other reactor and associated PCS programs. The historical costs have been escalated to 1983 dollars using U.S. Department of Labor data. Costs for certain items, such as thermoelectric converters, Brayton PCSs, etc., have been based on recent manufacturers' cost estimates.

A generic schedule was used as a baseline for defining subsystem development steps and the time necessary to develop, qualify, and acceptance test a subsystem. This schedule, shown in Figure II-1, is based on actual schedule data from the SNAP-2, -8, and -10A programs.

In interpreting the subsystem technology projections, it is important to understand which cost elements are included in the projections and which are not. The following cost elements are included:

- Subsystem scoping and development tests
- Subsystem design and engineering
- Subsystem fabrication and ground demonstration testing
- Subsystem component qualification testing

Figure II-1. Generic Development Schedule

- Fabrication and delivery (to a system prime contractor) of a subsystem for incorporation into a ground demonstration system for qualification testing
- Fabrication and delivery (to a system prime contractor) of three flight subsystem mockups for system qualification testing
- Fabrication and delivery (to a system prime contractor) of three acceptance-tested flight subsystems
- All ancillary and support costs for the above, including program management, program administration, quality assurance, etc.

The following cost elements are not included:

- Prime system contractor activities
- System design, and qualification and acceptance testing
- Test facilities and major manufacturing facilities
- Aerospace safety program activities
- Launch equipment and support activities
- Ground demonstration system testing
- Flight system ground demonstration testing

1. Reactor Subsystem

The reactor subsystem technology projections for a representative thermal power level associated with a 100-kWe class system are shown in Figure II-2. The projections fall into three separate and distinct levels of technology: current technology (ready for use in a space reactor electric system by the late 1980s or early 1990s); improved technology (ready by the mid-1990s) that would have higher operating temperatures, improved materials, and longer operating lifetimes; and advanced technology (available by the late 1990s) that would have still higher operating temperatures, still more advanced materials, and still longer lifetimes. As expected, the costs increase with increasing development time.

Figure II-2. Reactor Subsystems Technology Projections

2. Shielding Subsystem

The technology projections for the shielding subsystem are shown in Figure II-3. As noted in Section II.A.2, the shielding subsystem is considered to be developed to a reasonable state of readiness, and therefore the cost and associated development time required for any additional improvements would be minimal compared to that for the reactor subsystem.

3. Primary Heat Transport Subsystem

The technology projections for the primary heat transport subsystem are shown in Figure II-4. These projections also fall naturally into the three levels – current technology, improved technology, and advanced technology.

The major factor driving the primary heat transport technology projections is the increase in reactor subsystem operating temperatures that would be associated with advancements in that technology. Increasing temperatures would require development work on heat transport subsystem pumps, with some limited work on volume accumulators for liquid-metal subsystems and gas recirculators for gas-cooled subsystems. Development projections for heat pipes as a potential improvement in space reactor electric system heat transport are also shown.

4. Power Conversion and Processing Subsystem

Technology projections for 100-kWe class dynamic and static PCSs are shown in Figures II-5 and II-6, respectively.

For dynamic PCSs, the typical cost versus time projection seen for the reactor subsystem and the heat transport subsystem is readily evident. The technology projections for the dynamic PCS include not only increased

Figure II-3. Shielding Subsystems Technology Projections

9467-180

Figure II-4. Primary Heat Transport Subsystems Technology Projections

Figure II-6. Static Power Conversion Subsystems Technology Projections

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operating temperature but also increased subsystem device efficiency and overall subsystem lifetime.

Technology projections for the organic Rankine have not been included in the envelope of projections associated with other types of dynamic PCSs. The reason is that the organic Rankine cycle is limited in operating temperature to approximately 670 K (750°F) due to thermal decomposition of the organic working fluid. This technology block would seem to preclude significant advancement of the organic Rankine subsystem.

Figure II-6 shows the technology projections for 100-kWe class static PCSs (specifically, thermoelectrics and thermionics). This technology projection is seen to depart significantly from the trends associated with the reactor subsystems, heat transport subsystems, and dynamic PCSs. The trend of cost versus time is nearly flat for the static PCS technology projections because the development costs for evolving from current to improved to advanced technology levels are small relative to the production costs of the converters. This results from development being performed on a converter scale rather than a full subsystem scale. The production costs literally overwhelm the development costs, and hence the projected increase in overall subsystem costs with increasing technology level is minor. However, the value of this — on a system level — must still be evaluated.

5. Heat Rejection Subsystem

Figure II-7 shows the technology projections for the heat rejection subsystem. Note that the projected cost for developing an advanced technology heat rejection subsystem is based on an estimate for a liquid droplet radiator using a titanium structure. The need for developing such advanced technology for the radiator subsystem remains to be established (during the system-level assessment).

Figure II-7. Heat Rejection Subsystems Technology Projections

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Current technology for the heat rejection subsystem and those increases in performance associated with improved technology (specifically higher operating temperatures, alternate materials, and the development of condensing radiators as might be associated with some of the Rankine PCSs) are considered achievable at relatively low costs compared with those for the reactor and the dynamic PCSs.

6. Conclusions – Technology Projections

When we began this subsystem technology assessment, three levels of technology were defined – current, improved, and advanced. The results of the assessment have necessitated a redefinition of the time frames for these three levels of technology (due largely to the need to reestablish the technology infrastructures):

Original Guidelines			Revised Definition
Level of Technology	Mission Need	Deployment Time	Deployment Time
Current	Near term	Late 1980s	Late 1980s thru early 1990s
Improved	Mid term	Early 1990s	Mid 1990s
Advanced	Far term	Beyond 1995	Late 1990s

To some extent, "current technology" is "current" in name only. The government and industrial infrastructures that existed in 1973 have been largely dissipated since then. Rebuilding that infrastructure, i.e., "picking up the pieces," can be a long, time-consuming process. For purposes of this report, it has been assumed that the infrastructure can be reestablished in a timely fashion so as to support deployment of current technology subsystems in the time frame indicated.

System costs will obviously depend on (1) specific selections for the reactor, shielding, primary heat transport, power conversion, and heat rejection subsystems and (2) the level of technology to be used. Other cost elements, such as prime contractor and safety activity cost, would be included.

Estimation of the system costs is, strictly speaking, beyond the scope of this subsystem assessment. Nevertheless, judgments have been made, and the rough-order-of-magnitude (ROM) estimates for a 100-kWe system are as follows:

Technology Level	Estimated System Costs (billions of 1983 dollars)
Current technology	1/2 to 1-1/2
Improved technology	3/4 to 2-1/4
Advanced technology	1-1/2 to >3

Semicontinuous multimegawatt systems could probably be available in the early 1990s, with more advanced versions following in the middle to late 1990s. Savings associated with the lack of need for heat rejection subsystems would be offset by higher costs for the open-cycle Brayton PCS. The costs for these semicontinuous multimegawatt systems would, therefore, roughly approximate the estimates quoted above for 100-kWe systems available in the same time frames.

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C. DEVELOPMENT COMPATIBLE SUBSYSTEM TECHNOLOGIES

1. Discussion

Of the five major subsystems, two dominate from the standpoints of technical complexity, cost, and schedule: the reactor subsystem and the power conversion (and processing) subsystem. These two comprise more than 80% of the estimated cost for any total subsystem development program. The schedules for the remaining three subsystems (shield, primary heat transport, and heat rejection) support the schedules for the reactor and PCSs.

Table II-4 summarizes the development compatibility of the reactor and power conversion subsystems. Coded symbols are used to identify the compatibility or noncompatibility on technical and schedular bases. This table summarizes both the technical and schedular development compatibility among nine generic reactor subsystems and seven generic PCSs. This encompasses the total range of interest from today through the turn of the century.

Table II-4, though a little time consuming to understand, provides a real framework for a technology roadmap that can be used to focus out-year development planning. This planning will take place during the forthcoming system-level assessment. Table II-4 has been "built from the bottom up" using technology projections that reflect what can be done, not necessarily what should be done. During the system-level assessment, the table will be modified "from the top down" to define what needs to be done (knowing, from this subsystem technology assessment, what can be done). This modification will be the technology roadmap for planning out-year development on space reactor electric systems.

In Section IV, technology assessments are made for each of the subsystems. These technology assessments are presented in the form of "circle charts" as an aid to the reader in understanding the projected technology

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TECHNOLOGY LEVEL			TEMPERATURE [K (°F)]	TD (%)	LIFE (year)	POWER PER UNIT (kWt)	COMMENTS
C ^a	I ^b	A ^c					

THERMOELECTRICS						
✱		1250 (1790)	15 16	7	MODULAR	BINARY DOPED SiGe
✱		858 (1085)	15 16	5		LEAD TELLURIDE COMPACT CON- VERTER
✱		850-820 (950 1015)	17 18	7		TAGS CONVERTER
✱		1350 (1970)	16 17	2		BINARY DOPED SiGe* (TERNARY DOPED SiGe IS AN ALTERNATE
✱	✱	1550 OR (2343) 1350 (1970)	20 22 OR 18 19	2 OR 7		HI PERFORMANCE CONVERTER (U ₂ S ₃ OR Bi ₂ Cl ₃) OR TERNARY DOPED SIDE

THERMIONICS					
✿	1800 (2780)	14-20	1	MODULAR	IN CORE UO ₂ WITH W EMITTER
		20-24	2		IN CORE VENT TO GAP OXY D CONV
✿	1800 (2780)	25-27	2		OUT O ₂ -CORE OXY D CONV
✿	1600 1800 (24-25) (2780)	25-30	2 7		IN CORE ADV CONVERTER
		28-34	2 5		OUT O ₂ CORE ADV MAT LS

1100 (1520)	38 42	2	30
	1200 (1700)	43 45	3	60
		1300- 1400 [1800- 2000]	48 56	4	100-200

630 (575)	42 45	1	20
	640 (590)	47 51	2	50
		660 670 (725 745)	52 58	3 4	100 200

●		980 (1300)	22 25	1	35
	●	960 (1300)	28 32	2	90
		1000 (1340)	36 40	3	120

1110 (1540)	55-56	1	20 30
	1280 (1845)	45-50	1	20 30
	1330- 1380 (1930- 2025)	52-58	2,3	50 100

970 (1785)	60-64	1	16
	1010 (1360)	67-70	4	50
	1050 (1430)	70-75	7	100	

LIQUID METAL COOLED
THERMAL REACTORS

WATER-COOLED REACTOR	U-Zr	HASTELLOY N	520 (1200)
☼			
☼	U-Zr	HASTELLOY N WITH IMPROVED BARRIER	1000 (1340)
☼	U-Zr	SEPARATE FUEL MODERATOR COOLING SYSTEMS	1100 (1020)

INERT GAS COOLED REACTORS

		GRAPHITE PARTICLE PURL	NA	1200 (1700)	3
●	●	●	NA	1200 (1800)	5

GAS COOLED FAST REACTORS

●		UO ₂ BeO	MASTELLOY X	200 (1700)	1
●	●	UO ₂	REFRACTORY	1300 (1800)	2
●	●	U ₂ OR GASET	REFRACTORY	1500 (2240)	3

LIQUID METAL COOLED FAST REACTORS[illegible]

BOILING LIQUID METAL
FAST REACTORS

BOILING K FAST REACTORS
UO ₂	STAINLESS STEEL	1110 11540	3
	UN	REFRACTORY	1365 (2000)	7

THERMIONIC REACTORS

THERMIONIC REACTORS				
☛		UO ₂	REFRACTORY	1800 (2760) 1
	☛	UO ₂ Mo FOIL	REFRACTORY	1800 (2760) 2
	☛	UO ₂ -Mo PARTICLE	REFRACTORY	1800 (2760) 5 7

HEAT PIPE REACTORS

HEAT PIPE REACTOR
UO ₂	NA/SUPERALLOY HEAT PIPE	1100 (1520)	2
UO ₂	LMo Re HEAT PIPE	1500 (2240)	2
IMPROVED UO ₂ OR PARTICLE	HEAT PIPE OR LI/WIRE HEAT PIPE	1500 (2240)	4

STATIC PCS SUBSYSTEMS

DYNAMIC PCS SUBSYSTEMS

LEGEND

C^a CURRENT TECHNOLOGY (LATE 80 s TO EARLY 90 s)
I^b IMPROVED TECHNOLOGY (MID 90 s)
A^c ADVANCED TECHNOLOGY (LATE 90 s)

SCHEDULE COMPATIBILITY WITH
REASONABLE (WITHIN 170 K (~300°F))
TECHNICAL COMPATIBILITY

SCHEDULE COMPATIBILITY WITH ACHIEVABLE BUT POOR TECHNICAL COMPATIBILITY

☐ SCHEDULE INCOMPATIBILITY
☐ TECHNICAL COMPATIBILITY

☐ SCHEDULE INCOMPATIBILITY WITH
REASONABLE TECHNICAL COMPATIBILITY

SCHEDULE INCOMPATIBILITY WITH ACHIEVABLE BUT POOR TECHNICAL COMPATIBILITY

Table II-4. Development Compatibility Matrix for the Two Key Subsystems (Reactor and power conversion subsystems)

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steps that are judged to be reasonable versus time for each of the various subsystems. Using Table II-4 and these circle charts, Figures II-8, II-9, and II-10 illustrate potential systems that could be utilized for near-, mid-, and far-term applications. The potential systems are depicted vertically in each figure and are formed by sets of development-compatible (i.e., technically and schedularly compatible) subsystems. There are more potential far-term systems than either near- or mid-term systems because, for the advanced technology projected for the late 1990s, there are more potential options for the various subsystems.

The estimated dates when the systems would be available for flight testing are shown at the bottom of each column. The time frames by which subsystems technology advancement steps are judged to be attainable are dictated by the time required for resolving the key technical issues for the subsystems (see Section IV).

Figures II-8 through II-10 are intended as an aid to the reader's perception of potential systems that could be deployed in the near, mid, and far terms. It is to be recognized that these systems are indicative of what could be done, not necessarily what should be done. That judgment must await the results of the system-level assessments to be performed in the latter half of FY 1983.

2. Conclusions – Development Compatibility

Close inspection of Table II-4 reveals that most of the reactor and power conversion subsystems that could be flight qualified in the late 1980s through the mid-1990s are projected to be developed for the 920 to 1250 K (1200 to 1800°F) operating temperature range. Advanced subsystems for the late 1990s would be targeted for higher operating temperatures.

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REACTOR SUBSYSTEM	NaK COOLED ZrH REACTOR			HTGR REACTOR			GAS COOLED FAST REACTOR			Na OR K COOLED FAST REACTOR			BOILING K FAST REACTOR	THERMIONIC REACTOR	HEAT PIPE REACTOR			
POWER CONVERSION SUBSYSTEM	LOW TEMPERATURE THERMOELECTRICS		FREE PISTON STIRLING	HIGH TEMPERATURE THERMOELECTRICS	CLOSED CYCLE BRAYTON TURBOELECTRIC	POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	LOW TEMPERATURE THERMOELECTRICS		FREE PISTON STIRLING	LOW TEMPERATURE THERMOELECTRICS		FREE PISTON STIRLING	POTASSIUM RANKINE TURBOELECTRIC	THERMIONIC CONVERTER	HI TEMPERATURE THERMOELECTRICS (RADIATIVE COUPLING)	CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	FREE PISTON STIRLING (HEAT EXCHANGER REQUIRED)
SHIELD SUBSYSTEM	LiH NEUTRON SHIELD PASSIVE COOLING			LiH NEUTRON SHIELD ACTIVE COOLING			LiH NEUTRON SHIELD PASSIVE COOLING			LiH NEUTRON SHIELD PASSIVE COOLING			Li NEUTRON SHIELD ACTIVE COOLING	LiH NEUTRON SHIELD PASSIVE COOLING	LiH NEUTRON SHIELD ACTIVE COOLING			
HEAT TRANSPORT SUBSYSTEM	PUMPED LIQUID METAL			PUMPED GAS			PUMPED GAS			PUMPED LIQUID METAL			PUMPED LIQUID METAL	PUMPED LIQUID METAL	Na/SUPERALLOY HEAT PIPES			
HEAT REJECTION SUBSYSTEM	DIRECT CONDUCTING RADIATOR		NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR	GAS RADIATOR	CONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR		GAS RADIATOR	DIRECT CONDUCTING RADIATOR		NONCONDENSING RADIATOR	CONDENSING RADIATOR	NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR	GAS RADIATOR	CONDENSING RADIATOR	NONCONDENSING RADIATOR
ESTIMATED AVAILABILITY DATE	1987 1989	1989 1990	1989 1991	1991 1992	1991 1992	1991 1992	1987 1989	1987 1989	1989 1991	1990-1991	1990 1991	1990 1991	1991 1992	1988-1990	1991 1992	1991 1992	1991 1992	1991 1992

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REACTOR SUBSYSTEM	NaK COOLED ZrH REACTOR		HTGR REACTOR			GAS COOLED FAST REACTOR			FIXED BED REACTOR			Na OR K COOLED FAST REACTOR			Li COOLED FAST REACTOR			THERMIONIC REACTOR	HEAT PIPE REACTOR
	<div>IMPROVED H₂ BARRIER 1000 K (1340°F) OUTLET 1500 kWt 7 year LIFE</div> <div>1991 1993</div>	<div>IMPROVED H₂ BARRIER 1000 K (1340°F) OUTLET 1500 kWt 7 year LIFE</div> <div>1991 1993</div>	<div>IMPROVED ZrC COATING 10,000 kWt 1300 K (1880°F) OUTLET 5 year LIFE</div> <div>1993 1995</div>	<div>IMPROVED ZrC COATING 10,000 kWt 1300 K (1880°F) OUTLET 5 year LIFE</div> <div>1993 1995</div>	<div>IMPROVED ZrC COATING 10,000 kWt 1300 K (1880°F) OUTLET 5 year LIFE</div> <div>1993 1995</div>	<div>REFRACTORY CLAD UO₂ FUEL FAST REACTOR 2000 kWt 1300 K (1880°F) OUTLET 2 year LIFE</div> <div>1993 1995</div>	<div>REFRACTORY CLAD UO₂ FUEL FAST REACTOR 2000 kWt 1300 K (1880°F) OUTLET 2 year LIFE</div> <div>1993 1995</div>	<div>REFRACTORY CLAD UO₂ FUEL FAST REACTOR 2000 kWt 1300 K (1880°F) OUTLET 2 year LIFE</div> <div>1993 1995</div>	<div>FBR INCONEL FRIT 4000 kWt 1100 K (1520°F) OUTLET 3 year LIFE</div> <div>1992 1994</div>	<div>FBR INCONEL FRIT 4000 kWt 1100 K (1520°F) OUTLET 3 year LIFE</div> <div>1992 1994</div>	<div>FBR INCONEL FRIT 4000 kWt 1100 K (1520°F) OUTLET 3 year LIFE</div> <div>1992 1994</div>	<div>Na OR K COOLED SUPERALLOY CLAD UO₂ 1100 K (1520°F) OUTLET 2000 kWt 7 year LIFE</div> <div>1992 1994</div>	<div>Na OR K COOLED SUPERALLOY CLAD UO₂ 1100 K (1520°F) OUTLET 2000 kWt 7 year LIFE</div> <div>1992 1994</div>	<div>Na OR K COOLED SUPERALLOY CLAD UO₂ 1100 K (1520°F) OUTLET 2000 kWt 7 year LIFE</div> <div>1992 1994</div>	<div>Li COOLED REFRACTORY CLAD UN 1350 K (1970°F) OUTLET 2000 kWt 2 year LIFE</div> <div>1993 1995</div>	<div>Li COOLED REFRACTORY CLAD UN 1350 K (1970°F) OUTLET 2000 kWt 2 year LIFE</div> <div>1993 1995</div>	<div>Li COOLED REFRACTORY CLAD UN 1350 K (1970°F) OUTLET 2000 kWt 2 year LIFE</div> <div>1993 1995</div>	<div>UO₂-Mo FOIL FUEL, OXYGENATED CONVERTER WITH VENT TO GAP 2 year LIFE η_D = 20-24%</div> <div>1992 1993</div>	<div>Li/MoRe HEAT PIPES UO₂ FUEL 1500 kWt 1500 K (2240°F) EVAPORATOR 2 year LIFE</div> <div>1994-1996</div>
POWER CONVERSION SUBSYSTEM	<div>MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>FREE PISTON STIRLING</div>	<div>HIGH TEMPERATURE THERMOELECTRICS</div>	<div>CLOSED CYCLE BRAYTON TURBOELECTRIC</div>	<div>POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>HIGH TEMPERATURE THERMOELECTRICS</div>	<div>CLOSED CYCLE BRAYTON TURBOELECTRIC</div>	<div>POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>CLOSED CYCLE BRAYTON TURBOELECTRIC</div>	<div>MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>FREE PISTON STIRLING</div>	<div>CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>FREE PISTON STIRLING</div>	<div>HIGH TEMPERATURE THERMOELECTRICS</div>	<div>CLOSED-CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)</div>	<div>THERMIONIC CONVERTER</div>	<div>HIGH TEMPERATURE THERMOELECTRICS</div>
	<div>T = 980 K (1300°F) η_D = 28-32% LIFE = 2 YEARS Pe = 90 kWe</div> <div>1991 1993</div>	<div>T = 1010 K (1360°F) η_D = 67-70% 4-YEAR LIFE Pe = 50 kWe</div> <div>1992 1994</div>	<div>BINARY DOPED SiGe* T_H = 1350 K (1970°F) 2 year LIFE η_D = 16-17%</div> <div>1992 1993</div>	<div>NIOBIUM ALLOY MATERIALS T = 1200 K (1700°F) LIFE = 3 years Pe 60 kWt η_D = 43-45%</div> <div>1991 1993</div>	<div>T = 1280 K (1845°F) η_D = 45-50% LIFE = 1 year Pe = 20-30 kWt</div> <div>1993 1994</div>	<div>BINARY DOPED SiGe* T_H = 1350 K (1970°F) 2 year LIFE η_D = 16-17%</div> <div>1992 1993</div>	<div>NIOBIUM ALLOY MATERIALS T = 1200 K (1700°F) LIFE = 3 years Pe 60 kWt η_D = 43-45%</div> <div>1991 1993</div>	<div>T = 1280 K (1845°F) η_D = 45-50% LIFE = 1 year Pe = 20-30 kWt</div> <div>1993 1994</div>	<div>NIOBIUM ALLOY MATERIALS T = 1200 K (1700°F) LIFE = 3 years Pe 60 kWt η_D = 43-45%</div> <div>1991 1993</div>	<div>T = 980 K (1300°F) η_D = 28-32% LIFE = 2 YEARS Pe = 90 kWt</div> <div>1991 1993</div>	<div>T = 1010 K (1360°F) η_D = 67-70% 4-YEAR LIFE Pe = 50 kWt</div> <div>1992 1994</div>	<div>NIOBIUM ALLOY MATERIALS T = 1200 K (1700°F) LIFE = 3 years Pe 60 kWt η_D = 43-45%</div> <div>1991 1993</div>	<div>T = 980 K (1300°F) η_D = 28-32% LIFE = 2 YEARS Pe = 90 kWt</div> <div>1991 1993</div>	<div>T = 1010 K (1360°F) η_D = 67-70% 4-YEAR LIFE Pe = 50 kWt</div> <div>1992 1994</div>	<div>BINARY DOPED SiGe* T_H = 1350 K (1970°F) 2 year LIFE η_D = 16-17%</div> <div>1992 1993</div>	<div>NIOBIUM ALLOY MATERIALS T = 1200 K (1700°F) LIFE = 3 years Pe 60 kWt η_D = 43-45%</div> <div>1991 1993</div>	<div>T = 1280 K (1845°F) η_D = 45-50% LIFE = 1 year Pe = 20-30 kWt</div> <div>1993 1994</div>	<div>UO₂-Mo FOIL FUEL, OXYGENATED CONVERTER WITH VENT TO GAP 2 year LIFE η_D = 20-24%</div> <div>1992 1993</div>	<div>BINARY DOPED SiGe* T_H = 1350 K (1970°F) 2 year LIFE η_D = 16-17%</div> <div>1991 1993</div>
SHIELD SUBSYSTEM	<div>LiH NEUTRON SHIELD PASSIVE COOLING</div>		<div>Li NEUTRON SHIELD ACTIVE COOLING</div>			<div>Li NEUTRON SHIELD ACTIVE COOLING</div>			<div>LiH NEUTRON SHIELD ACTIVE COOLING</div>			<div>LiH NEUTRON SHIELD ACTIVE COOLING</div>			<div>LiH NEUTRON SHIELD ACTIVE COOLING</div>			<div>LiH NEUTRON SHIELD PASSIVE COOLING</div>	<div>LiH NEUTRON SHIELD ACTIVE COOLING</div>
	<div>LiH CAST IN SS VESSEL T_{MAX} < 930 K PASSIVE COOLING</div> <div>1986-1988</div>	<div>LiH CAST IN SS VESSEL T_{MAX} < 930 K PASSIVE COOLING</div> <div>1986-1988</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>	<div>LiH CAST IN SS VESSEL T_{MAX} < 930 K PASSIVE COOLING</div> <div>1986-1988</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991 1993</div>
HEAT TRANSPORT SUBSYSTEM	<div>PUMPED LIQUID METAL</div>		<div>PUMPED GAS</div>			<div>PUMPED GAS</div>			<div>PUMPED GAS</div>			<div>PUMPED LIQUID METAL</div>			<div>PUMPED LIQUID METAL</div>			<div>PUMPED LIQUID METAL</div>	<div>Li/Mo Re HEAT PIPE</div>
	<div>OUTLET TEMPERATURE 1145 K (1600°F) SUPER ALLOY Na, Li EM PUMP</div> <div>1985-1987</div>	<div>OUTLET TEMPERATURE 1145 K (1600°F) SUPER ALLOY Na, Li EM PUMP</div> <div>1985-1987</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1988-1990</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1988-1990</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1988-1990</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1988-1990</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1988-1990</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1988-1990</div>	<div>OUTLET TEMP 1145 K (1600°F) SUPER ALLOY NOBLE GAS BLOWER</div> <div>1985-1987</div>	<div>OUTLET TEMP 1145 K (1600°F) SUPER ALLOY NOBLE GAS BLOWER</div> <div>1985-1987</div>	<div>OUTLET TEMP 1145 K (1600°F) SUPER ALLOY NOBLE GAS BLOWER</div> <div>1985-1987</div>	<div>OUTLET TEMPERATURE 1145 K (1600°F) SUPER ALLOY Na, Li EM PUMP</div> <div>1991 1993</div>	<div>OUTLET TEMPERATURE 1145 K (1600°F) SUPER ALLOY Na, Li EM PUMP</div> <div>1991 1993</div>	<div>OUTLET TEMPERATURE 1145 K (1600°F) SUPER ALLOY Na, Li EM PUMP</div> <div>1991 1993</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED)</div> <div>1993 1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED)</div> <div>1993 1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED)</div> <div>1993 1995</div>	<div>OUTLET TEMPERATURE 922 K (1200°F) STAINLESS STEEL NaK EM PUMP</div> <div>1985-1987</div>	<div>EVAPORATOR TEMPERATURE 1500 K (2240°F) Mo Re Li 13 kW/cm²</div> <div>1991 1993</div>
HEAT REJECTION SUBSYSTEM	<div>CONDENSING RADIATOR</div>	<div>NONCONDENSING RADIATOR</div>	<div>DIRECT CONDUCTING (DEPLOYABLE)</div>	<div>DEPLOYABLE GAS RADIATOR</div>	<div>CONDENSING RADIATOR</div>	<div>DIRECT CONDUCTING</div>	<div>GAS RADIATOR</div>	<div>CONDENSING RADIATOR</div>	<div>DEPLOYABLE GAS RADIATOR</div>	<div>CONDENSING RADIATOR</div>	<div>DEPLOYABLE GAS RADIATOR</div>	<div>GAS RADIATOR</div>	<div>CONDENSING RADIATOR</div>	<div>NONCONDENSING RADIATOR</div>	<div>DIRECT CONDUCTING RADIATOR</div>	<div>GAS RADIATOR</div>	<div>CONDENSING RADIATOR</div>	<div>NONCONDENSING RADIATOR</div>	<div>DIRECT CONDUCTING RADIATOR</div>
	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>NONCONDENSING RADIATOR</div> <div>1986-1987</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1992</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1990</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1992</div>	<div>GAS-COOLED MODERATE SIZE RADIATOR</div> <div>1986-1988</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1990</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1990</div>	<div>GAS-COOLED MODERATE SIZE RADIATOR</div> <div>1986-1988</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>NONCONDENSING RADIATOR</div> <div>1986-1987</div>	<div>DIRECT CONDUCTING RADIATOR</div> <div>1985-1986</div>	<div>GAS-COOLED MODERATE SIZE RADIATOR</div> <div>1986-1988</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>NONCONDENSING RADIATOR</div> <div>1986-1987</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1992</div>
ESTIMATED AVAILABILITY DATE	1991 1993	1992 1994	1993 1995	1993 1995	1993 1995	1993 1995	1993 1995	1993 1995	1992 1994	1992 1994	1992 1994	1992 1994	1992 1994	1992 1994	1993 1995	1993 1995	1993 1995	1991 1993	1994 1996

Figure II-9. Improved Technology for Mid-Term Needs Development Compatible Subsystems

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REACTOR SUBSYSTEM	NaK COOLED ZrH REACTOR		GAS COOLED FAST REACTOR						FIXED BED REACTOR		LI COOLED FAST REACTOR											
	<div>SEPARATE FUEL/MODERATOR COOLING 1100 K (1520°F) FUEL OUTLET 2000 kWt 7 year LIFE</div> <div>1997-1999</div>	<div>SEPARATE FUEL/MODERATOR COOLING 1100 K (1520°F) FUEL OUTLET 2000 kWt 7 year LIFE</div> <div>1997-1999</div>	<div>IMPROVED UC UN OR CERMET FUELS 10 000 kWt 1500 K (2240°F) OUTLET 3 year LIFE</div> <div>1996-1998</div>	<div>IMPROVED UC UN OR CERMET FUELS 10 000 kWt 1500 K (2240°F) OUTLET 3 year LIFE</div> <div>1996-1998</div>	<div>IMPROVED UC UN OR CERMET FUELS 10 000 kWt 1500 K (2240°F) OUTLET 3 year LIFE</div> <div>1996-1998</div>	<div>IMPROVED UC UN OR CERMET FUELS 10 000 kWt 1500 K (2240°F) OUTLET 3 year LIFE</div> <div>1996-1998</div>	<div>IMPROVED UC UN OR CERMET FUELS 10 000 kWt 1500 K (2240°F) OUTLET 3 year LIFE</div> <div>1996-1998</div>	<div>IMPROVED UC UN OR CERMET FUELS 10 000 kWt 1500 K (2240°F) OUTLET 3 year LIFE</div> <div>1996-1998</div>	<div>FBR HIGH TEMPERATURE FRIT 50 000 kWt 1300 K (1880°F) OUTLET 5 year LIFE</div> <div>1998-1999</div>	<div>FBR HIGH TEMPERATURE FRIT 50 000 kWt 1300 K (1880°F) OUTLET 5 year LIFE</div> <div>1998-1999</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE</div> <div>1995-1996</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1500 K (2240°F) OUTLET MORE COMPACT 3000 kWt 3 year LIFE</div> <div>1997-1999</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE</div> <div>1995-1996</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1500 K (2240°F) OUTLET MORE COMPACT 3000 kWt 3 year LIFE</div> <div>1997-1999</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE</div> <div>1995-1996</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1500 K (2240°F) OUTLET MORE COMPACT 3000 kWt 3 year LIFE</div> <div>1997-1999</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE</div> <div>1995-1996</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1500 K (2240°F) OUTLET MORE COMPACT 3000 kWt 3 year LIFE</div> <div>1997-1999</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE</div> <div>1995-1996</div>	<div>LI-COOLED REFRACTORY CLAD UN ADDITIVES 1500 K (2240°F) OUTLET MORE COMPACT 3000 kWt 3 year LIFE</div> <div>1997-1999</div>		
POWER CONVERSION SUBSYSTEM	MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	FREE PISTON STIRLING	HIGH TEMPERATURE THERMOELECTRICS		CLOSED CYCLE BRAYTON TURBOELECTRICS		POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)		CLOSED CYCLE BRAYTON TURBOELECTRIC		HIGH TEMPERATURE THERMOELECTRICS				CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)				POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)			
	<div>T = 1000 K (1340°F) η_D = 36-40% LIFE = 3 YEARS Pe = 120 kWt</div> <div>1995-1997</div>	<div>T = 1050 K (1430°F) η_D = 70-75% 7 YEAR LIFE Pe = 100 kWt</div> <div>1995-1997</div>	<div>RARE EARTHSULFIDES OR BORON CARBONS T_H = 1550 K (2330°F) 2 year LIFE η_D = 20-22%</div> <div>1998-2000</div>	<div>SiGe (GaP) T_H = 1350 K (1970°F) 7 year LIFE η_D = 18-19%</div> <div>1998-2000</div>	<div>REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η_D = 48-52%</div> <div>1995-1997</div>	<div>CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η_D = 53-56%</div> <div>1999-2001</div>	<div>T = 1330 K (1935°F) η_D = 52-55% LIFE = 2 years Pe = 50 kWt</div> <div>1996-1997</div>	<div>T = 1380 K (2025°F) η_D = 56-58% LIFE = 3 years Pe = 100 kWt</div> <div>1999-2001</div>	<div>REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η_D = 48-52%</div> <div>1995-1997</div>	<div>CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η_D = 53-56%</div> <div>1999-2001</div>	<div>SiGe (GaP) T_H = 1350 K (1970°F) 7 year LIFE η_D = 18-19%</div> <div>1998-2000</div>	<div>SiGe (GaP) T_H = 1350 K (1970°F) 7 year LIFE η_D = 18-19%</div> <div>1998-2000</div>	<div>REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η_D = 48-52%</div> <div>1998-2000</div>	<div>REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η_D = 48-52%</div> <div>1995-1997</div>	<div>CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η_D = 53-56%</div> <div>1999-2001</div>	<div>CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η_D = 53-56%</div> <div>1999-2001</div>	<div>REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η_D = 48-52%</div> <div>1995-1997</div>	<div>T = 1330 K (1935°F) η_D = 52-55% LIFE = 2 years Pe = 50 kWt</div> <div>1996-1997</div>	<div>T = 1380 K (2025°F) η_D = 56-58% LIFE = 3 years Pe = 100 kWt</div> <div>1999-2001</div>	<div>T = 1380 K (2025°F) η_D = 56-58% LIFE = 3 years Pe = 100 kWt</div> <div>1999-2001</div>	<div>T = 1330 K (1935°F) η_D = 52-55% LIFE = 2 years Pe = 50 kWt</div> <div>1996-1997</div>	
SHIELD SUBSYSTEM	LiH NEUTRON SHIELD ACTIVE COOLING		MOLTEN LiH NEUTRON SHIELD						MOLTEN LiH NEUTRON SHIELD		MOLTEN LiH NEUTRON SHIELD											
	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991-1993</div>	<div>LiH SHIELD T_{MAX} < 930 K ACTIVE COOLING</div> <div>1991-1993</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	
HEAT TRANSPORT SUBSYSTEM	PUMPED LIQUID METAL		PUMPED GAS						PUMPED GAS		PUMPED LIQUID METAL											
	<div>OUTLET TEMP 1145 K (1600°F) SUPER ALLOY NOBLE GAS BLOWER</div> <div>1989-1991</div>	<div>OUTLET TEMP 1145 K (1600°F) SUPER ALLOY NOBLE GAS BLOWER</div> <div>1989-1991</div>	<div>OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1991-1993</div>	<div>OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1991-1993</div>	<div>OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1991-1993</div>	<div>OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1991-1993</div>	<div>OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1991-1993</div>	<div>OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER</div> <div>1991-1993</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1988-1990</div>	<div>OUTLET TEMP 1310 K (1900°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1988-1990</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	
HEAT REJECTION SUBSYSTEM	CONDENSING RADIATOR	NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR		DEPLOYABLE GAS RADIATOR		CONDENSING RADIATOR		ADVANCED RADIATOR		DIRECT CONDUCTING RADIATOR				DEPLOYABLE GAS RADIATOR				CONDENSING RADIATOR			
	<div>CON DENSING RADIATOR</div> <div>1987-1989</div>	<div>NONCON DENSING RADIATOR</div> <div>1986-1987</div>	<div>DIRECT CONDUCTING CARBON/ CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>DIRECT CONDUCTING CARBON/ CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>GAS- COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS- COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>CON DENSING RADIATOR</div> <div>1987-1989</div>	<div>CON DENSING RADIATOR</div> <div>1987-1989</div>	<div>ADVANCED CONCEPTS LIQUID DROPLET MOVING BELT ELECTROSTATIC</div> <div>1995-1997</div>	<div>ADVANCED CONCEPTS LIQUID DROPLET MOVING BELT ELECTROSTATIC</div> <div>1995-1997</div>	<div>DIRECT CONDUCTING RADIATOR</div> <div>1985-1986</div>	<div>DIRECT CONDUCTING RADIATOR</div> <div>1985-1986</div>	<div>DIRECT CONDUCTING RADIATOR</div> <div>1985-1986</div>	<div>DIRECT CONDUCTING RADIATOR</div> <div>1985-1986</div>	<div>GAS- COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS- COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS- COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS- COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>CON DENSING RADIATOR</div> <div>1987-1989</div>	<div>CON DENSING RADIATOR</div> <div>1987-1989</div>	<div>CON DENSING RADIATOR</div> <div>1987-1989</div>	<div>CON DENSING RADIATOR</div> <div>1987-1989</div>
ESTIMATED AVAILABILITY DATE	1997-1999	1997-1999	1998-2000	1998-2000	1996-1998	1999-2001	1996-1998	1999-2001	1998-1999	1999-2001	1998-2000	1998-2000	1998-2000	1998-2000	1996-1998	1999-2001	1999-2001	1999-2001	1997-1999	1997-1999	1999-2001	1999-2001

Figure 11-10. Advanced Technology for Far-Term Needs — Development Compatible Subsystems
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REACTOR SUBSYSTEM	BOILING K FAST REACTOR		THERMIONIC REACTOR		HEAT PIPE REACTOR											
	<div>BOILING K COOLED REFRACTORY CLAD UN 1365 K (2000°F) OUTLET 5000 kWt 7 year LIFE</div> <div>1995-1997</div>	<div>BOILING K COOLED REFRACTORY CLAD UN 1365 K (2000°F) OUTLET 5000 kWt 7 year LIFE</div> <div>1995-1997</div>	<div>UO₂ Mo CERMET FUEL SUBLIMED Mo COLLECTOR, IMPROVED INSULATOR 2.4 year LIFE $\eta_D = 25-27\%$</div> <div>1994-1996</div>	<div>PARTICLE FUEL SUBLIMED Mo COLLECTOR, IMPROVED INSULATOR 4.7 year LIFE $\eta_D = 28-30\%$</div> <div>1997-1999</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>
POWER CONVERSION SUBSYSTEM	POTASSIUM RANKINE TURBOELECTRIC		THERMIONIC CONVERTER		HIGH TEMPERATURE THERMOELECTRICS				CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)				POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)			
	<div>T = 1330 K (1935°F) $\eta_D = 52-55\%$ LIFE = 2 years Pe = 50 kWt</div> <div>1996-1997</div>	<div>T = 1380 K (2025°F) $\eta_D = 56-58\%$ LIFE = 3 years Pe = 100 kWt</div> <div>1999-2001</div>	<div>UO₂-Mo CERMET FUEL, SUBLIMED Mo COLLECTOR, IMPROVED INSULATOR 2.4 year LIFE $\eta_D = 25-27\%$</div> <div>1994-1996</div>	<div>PARTICLE FUEL SUBLIMED Mo COLLECTOR, IMPROVED INSULATOR 4.7 year LIFE $\eta_D = 28-30\%$</div> <div>1997-1999</div>	<div>RARE EARTH SULFIDES OR BORON CARBONS T_H = 1550 K (2330°F) 2 year LIFE $\eta_D = 20-22\%$</div> <div>1998-2000</div>	<div>SiGe (GaP) T_H = 1350 K (1970°F) 7 year LIFE $\eta_D = 18-19\%$</div> <div>1998-2000</div>	<div>SiGe (GaP) T_H = 1350 K (1970°F) 7 year LIFE $\eta_D = 18-19\%$</div> <div>1998-2000</div>	<div>RARE EARTH SULFIDES OR BORON CARBONS T_H = 1550 K (2330°F) 2 year LIFE $\eta_D = 20-22\%$</div> <div>1998-2000</div>	<div>REFRACTORY MATERIALS T = 1300 K (1800°F) LIFE = 4 years Pe 100 kWt $\eta_D = 48-52\%$</div> <div>1995-1997</div>	<div>CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt $\eta_D = 53-56\%$</div> <div>1995-1997</div>	<div>CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt $\eta_D = 53-56\%$</div> <div>1999-2001</div>	<div>REFRACTORY MATERIALS T = 1300 K (1800°F) LIFE = 4 years Pe 100 kWt $\eta_D = 48-52\%$</div> <div>1997</div>	<div>T = 1330 K (1935°F) $\eta_D = 52-55\%$ LIFE = 2 years Pe = 50 kWt</div> <div>1996-1997</div>	<div>T = 1380 K (2025°F) $\eta_D = 56-58\%$ LIFE = 3 years Pe = 100 kWt</div> <div>1999-2001</div>	<div>T = 1380 K (2025°F) $\eta_D = 56-58\%$ LIFE = 3 years Pe = 100 kWt</div> <div>1999-2001</div>	<div>T = 1330 K (1935°F) $\eta_D = 52-55\%$ LIFE = 2 years Pe = 50 kWt</div> <div>1996-1997</div>
SHIELD SUBSYSTEM	MOLTEN LiH NEUTRON SHIELD		LiH NEUTRON SHIELD PASSIVE COOLING		MOLTEN LiH NEUTRON SHIELD											
	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>LiH CAST IN SS VESSEL T_{MAX} < 930 K PASSIVE COOLING</div> <div>1986-1988</div>	<div>LiH CAST IN SS VESSEL T_{MAX} < 930 K PASSIVE COOLING</div> <div>1986-1988</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>
HEAT TRANSPORT SUBSYSTEM	PUMPED LIQUID METAL		PUMPED LIQUID METAL		HEAT PIPES											
	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 922 K (1200°F) STAINLESS STEEL NaK EM PUMP</div> <div>1985-1987</div>	<div>OUTLET TEMPERATURE 922 K (1200°F) STAINLESS STEEL NaK EM PUMP</div> <div>1985-1987</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>
HEAT REJECTION SUBSYSTEM	CONDENSING RADIATOR				DIRECT CONDUCTING RADIATOR				DEPLOYABLE GAS RADIATOR				CONDENSING RADIATOR			
	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>NONCONDENSING RADIATOR</div> <div>1986-1987</div>	<div>NONCONDENSING RADIATOR</div> <div>1986-1987</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>
ESTIMATED AVAILABILITY DATE	1996-1998	1999-2001	1994-1996	1994-1996	1998-2000	1998-2000	1998-2000	1998-2000	1998-2000	1999-2001	1999-2001	1998-2000	1998-2000	1999-2001	1999-2001	1998-2000

Figure II-10. Advanced Technology for Far-Term Needs - Development Compatible Subsystems
(Sheet 2 of 2)

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The incentive for achieving higher operating temperatures is the higher Carnot efficiency and higher system power-to-mass ratio generally associated with higher temperatures. Historically, however, extracting added performance by using extreme cycle temperatures has given rise to more technical and programmatic problems than any other design variable. Higher operating temperatures are the sworn enemy of highly reliable, long-lifetime systems.

Based on these observations — the inherent difficulty of projecting long-term needs and the high risk of advancing too quickly to high operating temperatures — the conclusion of the development compatibility assessment is that

A desirable target for the development of reactor and power conversion subsystems for the foreseeable future (the next 5 to 10 years) is in the temperature range of 920 to 1250 K (1200 to 1800°F).

This conclusion is made without the benefit of firm system functional requirements. But, if reaffirmed by the system-level assessment, it should materially reduce the development costs and schedules vis-à-vis development of advanced technology systems (see Section II.B).

III. STUDY BASIS

A. ASSESSMENT TECHNIQUES

To categorize the information collected, the reactor electric system was divided into five subsystems as shown in Table III-1. The technology assessment was conducted at the subsystem level, to first assure that all subsystems were thoroughly understood. A system-level study is the next step and will build on the insights provided by this subsystem understanding. The main objective of the current effort was to characterize the technology on the subsystem level and derive "development-compatible" subsystem technologies that are technically and schedularly compatible. Actual subsystem selection can now be conducted at the system level using decision modeling techniques.

The information used to assess subsystems for a space reactor electric system was obtained primarily from literature surveys and personal visits to organizations considered to have an applicable technology base. The overall methodology used in obtaining and using this information is illustrated in Figure III-1.

The basic objectives and guidelines were first set; these are shown in Table III-2. Guidelines for power level were rather broad since mission studies have not been completed to narrow down the range. Two types of power with broad ranges were considered: a steady-state power (20 to 400 kWe) and a "semicontinuous" power (1 to 100 MWe). For deployment, the three levels of technology shown in the table were defined. The current technology level was defined as subsystems where a strong technology base exists (development has proceeded to the point where near-term mission needs could be met by deployment in the late 1980s). The improved and advanced technology levels were oriented to mid-term and far-term mission needs for deployment in the early 1990s and beyond 1995. The deployment times included not only subsystem development but also subsystem qualification for flight and for flight system production and acceptance testing. And, to aid in schedule planning, a full-scale engineering development start date in late 1984 was used.

TABLE III-1
SUBSYSTEMS OF THE REACTOR ELECTRIC SYSTEM

Subsystem	Components/Equipment
Reactor	Fuel, cladding, working fluid, vessel, reflectors, core portion of heat pipes, power control mechanisms
Shielding	Gamma shield, neutron shield
Primary heat transport	Piping, heat pipes, pumps, working fluid, radiative subsystem transfer surfaces, volume accumulators, insulation
Power conversion and processing	Thermoelectric elements, turboelectric equipment, reciprocating cycle equipment, thermionic converters, power processing, power control, command-control-communications
Heat rejection	Radiative surface, radiative surface generator or structure, secondary heat transport equipment, micrometeoroid protection, special coatings, survival or masking equipment

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Figure III-1. Subsystem Technology Assessment Methodology

TABLE III-2
SUBSYSTEM TECHNOLOGY ASSESSMENT — BASIC OBJECTIVE AND GUIDELINES

Basic objective: Assist DOE by assessing the technology of subsystems/components/equipment suitable for use in space reactor electric systems. This information will be used as building blocks to arrive at complete power systems through a system-level assessment.

Guidelines: (1) **Power level:** For preliminary guidance, a steady-state power level range from 20 to 400 kWe should be considered. A pulse, or "semicontinuous," power range from 1 to 100 MWe should also be considered.

(2) **Deployment:** For this assessment, three levels of technology are defined; these levels are tied to the deployment times shown below.

<u>Level of Technology</u>	<u>Mission Need</u>	<u>Deployment Time^{a,b}</u>
Current	Near term	Late 1980s
Improved	Mid term	Early 1990s
Advanced	Far term	Beyond 1995

(3) **Schedule:** Assume planning purposes that the start of full-scale engineering development will be late 1984.

^aDeployment time should be considered as in-space initial operating capability (IOC) and thus should include subsystem development and qualification times, and flight system production and acceptance testing times.

^bA conclusion of this assessment is that the deployment times should be extended to the early 1990s for current technology, the mid-1990s for improved technology, and the late 1990s for advanced technology.

As highlighted in Section II.B, a major conclusion from this assessment is that it may be necessary to extend the use of current technologies into the early 1990s, improved technologies into the mid-1990s, and advanced technologies into the late 1990s.

The next step in the assessment was to conduct a literature search for work previously conducted on space reactor electric subsystem technologies. The key documents identified were retrieved and reviewed. From the information obtained, we compiled a list of organizations that possessed or had developed applicable technology. These organizations were then contacted to obtain names of key personnel who had knowledge of the technology and to determine their participation in the technology assessment. We requested technology information from these organizations, as shown in Table III-3. This information included subsystem, component, and equipment description; performance; technology status; critical technologies; schedule; cost; and references. Teams of Rockwell engineers visited those organizations that possessed a significant technology data base. In these visits, information was collected and the technologies discussed. Rockwell personnel tried to understand what currently exists and what logically could be projected to exist in the future. The information collected from these organizations, along with that obtained from the literature search and that available within Rockwell's Energy Systems Group, was used to perform the subsystem technology assessment. The results were compiled for this report.

The assessment was performed by the same Rockwell engineers that visited the organizations. Subsystem technologies were reviewed and analyzed to determine when they had been worked on and how far they had progressed in the development cycle. A numerical method of rating this progression was established, as shown in Table III-4. This rating, called the "Subsystem Technology Descriptive Rating," refers to the development status of the subsystem technology when it was worked on (not the status today). As will become evident in Section IV, the rating (in pictorial form, see Table III-4) is used as an aid for the reader. Whenever a specific subsystem is being described, the text is preceeded by the pictorial descriptive rating.

TABLE III-3

DESIRED INFORMATION FOR ASSESSING SUBSYSTEMS/COMPONENTS/EQUIPMENT

Subsystems Defined:	(1) Reactor, (2) shielding, (3) primary heat transport, (4) power conversion and processing, and (5) heat rejection.
Description:	Describe each subsystem, component, and piece of equipment, including name, size, mass, materials, working fluids, etc.
Performance:	Project the performance of each subsystem/component/equipment, including temperature, flow rate, mass, efficiency, life expectancy, etc. Estimate the expected performance improvement (level of technology) that could reasonably be achieved by each of the deployment times in Table III-1.
Technology Status:	Summarize the subsystem/component/equipment technology status, including the development performed to date and that required through ground engineering testing and then through flight qualification. Provide the vintage of your technology and also the development required to reach each level of technology shown in Table III-1.
Critical Technologies:	Prioritize the critical technologies that need to be developed at each level of technology for each subsystem/component/equipment.
Schedule:	Provide a schedule from the start of full-scale engineering development through flight qualification for each deployment time. The schedule should include for each subsystem/component/equipment: development testing, flight qualification, flight production, and acceptance testing.
Cost:	Provide any information that is pertinent in establishing overall costs for the development, qualification, and production of the subsystem/component/equipment.
References:	Provide a list of references and any available reports or handouts describing the subsystem/component/equipment. Of particular interest are those establishing the actual technology developed to date.

TABLE III-4
SUBSYSTEM TECHNOLOGY DESCRIPTIVE RATING

EXAMPLE: DESCRIPTIVE RATING = 5.5

<u>RATING</u>	<u>DESCRIPTION</u>
0	IDEA (NO STUDY)
1	EARLY CONCEPT (SIMPLE STUDIES)
2	CONCEPT (IMPROVED STUDIES)
3	CONCEPTUAL DESIGN (COMPONENT DEVELOPMENT NEEDS IDENTIFIED)
4	EARLY PRELIMINARY DESIGN (SCOPING COMPONENT DEVELOPMENT TESTS PERFORMED)
5	PRELIMINARY DESIGN (COMPONENT DEVELOPMENT TESTS WELL UNDERWAY)
6	FINAL DESIGN (CRITICAL TECHNOLOGY DEVELOPMENT COMPLETE AND DESIGN APPROVED FOR FABRICATION)
7	FABRICATED (ASSEMBLED AND CHECKED OUT)
8	GROUND TESTED (SIGNIFICANT TESTING DONE AT POWER AND TEMPERATURE)
9	FLIGHT QUALIFIED (COMPONENTS AND SUBSYSTEM QUALIFIED IN SYSTEM ENVIRONMENT)
10	FLOWN AND OPERATED (ACCEPTANCE TESTED AND PRODUCED POWER IN SPACE)

The assessment then continued by defining generic subsystem technologies and determining their current status, considering such important factors as past development status, vintage of the technology, and status of the technology infrastructure. For many technologies, it was found that "picking up the pieces" to reestablish the infrastructure would be time consuming. Once the current status had been determined, the team members identified the logical steps in development for each generic subsystem technology, and the key technical issues which paced these steps. Again, a pictorial method was used to display the steps, as shown in Figure III-2. Qualitative technology improvement was plotted against the fiscal year when development and qualification of the generic subsystem technology would be complete. These pictorials appear in Section IV. Each circle depicts a technology step, and the specifics of the step are defined inside the circle. The size of the circle approximates the range on completion time. The technical issues that must be resolved to reach each step are discussed in the accompanying text.

Costs and schedules were then estimated for each defined technology step for each generic type of subsystem. The cost and schedule estimating activity was based on both historical data from the Systems for Nuclear Auxiliary Power (SNAP) program and current data for subsystem technologies. Time-dependent cost trends were established from this data base. A methodology for estimating the cost and schedule needs of projected technologies was developed, and development costs and schedules were estimated for subsystem technologies applicable through the turn of the century. These costs and schedules are for the development of the subsystem, the flight qualification of the subsystem, and the delivery of acceptance-tested flight subsystems for use in a flight system. For convenience, the cost estimates are based on subsystems that would support a 100-kWe class reactor electric system. These costs are tabulated in Section V.

The development compatibility of subsystems was then determined both for technical and schedular compatibility. Groups of such subsystem technologies

Figure III-2. Subsystem Technology Assessment Pictorials

were identified. These groups are tabulated and presented pictorially in Section VI for current, improved, and advanced technologies. The key technical issues associated with each generic technology are also tabulated. This provides a focus for the upcoming system-level assessment.

B. DATA BASE

The information presented in this report is based on available documentation and data collected through visits to various organizations.

1. Organizations Visited

Thirty-two organizations were contacted to determine the extent to which they had technology applicable to space reactor electric systems. Of the organizations contacted, 23 (plus ourselves) were visited to assure that Rockwell personnel understood the technology and treated it accurately. Table III-5 presents the organizations contacted and visited. Also included are the areas discussed with each organization.

2. Documentation Collected

Approximately 300 documents were obtained and reviewed. These documents were gathered primarily from the literature searches and from the organizations visited. In addition, Rockwell has about 800 retrievable documents that were generated as part of the Systems for Nuclear Auxiliary Power (SNAP) programs. Key documents from this collection were reviewed for this report. A list of the primary documents and contacts used for this subsystem technology assessment is presented in the References.

TABLE III-5
ORGANIZATIONS VISITED AND AREAS OF DISCUSSION
(Sheet 1 of 2)

Organizations Contacted	Visited	Areas of Discussion				
		Reactor	Shielding	Primary Heat Transport	Power Conversion and Processing	Heat Rejection
Aerojet Energy Conversion Company	X	Nuclear rocket, ML-1 (gas cooled)			Mercury Rankine, organic Rankine, Brayton, thermoelectric	
Argonne National Laboratory						
Battelle Memorial Institute	X	Fuel irradiation, UO ₂ , UN		Material evaluation		Material evaluation
Battelle Pacific Northwest Laboratories	X	Fuel irradiation		Liquid metal corrosion loops		
Boeing Aerospace Company	X				Power processing and distribution	
Brookhaven National Laboratory	X	Gas cooled, particle fueled				
GA Technologies, Inc.	X	In-core thermionic, gas cooled, coated particle fuel			Thermionic, thermoelectric	
Garrett Turbine Engine Company	X			Heat exchangers and ducting	Brayton, potassium Rankine	Heat exchangers and ducting
General Electric Company - King of Prussia	X				Thermoelectric, potassium Rankine	
General Electric Company - Sunnyvale	X	Liquid metal fast breeder, reactor safety				
Grumman Aerospace						
Hughes Aircraft						
Jet Propulsion Laboratory	X				Thermoelectric, power processing	
Lawrence Livermore National Laboratory	X	Gas-cooled BeO, liquid metal fast reactor, UN		Analysis capabilities		Analysis capabilities
Los Alamos National Laboratory	X	Heat pipe fast reactor, nuclear rocket		Heat pipes	Thermoelectric	Heat pipe radiator

TABLE III-5
ORGANIZATIONS VISITED AND AREAS OF DISCUSSION
(Sheet 2 of 2)

Organizations Contacted	Visited	Areas of Discussion				
		Reactor	Shielding	Primary Heat Transport	Power Conversion and Processing	Heat Rejection
Mechanical Technology Incorporated	X				Stirling engine, bearings, organic Rankine, Brayton	
NASA-Lewis Research Center	X			Gas ducting	Brayton, foil bearings, materials, potassium Rankine	Emissivity coatings
Oak Ridge National Laboratory	X	Gas cooled, refractory metal, fuel irradiation	Shield analysis and testing	Heat exchangers, insulation, heat pipes, liquid metal pumps	Potassium Rankine	Condensing radiators, heat pipes
Pratt and Whitney						
Rasor Associates	X	In-core thermionic			Thermionic	
Rockwell International Corporation - Energy Systems Group	X	UZrH, fuels, design, SNAP	LiH shields	Liquid metal loops	SNAP	SNAP radiators
Rockwell International Corporation - Rocketdyne	X				Open-cycle Brayton	
Sundstrand Energy Systems	X				Organic Rankine	
Teledyne Energy Systems	X				Thermoelectric	
Thermacore, Inc.	X			Heat pipes		Heat pipes
Thermo Electron Corporation	X			High temperature insulation, heat pipes	Thermoionic, thermoelectric	Emissivity coatings
TRW Aircraft Components Group						
TRW Space and Technology Group						
Vought Corporation						
University of Washington	X					Liquid droplet radiator
Westinghouse Electric Corporation						
Westinghouse Hanford Company	X	Fuel development		Liquid metal loops

IV. TECHNOLOGY ASSESSMENT

A. REACTOR SUBSYSTEM

1. Background

From the earliest days of nuclear power, reactors have been considered for a wide variety of applications to vehicle propulsion (airplanes, rockets, ships, etc.) and electrical power production for spacecraft and terrestrial locations. Many of the reactor concepts derived for these applications are described below. The main appeal of reactors for such applications, and in particular for space applications, is the ability to obtain a large amount of energy from a compact source with a minimum amount of fuel.

Many of the reactor concepts derived during the last several decades of reactor development did not materialize into proven designs. This often was due to a redirection of emphasis or funding. Many others did evolve into real systems and were subjected to full reactor testing. In the case of SNAP-10A, for example, the reactor-powered system was launched and operated in space.^{IV-1} But even though many of the concepts fell short of full development, these programs contributed to a large technology data base that future space reactor programs may draw on.

2. Reactor Design Considerations

The primary objective in space reactor design is to provide as much heat at as high a temperature as possible, while being compatible with power conversion and other subsystems. The heat must be provided in the smallest and lightest configuration possible with a high degree of reliability. The design process involves selecting compatible components, as discussed below, that when combined into a reactor subsystem will provide the power level, outlet temperature, and lifetime to meet this objective.

a. Fuel Assembly

The reactor design begins with the selection of a particular coolant, fuel type, and geometry. Fuel materials of interest are hydrided UZr alloy, UO_2 , UC, UN, and cermets. Geometries may be cylindrical rods or pins, spheres, flat wafers or plates, wires or ribbons, or other special shapes. Fuel pins normally have the fuel material encased in metallic cladding that is sealed at each end with a plug. The fuel inside the cladding may be solid rods, pellets, powder, or spheres. Particulate fuel, which consists of graphite-coated fuel kernels in small-diameter (500- μm) microspheres, may be used in the loose form or may be imbedded in a graphite matrix to form a solid fuel element. Unclad UO_2 wafers stacked around heat pipes are being proposed for the SABRE design.^{IV-2} Wires or ribbons consisting of a metallic cladding bonded to a fuel core by extrusion or rolling operations have been considered for high-temperature gas-cooled reactors. Hollow hexagonal-shaped cermet fuel was used in another gas-cooled reactor design.

The function of a fuel assembly is to contain the fuel and fission products to prevent contamination of the primary system and to maintain the integrity of the core. In some instances, the stability of the fuel depends on the maintenance of a barrier between the fuel and coolant.

Tradeoffs between neutronic and thermal/hydraulic considerations dictate the size of the core. A sufficient amount of fuel must be assembled into the core to maintain criticality for the desired lifetime. Consideration must be given to the change in reactivity due to temperature changes, the buildup of fission product poisons, and fuel depletion. Space must be provided in the core for the coolant to remove the heat. This dilution of the core requires additional fuel to offset the effects of parasitic absorption in the coolant and the increased leakage because of the reduced core density.

b. Control

Reactivity control of space reactors is usually provided by reflector control using drums in the radial reflector. The drums are used to regulate neutron leakage from the core. The configuration may be in the form of full or partial cylinders, shutters, or windows. Cylindrical drums sometimes use a poison such as B_4C on one side to vary the amount of neutron absorption in the reflector. Shutters and windows merely open holes in the reflector through which neutrons may leak into space.

In addition to reflector control, some reactor concepts use a burnable prepoison to aid in reactivity control. As fission product poisoning builds up, the prepoison is depleted to offset some or all of the reactivity effect. This method is not appropriate in fast spectrum reactors because of the small cross section of most elements for fast neutrons. This is not a particular disadvantage to fast reactors because Xe and Sm poison effects are not the problem in fast reactors that they are in thermal reactors.

The total amount of reactivity control provided must be greater than the requirements for temperature defect, fission products, and fuel depletion. In large thermal reactors, it becomes increasingly difficult to provide sufficient control over a long lifetime using reflector control elements. In these instances, poison rods in the core, as in conventional power plant reactors, must be considered. Although not impossible, the use of in-core control rods in a space reactor is not a proven technology.

The movement of control drums is provided by pneumatic or electric motors, with the latter being the most common. The rotational or translational motion is provided through appropriate gear or screw drives. Drive motors and drive trains were developed on the SNAP UZrH reactor program that can survive the rigors of space operations, but the demands of temperatures higher than those used in the zirconium-hydride reactor programs require further development. The same drums used for control are also used for rapid

shutdown (scram), actuated by stored energy and separate logic networks. The requirements for part-load rejection in large power systems may require the development of improved drive systems.

c. Heat Removal

The normal mode of heat removal from the core is by convection. This may be by way of pumped liquid metals or gases or by evaporation in a heat pipe. Direct conduction from the core through thermoelectric (TE) modules and then radiation to space were considered in SNAP-10^{IV-3} and the Russian ROMASHKA.^{IV-4} The power limitation of such concepts became apparent very early, and pumped systems took over. The mode of cooling in-core thermionic reactors may be thought of as a combination of convection cooling and electron cooling. Some of the energy produced is removed directly from the reactor by the electrical current produced by the thermionic elements. The remainder is removed by a pumped coolant.

The liquid metals seriously considered for space reactor coolants are NaK, K, and Li. Sodium, which is the coolant used for LMFBRs, never offered any particular advantages over NaK for space reactor applications. The UZrH reactors all used eutectic NaK 78 as the primary coolant. SNAP-50^{IV-5} and other large fast reactor concepts all considered Li because of their high operating temperatures. One concept, the MPRE,^{IV-6} considered a boiling K reactor with direct input of K vapor into a Rankine-cycle turbine.

The gases considered for coolants are H₂, He, and He-Xe mixtures. The early aircraft nuclear propulsion (ANP) reactors^{IV-7} and the nuclear ramjet (Pluto)^{IV-8} all used air as a coolant. This obviously would not be appropriate for space reactors. Hydrogen was the coolant used in the nuclear rocket (Rover) program^{IV-9} primarily because the coolant was also the propellant. Most closed-cycle gas-cooled reactors either use He or a He-Xe mixture.

d. Structure

The reactor core support structure serves to hold the fuel assemblies in place against the hydraulic forces of the coolant during operation in space and against the handling and launch forces during transport into space. Whether it is necessary to hold the reactor together during reentry is a problem that has not yet been fully resolved.

The structure consists of grid plates and/or tie rods that position and hold the fuel in place within the reactor vessel. In the rotating bed reactor (RBR) and fixed bed reactor (FBR) concepts, the fuel is held in place by porous frits.^{IV-10} Other parts of the in-reactor structure serve as fluid plenums and flow baffles to guide the coolant into and out of the core.

Other support structure is required to position and support the reflector and control drums around the core vessel and to support the entire reactor subassembly on the spacecraft. The reflector support structure sometimes contains a mechanism for releasing or ejecting the reflector as a means of ultimate reactor shutdown.

e. Containment

The reactor containment may be subdivided into containment of the fuel and containment of the core and primary coolant. The cladding on metal-clad fuel or the graphite coating on particulate fuel serves as the fuel containment. The cladding or coating serves to protect the fuel from the coolant and to contain fission products. In the UZrH reactor, the cladding has a ceramic coating which also serves as a containment for the hydrogen moderator in the fuel. In the SABRE reactor, the barrier between the fuel and the coolant is provided by the walls of the heat pipe.

The containment for the core and primary coolant (if applicable) is provided by the reactor vessel and primary heat transport subsystem (piping, valves, heat exchangers, turbine). The heat pipes serve this purpose in

SABRE. It has a reactor vessel, but the vessel does not serve as primary containment for either the core or the primary coolant because the vessel is vented to space.

f. Materials

Candidate materials for cladding and structural applications for space nuclear reactors fall into three classes of alloys: (1) stainless steels, (2) nickel-base alloys, and (3) refractory alloys. The stainless steels have been extensively studied over the past 20 years under the fast breeder and thermal reactor programs. A large data base exists for these alloys at modest operating temperatures. The nickel-base alloys can operate at higher temperatures, but their performance has not been as well characterized as the stainless steels. Refractory alloys offer the greatest potential for the high operating temperatures needed for space reactor applications. Advanced development programs are providing important mechanical property data for use in selecting optimum refractory alloys. Table IV-1 summarizes the performance and limitations of each group. Each alloy class is described below.

(1) Stainless Steels

The austenitic stainless steels have the largest data base, although most of it applies to the LMFBR temperature regime. The irradiation fluence levels expected for space applications should not seriously degrade the stainless steels. These alloys are temperature-limited by strength to about 1100 K (1520°F), and liquid metal coolant compatibility will reduce this limit by about 200 K (360°F).

(2) Nickel-Base Alloys

The temperature limit is several hundred degrees (K) higher for nickel-base alloys than for the stainless steels, but mass transfer in liquid metals is greater. Compatibility with the proposed fuel systems needs further

TABLE IV-1
EXPERIENCE WITH REACTOR ALLOYS

Alloy	Typical Applications	Fluence ($E > 0.1 \text{ MeV}$) (n/cm^2)	Fuel Compatibility			Coolant Compatibility ^a					Vacuum Compatibility, Temperature at 1×10^{-7} Torr Vapor Pressure (K)	Fabric- ability	Avail- ability ^b
			UO ₂	UC	UN	Na	Li	NaK	K	He			
<u>Stainless Steels</u>													
316 SS	Piping, structure	1×10^{23} at 850 K	A to 1143 K	A to 1143 K	A to 1573 K	A to 973 K	A to 773 K ^c	A to 973 K	A to 973 K	A to 1000 K	1175	E	C
D 9	Fuel cladding	1.3×10^{23} at 723 K	N	N	N	N	N	N	N	N	1175	D	D
(But expected to be similar to Type 316 SS)													
<u>Nickel Base Alloys</u>													
Nimonic PE16	Piping, fuel cladding	1×10^{23} at 923 K	A to 1000 K	N	N	A to 973 K	UA ^c >723 K	A to 973 K	A to 973 K	A to 1173 K	1175	E	C
Inconel 718	Piping, fuel cladding	8×10^{22} at 850 K	A to 1000 K	N	N	A to 970 K	UA ^c >723 K	A to 973 K	A to 973 K	A to 1170 K	1175	E	C
Inconel 625	Piping, fuel cladding	1.6×10^{22} at 727 K	N	N	N	A to 970 K	UA ^c >723 K	A to 973 K	A to 973 K	A to 1100 K	1175	E	C
<u>Refractory Alloys</u>													
Mo	Barrier, reflector	6×10^{22} at 1400 K	A to 1500 K	N	N	A to 1500 K	A to 1700 K	A to 1500 K	A to 1500 K	A to 2000 K	1975	P	C
Mo-14 Re	Fuel cladding	10^{22} at 1400 K	A to 2400 K	N*	N*	N*	N*	N*	N*	A to 1600 K	1975	D	D
Mo-50 Re	Fuel cladding	6×10^{22} at 1400 K	A to 2144 K	A to 1811 K	A to 2144 K	A to 1400 K	A to 1920 K	N*	N*	A to 1600 K	1975	I	C*
TZM or TZr	Structure	10^{22} at 1400 K	A to 2144 K	A to 1811 K	A to 2144 K	N*	A to 1920 K	A to 1473 K	A to 1530 K	A to 2200 K	1975	P	C
ASTAR 811C	Piping, structure	N	Q at 1600 K	N	N	N*	N*	N*	N*	A to 2100 K	1865	D	D
T 111	Piping, structure	1×10^{20} at 1673 K 2×10^{22} at 900 K	UA at 1600 K	UA at 1644 K	UA at 1572 K	A to 1473 K	A to 1643 K	A to 1473 K	A to 1473 K	A to 1600 K	1865	A	C
W-25 Re	Fuel cladding	1×10^{20} at 1573 K [†] 3.7×10^{20} at 1173 K	A at 1644 K	N	A to 1673 K	N*	A to 1643 K	N*	N*	A to 1800 K	2350	I ^d	C*
Nb-1 Zr	Piping, structure	5×10^{21} at 940 K 4×10^{22} at 720 K	UA at ^e 1273 K	Q at 1373 K	UA at ^e 1473 K	A to 1473 K	A to 1473 K	A to 1473 K	A to 1473 K	A to 1600 K	2140	A	C
W	Barrier	4×10^{22} at 720 K	A at 2000 K	A at 2000 K	N* at 2000 K	N* at 1700 K	N* at 1700 K	N* at 1700 K	N* at 1700 K		2520	P	C

^a Impurity levels for all coolants must be controlled especially for C, O, N, and H.

^b Has been produced commercially in the past but may not be in production now.

^c Data from the fusion program indicate 450 to 500 K may be the ceiling for a 7-year life system.

^d Fabricable with difficulty due to high temperatures involved.

^e Fuel pins containing 5 to 10 mil CVD W barriers have been irradiated with success.

Notes

A = Acceptable

UA = Unacceptable

N = No data found this time

Q = Questionable

N* = Expect to be A

† = Irradiated as cladding on fuel

E = Excellent

P = Poor, not weldable

D = Under development

C = Commercially available

I = Insufficient data for design applications

C* = Commercially available but not stocked

study. Nickel alloys tend to be more resistant to irradiation swelling, but, because of their more complex microstructures, fracture ductilities are generally reduced. For example, Nimonic PE 16 is a strong candidate for LMFBR applications but its ductility drops to near zero at 973 K (1291°F) after irradiation.

(3) Refractory Alloys

Refractory alloys have the highest temperature limit and hence have the greatest potential for space reactor applications at 1400 to 1700 K (2060 to 2600°F). Most of the data are for pure Mo, TZM, and molybdenum-rhenium alloys with some on tantalum alloys (ASTAR 811C and T111). Mo-14 Re alloy appears to offer the best combination of irradiation resistance and compatibilities with the relevant environments. However, this alloy is still in development.

3. Reactor Descriptions

a. Liquid-Metal-Cooled Reactors

(1) SNAP-2

The SNAP-2 program^{IV-11} was an outgrowth of studies made in 1951 by the Rand Corp. concerning the feasibility of nuclear reactor power supplies for satellite auxiliary power requirements. In 1956, specific objectives were established for a 3-kWe system using a UZrH reactor coupled to a mercury Rankine power conversion subsystem (PCS). UZrH was selected to minimize the U²³⁵ inventory for the reactor subsystem (U²³⁵ was a priority material for weapons use when SNAP-2 objectives were established). The contract for reactor development was awarded to Atomics International (AI), and Thompson Ramo Wooldridge was awarded the PCS development contract.

SNAP-2 reactor development at AI proceeded from a critical experiment to establish neutronic behavior through the power operation of two ground tests — the SNAP Experimental Reactor (SER) and the SNAP-2 Developmental Reactor (S2DR). The SER was on test for 10,000 h during the period from September 1959 to November 1960, and S2DR was on test for 11,000 h during the period from April 1961 to December 1962.

Despite the success of these ground demonstrations, emphasis on a 3-kWe system had been reduced, and the requirement for a SNAP-2 flight test was canceled in 1963. Because the SNAP-2 reactor had demonstrated capability in excess of that required for the SNAP-10 program, it was adopted for use with a TE PCS. This combination of the SNAP-2 reactor with the SNAP-10 PCS became known as SNAP-10A.

(2) SNAP-10A

The objective of the SNAP-10A program (summarized in Ref. IV-12) was to develop a nuclear reactor power system capable of producing a minimum of 500 We for a period of 1 year in a space environment. SNAP-10A was the first reactor-powered electrical system to be flight tested in earth orbit. The flight system, FS-4, ran for 43 days before an electrical malfunction in the Agena spacecraft caused the reactor to automatically shut down. An identical ground system, FS-3, ran continuously for 10,000 h before being shut down for examination.

The core of the SNAP-10A reactor consisted of uranium-zirconium alloy fuel elements clad in stainless steel.^{IV-13} The uranium was fully enriched in U^{235} and comprised 10% by weight of the fuel. The fuel material was hydrided to a hydrogen concentration of 6.35×10^{22} atoms of hydrogen/cm³, which is slightly less than the hydrogen concentration in cold water, and thus the elements also served as moderator. The fuel elements were positioned and

constrained in the core by upper and lower grid plates. The grid plates were supported by a fixed ring at the bottom and by holddown springs at the top. Beryllium side reflectors were used in the void between the hexagonal fuel element array and the circular reactor vessel. The reactor vessel top head acted both as a vessel closure and as a restraining structure for the core holddown springs. The reactor vessel was surrounded by a beryllium structure that functioned as the reflector. Four control drums, which formed a part of the reflector assembly, provided both the fine and coarse control required for SNAP-10A reactor operation. Control devices such as electric motors, springs, and sensors were attached directly to the reflector assembly. Controlling elements such as timers and reactor controller were located in the instrument compartment.

The SNAP-10A reactor subsystem consisted of the reactor structure, reflector assembly, reactor control equipment, and reactor core. It is shown along with the radiation shield in Figure IV-1. Basic design parameters and nominal operating conditions for the SNAP-10A reactor subsystem are given in Table IV-2.

(a) Reactor Structure and Core

The reactor structure and core consisted of a reactor vessel assembly, 1 top head, 2 grid plates, 6 internal reflectors, 12 holddown springs, 37 fuel elements, and 4 support legs. The reactor core and vessel (including grid plates) weighed 76 kg (167 lb). Approximately 3 kg (7 lb) of this was NaK coolant.

The reactor vessel assembly consisted of a deep drawn and spun reactor vessel to which a support ring, two NaK inlet pipes, and reflector support bracketry were welded. The assembly was constructed entirely of type 316 stainless steel. The reactor vessel had an inside diameter of 22.5 cm (8.875 in.) and an overall length of 40.6 cm (16 in.). It was essentially a right circular cylinder with the lower closure an inverted dome. Wall thicknesses varied from 0.8 mm (0.032 in.) in the core region to 3.2 mm (0.125 in.)

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Figure IV-1. SNAP 10A Reactor
and Shield

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IV-11

TABLE IV-2
SNAP-10A REACTOR DESIGN SUMMARY

Nominal Operating Conditions		Reflector Control Elements	
Reactor thermal power (kWt)	39.5	Number	4
Net electrical power (W) — minimum	525	Material	Be
Operating life	1 yr	Nominal thickness [cm (in.)]	5.1 (2)
NaK outlet temperature [K (°F)]	804 (987)	Internal Reflectors	
NaK inlet temperature [K (°F)]	733 (859)	Material	Be
NaK operating pressure [kPa (psia)]	34 (5)	Cladding	None
Maximum fuel temperature [K (°F)]	843 (1058)	Nuclear Parameters	
Maximum cladding temperature [K (°F)]	832 (1037)	Average core thermal flux (n/cm ² ·s)	2.5x10 ¹¹
Reactor Design		Mean fission energy (eV)	0.12
Fuel Elements		Isothermal temperature coefficient at nominal operating conditions (Δ/°F)	-0.29
Number	37	Uranium loading [kg U ²³⁵ (lb)]	4.75 (10.5)
Fuel alloy (wt. % U in Zr)	10	Effective delayed neutron fraction	0.008
Degree of hydriding N _H (10 ²² atom/cm ³)	6.35	Mean prompt neutron lifetime (μs)	6.5
Cladding material	Hastelloy-N	Burnable Poison	
Fuel diameter [cm (in.)]	3.07 (1.21)	Material	Sm ₂ O ₃
Cladding diameter OD [cm (in.)]	3.18 (1.25)	Preposition reactivity worth (\$) ^a	-1.60
Cladding thickness [cm (in.)]	0.38 (0.015)	Initial cold excess reactivity (\$) ^a	+3.00
Active fuel length [cm (in.)]	31.1 (12.25)	Total lifetime reactivity loss (\$) ^a	-2.58
Core Vessel		Xenon equilibrium	-0.17
Material	316 SS	Samarium burnout	+0.17
Internal diameter [cm (in.)]	22.5 (8.875)	Temperature and power defect	-2.25
Length [cm (in.)]	39.6 (15.6)	Hydrogen loss	-0.03
Minimum wall thickness [mm (in.)]	0.81 (0.032)	Hydrogen redistribution	-0.20
Inlet Plenum Grid Plate (lower)		Burnup and fission product poisons	-0.10
Material	316 SS	Contingency (\$) ^a	+0.42
Thickness [cm (in.)]	1.27 (0.5) (sandwich construction)	Total control drum worth with no shims (\$) ^a	8.60
Outlet Plenum Grid Plate (upper)		Total shim capability (\$) ^a	1.26
Material	316 SS	Weight Summary [kg (lb)]	
Thickness [cm (in.)]	0.32 (0.125)	Reactor core and vessel assembly	167 (367)
Reflector		Reflector-control assembly	107 (235)
Material	Be	Shield assembly	217 (477)
Configuration	Cylindrical sleeve	Electronic equipment	8 (18)
Nominal thickness [cm (in.)]	5.1 (2)	Total nuclear system	499 (1098)

^aOne dollar = 0.008 Δk/k

in areas to which attachments were welded. The fuel elements were supported and positioned in the reactor vessel by the two grid plates. The upper grid plate had a diameter of 22.4 cm (8.83 in.); the lower plate had a diameter of 22.2 cm (8.75 in.). The upper plate was a single piece of material 3.2 mm (0.125 in.) thick. The lower plate was a brazed assembly consisting of a 1.5-mm- (0.06-in.-) thick baffle plate and a 1.5-mm- (0.06-in.-) thick orifice plate with spacers between them. The overall thickness of the lower grid plate was 12.7 mm (0.5 in.). The plates had 37 holes arranged on a 32-mm (1.26-in.) triangular pitch for positioning the fuel elements. The plates were also perforated with 72 holes on a 18.5-mm (0.728-in.) triangular pitch for coolant flow. The diameter of the flow holes in the upper grid plate and in the baffle plate of the lower grid plate was 9.5 mm (0.375 in.). The diameter of the flow holes in the orifice plate varied from 6.4 mm (0.25 in.) surrounding the central fuel element to 4.8 mm (0.188 in.) at the outermost ones. Six pins were attached to each plate for positioning the internal reflectors. Twelve pins were attached to the upper grid plate for location of the core holddown springs.

Six beryllium internal side reflectors were positioned between the grid plates; pins in the grid plates mated with holes in the reflectors. The reflectors were 31.6 cm (12.5 in.) in length and shaped to fit fuel element and reactor vessel wall contours. The internal reflectors had no major structural function.

The reactor vessel top head was the final closure for the reactor vessel. In finished form, it was a 150° cone with a NaK outlet at the center and a flange at the outer edge for attachment to the reactor vessel. The wall thickness of the conical section was 2.4 mm (0.095 in.). Sockets were forged in the head for locating the core holddown springs and also for attaching the pump support outriggers. The flange had an extended lip that was welded to the reactor vessel. This simplified the final closure operation and provided a configuration that facilitated top head burnoff during atmospheric reentry.

Twelve core holddown springs were compressed between the reactor vessel top head and the upper grid plate. These springs held the core in place during launch operations but allowed relative thermal expansion between the core and the reactor vessel. The springs were fabricated from Rene 41 wire and exerted a force of 167 N (37.5 lb) each as installed.

The reactor was supported on the converter structure by four support legs formed from 0.8-mm (0.032-in.) 6AL-4V titanium alloy and welded. The legs were designed to support the reactor structure, with NaK pump and reflectors attached, during launch. The legs permitted relative thermal expansion between the reactor and converter structures. The support legs were attached to both reactor and converter structures with blind bolts.

(b) Fuel Elements

The reactor fuel material was a hydrided zirconium-uranium alloy that served both as fuel and moderator. The alloy contained 10 wt. % uranium and was hydrided to a hydrogen concentration of 6.35×10^{22} atoms of hydrogen/cm³. The uranium was fully enriched in U²³⁵. This material was formed into rods of 30.7 mm (1.21 in.) diameter and 31.1 cm (12.25 in.) length and was canned in Hastelloy-N cladding tubes. The cladding tubes were 31.75 mm OD and had a wall thickness of 0.38 mm (0.015 in.). Internal surfaces of the cladding tubes were coated with a 0.076-mm (0.003-in.) layer of ceramic glass material. This coating acted as a barrier to hydrogen leakage from the fuel. The ends of the fuel elements were sealed with end caps of Hastelloy-N material welded to the cladding tube. The end pins that positioned and held the elements between the grid plates were an integral part of the end caps. Each fuel element weighed 1.5 kg (3.4 lb) and had an overall length, including end pins, of 32.6 cm (12.82 in.).

Each unclad fuel rod contained approximately 128 g of U²³⁵, 11.8 g of U²³⁸, 24.6 g of hydrogen, and 1215 g of zirconium. Total weight was approximately 1380 g (3 lb). A small amount of carbon (approximately 0.15 wt. %) was added as a grain refiner to produce a stronger rod. The fuel was a hard,

semibrittle material with a machined surface and a metallic appearance. All edges of the fuel rod were rounded to prevent damage to the ceramic hydrogen barrier during assembly and handling of the fuel element.

The ceramic glass hydrogen barrier prevented excessive loss of hydrogen from the fuel rod operating in the high-temperature and high-radiation environment of the reactor core. The integrity of this barrier had to be maintained. Hence, the fuel elements had to be handled and shipped with suitable precautions to prevent damage due to excessive shock or vibration.

The ceramic glass barrier was composed primarily of oxides of aluminum, silicon, titanium, manganese, and barium with smaller amounts of sodium, lithium, and potassium. This ceramic coating was generally applied in three firings. In the last two firings, a small quantity of samarium oxide (Sm_2O_3) was incorporated into the coating. The nominal samarium oxide loading was 3 mg/cm (8 mg/in.). The samarium is a burnable poison. Depletion of this material inserted reactivity at the proper rate to provide more than 1 year of stable reactor operation.

One end cap of the fuel element was welded to a cladding tube before the ceramic barrier was applied. After the ceramic barrier had been fired, a fuel rod was inserted into the cladding tube. The cladding tube was then sealed against hydrogen loss by the insertion of a ceramic-coated blend cap. The hydrogen barrier was made complete by locally heating the cladding tube and blend cap to blend the ceramic surfaces together. Following the blending operation, an end cap was welded to the cladding tube, covering the blend cap. No end play existed between the fuel rod and the cladding tube.

(c) Reflector and Control Assembly

The reflector consisted of a beryllium sleeve, approximately 5 cm (2 in.) thick, surrounding the reactor core. The inside of this sleeve was a cylinder, while the outside was made up of plane sections approximating a cylinder.

Four semicylindrical cavities, spaced at 90° intervals, in the external surface of the reflector were utilized for control purposes. Reactor control was provided by four cylindrical segment beryllium drums. The drums were rotated into the semicylindrical cavities in the reflector block to vary neutron leakage from the core. The rotation angle at the drums from full in to full out was 135° . With the control drums in the full-out position, the reactor was adjusted to be $\$5.60$ subcritical at room temperature.

The drums were cut from a cylinder of 8.9-cm (3.5-in.) radius. The bare (unshimmed) drum segment had a maximum thickness of 4.5 cm (1.8 in.). Each drum could be increased in thickness by attaching as many as three 0.32-cm- (0.125-in.-) thick shims to the back or flat side of the drum. The length of each drum was 25.7 cm (10.125 in.).

When the startup command was received, squibs were fired in actuators adjacent to each control drum. This action pulled a pin, releasing the drum. The two coarse control drums were spring-loaded and, when released, were immediately rotated inboard. This added $\$4.30$ of reactivity. Fifty seconds after the startup command was received, the two fine control drums took their first step. They then moved a 0.5° step each 150 s until reactor operating temperature was reached. It took approximately 7 h after startup before criticality was reached, and approximately 2 h more before operating temperature was reached.

The temperature switch located in the outlet NaK line was set for 816 ± 6 K ($1010 \pm 10^\circ\text{F}$) to control the reactor during initial full-power operation. When the outlet temperature dropped below the set point, the fine control drums would take another step inward. This continued until 72 h after the startup command, at which time the controller was commanded "off."

The control and reflector assembly was mounted on the reactor in two halves. Each half was supported on two hinges located at the lower end of the assembly. The halves were rotated into position on the hinges until stops located near the top of the reactor were contacted. The complete assembly was

held in position by a retaining band around the upper end of the reflector halves.

The reactor could be shut down by ejecting the control and reflector assembly. Ejection was accomplished by compression springs located at the bottom of the reflector halves. When the retainer band was released, the reflector halves were pivoted outward around their support hinges by the spring force.

Before ejection of the thermal shield, the reflector halves could be rotated outward at least 7° by releasing the retaining band. After the thermal shield was ejected, the reflector halves could be completely ejected from the reactor. However, the two reflector halves would still be connected to the reactor by the electrical wiring. Seven degrees outward rotation of the reflector reduced reactivity by about \$17, and complete reflector ejection reduced reactivity approximately \$21.

Three distinctly different signals were used to initiate reflector ejection. All three signals activated an electrical device which released the reflector retainer band. An electrical signal could be provided by (1) ground command through an umbilical prior to liftoff, (2) telemetry, or (3) a failure-sensing device (low converter output voltage). The failure-sensing device started a 1-min timer, which reset if the malfunction signal was removed. Thus, a momentary voltage fluctuation would not cause an inadvertent shutdown. If the signal persisted, a 1-h delay timer was started. During the 1-h delay, pertinent data were recorded. At the conclusion of the delay period, the timer triggered the electrically actuated band release device (EABRD).

The EABRD consisted of a hollow Rene-41 cylinder and an electrical heater, which was contained within the cylinder. The cylindrical body was incorporated directly into the retaining band, being attached by special end fittings, and formed an integral part of the band. When a reflector ejection

signal was received, voltage was applied to the heater, which raised the temperature of the cylinder to about 1200 K (1700°F). The band tension springs easily pulled the body apart in its weakened condition at high temperature, thus severing the band and allowing reflector removal.

A thermal-mechanical device was also provided to separate the reflector retaining band in the event the EABRD did not function. This device, called the temperature actuated band release device (TABRD), was actuated by a drop in temperature of the NaK in the outlet line adjacent to the reactor. The TABRD was approximately 1 ft long and worked on the principle of differential thermal expansion of two materials. In this design, a stainless steel rod moved relative to a molybdenum tube. The two were rigidly joined at one end by a notched stud, which also served as a shear pin between two overlapping end fittings of the reflector retaining band and a bolt. The notch in the stud was in line with the mating surfaces of the band fittings. While the temperature was increasing, the steel rod, which had a much higher coefficient of thermal expansion than the molybdenum, was allowed to grow freely relative to the tube. On reaching a preset temperature (770 ± 14 K ($925 \pm 14^\circ\text{F}$)), an external-type snap ring would close behind a stepped collar. Thereafter, when the temperature decreased below the preset temperature, the stainless steel rod would be forced in tension. When the temperature reached approximately 530 K (500°F), the tension would become great enough to break the stud at the notched section and release the retainer band. A temperature drop that would cause the TABRD to release the retainer band could occur in one of two ways: (1) NaK flow stoppage or (2) loss of NaK.

The retainer band was also designed to release during reentry heating if reflector ejection had not already occurred. The brazed joint in the electrically actuated band release device would melt at about 91,000 m (300,000 ft), thereby releasing the reflectors. If for some reason this did not occur, the entire band would melt at about 87,000 m (285,000 ft), resulting in reflector ejection.

(d) Control Drum Drive Actuators

The control drum drive actuator was required to rotate the drum in small ($1/2^\circ$) increments. A stepper motor was used, together with a single 13.8:1 gear set. The motor was a synchronous machine, with a permanent magnet rotor and a two-phase stator. The rotor and stator were notched to form 50 and 48 teeth on each, respectively. The mismatch produced the effect of 200 poles. Therefore, with each change in polarity in the stator windings, the rotor would rotate 1.7° . Each phase of the stator was divided into two windings of opposite polarities. This was done to allow simple ON-OFF switching to produce the effect of phase polarity reversal. At any time during actuation, two stator windings were always energized. The direction of rotation was determined by the sequence in which the windings were energized. The actuator was designed to perform four steps in order to rotate the drum $1/2^\circ$. A four-step sequence was used to allow the electronic circuits in the controller to return to a stable condition.

A solenoid-actuated brake was mounted in the upper end bell of the actuator, with the solenoid armature spring-loaded against a disc on the rotor shaft. The solenoid winding was in series with the stator, being connected to the common return of the four stator windings. Therefore, the brake was released when power was applied to the actuator and was reapplied when the actuator was deenergized.

(3) SNAP-8

The SNAP-8 program (summarized in Ref. IV-14) was directed toward the development of lightweight nuclear reactors that would operate in space for longer than 10,000 h at a power level of 600 kWt. Toward this objective, two reactors were designed, built, and ground tested. The experimental reactor

(S8ER) was an engineering test of a prototype core but did not have flight-developed hardware. The developmental reactor (S8DR) was a nuclear subsystem test to verify the operability of the integrated flight configuration.

(a) SNAP-8 Experimental Reactor

(i) Reactor

The S8ER reactor consisted of a homogeneously combined uranium fuel and hydrogen moderator with movable beryllium reflectors for control and liquid NaK as the heat transfer medium. The subsystem was a direct extension of the technology developed and tested for the lower powered SNAP-2 reactors. The reactor contained 211 fuel-moderator elements arranged in a triangular pattern on 14.5-mm (0.57-in.) centers. The assembly resulted in a hexagonal right cylinder, with slightly chopped corners, 20.3 cm (8 in.) across flats and 36.75 cm (14.47 in.) long. Design and operating characteristics are summarized in Table IV-3. Figure IV-2 shows a section of the reactor and external reflector assembly. The S8ER released 5×10^6 kW·h of energy during 500 days of operation. This included 100 days at 600 kWt and 365 days above 400 kWt at 978 K (1300°F) outlet temperature. Power tests were terminated in April 1965.

Each fuel element consisted of an individual fuel rod, a hydrogen diffusion barrier containing burnable poison, exterior cladding, end caps, and grid plate indexing pins (see Figure IV-3).

The rods were composed of enriched (93.15% U^{235}) fuel and hydrogen moderator in the form of a solid zirconium-uranium alloy (10% uranium) hydrided to a hydrogen density of 6×10^{22} atoms/cm³. The rods were 13.5 mm (0.532 in.) in diameter and 0.356 m (14 in.) long. The hydrogen diffusion barrier was a 0.076-mm- (0.003-in.-) thick ceramic coating applied on the inside surface of the cladding to prevent the loss of the hydrogen moderator. A Sm_2O_3 burnable poison (1.3 mg/cm (3.3 mg/in.)) was added to the ceramic coating to compensate for the reactivity loss caused by the buildup of

TABLE IV-3
S8ER DESIGN DATA SUMMARY
(Sheet 1 of 2)

<u>Design Parameters</u>	
Design power level	600 kWt
Life	10,000 h
Primary coolant	NaK 78 eutectic
Number of fuel elements	211
Fuel loading [kg (lb)]	6.56 (14.4) ^a
Flow rate [liters/s (gpm)]	8.8 (140)
Reactor inlet temperature [K (°F)]	867 (1100)
Reactor inlet pressure [kPa (psia)]	267 (38.7)
Reactor outlet temperature [K (°F)]	978 (1300)
Reactor outlet pressure [kPa (psia)]	254 (36.8)
NaK inventory [kg (lb)]	143 (315)
<u>Fuel Elements</u>	
Composition	Hydrided Zr-U alloy
Hydrogen concentration	6×10^{22} atoms/cm ³
Alloy	90 wt. % Zr - 10 wt. % U
Cladding (Hastelloy N) [mm (in.)]	0.25 (0.01) walls and 6.35 (0.25) end caps
Fuel rod diameter [mm (in.)]	13.5 (0.532)
Fuel rod length [m (in.)]	0.356 (14.0)
H ₂ diffusion barrier thickness [mm (in.)]	0.076 (0.003)
Burnable poison	48 mg of SM ₂ O ₃ per element
Maximum cladding temperature at design power level [K (°F)]	1056 (1440)
Maximum fuel temperature at design power level [K (°F)]	1112 (1542)

^a93.15% enrichment

TABLE IV-3
S8ER DESIGN DATA SUMMARY
(Sheet 2 of 2)

<u>Reactor Core</u>	
Lattice spacing [mm (in.)]	14.5 (0.57) center to center
Core vessel size [m (in.)]	0.237 (9.34) OD by 0.533 (21) nominal height
Core vessel material	316 SS
Core vessel wall thickness [mm (in.)]	1.59 (0.0625)
Upper grid plate material	316 SS
Upper grid plate diameter [m (in.)]	0.233 (9.2)
Upper grid plate thickness [mm (in.)]	8.73 (0.34)
Upper grid plate coolant passages [mm (in.)]	420 3.97 (0.16) diameter holes
Lower grid plate material	Hastelloy C
Lower grid plate diameter [m (in.)]	0.233 (9.2)
Lower grid plate thickness [mm (in.)]	7.94 (0.31)
Lower grid plate coolant passages [mm (in.)]	420 3.18 (5/32) diameter holes
Internal reflector	18 BeO filler pieces
Internal reflector cladding	Hastelloy N
<u>Control and Safety Elements</u>	
Number of elements	6
Material	Beryllium
Lengths [m (in.)]	0.368 (14.5)
Drum radius of curvatures [m (in.)]	0.119 (4.69)
Nominal thickness [mm (in.)]	76 (3)
Weight of each control element (including support and shim) [kg (lb)]	12 (26.5)
Radial bushing clearance (self-aligning) [mm (in.)]	0.10 (0.004) to 0.15 (0.006) diametral
Friction torque [J (in.-lb)]	1.13 (10)

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Figure IV-2. S8ER Core and Reflector

9467-14

Figure IV-3. S8ER Fuel Element

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fission-product poison. The cladding consisted of 13.7-mm- (0.43 in.-) ID Hastelloy N tubing with a 0.25-mm (0.01-in.) wall thickness. The cladding envelope was completed by Hastelloy end caps seal-welded to the tubing. The manufacturing process ensured that a continuous diffusion barrier was provided on the inside surface of the cladding. The finished fuel element was 14.2 mm (0.56 in.) in diameter and 0.368 m (14.47 in.) long (excluding the positioning pins located at the ends of each element).

Grid plates supported and positioned the fuel elements and internal reflector filler pieces in the core. The lower grid plate was made of Hastelloy C and the upper grid plate of type 316 stainless steel. Both grid plates were pierced by 211 holes spaced in a triangular array on 14.5-mm (0.57-in.) centers to position and hold the fuel elements in the core. The grid plates were also pierced with 420 holes for coolant passage. Additional holes were located on the periphery of each grid plate for positioning and holding internal reflector filler pieces and core tie rods.

The internal reflector filler pieces occupied the space on the periphery of the core between the hexagonal fuel element pattern and the cylindrical reactor-vessel shell. They were made of beryllium oxide clad with Hastelloy N.

A type 316 stainless steel coolant baffle plate was located in the inlet plenum of the reactor between the lower grid plate and the coolant inlet line. A pattern of openings in the plate had been established empirically by the SNAP-8 core hydraulic experiments to adjust the coolant flow profile through the core to match the calculated radial power distribution and minimize the fuel cladding and hydrogen-barrier temperatures during full-power operation.

The core vessel was constructed of a cylinder of type 316 stainless steel, 0.233 m (9-3/16 in.) in ID and 0.394 m (15-1/2 in.) in length with a 1.59-mm- (1/16-in.-) thick wall. The lower end of the cylinder was closed with a reversed dish 3.18 mm (1/8 in.) thick and approximately 0.102 m (4 in.) deep. Coolant entered the cylindrical portion of the lower head via a side

inlet nozzle and a 50.8-mm (2-in.) stainless steel pipe. The upper end of the vessel was closed by means of a 6.35-mm- (1/4-in.-) thick cylinder, with the lower end welded to the top of the reactor vessel wall and the upper end welded to the bottom of the shutdown-shield assembly above the core. This cylinder served as the upper coolant plenum, with coolant discharging through a side nozzle and a 50.8-mm (2-in.) stainless steel pipe. Seven thermocouples were located in the plenum below the dished head to measure temperature distributions in the coolant leaving the core. Three thermocouples were located in the exit port to measure bulk-coolant exit temperature.

(ii) Reflector and Controller

The core vessel was surrounded by a 76.2-mm (3-in.) nominal thickness of beryllium, which formed the external radial reflector. The reflectors were 0.368 m (14.5 in.) long and were divided into six stationary and six movable sections. The portions of the reflector that formed the six movable sections (control drums) were segments of right circular cylinders with the axis outside the reflector assembly. Upper and lower bushings were located on the axis of each segment, permitting rotation of the control drums. Reactor control was accomplished by rotating the control drums toward or away from the core to change the neutron economy and reactivity of the core. The control drums were also used as safety elements to scram the reactor. Removable shim plates were provided on the control drums to permit adjusting the control drum worth before reactor startup.

The reflectors were supported by two stainless steel support structures attached to a flange on the transition cylinder that connected the core vessel to the shutdown shield. The support structures were also pinned to brackets attached to the bottom of the core vessel to provide radial support. The movable control drums were connected to the support structures by means of the upper and lower bushings.

The bushings were selfaligning, which permitted operation with limited misalignment due to thermal expansion. The upper bushings carried both radial and thrust loads. The lower bushings carried only a radial load.

The control mechanisms (shown in Figure IV-4) that positioned and scrambled the control drums were located above the shutdown shield, and each control drum had its individual drive system. The major components of the control system were a drive shaft, an electromagnetic clutch, a scram spring, a reversible ac motor, a pneumatic snubber, and a position indicator.

Each drive shaft was ball-bearing mounted in a removable sleeve housing mounted in the shutdown shield. The connection between this drive shaft and the control-drum pivot shaft was made via two torsionally rigid, flexible couplings and an intermediate shaft. This arrangement made it possible to maintain angular registry while allowing relative motion between the drive shaft and the control drum caused by the differential thermal expansion of components.

The drive shaft was keyed to an electromagnetic, rotary scram clutch and to a scram drive arm. It was also connected by a flexible coupling (with zero backlash) to the shaft of a position indicator transmitter. Thus, the indicator sensed the angular displacement of the control drum directly. The inner end of a flat, spiral torsion scram spring was connected to the scram drive arm so that the spring torque always tried to rotate the reflector toward the scram position. The solenoid portion of the scram clutch was keyed to the drive shaft, and the armature section was attached to a worm gear meshed with the drive worm. The drive worm was connected through a torque-limiting coupling to a reversible ac-drive motor.

The two sections of the scram clutch were connected with crown tooth face gears, providing a creepless engagement. When the electromagnet was deenergized, the connection between the two halves of the clutch would be broken, and the force of the scram spring would drive the reflector to the full-out position. With the clutch energized, the reflector could be positioned by operating the drive motor. Because nonreversible worm gearing was used, the scram spring could not drive the drive motor. However, the force of the scram spring on the drive worm eliminated any backlash in the worm gearing.

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Figure IV-4. S8ER Control Drive and Scram Mechanisms

Dual, electrical limit controls and physical stops limited the travel of the reflector in both the "in" and "out" directions. A pneumatic snubber was used to slow the control drum at the end of the scram rotation. Mechanical keys were provided to lock each reflector shaft in the least reactive position during maintenance work. These keys engaged milled slots on the shaft and were bolted to the top of the shutdown shield. Electrical connections at the control console allowed only one of six drums to be positioned at any time. Actuating a set-back switch, however, would drive all six drums out simultaneously. Additional circuitry was provided to drive any control drum to its full-out position should the spring fail to scram the drum. Only four of the six drums were needed to shut down the reactor.

(b) SNAP-8 Developmental Reactor

(i) Reactor

The SNAP-8 developmental reactor assembly included the fuel elements, the core vessel assembly, and the reflector and control components assembly (Figure IV-5). The 211 fuel elements were supported in the core vessel by an upper and a lower grid plate. The core had an active height of 0.427 m (16.825 in.) and an equivalent diameter of 0.221 m (8.7 in.). The reflector assembly was composed of six movable semicylindrical control drums and a stationary beryllium reflector. The principal parameters and characteristics of the S8DR system are summarized in Table IV-4. The reactor was designed for 600 kW of power at an outlet temperature of 978 K (1300°F). Alternate operating conditions were 1 MW and 867 K (1100°F). S8DR was operated at 600 kW for 6700 h and at 1 MW for 430 h.

The S8DR core contained 211 fuel elements cooled by NaK-78 flowing axially between them. Each fuel element consisted of a cylindrical rod of fuel moderator, clad in Hastelloy N tubing.

9467-10

Figure IV-5. S8DR Ground Test Assembly

TABLE IV-4
SNAP-8 DEVELOPMENTAL REACTOR CHARACTERISTICS
(Sheet 1 of 3)

<u>General</u>	
Design power level (kWt)	600
Reactor outlet temperature [K (°F)]	978 (1300)
Reactor inlet temperature [K (°F)]	867 (1100)
Average power density [kW/liter (kW/ft ³)	37 (1048)
Maximum power level [MWt]	1
Reactor outlet temperature [K (°F)]	867 (1100)
Reactor inlet temperature [K (°F)]	717 (830)
Design life (h)	12,000
Primary coolant	NaK
Number of fuel elements	211
Fuel loading [kg (lb)]	8.2 (18) U enriched to 93.15% U ²³⁵
Reflector nominal thickness [cm (in.)]	13 (5)
Control	6 rotating Be reflector drums
<u>Reactor Core</u>	
Core vessel size ^a [m (in.)]	0.235 (9.24) ID by 0.818 (32.2) height to outlet line
Material	316 stainless steel
Wall thickness [mm (in.)]	2.3 (0.091)
Grid plates	
Coolant passages	420 [3.97-mm (0.165-in.) diameter]
Spacing of fuel element positioning holes [mm (in.)]	14.6 (0.573) (triangular lattice)
Spacing of coolant holes [mm (in.)]	8.4 (0.331) (hexagonal lattice)
Upper grid plate	
Material	316 stainless steel
Diameter [m (in.)]	0.234 (9.217)
Thickness [mm (in.)]	21.6 (0.85)
Lower grid plate	
Material	Hastelloy C
Diameter [m (in.)]	0.234 (9.217)
Thickness [mm (in.)]	16.5 (0.65)
Internal reflector material	
	30 stainless-steel-clad BeO and 12 solid stainless steel pieces
Cladding thickness [mm (in.)]	0.76 (0.03)
Cladding material	316 stainless steel

^aAll dimensions are at room-temperature conditions unless otherwise specified.

TABLE IV-4
SNAP-8 DEVELOPMENTAL REACTOR CHARACTERISTICS
(Sheet 2 of 3)

<u>Fuel Elements</u>	
Fuel rods	
Composition	Hydrided ZrU alloy
Fuel	10.5 wt. % U
Enrichment	93.15 wt. % ^{235}U
Hydrogen concentration	6.05×10^{22} atoms/cm ³
Diameter [mm (in.)]	13.4 (0.529)
Length [m (in.)]	0.427 (16.825)
Axial hydrogen gap [mm (in.)]	6.1 (0.24)
Radial hydrogen gap [mm (in.)]	0.09 (0.0035)
H ₂ diffusion barrier [mm (in.)]	0.05 (0.002) ceramic coating on inside of cladding surface
Burnable poison	58-mg Sm ₂ O ₃ /element (in ceramic)
Cladding (Hastelloy N) [mm (in.)]	13.7 (0.54) ID, 0.287 (0.011) thick
Fuel elements	
Diameter [mm (in.)]	14.3 (0.563)
Length [m (in.)]	0.44 (17.325) (excluding positioning pins)
Peak fuel temperature at design power level [K (°F)]	1080 (1485)
Peak cladding temperature at design power level, K (°F)	1015 (1367)
Average fuel burnup (12,000 h at 600 kW)	0.22 metal at. %
<u>Control Drums</u>	
Number of drums	6
Material	Beryllium
Length [m (in.)]	0.39 (15.375) (total reflector length = 0.47 (18.5))
Drum radius of curvature [m (in.)]	0.114 (4.5)
Nominal thickness [m (in.)]	0.130 (5.125)
Drive means	Electromagnetic stepping motor
Rotating range	0 to 109°
Full-in stop	1.0 ± 1.0°
Full-out stop	135 ± 2.0
Drive-system backlash	±0.05°
Scram power	Torsional spring

TABLE IV-4
SNAP-8 DEVELOPMENTAL REACTOR CHARACTERISTICS
(Sheet 3 of 3)

<u>Nuclear Characteristics</u>	
Mean prompt neutron lifetime	8.4 μ s
Effective delayed neutron fraction	0.0080
Median fission energy	0.154 eV
Thermal flux at 600 kW (<1 eV)	3×10^{12} n/cm ² ·s
Radial peak to average power	1.29
Axial peak to average power	1.36
Clean, wet excess reactivity	\$14.40
NaK reactivity worth	\$0.25
Total isothermal temperature	-0.10¢/K (-0.18¢/°F) at 294 K (70°F)
Coefficient of reactivity	-0.11¢/K (-0.20¢/°F) at 922 K (1200°F)
Power coefficient of reactivity	-0.035¢/kW
Rise to 600-kW power and 922 K (1200°F) core average temperature	-\$2.15
Reactivity inventory (12,000 h)	
Hot end-of-life excess (600 kW)	\$5.45
Samarium burnout	\$2.90
Fission product poisoning	
Xe	-\$1.05
Sm	-\$1.02
Other	-\$1.17
Axial hydrogen redistribution	-\$0.65
Hydrogen loss	-\$5.15
U ²³⁵ burnup	-\$0.66
Control drum worth	
Total — 6 drums	\$21.06
Single drum	\$3.90
Maximum differential worth	5.75¢/deg
<u>Thermal and Hydraulic Characteristics</u>	
Average heat flux (at 600 kW) [kW/m ² (Btu/h ft ²)]	148 (47,000)
Fuel-to-coolant heat transfer area [m ² (ft ²)]	4.1 (43.6)
Mass flow through core (600 kW and 110 K Δ T) [kg/s (lb/h)]	6.2 (47,800)
Average velocity through core, hot [m/s (ft/s)]	1.4 (4.6)
Mass flow through core (1000 kW), [kg/s (lb/h)]	7.6 (60,000)
Average velocity through core, hot [m/s (ft/s)]	1.7 (5.6)

The fuel-moderator material was composed of a hydrided alloy of zirconium and fully enriched ($93.15 \pm 0.15\%$ U^{235}) uranium. The uranium content constituted 10.5 ± 0.3 wt. % of the unhydrided alloy. This material also contained 0.15 ± 0.05 wt. % carbon, added in the form of zirconium carbide (ZrC) prior to the initial arc melting of the zirconium-uranium alloy. This carbon additive resulted in a refinement of the alloy grain structure and a consequent improvement in the production yield during the hydriding process.

The 211 fuel rods were hydrided to have N_H^* values between 5.97 and 6.15, with an average for all 211 rods of 6.05. At the operating temperatures characteristic of S8DR, hydrogen pressures in the range to 0.1 or 0.2 MPa were produced in the annulus between the fuel rod and the cladding. To minimize hydrogen loss from the fuel element, the interior surface of the cladding was coated with a ceramic material. This coating reduced the permeation of hydrogen through the cladding wall by a factor of about 1000. The ends of each cladding tube were enclosed by Hastelloy N end caps. These end closures were welded to the tubing. The ceramic coating on the final end closure was fused to that on the tubing to form a continuous hydrogen seal.

The active length of the S8DR fuel element (i.e., the length of the fuel rod) was 0.427 m (16.825 in.). An axial gap of 6.1 mm (0.24 in.) between the top surface of the fuel rod and the ceramic layer on the end cap was designed to accommodate axial fuel swelling and to minimize the heat transfer effect due to fission gas accumulation. The 13.4-mm- (0.529-in.-) diameter fuel rod was separated from the 0.05-mm- (0.002-in.-) thick ceramic layer on the cladding by a nominal hot diametral gap of 0.19 mm (0.007 in.) that served to accommodate fuel swelling. The Hastelloy N tubing had an ID of 13.7 mm (0.540 in.) and a nominal thickness of 0.287 mm (0.011 in.).

* N_H is defined as the number of hydrogen atoms ($\times 10^{-22}$) per cubic centimeter of the hydrided zirconium-uranium alloy.

The core vessel assembly included the reactor core vessel, the fuel elements, the upper and lower grid plates, the flow distributor and flow baffle, the internal reflectors, and the support structure for the reactor.

The reactor core vessel was designed to minimum weight for space application. The basic design was modified to allow thermocouple penetrations to the upper plenum. The reactor core vessel was a right circular cylinder consisting of a lower plenum head, an intermediate cylinder, and an upper plenum head, all fabricated from type 316 stainless steel.

The lower plenum head had a flat external bottom and a convex internal bottom surface. This head had internal brackets for mounting the flow distributor, the baffle plate, and the lower grid plate. An external mounting ring was used to couple the core vessel to the support structure.

The intermediate cylinder, with a nominal wall thickness of 2.3 mm, was welded to the upper and lower plenum heads. It had internal lugs to support and position the upper grid plate and external brackets on which to mount the external reflector and controller.

The upper plenum head contained a coolant outlet nozzle, three holddown lugs for the upper grid plate, three support lugs for the thermocouple bridge, and 27 thermocouples to measure coolant temperature at the reactor outlet. This head was composed of a reducer section that connected the intermediate cylinder to a 127-mm (5-in.) Schedule 10 pipe tee, which contained the 50.8-mm (2-in.) outlet nozzle. The other end of the tee was connected to the double bulkhead thermocouple penetration assembly. The thermocouples in the upper plenum were made of Chromel-Alumel, insulated with MgO and sheathed in type 316 stainless steel. Each thermocouple sheath was brazed at both bulkheads where it penetrated the plenum head. Inside the plenum, the thermocouples were bundled in a 25.4-mm (1-in.) pipe and then distributed on a bridge to locations at the outlet of the core coolant channels. The core

vessel was designed for 0.34-MPa (35-psig) operation with an inlet temperature of 894 K (1150°F) and an outlet temperature of 1006 K (1350°F).

The grid plates supported and positioned the fuel elements and internal reflectors in the reactor. The upper grid plate (outlet plenum) was a 0.234-m- (9.217-in.-) diameter by 21.6-mm- (0.65-in.-) thick disc of type 316 stainless steel. Its lower edge was notched at three places to position it on the support lugs of the core vessel.

The lower grid plate (inlet plenum) was a 0.234-m- (9.217-in.-) diameter by 16.5-mm- (0.65-in.-) thick disc of Hastelloy C. The plate was bolted to the reactor vessel support lugs through three holes drilled through the peripheral area of the plate.

Both grid plates contained 211 6.35-mm- (0.25-in.-) diameter holes located in a triangular array to provide for positioning of the fuel elements. Located on a hexagonal lattice around the fuel element positioning holes were 420 3.97-mm- (5/31-in.-) diameter holes for coolant flow. These were straight-through holes in the bottom grid plate, but they tapered outward 10° from bottom to top in the upper grid plate. Each grid plate contained an additional 42 holes on the outer periphery to position the internal reflectors.

The grid plates were designed to withstand a 35-g launch load with a 1.4 safety factor as specified for a flight-type reactor. This design provided a very high margin of safety for the S8DR operational requirements.

Hastelloy C was chosen for the lower grid plate material because of its low thermal expansion coefficient and resulting low-temperature coefficient of reactivity. Since at the outlet of the reactor a high temperature-coefficient of reactivity was desirable for self-regulation, the upper grid plate was fabricated of type 316 stainless steel.

The coolant flow distributor and baffle plate were located in the inlet plenum below the lower grid plate. The flow distributor was a basket-shaped piece that split the flow coming into the lower plenum, prevented vortexing, and provided a constant pressure profile across the lower face of the flow baffle plate. The distributor had a wedge-shaped piece that faced the coolant inlet nozzle and two rows of 32 holes each around the sides. The unit was manufactured of 3.2-mm (1/8-in.) type 316 stainless steel. The pressure drop across the distributor was about 3450 Pa (0.5 psi). This did not give rise to any significant stresses in the material.

The coolant flow baffle plate was fabricated of type 316 stainless steel. It had a diameter of 0.234 m (9.2 in.) and a thickness of 12.7 mm (0.5 in.). To give the desired flow distribution through the core, 73 holes were drilled through the plate in a configuration. The exact configuration for the baffle plate was determined by hydraulic tests. The pressure drop through the baffle plate of 13,800 Pa (2 psi) did not result in any significant stresses.

The reactor core vessel contained 42 internal reflectors to fill the void between the hexagonal-shaped fuel array and the circular vessel. Thirty of the internal reflectors were BeO pieces clad in type 316 stainless steel. The 12 smallest reflector pieces were solid type 316 stainless steel.

The internal reflectors were designed for operation at 1006 K (1350°F). Space had been allowed inside the cladding both for BeO swelling and for helium buildup from neutron reactions with the beryllium during reactor operation.

(ii) Reactor Reflector and Control Drums

The reactor core vessel was surrounded by a beryllium reflector that was divided into two stationary and six movable sections. Reactor control was provided through the movable sections. Figure IV-6 shows a drawing of the reactor with reflector and drive assembly.

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Figure IV-6. Schematic of S8DR Reflector and Drive Assembly

The beryllium reflector assembly has a nominal thickness of 0.130 m (5.125 in.) and a length of 0.47 m (18.5 in.). The reflector assembly was divided axially into two halves. Each half consisted of a stationary reflector and three control drums together with drives, bearings, and support brackets. The reflector assemblies were attached to and supported by brackets that were an integral part of the reactor core vessel. Latching mechanisms were provided at the upper end of the assembly to lock the reflector halves together.

The length of the reflector assembly for the S8DR was increased over that used for the S8ER to provide additional reactivity. This increase was provided by stationary sections above and below the control drums (which were essentially the same as the S8ER drums).

The drums were shaped as segments of right-circular cylinders with 0.114-m (4.5-in.) radii. The basic drum had a length of 0.389 m (15.3 in.) and a maximum thickness of 54 mm (2.125 in.). A 76-mm (3-in.) beryllium shim was attached to the back of each drum to make up the total 0.130-m (5.125-in.) thickness. Mechanical stops for the drum were positioned at $1^\circ \pm 1^\circ$ at the full-in position and at $135^\circ \pm 2^\circ$ at the full-out position. Each drum contacted a snubber at 109° from full-in.

All beryllium surfaces were anodized to give them high emissivity. The outside beryllium surfaces (those not facing the reactor vessel) were coated with an additional high-emissivity coating since the emissivity of the anodized coating had been found to degrade slightly at the high temperature and vacuum conditions of the S8DR environment. The reflector assembly was designed to limit the beryllium temperatures to less than 1006 K (1350°F).

The anodized coating on the similar S8ER reflectors served also to inhibit corrosion. Since all high-temperature operation for S8DR would be performed with the reactor in a 1.3×10^{-3} Pa (10^{-5} Torr) vacuum, no beryllium corrosion was anticipated.

Each control drum rotated in a pair of upper and lower self-aligning journal bearings. Each bearing had the journal bushings held in a spherical cartridge that, in turn, moved in a socket to provide self-alignment. Contact surfaces between the ball and the socket were coated with aluminum oxide and then covered with a molybdenum disulfide base dry-film lubricant. The balls were made of P-5 carbon-graphite. The shafts that ran in the ball were also coated with dry-film lubricant. The bearings were designed for operation at 894 K (1150°F) in an environmental pressure of 1.3×10^{-3} Pa (10^{-5} Torr) or less.

The maximum beryllium reflector temperature calculated for 600-kW operation at a 978 K (1300°F) core outlet temperature was 944 K (1240°F). At 1 MW and an 867 K (1100°F) outlet temperature, the maximum calculated beryllium temperature reached 1004 K (1348°F), just below the design temperature. These peak temperatures would occur in the stationary beryllium cusps. The maximum control drum temperature was estimated to be 968 K (1283°F) for 1-MW operation.

The maximum control drum bearing temperature during operation at 600 kW and 978 K (1300°F) outlet temperature was calculated to be 856 K (1080°F). At 1 MW and an outlet temperature of 867 K (1100°F), the maximum bearing temperature was calculated to be 917 K (1190°F). Bearing qualification tests were run at 950 K (1250°F) to cover this condition. The carbon-graphite balls were restricted to operation below 533 K (500°F) when in an oxidizing atmosphere. Maintenance of the 1.3×10^{-3} Pa (10^{-5} Torr) vacuum was necessary at full-power, full-temperature operation to prevent the balls and also the dry-film lubricant coating from oxidizing.

The control drum actuator, an 8-pole reluctance motor, produced torque by the interaction of magnetic forces between the stationary teeth on the fixed poles and the notched teeth cut into the cylindrical rotor. The field coils for the poles were sequentially energized to produce a revolving magnetic field, which turned the rotor. The alignment of the stator-pole teeth was such that, as each pole was energized, the rotor teeth were in a position to provide maximum torque.

The brake contained a solenoid assembly which held the actuator rotor fixed while deenergized and released it for movement when energized. The braking disk attached to the actuator rotor contained 100 teeth; these mated with the 50 teeth on the bellows-supported disk when the brake was applied. One step of the actuator was a turn through 3.6° , or one tooth pitch. This was geared through a 13.8 reduction to the control drum, which made a 0.26° step.

The actuator assembly was designed to operate at 825 K (1025°F) in a pressure of 1.3×10^{-3} Pa (10^{-5} Torr) or less. To minimize differential expansion problems over the operating temperature range of the actuator, a single material was used for fabricating structural and magnetic components. The material chosen, Hiperco 27, is a cobalt-iron alloy that has the required structural and magnetic properties. The exception was Inconel X, which was chosen for the flexure assembly of the brake. Bearing surfaces were coated with aluminum oxide and run in carbon-graphite bearings. The magnetic coils were wound with stainless-steel-clad copper wire. The wire was insulated with a glass yarn and encapsulated with a ceramic encapsulant. The brake teeth were coated with MoS_2 dry-film lubricant to prevent adhesion during high-temperature and vacuum operation.

The maximum actuator operating temperatures were calculated to be 778 K (940°F) for operation at 600 kW and 978 K (1300°F) average core outlet temperature and 781 K (945°F) for operation at 1 MW and 867 K (1100°F) outlet temperature. Operation at a pressure of 1.3×10^{-3} Pa (10^{-5} Torr) or less was required to prevent oxidation of the carbon-graphite bushings and the dry-film lubricant.

Coil pairs, which operated simultaneously, were wired in parallel so that the failure of one coil would not prevent the operation of the actuator.

The drive and scram mechanism for S8DR provided the link between the actuator and the control drum. Each unit consisted of a drive gear, two concentric shafts, an electromagnetic clutch, and a scram spring.

The scram spring and its release clutch operated through concentric shafts. The outer shaft was attached to the lower half of the clutch and extended downward to the gear, which was driven by the actuator. The inner shaft was connected to the upper half of the clutch, which contained the magnet coils, and extended down to attach to the control drum through a flexible coupling. The lower portion of the clutch assembly contained a clock-type scram spring. The outside end of the scram spring was connected to the spring housing, and the inside end was attached to the inner shaft. An interlock prevented the drums from operating without the scram spring first being wound and the clutch energized. The clutch magnet was faced with teeth that mated with the teeth on the lower housing when the clutch closed.

During normal operation, the entire clutch assembly was locked together and rotated as a single unit. The drive path from the actuator was through the rotor pinion to the large-diameter drive gear, up through the outer tube, through the clutch, and down through the inner shaft to the drum.

To scram a drum, power to the clutch magnet was interrupted, and the clutch disengaged by gravity and by the bellows spring on the lower housing. With the clutch open, the scram spring unwound to rotate the control drum to the full-out position. The spring was rewound by driving the actuator in the drum-out direction with the clutch open. With the drum already out against the snubber, the actuator drove only the outer shaft and spring housing, thus rewinding the spring. A limit switch on the spring housing was tripped when the spring was fully wound. This allowed the clutch to be energized and closed for normal drum operation.

Should the actuator fail, there was the capability for manually setting the control drum to a predetermined position. This shimming could be accomplished by shutting down the reactor, entering the vacuum vessel, and remotely cranking the control drum to the required position. This operation was administratively controlled such that the shutdown criteria would not be compromised.

The scram mechanism was fail-safe because a power failure would automatically open the clutch and result in control drum scram. Also, in the event of burnout of the clutch coil, the mechanism would automatically scram the drum. Because of the lower operating temperature of the scram clutches as compared to the actuator mechanisms, these units were not redesigned to replace the chrome-plated windings.

The snubber assembly contained a spring to absorb the control drum energy when scrambled. An arm on the drum drive shaft struck the snubber rack which pushed against the snubbing spring. A spring-loaded pawl engaged the teeth of the rack as it was pushed in and prevented the spring from bouncing the drum back from the snubber region. After the drum was driven out of the snubber region, it tripped the pawl to release the snubber rack.

Three limit switches were mounted on a bracket attached to each snubber assembly. One switch indicated when the drum was full-in, one when the drum was full-out, and the third indicated when it was beyond full-out and in the snubber zone.

The position sensor, a rotary variable reluctance device, was attached to the drive mechanism. The sensor operated from a constant ac excitation and generated an amplitude and a phase-varying output that were proportional to the angular position of the sensor. The output signal from the sensor was transmitted to a demodulator, which converted it into a 0- to 5-Vdc signal. This dc signal was used by the coarse position indicator to show drum position and by the precision drum-position indicator to verify that the drum had moved.

(4) SNAP 5-kWe Reactor

The reactor designed for integration into the 5-kWe reactor-thermoelectric system is illustrated in Figure IV-7. The reactor was designed for five years of continuous operation at 110 kWt. The design parameters are summarized in Table IV-5. The complete reactor design is summarized in Ref. IV-15. The reactor consisted of:

- 1) A 0.27-m- (10.5-in.-) diameter by 0.41-m- (16-in.-) long (fuel length) core made up of 85 25.4-mm- (1-in.-) diameter fuel elements packed in a triangular array, plus 12 internal reflector segments to fill the void space between the fuel element array and the circular reactor vessel.
- 2) Inlet and outlet grid plates plus a corner liner to support the fuel elements within the vessel and properly control the flow through the element array.
- 3) The reactor vessel, which was the containment for the NaK coolant and served as the basic structure of the reactor assembly.
- 4) The reflector-control assembly, a tapered beryllium reflector assembled in two identical halves, each having an axially sliding control segment, associated drive, and actuator. The reflector mounting hardware allowed the reflector halves to rotate 15° outward for end-of-life shutdown.

The NaK 78 coolant was fed to the reactor through two pipes that traversed the reflector through grooves in the outer face at the split line between the two halves. Flow was directed within the inlet plenum by vanes and baffling to produce a flat pressure profile over the inlet grid plate. After passing through the core, the flow exited from the outlet plenum through two nozzles oriented 15° off the reflector split-line axis.

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Figure IV-7. SNAP 5-kWe Reactor

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TABLE IV-5
SUMMARY OF 5-kWe REACTOR DESIGN

Thermal power (kWt)	110
Number of fuel elements	85
Active fuel length [mm (in.)]	406 (16)
Lattice pitch [mm (in.)]	26 (1.025)
Inter-element spacing [mm (in.)]	0.508 (0.02)
Cladding thickness [mm (in.)]	0.635 (0.025)
Glass thickness [mm (in.)]	0.064 (0.0025)
Gap thickness [mm (in.)]	0.152 (0.006)
Fuel diameter [mm (in.)]	23.8 (0.938)
Active core diameter [mm (in.)]	252 (9.92)
Active core volume [liters (ft ³)]	20.3 (0.72)
Core vessel OD [mm (in.)]	269 (10.6)
Vessel-reflector clearance [mm (in.)]	1.52 (0.06)
Reflector midplane thickness [mm (in.)]	60.6 (2.386)
Reactor midplane envelope [mm (in.)]	393 (15.492)
Taper angle (deg)	8.5
Segment stroke [mm (in.)]	102 (4.0)
Movable angle (deg)	235
Fissile mass (kg U ²³⁵)	9.0
N _H in fuel (10 ²² atoms/cm ³)	5.72

A unique aspect of this reactor, relative to all previous SNAP reactors, was the tapered reflector with sliding control segments. All previous SNAP reactors used rotating drums for control. The primary advantage of this new design was the minimization of the diameter of the neutron shield required to shadow the payload.

(a) Core Assembly

The core was designed as a unitized assembly of the 85 fuel elements, the inlet and outlet grid plates, the fuel element locking rods, the internal reflectors, and a core liner. The unit was bench assembled and had poison splines in the coolant channels to ensure subcriticality; it then was inserted into the reactor vessel. Of the 85 fuel elements in the core assembly, 27 were right-hand finned, 27 were left-hand finned, and 31 were unfinned (neutral). The elements were arrayed in a triangular lattice as shown on Figure IV-8.

(b) Fuel Element

Each fuel element consisted of a segmented (five piece) fuel rod of hydrided 10% uranium-zirconium alloy sealed within a Hastelloy-X cladding. A burnable poison was included as a vapor-deposited coating of gadolinium oxide applied to the circumferential surface of the fuel slugs. A ceramic coating on the inside surfaces of the cladding was the primary hydrogen retention barrier. Basic dimensional data for the fuel element are given in Table IV-4.

The finned elements had three fins, spaced 120° apart, that spiraled through three turns on a 127-mm (5-in.) pitch. These 3.0-mm- (0.12-in.-) wide fins were formed by hobbing a heavy wall starting tube; the hobbing left a fin height of 1.3 mm (0.05 in.). Then six flats were milled to a fin height of 0.25 mm (0.01 in.), giving the cross section as shown in Figure IV-9.

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Figure IV-8. Core Plan for SNAP 5-kWe Reactor

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Figure IV-9. Fuel Element Cross Section for SNAP 5-kWe Reactor

A vibration suppressor was installed between the fuel and the cold end cap, as illustrated in Figure IV-9. This device, which resembled a collar button, prevented any axial movement of the fuel rod under the dynamic loadings incurred during launch. It was designed to yield by collapse of the prebent column to accommodate axial fuel growth during reactor operation, without overstressing the cladding tube.

The hot end cap was basically a spherical head closure, butt welded to the cladding tube. The end pins had a 15° chamfer to provide a lead-in for the grid plate during core assembly. The pin diameter on all finned elements and on the 12 outer neutral elements was sized for 0.5 mm (0.02 in.) diametral clearance in the grid plate hole; on the inner neutral elements, the pin was sized for a close fit.

The cold end cap was a flat head closure with an interference fit between the coated tube inside surface and the coated outside surface of the engaging socket portion of the cap. After the cap was inserted, the engaged section of the cap and tube was heated to the softening temperature of the ceramic to blend the coating into a continuous hydrogen barrier. The cold end pins were all sized for a close fit engagement in the grid plate holes. This close fit engagement was limited to a 3.2 mm length engagement adjacent to the end surface of the cap. The remainder was reduced in diameter to facilitate the assembly of the finned elements adjacent to other finned elements.

(c) Grid Plates

The inlet grid plate was the basic structural element of the core assembly. The fuel elements were secured to the grid plate by the locking rods, which threaded through holes in the fuel element and pins. Each locking rod secured a complete row of elements across the core and thereby also performed a fuel element orientation functionally maintaining the correct relationship between the flattened fins on the clad outside surfaces of adjacent elements.

The inlet grid plate was designed as a brazed sandwich structure made up of two 1.6-mm- (0.062-in.-) thick facing sheets, an outer rim, and 85 spacer posts located at the fuel element axis positions. The spacer posts and the outer rim acted as shear transfer elements and thereby caused the two facing sheets to function as though they were integral parts of a solid plate. This sandwich, with a 9.6-mm (0.376-in.) separation between the facing sheets, had a strength equivalent to a 9.7-mm- (0.38-in.-) thick solid plate, and a stiffness equivalent to 10.4-mm- (0.41-in.-) thick solid plate. The sandwich plate approach thereby produced a significant weight saving.

The outlet grid plate provided only lateral spacing and support for the fuel elements. This plate was, therefore, of minimum thickness (1.6 mm, or 0.622 in.). An integral rim at the outer edge was included in the design to give the plate flatness stability and to provide a large engagement with the core liner.

Both grid plates were fabricated from type 316 stainless steel; the upper grid plate assembly was furnace brazed using a nickel base braze alloy. The coolant passage holes were 6.35 mm (0.25 in.) in diameter, except for 12 of the outer boundary channel holes, which were 4.83 mm (0.19 in.) in diameter due to space limitations. Each plate also had 12 4.32-mm- (0.17-in.-) diameter holes near the periphery for engaging the internal reflector positioning pins.

The use of the brazed sandwich structural grid plate was based on experience with almost identical designs in the SNAP-10A and SNAP-2 DR reactors.

(d) Internal Reflectors

Twelve internal reflector segments filled the spaces between the noncircular boundary of the fuel element array and the cylindrical core liner. These consisted of BeO shapes contained in sealed stainless steel enclosures. The surfaces facing the boundary fuel elements were configured to provide flow

rates in individual boundary channels; this minimized the possible temperature difference across the boundary fuel elements.

Fast neutron irradiation of BeO caused material swelling and helium generation, which produced a pressure buildup within the stainless steel container. The design therefore included a 38.1-mm (1.5-in.) axial gas space to limit the pressure at end-of-life to 124 kPa (18 psi), equal to the NaK coolant pressure if both NaK volume accumulator bellows were to fail. The internal reflectors, initially evacuated, therefore had a negative internal pressure throughout reactor lifetime.

Each internal reflector was held in position by a pin at each end that engaged the grid plates. Because reflector location relative to the boundary elements was critical in establishing the boundary channel flow areas, these pins were custom located during the core assembly sequence.

(e) Core Liner

The core liner was a 0.38-mm- (0.015-in.-) thick stainless steel cylinder closely fitted around the internal reflectors. Its primary function was to limit core bypass flow to a very low value (less than 0.5%) to minimize the bowing stresses induced in the boundary fuel elements by cross-element temperature differentials, while still allowing a generous clearance for insertion of the unitized core assembly into the reactor vessel.

(f) Control Segments

Reactivity control was provided by two sliding reflector segments. The movable segments subtended 235° of the reactor circumference and opened 102-mm (4-in.) windows about the vertical center of the core. The sliding segments moved along the 8.5° taper angle rather than parallel to the core vessel surface. This also opened gaps between the reflector and the core vessel and between the sliding segment and the fixed reflector; these enhanced the control worth.

(5) SNAP-50/SPUR

The design of the SNAP-50 space power unit reactor (SPUR) by Pratt & Whitney Aircraft started in 1961 at a power level of 8 MWt when conceptual studies of the reactor and power conversion indicated that a single optimum reactor design was capable of providing the thermal energy necessary for systems in the 300-kWe to 1-MWe range. The reactor thermal power required ranged from 2 to 8 MW. This approach was subsequently revised because experimental data, which became available after this decision, indicated that the fuel and fuel cladding performance assumptions were too optimistic to achieve 8-MWt capability within the weight limitations of a 300-kWe system. Consequently, the design power level for the reactor was reduced to 2.2 MWt, which was adequate for a 300-kWe application leading to the PWAR-20 reactor and shield concept as shown in Figure IV-10. This concept is summarized in Ref. IV-16.

The design philosophy for the PWAR-20 was to rely on conservative assumptions whenever this was consistent with the system objectives rather than to rely on as yet undemonstrated development breakthroughs. To a large extent, this approach was successful in developing a reactor concept that promised to meet the SNAP-50 development goals. The PWAR-20 design would have provided a coordinated basis for design and analysis aimed at completing the development and demonstration of a high-performance reactor and shield at the 2.2-MWt power level.

(a) Description of the PWAR-20

(i) General Concept

The PWAR-20 concept is shown in Figures IV-10 and IV-11. Design data are summarized in Table IV-6. The reactor was lithium-cooled and had a fast

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Figure IV-10. PWAR-20 (SNAP-50/SPUR) Reactor and Shield Concept

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Figure IV-11. PWAR-20 (SNAP-50/SPUR) Reactor Core Concept

TABLE IV-6
SNAP-50/SPUR DESIGN DATA SUMMARY
(Sheet 1 of 2)

Reactor	
Design power (MWt)	2.2
Full-power lifetime (h)	10,000
Core coolant outlet temperature [K (°F)]	1367 (2000)
Core coolant temperature rise [K (°F)]	56 (100)
Flow rate [kg/s (lb/s)]	9.6 (21)
Materials	
Fuel	95% dense UN
Structure	Cb-1 Zr-0.1 C, type 316 SS
Fuel clad	Cb-1 Zr-0.1 C
Coolant	Li (99.9% Li-7)
Side reflector	BeO
End reflector	Depleted UN
Fuel cladding barrier	W-25Re
Side Reflector Cooling	Thermal radiation to space
Reactor Dimensions [m (in.)]	
Pressure vessel length	1.32 (51.8)
Reactor outside diameter	
Reflector segments OD, full-in (min.)	0.553 (21.8)
Reflector segments OD, launch position (max.)	0.956 (37.6)
Core L/D	0.864
Side reflector thickness (BeO)	0.102 (4.0)
Reactor Weight [kg (lb)]	784 (1725) Wet
Fuel Elements	
Fuel density (g/cm ³)	13.58
U-235 enrichment	93%
U-235 loading [kg (lb)]	104 (229)
OD [mm (in.)]	6.35 (0.25)
Cladding thickness [mm (in.)]	0.381 (0.015)
Clad barrier thickness [mm (in.)]	0.127 (0.005)
Radial fuel-to-clad clearance at assembly, nominal [mm (in.)]	0.0381 (0.0015)
Spacing between fuel elements [mm (in.)]	6.60 (0.26)
Fuel element spacers	
Type	Cb-1 Zr-0.1 C disks (with central hole)
Number of fuel elements	6 (including 2 end plugs)
Diameter [mm (in.)]	6.60 (0.26)
Length, including barrier disks [mm (in.)]	12.7 (0.5)

TABLE IV-6
SNAP-50/SPUR DESIGN DATA SUMMARY
(Sheet 2 of 2)

Fuel Elements (continued)	
Fuel length [m (in.)]	0.256 (10.1)
End reflector length (depleted UN at shield end) [mm (in.)]	76 (3.0)
Number of fuel elements in core	1603
Number of fuel element cans in core	
Hexagonal	13 (91 fuel elements each)
Partial	6 (70 fuel elements each)
Distance across flats	
Hexagonal can [mm (in.)]	65.5 (2.58)
Can spacing [mm (in.)]	2.54 (0.1)
Can spacers	
Type	TiC buttons on hex ring
Location	Core midplane on 7 cans
Can wall thickness [mm (in.)]	0.864 (0.0034)
Core and Pressure Vessel, Nominal Dimensions [mm (in.)]	
Effective core diameter	297 (11.7)
Radial distance between effective core and side reflector BeO	28.2 (1.11)
Effective thickness, coolant flow annulus	11.1 (0.438)
Pressure vessel ID	319 (12.6)
Pressure vessel thickness	6.53 (0.257)
Pressure vessel OD	332 (13.1)
Pressure vessel-to-jacket gap	5.08 (0.20)
Preheat jacket thickness	1.52 (0.06)
Preheat jacket OD	345 (13.6)
Side Reflector, Nominal Dimensions [mm (in.)]	
Inner can wall ID, full-in	350 (13.8)
Reflector thickness	
Total	105 (4.12)
BeO	102 (4.0)
Length	434 (17.1)
Can wall thickness	1.57 (0.062)
BeO block size	12.7 x 19.1 to 12.7 x 12.7 (0.5 x 0.75 to 0.5 x 0.5) tapered (102-mm [4-in.] length)

neutron energy spectrum: the median energy of neutrons causing fission was approximately 0.35 to 0.55 MeV. Columbium alloy was used for the pressure vessel, core support structure, and fuel element cladding, and stainless steel was used for all support structures that operated below 978°K (1300°F).

The fuel was enriched uranium in the form of high-density UN. The UN fuel and depleted UN end reflectors were contained in sealed columbium alloy tubes. These fuel elements also contained gaseous fission product accumulation spaces at the cool end of the core. The clad of each fuel element was separated from its neighbor by placing Cb alloy disk spacers along the length of the fuel element.

The fuel elements were grouped like "straws in a box" into 19 fuel can subassemblies; these were supported and spaced by upper and lower core plates to form the reactor core. Ceramic spacing buttons were used to maintain the 2.5-mm (0.1-in.) spacing between cans at the fuel midplane.

A Cb alloy shroud formed an annular passage around the pressure vessel for the flowing helium required in preheating and removal of decay heat.

Six pivoted reflector segments were arranged so that, in the fully closed position, they completely surrounded the reactor core. These segments consisted of columbium alloy containers filled with small blocks of beryllium oxide; they were connected to the reactor and reflector support structure by means of flexure bearings at each end. Control of the reactor was achieved by rotating the reflector segments to change the amount of neutron leakage.

Each segment was individually driven by a collinear rotating hermetic can drive consisting of a stepping motor and a harmonic gear reduction unit. The motors and harmonic drives were placed behind the neutron shield and were connected to their respective reflector segments by shafts that penetrated the shield.

The reactor was controlled by an ON-OFF type of controller. Startup was accomplished with a preprogrammed startup controller until a set reactor coolant outlet temperature of 1100 to 1200 K (1500 to 1700°F) had been reached. At this time, the control was switched to the reactor controller, which used the lithium outlet temperature as the controlled variable.

(ii) Pressure Vessel and Core

The pressure vessel and core assembly are shown in Figure IV-11, and a typical fuel element is shown in Figure IV-12. A W-Re compatibility barrier separated the fuel from the cladding and disk spacers, and each fuel element was separated from adjacent members by columbium alloy disk spacers that were welded to and formed part of the fuel cladding. By providing thick points of contact, these spacers eliminated the question of fuel Cb alloy cladding damage due to self-welding and metal pull-out. Central holes in the four interior spacers interconnected all regions of the element to allow the use of a single fission gas accumulation volume located at the cool end. With the disk spacer pin design, axial clearance between the fuel and disk spacers had to be provided to accommodate differential thermal growth and fuel expansion due to irradiation. W-Re springs were provided in each fuel section to hold the fuel against the midplane spacer under launch and operational inertia loading. A single depleted UN end reflector located between the core and shield was also included within each fuel element. The primary function of this end reflector was to reduce nuclear heating in the neutron shield.

The fuel elements were arranged in a triangular pitch and were contained in 19 fuel cans. Within each can, they were trapped between an upper and lower grid plate, similar to straws in a box. Of these 19 cans, 13 were hexagonal in cross section and contained 91 elements, and 6 were partial hexagonal in cross section and contained 70 fuel elements. The partial fuel cans were used to obtain the desired cylindrical-shaped outer boundary. To provide the capability for unloading fuel in place in a ground test, the cans were spaced 2.5 mm (0.1 in.) apart to prevent self-welding between cans. This spacing was

Figure IV-12. PWAR-20 (SNAP-50/SPUR) Reactor Fuel Element

the minimum practical thickness for TiC spacers used to maintain core midplane spacing.

A rigid core design was provided by the two core plates that radially supported the ends of the fuel cans and the button-type spacers between the cans at the core midplane. The buttons were attached to hexagonal columbium alloy rings that were placed around the center fuel can and the six outer cans to limit positive reactivity changes due to fuel can bowing. Ceramic or cermet journals in the upper core plate were used to prevent self-welding and allow axial expansion. Axial fuel element support was provided by mechanically attaching the fuel can nozzles to the lower core plate. The core was supported axially by the core support cone, which was mechanically attached to the lower head of the pressure vessel. Radial support at the core upper end was provided by lugs on the core plate that fit closely to the pressure vessel inside diameter.

Lithium coolant entered and left the pressure vessel at the lower head. The coolant flowed upward from the inlet through an annular passage formed by the outer boundary of the core and the pressure vessel wall, thus cooling the core boundary and pressure vessel before entering the core from the upper plenum. Flow control orifices were provided in the exit nozzles of all but the center fuel can. These orifices were to distribute the coolant flow through the core to obtain a nominal 1370 K (2000°F) mixed-mean coolant outlet temperature from each can.

The pressure vessel was enclosed by a hermetically sealed columbium alloy preheat jacket, the purpose of the jacket was to channel high-temperature helium to preheat and maintain the temperature of the core and pressure vessel at 810 K (1000°F) during ground launch preparation operations. This jacket provided the inert gas atmosphere required for the oxidation protection of the pressure vessel and could also be used as a duct for forced-convection cooling during after-heat removal after reactor testing.

(iii) Reactor Control

Each of the six reflector segments were made of 0.102-m- (4-in.-) long blocks of sintered BeO nested in a columbium alloy container. The container was vented to allow the helium generated in the BeO to pass to space. A protective coating on both the inside and outside of the columbium was required to prevent oxidation during reactor preheating in air.

All of the BeO surrounding the core was in the movable segments, thus producing the greatest control range with this configuration. The reflector segments, which were connected to the support structure by means of flexure bearings at each end, controlled the reactor by rotating in or out on these bearings, thereby controlling the neutron leakage. These bearings were located so that their maximum temperature during full-power operation was limited to 810 K (1000°F).

Each reflector segment was actuated by a collinear control drive mechanism which was located behind the reactor shield where the maximum temperature would be 590 to 700 K (600 to 800°F). The neutron dose level at the drums was approximately 1×10^{17} nvt (>1 MeV). The mechanism was contained to provide an inert gas atmosphere for all components.

The design consisted of a direct current 1.8° stepping motor, similar to the Superior Electric SS-250, directly coupled to a harmonic gear reducer, and a hermetically sealed inert atmosphere container. Hermetic sealing was provided by an external rotating can that terminated in a horizontal bellows at the base. The stepping motor had the following operating characteristics:

• Pulse rate range	4 to 48 pulses/s in
4-pulse increments
• Angular rotation/pulse	1.8°
• Measured torque output at
672 K (750°F)	0.01 kg·m (150 oz·in.)

A single-phase 160:1 harmonic gear transducer transmitted rotation and torque to the reflector segments.

The estimated control segment worth was 10.0¢/deg, and the 1¢/s design point shim rate was obtained at 8 pulses/s. The range of shim rates available with this design by changing motor pulse rates was 0.5 to 6.0¢/s. The calculated maximum torque load on the motor was 0.004 kg·m (55 oz·in.), which is approximately 30% of the measured motor high-temperature torque capability.

Absolute reflector position was continuously monitored for diagnostic purposes by use of a direct-coupled rotary differential transformer.

The reactor controller concept consisted of starting the reactor on a preprogrammed schedule using a startup controller to bring the reactor critical and increase power until the reactor core reached a preset temperature of 1100 to 1200 K (1500 to 1700°F). Reactor control was then switched from the startup controller to a steady-state reactor controller. The steady-state controller maintained a set reactor coolant outlet temperature through a control system in which temperature error signals were used to actuate the control drives. The measured core coolant outlet temperature was compared with a requested value to generate an error signal. This temperature error signal was acted on by lead lag compensation circuits. When of sufficient magnitude, the compensated temperature error signal would actuate the control drive motor. The motor drives the reflector through a harmonic gear train.

(iv) Support Structure

The reactor support structure was designed to allow the core and pressure vessel assembly to be inserted into a fully assembled and tested reflector control drive assembly. The reflector support structure consisted of upper and lower stainless steel alloy support rings joined by six equally spaced columbium alloy pylons to form a "bird cage" structure. Flexure bearings, which supported the movable reflector segments, were mounted on the

support rings. The reflector structure supported the core and pressure vessel through 12 columbium alloy upper fingers and 12 lower fingers.

The reactor assembly was supported from the shield by six stainless steel struts connected to the lower reflector support ring. Loads were transmitted through the lithium hydride shield container to the main power plant support structure.

The control drives were supported from the lower head of the shield by stainless steel alloy struts. These struts were connected to a support ring to which the control drives were fastened with the stainless steel bolts.

(6) SPR-6

The SPR-6 was to be a 10-MWe, Rankine-cycle, nuclear electric power system for space propulsion. The conceptual design was prepared by the Lawrence Radiation Laboratory in the late 1960s as a follow-on to the SNAP-50 program. It is more completely described in Ref. IV-17. No reactor of this concept was ever built.

(a) Reactor Core

In this conceptual design, the reactor core is composed of a close-packed bundle of fuel elements supported on the downstream end by a 30-cm- (11.8-in.-) thick tungsten support structure and on the upstream end by a spline attachment providing for axial and radial growth. The downstream support structure also acts as a radiation shield. Figure IV-13 is an artist's sketch of the SPR-6 reactor. General characteristics are given in Table IV-7.

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Figure IV-13. Artist's Sketch of the SPR-6 Reactor

TABLE IV-7
CHARACTERISTICS OF THE SPR-6 REACTOR

<u>Nuclear Parameters</u>	
Power level (MWt)	57
Lifetime (h)	10,000
Maximum fuel burnup (at. %)	3.3
Average fuel power density (W/cm ³)	575
Radial peak-to-average power density	1.12
Axial peak-to-average power density	1.40
Median neutron energy causing fission (MeV)	0.30-0.33
Maximum neutron flux in core above 1 MeV (n/cm ² ·s)	2.6×10^{14}
Prompt neutron lifetime, λ^* (s)	8.1×10^{-8}
Delayed neutron fraction, β_{eff}	0.0068
<u>General Dimensions [cm (in.)]</u>	
Fueled-core length	87.0 (34.3)
Fueled-core mean OD	62.1 (24.4)
Liquid-lithium-control-zone OD	22.7 (8.9)
Reactor-coolant-return-duct diameter	15.3 (6.0)
Pressure-vessel ID	65.0 (25.6)
Side-reflector ID	67.4 (26.5)
Side-reflector OD	71.4 (28.1)
<u>Hybrid Fuel Element Dimensions [cm (in.)]</u>	
Fuel radius	0.216 (0.085)
Fuel-element-cladding inside radius	0.225 (0.088)
Fuel-element-cladding outside radius	0.250 (0.098)
Coolant-tube inside radius	0.220 (0.087)
Coolant-tube outside radius	0.250 (0.098)
Number of fuel tubes	8064
Number of coolant tubes	4032
<u>Masses [kg (lb)]</u>	
Mass of uranium required	1242 (2732)
Mass of U-235 required	865 (1903)
<u>Thermal-Hydraulic Parameters</u>	
Reactor-coolant flow rate [kg/s (lb/s)]	191 (420)
Reactor-coolant outlet temperature [K (°F)]	1650 (2510)
Reactor-coolant temperature rise [K (°F)]	73 (131)
Reactor-coolant inlet pressure [MPa (psia)]	2.6 (380)
Reactor pressure drop [kPa (psi)]	90 (13)

In cross section, the active core is an annular region surrounding a central coolant-return passage and a liquid control region. The outer boundary of the core is 12 sided to accommodate the hexagonal packing pattern of the fuel elements. The fueled core is enclosed by thin polygonal W-25% Re membranes at the inner and outer radii.

A slight radially inward net pressure is produced on the outer membrane by venting the stagnant lithium fluid present between the core and the pressure vessel to the upstream pressure. A slightly radially outward net pressure exists across the inner membrane because of the core pressure drop. Thus, the core acts as a rigid column between the upstream and downstream supports, dynamically restrained by the side membranes and the radial pressure gradients. This concept of a dynamically restrained core prevents changes in the positions of the fuel elements and keeps the flow passages aligned.

Surrounding the core is a 1.2-cm- (0.49-in.-) thick cooled tantalum pressure vessel and a segmented 2-cm- (0.8-in.-) thick molybdenum side reflector. The reactor pressure vessel is clad on the inner surface with tungsten material so that the reactor coolant is exposed only to the UN fuel and tungsten base materials.

A separate cooling loop maintains the reactor pressure vessel, as well as the boiler-preheater pressure vessel, at temperatures below 1400 K (2060°F). Although this cooling loop adds some complexity to the subsystem, it makes possible a minimum reactor-vessel thickness and thus enhances the side-reflector control worth, reduces the overall subsystem mass, and simplifies fabrication of the pressure vessel. The cooling loop extracts heat from the reactor and boiler-preheater pressure vessels by conduction and transfers it to potassium liquid in the preheater.

A liquid-lithium, high-temperature primary radiation shield is created by an enlarged exit plenum. A W-25% Re structure, also present in the plenum, acts as the axial support for the core. A neutron flux attenuation of 10^{-2}

in this region reduces the problems of overheating in the main LiH radiation shield located outside the pressure vessel.

The reactor coolant is returned to the reactor by a 15.3-cm- (6-in.-) diameter central duct. An annular region 3.7 cm (1.5 in.) thick surrounding the central duct contains liquid-lithium control circuits. Pressure drop in the reactor coolant return duct is about 10% that in the core. The optimum cross-sectional area of the reactor coolant return duct plus the annular control region is about 15% of the fueled core area.

The radial power profile is flattened by varying the U^{235} enrichment in four annular zones. Radial peak-to-average power density ratio is 1.12. Axial U^{235} enrichment is uniform and yields the natural cosine-like power density with a peak-to-average value of 1.4.

(b) Hybrid Fuel Elements

The core contains hybrid fuel elements in an arrangement of close-packed, equal-diameter metal tubes (see Figure IV-14). Every third tube is unfueled and serves as a reactor coolant passage. Interstices between the tubes are filled with stagnant liquid lithium. Allowance for fuel swelling due to burnup is provided by a small gap between the fuel pellets and the cladding.

The hybrid fuel element provides cladding redundancy since the flowing lithium coolant is separated from the fuel by a double wall. Its pressure drop characteristics are intermediate between those of pin-type and tube-type fuel elements. Although a low fuel volume fraction results from using hybrid fuel elements, reactor criticality is still achievable.

The linear swelling of the fuel is assumed to be equal to twice the burnup fraction. Establishment of an average 2% burnup limit appeared reasonable in light of available material data. Fuel swelling up to twice that assumed could be accommodated by the gap provided.

(a) FUEL ELEMENT CONFIGURATION

- URANIUM MONONITRIDE FUEL
- TUNGSTEN 25% RHENIUM CLADDING
- LITHIUM COOLANT
- STAGNANT LITHIUM

(b) 30° SEGMENT OF HYBRID FUEL ELEMENT

(c) HYBRID FUEL ELEMENT TEMPERATURE
RISE ABOVE MEAN COOLANT TEMPERATURE

9467-22

Figure IV-14. SPR-6 Hybrid Fuel Element

The swelling allowance gap is initially filled with liquid lithium bonding fluid. Fission gases are removed continuously from the fuel rather than being allowed to collect. This implies a nominal circulation of the lithium bonding fluid and continuous separation of fission products. By comparison, if helium rather than lithium were used as the bonding agent, there would be a 130 K (230°F) temperature drop across the swelling allowance gap.

The temperature distribution of the hybrid elements does not vary significantly from that of a uniformly cooled rod. This is because the thermal conductivities of lithium and tungsten are high and nearly equal.

Heat transfer for these elements is based on a slug-flow model. If the coolant is assumed to flow with uniform velocity through the passage and if there is assumed to be temperature continuity at the coolant-cladding interface, the problem reduces to a source-sink conduction problem. With this assumption, a constant Nusselt number of 8 is applicable for heat transfer in circular tubes.

Fuel element dimensions were selected on the basis of fuel thermal stress. Thermal stress is considered to be important during startup and shutdown of the reactor. The degree of fuel cracking that could be tolerated, however, is not known. Data on UN indicates a modulus of rupture strength of 70 MPa (10,000 psi) at 1650 K (2500°F). An allowable stress of half this value was applied in sizing the fuel elements.

Stresses in the cladding and coolant tubes due to bundling pressure are quite low. Distortion of the coolant tubes is avoided by making their wall thickness 20% greater than the fuel cladding thickness.

(c) Reactor Control

Nuclear ground safety before reactor operation is provided by initially filling the fueled core with lithium having 28% enrichment in the isotope Li^6 and then filling the remaining volume of the reactor coolant loop

with Li^7 . The increased concentration of Li^6 in the fueled core during ground handling would render the reactor subcritical by 10% Δk . This isotope separation would be maintained since lithium is solid below 450 K (350°F).

At the beginning of life, all lithium mixes together in the reactor coolant loop. This, in effect, removes Li^6 from the fueled core and renders the reactor near critical. This ground safety concept is used instead of mechanical devices in the reactor.

Redundancy in reactor control is provided by two independent controllers. The adjustable reactivity range required of each control system can be reduced since reactivity reserves and provisions for the failure of individual segments of each controller are accounted for by the backup controls. Each mechanism, by itself, must be capable of carrying the mission to completion.

A 2-cm- (0.8-in.-) thick molybdenum side reflector is used for the mechanical control system. Variations in reactivity are made by axially translating segments of the reflector. The segmentation plus individual controls for each segment provides a high degree of redundancy.

Molybdenum was chosen as the side reflector material over the slightly more effective BeO since it offered a simple construction of solid metal reflector segments. Beryllium oxide would require a metal container.

Lithium control tubes positioned in a 3.7-cm- (1.46-in.-) thick annulus between the central duct and the fueled core are used for the liquid control system. These tubes can be filled with or voided of Li^6 , which acts as the neutron-absorbing medium. Heat removal from the control region is accomplished through the interstitial coolant of the core; this coolant is circulated between adjacent control tubes. Control tubes are double walled as an added precaution to prevent any accidental transfer of Li^6 into the main coolant stream. The control tubes contain metal screens in the form of capillary wicks, similar to those in heat pipes. This allows the uniform distribution of the liquid lithium under zero-gravity conditions.

The quantity of lithium in the control tubes is varied with a control element loop. Lithium vapor leaves the capillary screens continuously by evaporation and passes out of the control tubes through a duct. The vapor is condensed and the resulting liquid is pumped back into the control tubes to complete the cycle. The rate of evaporation is determined by the rate of heat generation in the control tubes. If liquid is pumped into the control tubes faster than it is evaporated, the proportion of liquid will increase.

(7) NASA-Lewis Fast Reactor

NASA-Lewis conducted, in the 1969-1971 time period, a technology program on a compact fast reactor for space electric power use. Figure IV-15 shows the reactor and fuel pin configurations that were evolved. A cross-sectional view of the reactor is given in Figure IV-16, and the reactor's characteristics are summarized in Table IV-8. A more complete description of the concept is given in Reference IV-18.

A unique feature of the reactor is the use of movable fuel in the control drums. Each of the six drums contains 11 fuel pins. The opposite side of a drum contains a poison material. The reflectors are a molybdenum alloy.

The reactor core contains 181 fuel pins. These add to the 66 drum pins for a total of 247. The fuel is fully enriched UN clad with T-111 (Ta-8W-2Hf). A thin tungsten barrier is used on the inside of the T-111 to prevent reaction between the UN and T-111.

The reactor is cooled with lithium enriched in Li^7 . It is designed to operate at an outlet temperature of 1222 K (1740°F). The reactor is contained within a T-111 pressure vessel that is 57.7 cm (22.7 in.) in diameter and 68.5 cm (27 in.) high.

REACTOR CONFIGURATION CORE FUEL PIN

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Figure IV-15. NASA-Lewis Fast Reactor and Fuel Pin Configuration

Figure IV-16. Cross Section of Reactor Core

TABLE IV-8
NASA-LEWIS FAST REACTOR CHARACTERISTICS
(Sheet 1 of 2)

<u>General</u>	
Reactor power (MWt)	2.17
Operating life (h)	50,000
Coolant inlet temperature [K (°F)]	1165 (1637)
Average coolant outlet temperature [K (°F)]	1222 (1740)
Number of fuel pins	247
Fuel pin diameter [cm (in.)]	1.90 (0.75)
Active core length [cm (in.)]	39.6 (14.8)
Fuel loading, U-235 [kg (lb)]	182 (400)
Total mass of reactor [kg (lb)]	1600 (3520)
<u>Reactor Structure</u>	
Material	T-111
Pressure vessel OD [cm (in.)]	57.7 (22.7)
Pressure vessel wall thickness [cm (in.)]	0.635 (0.25)
Pressure vessel height [cm (in.)]	68.5 (27.0)
<u>Fuel Pin</u>	
Composition	UN
U-235 enrichment (%)	93.2
Clad material	T-111
Fuel pin OD [cm (in.)]	1.90 (0.75)
Fuel pin length [cm (in.)]	43.8 (17.2)
Clad thickness [cm (in.)]	0.147 (0.058)
Tungsten liner thickness [cm (in.)]	0.013 (0.005)
Fuel pellet OD [cm (in.)]	1.58 (0.622)
Nominal radial fuel-clad gap [cm (in.)]	0.011 (0.004)
<u>Fluid Flow</u>	
Coolant	Li-7
Mass flow [kg/s (lb/s)]	9.4
Coolant velocity [cm/s (ft/s)]	115 (3.8)
Plenum-to-plenum [kPa (psi)]	7 (1)
<u>Heat Transfer</u>	
Radial peak-to-average power	1.33
Axial peak-to-average power	1.23
Maximum clad temperature [K (°F)]	1250 (1790)
Maximum nominal fuel temperature [K (°F)]	1370 (2006)
Maximum hot spot fuel temperature [K (°F)]	1450 (2150)

TABLE IV-8
NASA-LEWIS FAST REACTOR CHARACTERISTICS
(Sheet 2 of 2)

<u>Control System</u>	
Materials	TZM, T-111, UN
Number of control drums	6
Number of fuel pins per drum	11
Drum OD [cm (in.)]	14.6 (5.75)
Drum length [cm (in.)]	55 (21.7)
Reactivity worth of drums (% $\Delta k/k$)	8.51
<u>Reflectors</u>	
Material	TZM
Side reflector thickness [cm (in.)]	7.6 (3)
End reflector thickness [cm (in.)]	5.08 (2)

A critical experiment was performed that confirmed the required critical mass and control drum worth.

(8) Nu ERA II

Following termination of the space reactor program in 1973, the Westinghouse Advanced Energy System Division (WAESD) continued to study advanced high-powered reactors for aircraft propulsion and other applications. The concept was called NuERA for nuclear extended-range aircraft, and the entire primary system was designed to be contained in a spherical containment vessel. A schematic of the reactor concept for Nu ERA II is shown in Figure IV-17. Specifications for the reactor are given in Table IV-9. A reactor test was never performed for this concept.

The reactor is lithium cooled and uses UN fuel clad with ASTAR 811C. It was designed to operate at a power level of 275 MWt for 10,000 h at a reactor outlet temperature of 1255 K (1800°F). Reactivity control was to be provided by reflector control drums and in-reactor shim rods. The reflector material between control drums was to be nickel.

(9) Medium Power Reactor Experiment (MPRE)

Studies at the Oak Ridge National Laboratory of systems for the production of electrical power in space vehicles indicated that fission-reactor and turbine-generator Rankine-cycle subsystems, utilizing potassium as the working

Figure IV-17. WAESD NuERA Lithium-Cooled, Pumped-Loop Reactor

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TABLE IV-9
WAESD NuERA II REACTOR SPECIFICATIONS
(Sheet 1 of 3)

<u>General Reactor Concept</u>	
Reactor power (MWt)	275
Design lifetime at full power (h)	10,000
Structural material	Niobium - 1% zirconium
Reactor coolant	Lithium 7
Coolant inlet temperature [K (°F)]	1033 (1400)
Coolant outlet temperature [K (°F)]	1255 (1800)
Fuel	Uranium nitride
Fuel-clad barrier	Tungsten
Fuel clad	ASTAR 811C
Fuel arrangement	Pins clustered in hexagonal array
Method of reactor control	Drums in reflector and shim rods in core
Number of control drums	12
Number of shim rods	7
Absorber material	$B_{10}C$
Moderator material	Yttrium hydride
Reflector material:	
Side	Nickel
End	Tungsten
<u>Reactor Configuration</u>	
Equivalent active core diameter [m (in.)]	0.762 (30)
Active core length [m (in.)]	0.584 (23)
End reflector thickness (each end) [cm (in.)]	8.9 (3.5)
Side reflector thickness nominal [cm (in.)]	22.9 (9.0)
Core support plate thickness [cm (in.)]	5.1 (2.0)
Core barrel thickness [cm (in.)]	0.38 (0.15)
Core barrel ID [cm (in.)]	83.8 (33)
Pressure shell OD [cm (in.)]	135.4 (53.3)
Pressure shell thickness [cm (in.)]	2.54 (1.0)
Pressure shell length [m (in.)]	0.254 (100)
<u>Fuel Cluster Can Configuration</u>	
Number of full fuel cluster cans in core	48
Number of partial fuel cluster cans in core	6
Fuel can shape	Regular hexagon
Fuel can distance across flats	9.31 (3.67 (full)
(outside) [cm (in.)]	7.24 (2.85) (partial)

TABLE IV-9
WAESD NuERA II REACTOR SPECIFICATIONS
(Sheet 2 of 3)

<u>Full Cluster Can Configuration (continued)</u>	
Fuel can thickness [cm (in.)]	0.081 (0.032)
Number of pins per full fuel cluster can	127
Number of pins per partial fuel cluster can	103
Total number of pins in core	6,714
Type of spacer	Spiral wire, 0.061 cm (0.024 in.) diameter, 20 cm (8 in.) pitch
Fuel pin OD [cm (in.)]	0.737 (0.29)
Fuel pin clad thickness [cm (in.)]	0.058
Fuel-clad barrier thickness [cm (in.)]	0.0025
Helium thermal bond thickness [cm (in.)]	0.006
Length of fuel pin, including end caps [cm (in.)]	96.5
Fuel pin fission gas chamber length [cm (in.)]	20.3 (8.0)
<u>Shim Rod Configuration</u>	
Number of shim rods	7
Shim rod shape	2 prisms with hexagonal cross section
Shim rod distance across flats [cm (in.)]	8.5 (3.35)
Combined length of shim rod sections [cm (in.)]	48.3 (19.0)
Maximum separation between shim rod sections [cm (in.)]	48.3 (19.0)
Number of poison pins per section	66
Number of moderator pins per section	42
Pin clad thickness [cm (in.)]	0.051 (0.02)
Boron carbide pellet diameter [cm (in.)]	0.584 (0.23)
Yttrium hydride pellet diameter [cm (in.)]	0.584 (0.23)
<u>Side Reflector Configuration</u>	
Core barrel OD [cm (in.)]	84.58 (33.3)
Inner assembly ID [cm (in.)]	84.73 (33.36)
Inner assembly effective OD [cm (in.)]	107.26 (42.2)
Outer assembly effective ID [cm (in.)]	107.62 (42.2)
Outer assembly OD [cm (in.)]	130.3 (51.3)
Reflector can thickness [cm (in.)]	0.203 (0.08)
Number of inner assemblies	12
Number of outer assemblies	12

TABLE IV-9
WAESD NuERA II REACTOR SPECIFICATIONS
(Sheet 3 of 3)

<u>Control Drum Configuration</u>	
Number of control drums	12
Drum can OD [cm (in.)]	22.35 (8.80)
Drum can ID [cm (in.)]	22.20 (8.74)
Diameter of drum center circle [cm (in.)]	107.44 (42.3)
Clearance between control drum and reflector [cm (in.)]	0.43 (0.17)
Extent of poison distribution around drum (deg)	150
Thickness of poison sector [cm (in.)]	3.49 (1.37)
Thickness of poison can [cm (in.)]	0.05 (0.02)
Total number of poison pins per drum	96
Clad thickness on poison pin [cm (in.)]	0.05 (0.02)
Diameter of B ₄ C pellets [cm (in.)]	0.91, 0.81, 0.74, 0.64 (0.36, 0.32, 0.29, 0.25)

fluid, were the most promising. Further studies led to the conclusion that the high reliability required could be obtained only with the simplest possible configuration, that is, a single-loop in which the potassium is boiled in the reactor core and the vapor flows directly to the turbine.

The two most vital questions associated with the feasibility of such a system were considered to be whether stainless steel is compatible with boiling potassium and whether the heat transfer characteristics of boiling potassium under forced convection would be adequate to remove the heat from a reasonably small core. Favorable results of the feasibility investigation led to the decision to proceed with a program to carry out a reactor experiment. A design power output of 1 MWt, or about 140 kWe, was chosen, in part because it seemed well suited to the needs of many important space missions, and in part because it was intermediate between the power levels of the SNAP-8 and SNAP-50 reactors. Because of this intermediate output, it was called the Medium-Power Reactor Experiment (MPRE).

MPRE was designed to investigate the feasibility of a boiling-potassium reactor system, which was conceived as a major simplification of the multiloop systems commonly used in concepts of nuclear reactors for space power systems. However, the MPRE program was terminated before a reactor experiment was built and tested.

A vertical section through the center line of the MPRE is shown in Figure IV-18. In this design, the reactor core consists of a bundle of fuel rods, each consisting of fully enriched UO_2 pellets contained in a type 316 stainless steel capsule with a 12.7-mm (0.500-in.) OD.

The fuel rods are arranged in an equilateral triangular pattern with a center-line spacing of 14.3 mm (0.5625 in.), so the coolant-passage volume fraction in the active lattice is 0.283. The core is surrounded by a beryllium oxide reflector that is 8.26 cm (3.25 in) thick on the side and 12.7 cm (5.00 in.) thick on the bottom. It is fabricated from a series of disks, each

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Figure IV-18a. Cross Section Through the MPRE Reactor Assembly

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Figure IV-18b. The MPRE Fuel Element

approximately 3.0 cm (1.2 in.) thick. A 7.6-cm- (3.0-in.-) long stack of BeO pellets above the UO_2 in each fuel rod serves as a portion of the top reflector. An additional 10.9 cm (4.3 in.) of void space above the BeO pellets provides a reservoir for fission-gas accumulation and contains a piston and molybdenum spring to restrain axial movement of the BeO and UO_2 pellets. The lower BeO reflector is divided into four independent quadrants that may be moved axially to adjust reactivity.

Above the core is a vapor separator designed to give a vapor exit quality of about 99%. Immediately below it is the expansion tank. The type 304 stainless steel pressure vessel encloses the entire vapor-separator region but passes between the core and the reflector, where its thickness is 6.4 mm (0.25 in.). Thus the side reflector, the movable bottom reflector, and all the parts of the control drive mechanism are outside the potassium circuit.

To avoid excessive coolant flow bypass around the core perimeter, the core is surrounded by a thin stainless steel flow baffle that conforms in shape to the fuel-rod array. The baffle also performs two additional functions: it constrains the radial movement of the fuel rods, and it defines a small annular coolant passage between the baffle and the core vessel for the return of coolant from the vapor separator to the core inlet. This flow also removes the heat generated by radiation in the vessel and the inner portion of the side reflector.

Surrounding the entire reflector is a thin neutron-absorbing layer — a 6.4-mm- (0.25-in.-) thick layer of 0.1% boron steel. This layer reduces activation of components (such as the control rod guides and supports). In addition, the boron layer isolates the reactor from its environment so that no accidental addition of reflecting material can increase the reactivity. In fact, adding water between the boron layer and the reactor shield would cause a small loss of reactivity.

The BeO blocks in the side reflector incorporate twelve 2.54-cm- (1-in.-) diameter holes centered on a circle coaxial with the core. By adding BeO rods in any or all of these holes, the reactivity can be adjusted after the reactor is installed in the facility but before its operation so as to minimize the excess reactivity available from control plug insertion. In this way, it should be possible to limit the maximum available reactivity in the hot, clean reactor to about 0.5%.

The fluid flow path through the reactor can be traced in the vertical section shown in Figure IV-18. Liquid potassium enters the core at the bottom, rises between the 12.7-mm- (0.5-in.-) diameter fuel elements, emerges from the core at a vapor quality of about 20%, and enters the vapor separator, where the liquid is skimmed off and discharged to the expansion tank directly beneath the vapor separator. The vapor passes radially outward at the bottom of the vapor separator and then upward around the outer perimeter and out the vapor exit pipe at the top. The liquid potassium from the boiler feed pump enters a set of four jet pumps operating in parallel beneath the expansion tank. These draw liquid from the expansion tank and discharge it so that it returns to the core inlet.

The principal geometric and performance data for the MPRE are presented in Table IV-10.

(10) Gulf General Atomic (GGA) Thermionic Reactor

An electric propulsion thermionic reactor was designed to produce 120 kWe with 162 UC-ZrC-fueled thermionic fuel elements (TFEs).^{IV-19}

TABLE IV-10
MPRE DESIGN PARAMETERS

<u>Reactor Operating Conditions</u>	
Thermal power (MW)	1
Coolant	Potassium (boiling)
Inlet temperature [K (°F)]	1106 (1530)
Outlet temperature [K (°F)]	1111 (1540)
Inlet pressure [kPa (psia)]	207 (30)
Outlet pressure [kPa (psia)]	200 (29)
<u>Reactor Geometry</u>	
Fuel element arrangement	Equilateral triangular pitch
Moderator	None in core
Diameter (equivalent) [cm (in.)]	23.3 (9.17)
Distance across corners [cm (in.)]	24.1 (9.50)
Length [cm (in.)]	29.5 (11.625)
Active core volume [liters (ft ³)]	12.58 (0.444)
Core free-flow area [cm ² (ft ²)]	0.84 (0.130)
Side-reflector thickness [cm (in.)]	7.62 (3.0)
<u>Fuel Element</u>	
Configuration	Rods
Fuel form	UO ₂ pellets in metal capsules
Number of fuel elements	241
Overall length [cm (in.)]	50.5 (19.875)
Fueled length [cm (in.)]	29.5 (11.625)
Fuel enrichment (%)	93
Cladding material	Type 316 stainless steel
Cladding thickness [mm (in.)]	0.5 (0.020)
Fuel element outside diameter [cm (in.)]	1.27 (0.50)
Center-line spacing [cm (in.)]	1.43 (0.5625)
Burnup at end of life (% of uranium atoms)	0.76
<u>Control Elements</u>	
Reactivity control method	Four movable BeO plugs (quadrants of the bottom reflector)
Control plug motion	Vertical displacement parallel to axis of reactor; total travel 8 in.
Vertical height of control plugs [cm (in.)]	21.6 (8.6)
Sheath material	Type 304 stainless steel
Combined worth of control plugs (% $\Delta k/k$)	5.0
<u>Pressure Vessel</u>	
Material	Type 304 stainless steel
Shell OD [cm (in.)]	26.7 (10.5)
Wall thickness [mm (in.)]	6.4 (0.25)
Overall length (with expansion tank and vapor separator) [cm (in.)]	86.4 (34)

Elevation and plan views of the reactor core are given in Figures IV-19 and IV-20. The core consists of 162 TFEs of about 3.3-cm (1.3-in.) OD arranged in a hexagonal lattice. The design of the UCZrC-fueled TFEs closely resembles that of the 6F series of prototype thermionic elements tested at GGA. A drawing of the 6F cell is given in Figure IV-21. The TFEs contain six thermionic diodes, each 5.1 cm (2 in.) in length, arranged axially and separated by 2.3-cm- (0.90-in.-) long intercell connector regions (see Figure IV-22). Axial reflector pieces of 10-cm- (4-in.-) long BeO are included on each end of the TFEs. Additional TFE data are given in Table IV-11.

Reactor control is provided by 11-cm-thick BeO shutters that open out to increase core leakage (Figure IV-20). The core and reflector shutters are separated by a niobium pressure vessel. NaK coolant flows in the spaces between the TFEs to remove waste heat from the core.

b. Gas-Cooled Reactors

(1) Kiwi

The Kiwi nuclear rocket reactors were the first designed and tested by LANL as part of the Rover project. The first of these, Kiwi-A, was tested in July 1959 at 70 MWt for 5 min; its primary purpose was to serve as a proof of principle.

The Rover reactor designs featured graphite-moderated, hydrogen-cooled cores. The fuel for the early Kiwi designs consisted of highly enriched UO_2 extruded in a hexagonal-shaped graphite fuel element. Each element contained 19 coolant channels. The elements were 1.9 cm (3/4 in.) across the flats and

Figure IV-19. Thermionic Reactor Elevation View

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Figure IV-20. Thermionic Reactor Plan View

9467-4

Figure IV-21. Arrangement of Components for F Series Cells

9467-3

Figure IV-22. Thermionic Fuel Element

ESG-DOE-13398

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TABLE IV-11

TFE DATA

a. Reactor and TFE Materials

Emitter	
Tungsten	Inner 0.7 mm (0.03 in.) vapor deposited from WF_6
	Outer 0.3 mm (0.01 in.) vapor deposited from WCl_x
Surface	(110) crystal orientation, 4.9 eV work function
Collector	Nb
Electric lead	Mo
Outer sheath	Nb-1% Zr
Fuel loading	90 UC-10 ZrC plus 4 wt. % W
Range of carbon/uranium ratio	$1.01 \leq C/U \leq 1.03$
Average theoretical density (%)	77
U^{235} inventory (total for reactor)	195 kg (429 lb)
Maximum emitter cavity loading	69% of theoretical density
Insulator seal	Brazed Nb- Al_2O_3 (Luclox)-Nb
Sheath insulator (tri-layer)	Gas pressure bonded Nb- Al_2O_3 -Nb

b. TFE Dimensions

Emitter dia [cm (in.)]	2.59 (1.02) ID - 2.79 (1.10) OD
Collector dia [cm (in.)]	2.86 (1.13) ID - 2.59 (1.02) OD
Alumina sheath insulator dia [cm (in.)]	2.59 (1.02) ID - 3.13 (1.23) OD
Inner sheath dia [cm (in.)]	3.13 (1.23) ID - 3.23 (1.27) OD
Outer sheath dia [cm (in.)]	3.23 (1.27) ID - 3.34 (1.31) OD
Interelectrode dia gap [cm (in.)]	0.029 (0.011)
Emitter area [cm ² (in. ²)]	49 (7.6)
Fuel length per cell [cm (in.)]	5.08 (2.0)
Unfueled length per cell [cm (in.)]	2.29 (0.9)
Length - six cells [cm (in.)]	44.20 (17.4)
Upper reflector length [cm (in.)]	10.16 (4.0)
Lower reflector length [cm (in.)]	10.16 (4.0)
End cell fittings & miscellaneous [cm (in.)]	3.70 (1.46)
Total length [cm (in.)]	68.20 (26.9)

132 cm (52 in.) in length. The core was made up of these elements supported by tie tubes to transmit the pressure load from the hot end to a support plate at the cold inlet end. The Kiwi-A reactor used a thick graphite reflector and gaseous hydrogen as the coolant. The Kiwi-B design used a beryllium reflector and had liquid hydrogen entering the reactor as the coolant. The reactor power level for Kiwi-B was also expanded to 1000 MW.

Problems encountered with the UO_2 -graphite fuel led to the development of BISO particulate fuel for Kiwi-B4E and all subsequent Rover reactors. This fuel consisted of highly enriched UC_2 kernels coated with pyrolytic carbon to form microspheres with diameters less than 1 mm (0.04 in.). These spheres were embedded in graphite to form the hexagonal fuel elements. Kiwi-B4E was operated for a maximum of 11 min at an outlet temperature of about 1900 K (3000°F). Stability problems caused by flow-induced vibrations led to the termination of the Kiwi development and the initiation of alternate rocket reactor designs.

(2) NRX

The initiation of the NERVA (Nuclear Engine for Rocket Vehicle Application) program saw the beginning of industrial participation in the Rover program by Westinghouse for reactor design. The Westinghouse reactor design, called NRX, was developed as an improved Kiwi-sized (i.e., 1000-MWt) reactor in a rocket engine configuration. The objective of NRX reactor development was to demonstrate reactor operation at 1100 MWt for 60 min. This goal was reached in December 1967 when NRX-A6 successfully ran for 62 min at 2220 K (3540°F) and 1100 MWt. Westinghouse participation in the NERVA program is summarized in Ref. IV-20.

The NRX, shown in Figure IV-23, used the same basic fuel element developed for the Kiwi reactor. To increase core life, the element was coated with niobium carbide to increase its corrosion resistance. The main emphasis in the NERVA program was to solve the mechanical problems that had been evident in the Kiwi design. That these problems were successfully solved is evidenced by the 62-min run of NRX-A6.

(3) Phoebus

The Phoebus rocket reactor design effort at LANL was divided in two parts. Phoebus-1 was a continuation of the effort to advance the technology of Kiwi-sized reactors, and Phoebus-2 was an effort to design a 5000-MWt reactor. The higher powered reactor was intended for use in manned missions to Mars, and the size selected was optimum for this purpose. In 1968, Phoebus-2A was successfully operated for 12 min at 4100 MWt at an exit temperature above 2200 K (3500°F).

(4) Pewee

Reduced interest in manned planetary missions following the Phoebus-2A test led to the development of a smaller rocket reactor named Pewee. The objective of Pewee was to demonstrate higher temperatures and a longer life capability at a reduced power level of about 500 MWt. Pewee, with only 400 fuel elements rather than the 1100 of Kiwi and NRX, was intended to serve as a less expensive test bed for rocket reactor development. In December 1968, Pewee-1 ran at 2555 K (4140°F) exit temperature and 514 MWt for 40 min. The core power density was 2340 MW/m^3 (66 MW/ft^3), which was then the highest ever achieved. It was later exceeded in the nuclear furnace, a small fuel test assembly.

9467-1

Figure IV-23. NRX Reactor

By the termination of the Rover program in 1973, demonstrations had been made of fuel life up to 109 min at 2500 K (4040°F) and of a power density of 4500 MW/m³ (127 MW/ft³). This was accomplished with fuel elements made from a composite of solid solution UC-ZrC particles in ZrC-coated graphite.

(5) Small Nuclear Rocket Engine (SNRE)

A short time after the completion of the NERVA program, a study was performed at LANL for a small nuclear rocket engine (SNRE).^{IV-21} Three different engine design concepts, designated α , β , and γ , were developed with each one based on essentially the same reactor configuration. The α and β reactors were to use fuel elements made up of a composite mixture of graphite and a solid solution of UC-ZrC. The γ reactor was to use a fuel element made up entirely of UC-ZrC.

The cross-sectional configuration of the reactor proper is shown in Figure IV-24 for an axial location about midway along the core. This illustration shows the relative locations of fuel elements (open hexagons) and support elements (hexagons with internal circles). In the general core array, each fuel element is symmetrically located among three support elements. This is the manner in which each fuel element is supported, i.e., by the tie tubes located in the three neighboring support elements. The total number of fuel elements is 564, and the number of support elements is 241.

The hexagonal fuel elements have 19 coolant channels, each with a diameter of 0.23 cm (0.09 in.), and the support elements have one large hole of 1.6 cm (0.63 in.) diameter. The elements are 0.89 m (35 in.) long and measure 1.91 cm (0.75 in.) from flat to flat. The total fuel-element mass is 433 kg (953 lb), and the total support element mass is 80.7 kg (178 lb). As shown in Figure IV-25, each fuel element is held in position by three adjacent support elements, and the corresponding hot-end support assembly. A sectional view of

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Figure IV-24. SNRE Reactor Core Cross Section

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Figure IV-25. Schematic of SNRE Fuel Elements, Support Elements, and Hot-End Support Hardware

the support-element assembly, with dimensions, is shown in Figure IV-26. The hot ends of both the fuel elements and the support elements may have brazed tips. These tips (of hot-pressed carbide-carbon composite) provide the elements with a support surface that will resist abrasion and other types of mechanical damage when in contact with the support assembly.

One of the several prospective geometries for the core periphery is shown in Figure IV-26. Reading outward from the fuel elements are: an insulator layer, a cooled inboard slat section, a metal wrapper, a cooled outboard slat section, and an expansion gap. Outside the expansion gap is the reflector system, which consists of a beryllium barrel and the reflector sectors themselves with the 12 control drums. There is a small coolant gap between the outside of the reflector and the reactor pressure vessel.

The beryllium barrel assembly consists of two beryllium cylinders 0.127 m (0.5 in.) thick and 0.388 m (15.3 in.) long. The assembly is captured between the aft and forward rings.

The reflector assembly is 0.864 m (34.0 in.) long, 0.147 m (5.8 in.) thick, has an ID of 0.673 m (26.5 in.). The reflector sectors are made of beryllium and, when installed, form a cylinder with 12 holes into which the control drums fit. The control drums are made of beryllium and have a boron-copper control plate attached to one side. The 12 drums are driven by six small actuators through a spline coupling. A solenoid-operated mechanism is provided to disengage the actuator in case of drum malfunction. In this event, a spring drives the drum to a predetermined position. The control-drum actuators are enclosed within the pressure vessel, and only penetrations for instrumentation, tie tube inlet lines, and tie-tube outlet lines are required.

9467-11

Figure IV-26. SNRE Tie Tube and Support Element Geometry

(6) Pluto — Tory II-C

The Tory II-C reactor was constructed and tested in a flight configuration suitable for the engine of a nuclear ramjet as part of the Pluto Program to develop a low-altitude, supersonic missile.^{IV-8} The reactor was successfully operated at full power of 500 MWt for approximately 5 min at an outlet temperature of 1450 K (2150°F). The length of the test was limited by the compressed air storage capacity which supplied the reactor coolant. A cursory examination following the test revealed no apparent damage to the reactor core. The cancelation of the project prevented a more extensive examination.

The Tory II-C reactor consisted of a core, a reflector, structural components, and a reactor duct as shown in Figure IV-27. The duct, a flanged circular cylinder, was 262 cm (103 in.) long and 145 cm (57 in.) in diameter and contained the fueled core, which was 130 cm (51 in.) long and 121 cm (47.5 in.) in diameter. Structural components included the support grid, tie rods, side support springs, peripheral shims, and base blocks.

About 47.5% of the gross reactor volume consisted of hollow, hexagonal beryllium oxide tubes. These comprised the homogeneously fueled moderators and reflectors that make up the reflected core. There were approximately 465,000 ceramic tubes having a typical length of 10 cm (4 in.). They were either unfueled or loaded to various degrees with highly enriched uranium dioxide. The hollow tubes were close-packed to provide a honeycomb pattern of

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Figure IV-27. Tory II-C Reactor Schematic

about 27,000 parallel flow channels running the length of the reactor. The flat-to-flat dimension of the hexagonal fuel tubes was 0.754 cm (0.297 in.), giving a fuel tube porosity of about 53%. Heat released in the fission of U^{235} was conducted to the channel walls and transferred by convection into the air passing through the channels. The ratio of the fueled channel-flow area to the overall cross-sectional area of the reactor was 0.33.

The core was reflected on all sides. The forward reflector was 24.6 cm (9.7 in.) thick, while the aft reflector was 6.1 cm (2.4 in.) thick. Both were essentially composed of hexagonal beryllium oxide tubes. The aft reflector contained chromium-cermet transition pieces; these manifolded seven fuel tube passages into one passage. The side reflector was composed of a ring of hexagonal BeO tubes in immediate contact with the core. The ring was nominally 5 cm (2 in.) thick. Nickel peripheral shims, nominally 2.5 cm (1 in.) thick, surrounded the BeO.

The forces on the reactor were accommodated by side and axial supports located in a high-pressure duct. The side support occupied the annulus between the reflected core and the duct. It consisted of 32 bands of corrugated springs; these exerted a uniform radial compressive force on the reflected core. The irregular radial boundary formed by the hexagonal tube columns meshed with the inner surface of thick nickel shims to assure a stable geometry. The corrugated springs held the nickel shims in place, but were flexible enough to accommodate the thermal expansion of the reflected core. Transverse inertia loads on the reflected core, which would arise during missile maneuvers, were transferred to the duct by 30 support rails bolted to the inside of the duct.

Axial loads were reacted at the front end of the duct. A support grid spanned the duct and was pinned to an interrupted ledge on the inside of the duct. Attached to the support grid were 121 tie rods; these passed through the reflected core and attached to base blocks at its rear. An axial compression was applied at each tie rod location by coil springs compressed between

the support grid and the front of the core. Thus, the bundle of almost one-half million parts comprising the reflected core was held in the desired position and geometry by side spring pressure and prestressed "hangers" or tie rods.

Nuclear control was achieved by moving hafnium control rods within the core. Twelve of these, called shim rods, were situated approximately 23 cm (9 in.) off axis; the remaining two were closer to the reflector. Of those two, one served as a vernier rod, the other as a safety rod. All move axially within especially large tie rods.

Normal shim rod movement was mechanically restricted to less than 7.6 cm/s (3 in./s). Scram time, however, was approximately 1.5 s. Four actuators located within the duct moved the shim rods; each actuator handled a set of three shim rods. The safety and vernier rods had separate actuators.

An independent safety assembly composed of six poison rods was also mounted within the ducting. This had been designed to yield a standby shutdown capability. It was to be used only when the reactor was at zero or low power, because the entry of these rods into the corresponding centrally located tie rods would restrict coolant flow in the tie rods. It was mechanically actuated from outside the duct to provide a high degree of reliability.

(7) Fixed Bed Reactor (FBR)

The fixed-bed reactor (FBR) is being developed as a gas-cooled reactor employing particulate fuel.^{IV-10} The 500- μ m- (0.02-in.-) diameter TRISO particulate fuel considered for use in this concept is to be derived from the fuel particles used in high-temperature gas-cooled reactors (HTGRs). The very large surface area of the particles would allow very high power densities,

high coolant-outlet temperatures, small temperature differences between fuel and coolant, and the very fast startup and shutdown of the reactor.

In the FBR design, illustrated in Figure IV-28, the particles are packed between two porous frits, forming a relatively thin annular fuel bed through which coolant flows radially. Neutrons are moderated in external reflectors outside and inside the annular fuel bed. The FBR can be operated in the continuous Brayton cycle mode with inert gas coolant (e.g., He) and a turbo-generator/compressor PCS. Depending upon further successful fuel development, the FBR has the potential for operation in the open-cycle mode with H_2 coolant, generating very high-power semi-continuous outputs. The same reactor might eventually operate bimodally, switching back and forth between open- and closed-cycle modes when desired.

The FBR might have the potential for power outputs up to 300 MWt in the He-cooled mode and 1000 MWt in the H_2 -cooled mode from a device 1 m (3.3 ft) in diameter by 1 m (3.3 ft) in height. Operating time can be long in the closed-cycle mode (months to years, depending on power) but is limited (to a few hours at most) in the open-cycle mode because of H_2 storage requirements.

The FBR could satisfy many potential applications. Two basic-size reactors might cover the continuous power range from 100 kWe to 100 MWe without serious weight penalty. Open-cycle, H_2 -cooled generation levels could be approximately five times the continuous levels. Coolant outlet temperature capability is high: 1500 K (2240°F) in the continuous mode and 2000 K (3140°F) in the open-cycle mode.

Present HTGR fuel particles can be used for closed-cycle operation. Modification of the particles to use ZrC coatings would be necessary for open-cycle operation, but the development of such particles is being considered for commercial purposes.

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Figure IV-28. Fixed Bed Reactor (FBR)

One of the main developments needed for the FBR appears to be a high-temperature (inner) frit. Refractory metal frits appear satisfactory for temperatures up to 1500 K (2240°F). Refractory ceramic frits appear to be practical for much higher temperatures but must be tested in representative reactor configurations.

(8) Rotating Bed Reactor (RBR)

For very high-power outputs and/or very high coolant outlet temperatures, the FBR would be superseded by a second reactor concept, the rotating bed reactor (RBR),^{IV-10} as illustrated in Figure IV-29. In the RBR, the annular fuel bed is held inside a rotating porous frit. Coolant passes radially through the frit at low temperature, then through the particle fuel bed, and finally exits at high temperature from a nozzle at one end of the RBR internal cavity.

The RBR would operate only in the open-cycle mode with H₂ coolant. The fuel particle bed can operate in either of three states — fully settled, partly fluidized, and fully fluidized. At a given coolant flow rate, the fuel bed state is determined by frit rotation speed. Higher speeds are required for settled-bed operation, lower speeds for fluidized-bed operation.

Output powers up to 5000 MWt can be generated with relatively small RBRs, (e.g., 1 m long by 1 m (3.3 ft) in diameter). RBR coolant outlet temperatures up to 3000 K (4940°F), the limit of materials capability, can be achieved.

The RBR would be used for direct thrust applications, where high-power and high-specific impulses ($I_{sp} \approx 1000$ s) are required and for generation of very large electric power outputs using magneto hydrodynamic (MHD) or high-temperature turbines.

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Figure IV-29. Rotating Fluidized Bed Rocket Engine

(9) 710 Reactor

The reactor design work performed under the 710 program^{IV-22} was divided into four time periods which differed significantly as to program objectives. From May 1962 to October 1963, work was performed on a reactor that was to be tested at high exit-gas temperatures of 2200 K (3500°F), both in a closed loop with neon as the coolant and in an open loop with hydrogen as the coolant. From October 1963 to July 1965, work was conducted on an experimental reactor that was to be tested for 100 h in a closed loop with neon as the coolant and reactor exit-gas temperatures of 2200 K (3500°F). From July 1965 to October 1966, work was oriented toward developing a reactor for a 200-kW, Brayton-cycle space power system with operating time requirements of 10,000 h and an exit-gas temperature of approximately 1480 K (2200°F). From October 1966 to October 1967, the 710 program was reduced to a fuel technology development program, and studies were performed on the potential of the 710 fuel system for Brayton-cycle, Rankine-cycle, and thermoelectric space power systems.

Three critical experiments were performed to determine the neutronic performance of the 710 reactor. No reactors were built or tested.

In concept, the 710 reactor is a fast-spectrum, refractory metal reactor in which a hexagonal cermet fuel element that has multiple tubular flow channels is used. This concept evolved with the objective of defining a high-performance reactor subsystem that could be used as a heat source for a wide range of mobile power generation and propulsion systems utilizing nonoxidizing working fluids. The motivation behind the development of a multipurpose concept was to minimize the overall reactor development costs for applications that require basically similar high-performance nuclear subsystems. It was recognized from the beginning that in some cases a reactor concept selected

specifically for an application might have better performance potential than a multipurpose concept, but it was believed that the performance margin could be made small enough that the additional development costs would override any difference in performance.

The requirements for a high-performance reactor subsystem were defined as high working fluid temperature, high core power density, and long operating life, commensurate with the detailed requirements of a specific application. Multiple restart and part-load operating capability were also defined as prime characteristics to be achieved by the reactor subsystem to obtain maximum operating flexibility.

The required working fluid, temperature, reactor power density, and life vary widely for different applications. For space power reactors using inert gases or liquid metals, the working fluid temperature may be limited to approximately 1480 K (2200°F). The operating life objective for these subsystems is 1 to 5 years, which limits allowable power densities to 80 to 500 W/cm³ (2 to 14 MW/ft³) of fuel. Marine or aircraft nuclear propulsion applications have intermediate life objectives, 1000 to 2000 h, and intermediate power densities, 1000 to 3000 W/cm³ (28 to 85 MW/ft³) of fuel. For a nuclear rocket in which hydrogen is the working fluid, the appropriate range of gas temperatures is 2500 to 3000 K (4000 to 5000°F), the power densities are in the order of 17,000 to 33,000 W/cm³ (48 to 93 MW/ft³) of fuel, but the operating life would be relatively short (1 to 10 h).

The combination of three basic ideas was visualized as yielding a concept that could meet the above requirements: a fast-spectrum reactor, a cermet fuel with a continuous refractory metal structure, and a tubular fuel element geometry. A fast-spectrum reactor concept was adopted in which the core is comprised only of fuel and structure to minimize the reactor size. Because of

the relatively low neutron absorption probability at high neutron energies, this type of reactor permits using refractory metals that have high strength at high temperature and adequate ductility for cyclic life requirements. A tubular concept based on a cermet fuel was selected for the fuel element (Figure IV-30). The cermet fuel was visualized as a continuous refractory metal structure surrounding the individual fuel particles that would provide at least partial fuel and fission product retention, high strength, and good thermal conductivity to the fuel body. To assure positive fission product and fuel retention for the overall element, the cermet body is completely enclosed with a cladding that is bonded to it. A tubular fuel element concept was considered advantageous since tubes provide a flow channel independent of other structural elements, resist distortion, and can be fabricated to close dimensional tolerances.

For a multipurpose reactor that can be used for applications requiring widely different operating conditions and different working fluids, the 710 tubular cermet fuel element offers advantages from the standpoint of structural strength, low dimensional tolerance effects, and dimensional stability.

The use of tubular cermet fuel elements results in a straightforward reactor concept as shown by a simplified model in Figures IV-31 and IV-32. The fuel elements are mechanically attached to a metallic front tubesheet using a single-point support. The tubesheet, in turn, is attached to the pressure vessel.

In designing such a reactor for high-performance operation, temperature flattening of the core is necessary to keep the coolant fluid temperature as close as possible to the maximum temperature capabilities of the fuel element materials. A typical procedure to accomplish this is to vary the fuel loading of the elements radially across the core. This procedure necessitates an

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Figure IV-30. 710 Reactor Fuel Element Concept

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Figure IV-31. Longitudinal View of the 710 Reactor Assembly

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Figure IV-32. Cross Section of 710 Reactor Assembly

increase in core size. With the tubular matrix element, temperature flattening can also be accomplished by varying the hydraulic diameter from fuel element to fuel element and, if necessary, within a given fuel element. This procedure requires less of a change in reactor size than does the fuel loading variation procedure.

Since the 710 reactor is a fast-spectrum concept, the metallic front tubesheet also serves as a reasonably efficient front reflector. In applications that require a rear reflector, a thicker header plate can be used at the back end of each fuel element.

In the configuration shown, a beryllium radial reflector is used. This material yields the highest reactivity worth per unit weight for this type of reactor. It also has good thermal and structural properties. Metallic, non-moderating reflectors could also be used for applications in which they would yield an overall design advantage.

The reflector is made up of cylindrical sections. Boron strips are attached to these cylinders, and reactivity control is achieved by cylinder rotation. The design is flexible, however, in regard to the placement of the reflector, the type of reactivity control, and the coolant circuit for the reactor components. The reflector can be located within or outside the pressure shell. Reactivity control can be achieved by the above-mentioned method, by sliding the reflector longitudinally, or by displacing it radially. Depending on reactor component cooling requirements, the coolant can pass through the reflector before entering the reactor, can be used to cool the pressure shell and reflector, or can be passed directly into the reactor. These options provide flexibility in tailoring the reactor design to a particular application. Key physical characteristics of a 220-MWt, hydrogen-cooled concept are given in Table IV-12.

TABLE IV-12
710 REACTOR CHARACTERISTICS AND DIMENSIONS

<u>Reactor Data</u>	
Power (MWt)	220
Diameter (equivalent) [cm (in.)]	31.1 (12.25)
Active length [cm (in.)]	30.5 (12.0)
Free flow area [cm ² (in. ²)]	94.3 (14.62)
Radial reflector material	Beryllium oxide
Radial reflector thickness [cm (in.)]	12.7 (5.0)
Core and pressure shell material	Hastelloy X
Thermal radiation shield material	Tungsten
Front reflector material	Beryllium oxide
Front reflector thickness [cm (in.)]	1.2 (4.0)
Core forward structural material	45 Mo - 55 Re (wt. %)
Number of rotating reflector control drums	9
Control absorber	B ₄ ¹⁰ C
Control segment arc (deg)	90
Control segment thickness [cm (in.)]	1.0 (0.394)
<u>Fuel Element Data</u>	
Shape	Hexagonal
Fuel material (vol %)	60 UO ₂ - 40 W
Structural material	Tantalum
Number	85
Distance across flats [cm (in.)]	3.14 (1.235)
Center-to-center distance [cm (in.)]	3.21 (1.265)
Container wall thickness [mm (in.)]	0.25 (0.010)
Aft header plate thickness [mm (in.)]	0.38 (0.015)
Number of coolant tubes	169
Tube hydraulic diameter [mm (in.)]	0.91 (0.036)
Tube wall thickness [mm (in.)]	0.15 (0.006)

(10) Aircraft Nuclear Propulsion (ANP) Reactors

(a) HTRE-1

The first reactor tested during the Aircraft Nuclear Propulsion (ANP) program^{IV-7} was designated the Heat Transfer Reactor Experiment-1 (HTRE-1). The reactor concept is illustrated in Figure IV-33.

The reactor consisted of a cylindrical aluminum water vessel penetrated by 37 tubular air passages in a hexagonal pattern 76.2 cm (30 in.) across the flats. Each passage contained a concentric ring fuel cartridge made of an 80 Ni - 20 Cr alloy impregnated with UO_2 . The active length of each cartridge was 73.7 cm (29 in.). The air passage tubes were lined on the inner surface with a thin sleeve of stainless-steel-jacketed insulation to reduce the direct transmission of heat into the water moderator. The control rod guide tubes also served as inlet tubes for the moderator water, which filled the entire reactor vessel except for the air passages and cooled the beryllium reflector and aluminum structure. The water pressure was only that required for pumping, and was maintained at a temperature of 344 K (160°F) by circulation to an external radiator; the fuel elements operated at a temperature of approximately 1200 K (1700°F), heating the air to about 1000 K (1350°F).

Each fuel element within the reactor generated the same power. This was accomplished by varying the tube spacing, with the maximum spacing occurring near the outside of the reactor, where the power would normally be low. Thus, more moderator was associated with each tube, and the thermal flux from tube to tube was equalized. The beryllium reflector also helped to maintain a sufficiently high flux in the outer tubes. The fine radial power distribution was flattened within a fuel cartridge by radial variation of the fuel loading from ring to ring.

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Figure IV-33. HTRE-1 Reactor

The reactor vessel was attached to the top shield plug and both were inserted as an integral assembly into the cavity in the facility shield. The control rod actuators were mounted on the top plate of the shield plug, as were the nuclear sensor supports, the neutron source actuators, the water inlet and outlet pipes, and the instrumentation leads for the reactor assembly.

HTRE-1 was operated above 200 kW for 486 h and at full power (20 MW) for 151 h. The first full power test took place in January 1956.

(b) HTRE-2

The need to test fuels and materials with increased performance and endurance capabilities over those demonstrated in HTRE-1 and the limited availability of in-pile test facilities led to the decision to modify the HTRE-1 reactor to accommodate large test specimens of more advanced reactors with which the capabilities and interactions of moderator, fuel elements, and structural materials could be evaluated. This modified reactor, designated HTRE-2, was used to test a variety of metallic and ceramic reactor components as well as to perform other special purpose tests.

The modification of the HTRE-1 reactor was started in early 1956. Modification was completed and the first specimen brought to test in July 1957. HTRE-2 was used to test further improvements or alternatives in reactor materials and components until the ANP program was terminated in 1961.

The HTRE-2 "parent core" was similar to the HTRE-1 core, except that the central air tubes had been removed and replaced by a hexagonal void 28 cm (11 in.) across the flats (see Figure IV-34). A corresponding opening was made in the top shield plug so that sections of advanced reactors could be inserted into the HTRE-2 parent core without requiring the removal of the core from the shield. The inserts were suspended from a small-diameter shield plug that filled the opening in the main shield plug. No special cooling air circuit was provided for the insert. The air was drawn from the common plenum chamber above the reactor.

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Figure IV-34. Artist's Conception of HTRE-2 Parent Reactor, Shield Plug, and Test Insert

Since it was expected that some of the inserts would contribute less to reactivity than the seven fuel cartridges that had been removed, an additional 10 cm (4 in.) of beryllium side reflector was added to the parent core to maintain an acceptable reactivity balance.

(c) HTRE-3

Simultaneous with the initiation of the HTRE-2 program, designs were started for a full-scale reactor test, designated Heat Transfer Reactor Experiment No. 3 (HTRE-3). Although basically similar to HTRE-1 and HTRE-2, HTRE-3 was dissimilar in three basic ways: (1) the reactor was mounted horizontally and was equipped with a flight-type shield; (2) a high-temperature moderator was used; and (3) the power plant was designed for simultaneous operation of two turbojet engines from a single heat source.

Hydrided zirconium had been developed to a sufficiently advanced state to be used to evaluate the aerothermal and mechanical design technology in a full-scale, solid-moderated reactor test. Thus, hydrided zirconium for the moderator and nickel-chromium for the fuel elements were the major materials selected for the HTRE-3 reactor. Europium oxide was used as the control rod poison primarily because of its high-temperature compatibility with the containment materials.

The HTRE-3 reactor core assembly is shown in Figure IV-35. The active core, 76 cm (30 in.) long and 130 cm (51 in.) in diameter, contained 150 cells inside a 7.6-cm- (3-in.-) thick beryllium reflector. Each cell consisted of a fuel cartridge inside a hydrided zirconium moderator element; the moderators were hexagonal on the outside and circular on the inside. All the reactor components were cooled by primary air. Each fuel cartridge had 19 stages made up of 12 concentric metallic rings. The UO_2 fuel, dispersed in a matrix of 80 Ni-20 Cr, was clad with 80 Ni-20 Cr stabilized with niobium.

Figure IV-35. HTRE-3 Core Assembly
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Power flattening was achieved by (1) varying the hydrogen content of the moderator for gross radial control, (2) shimming the fuel elements with boron steel for circumferential power control, (3) extending the moderator beyond the active core for longitudinal power control, and (4) varying fuel loading in the individual fuel rings for fine radial power control.

The moderator tubes were cooled to approximately 920 K (1200°F) by routing air through longitudinal slots in the inside surface. The fuel cartridge and moderator were separated by an insulation liner. Both the moderator tubes and the fuel cartridges were attached to the front tubesheet by disconnects and freely supported by the rear tubesheet to allow for thermal expansion.

The reactor control rods, located at the junction of three moderator cells, included 30 shim rods, 3 dynamic rods for power changes, and 15 safety rods normally out of the core except for shutdown.

The reflector was made of hexagonal beryllium sectors provided with longitudinal holes for cooling.

HTRE-3 was operated for 126 h at full power (30 MW) during the period from 1958 to 1960. In 1960, the reactor was inspected and found to be in excellent condition. An additional 21 h of testing was done to investigate full nuclear startup capability.

(11) Metallic Annular Rocket Reactor System (MARRS)

The Metallic Annular Rocket Reactor System (MARRS) study^{IV-23} was performed at Atomics International to investigate an alternative design for nuclear rockets.

In this design, the active core of the MARRS reactor reference design is annular in shape, having an OD of 61 cm (24 in.), an ID of 20 cm (8 in.), and a length of 81 cm (32 in.) (see Figure IV-36). The core consists solely of fueled tungsten-clad wire 0.9 mm (35 mils) in diameter and unfueled tungsten spacer wires averaging 0.34 mm (13.5 mils) in diameter.

The fueled wire itself consists of an inner fueled region and outer cladding. The fueled region is composed of 70 vol % UN and 30 vol % W. The cladding is made of tungsten alloyed with 3 to 5% rhenium. All of the fueled wires are 0.9 mm (35 mils) in OD, but the clad thickness ranges from 0.08 mm (3 mils) at the inner core radius to 0.18 mm (7 mils) at the side of the core. The diameter of the fueled region varies, then, from 0.74 mm (29 mils) down to 0.53 mm (21 mils).

The wire is manufactured by vapor depositing W on UN particles, placing this fuel in a braided W wire tube, then swaging and drawing the tube to the 0.9-mm- (35-mil-) OD size. There are 360 km (223 miles) of fueled wire in the core, weighing 1525 kg (3354 lb).

The core is constructed by first winding a layer of fueled wire on a mandrel, like thread on a spool, with the wires spaced 0.7 mm (27 mils) apart. Then spacer wires are placed across the fueled wire in the axial direction and located 12.7 mm (0.5 in.) apart. A second layer of fueled wire is then wound on the spacer wires, generally being continuous with the first fueled layer. A second set of spacer wires is then placed in position, another layer of fueled wire is added, and so on until a ring-shaped subdivision of the core is completed.

The geometry of such a fuel matrix has a fueled wire solidity of 41.5%, a spacer wire solidity of 1%, and a porosity for the hydrogen coolant of 57.5%. The heat transfer area provided by the fueled wires is $1870 \text{ m}^2/\text{m}^3$ ($570 \text{ ft}^2/\text{ft}^3$). Virtually all of the solid portion of the core is oriented

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Figure IV-36a. MARRS Nuclear Rocket Reactor — End View

9467-37

Figure IV-36b. MARRS Nuclear Rocket Reactor — Side View

in the circumferential direction so that the radially outward gas loads are taken in pure tension. This is analogous to the construction of some gun barrels and rocket cases.

The core is built up of 57 ring-shaped elements that are 10 cm (4 in.) in the axial direction and 2.5 cm (1 in.) in radial thickness. However, the innermost layer is made continuous along the 81 cm (32 in.) length to provide greater core strength.

No reactor of this type was ever built or tested.

(12) AVR

The AVR^{IV-24} is a 46-MWt ground-based demonstration reactor in West Germany; its purpose is to demonstrate the feasibility of an HTGR with spherical fuel elements and high operating temperatures. It is discussed here because derivatives of this reactor could have potential use in space reactor electric systems.

The primary system of the AVR is shown in Figure IV-37, and technical information is summarized in Table IV-13. A key feature of the AVR is its on-line refueling. The core is fueled with ~100,000 graphite spheres containing coated fuel particles. The spheres are 6 cm (2.4 in.) in diameter. During operation, the spheres are continuously withdrawn from the bottom of the reactor core, and other spheres are added at the top of the core so that the spheres circulate continuously. The burnup of the discharge pebbles is measured. Based on the burnup of each pebble, it is decided whether to discharge it or to return it to the top of the bed in one of five radial sectors. The spent elements removed from the cycle during this process are replaced by new pebbles. Helium flows in the upward direction through the pebble bed and then across the steam generator tubes so as to produce steam.

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Figure IV-37. AVR Reactor

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TABLE IV-13
AVR TECHNICAL DATA

Thermal power rating (MWt)	46.0
Electrical power, gross (MWe)	15.0
Electrical power, net (MWe)	13.0
Net efficiency (%)	28
Core power density [MW/m^3 (kW/ft^3)]	2.5 (71)
Core inlet temperature [K ($^{\circ}\text{F}$)]	548 (526)
Core outlet temperature [K ($^{\circ}\text{F}$)]	1223 (1741)
Maximum fuel temperature [K ($^{\circ}\text{F}$)]	1423 (2101)
Maximum fuel element surface temperature [K ($^{\circ}\text{F}$)]	1363 (1993)
Maximum hot gas temperature [K ($^{\circ}\text{F}$)]	1313 (1903)
Primary system pressure [MPa (psi)]	1.08 (157)
Core diameter [m (ft)]	3.0 (9.8)
Core height [m (ft)]	2.5 (8.2)
Reactor vessel type	Steel
Core helium flow direction	Up
Fuel type	HEU/Th
Heavy metal/fuel element (g)	
U^{235}	1
Th	5
Total	6
Discharge burnup ($\text{MW}\cdot\text{d}/\text{T}$)	137,000

So far, three types of fuel elements have been used (Figure IV-38): a first charge element of U.S. origin, the so-called "wallpaper-type" element, and an isostatically pressed element. In the case of the first charge element, pyrocarbon-coated fuel particles are mixed with graphite powder and inserted into a hollow graphite sphere. In the wallpaper-type element, the fuel particles are arranged on the inner surface of the hollow sphere. The interior is filled with a fuel-free graphite matrix. In the case of the pressed element, the fuel particles are pressed with graphite and binder to form a matrix and are surrounded by an additional fuel-free graphite zone of the same basic material.

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Figure IV-38. Fuel Types Employed in AVR

The AVR continues to be used as a test bed for HTGR fuel. In all, 11 different types of fuel elements have been used that differ in enrichment, U-Th ratio, chemical form of the particle, particle size, and coating. The maximum fuel burnup achieved has been 184,000 MW·d/T.

Full-power generation can be achieved with a helium core outlet temperature of 1040 K (1420°F), and in its early years of operation, the core typically operated with an outlet temperature of 1120 K (1560°F). Since 1974, the core outlet temperature has been increased to 1220 K (1740°F) to demonstrate acceptable long-term fuel performance at the elevated temperatures.

(13) ML-1

A nitrogen-cooled reactor designated the ML-1 was developed and tested by Aerojet-General Corporation during 1959 to 1965.^{IV-27} The reactor was designed as a mobile, trailer-mounted, power source for Army field use. It was required to deliver 3 MWt to the coolant at 920 K (1200°F) outlet temperature. The initial core design life was 3000 h, but this was subsequently increased to 10,000 h based on in-pile tests. This reactor is discussed because derivatives of it could have potential use in space reactor electric systems.

The ML-1 used a 19-pin fuel element as shown in Figure IV-39. The fuel pins had Hastelloy X cladding and incorporated $\text{UO}_2\text{-BeO}$ ceramic fuel. Fuel pin cladding material evaluations performed early in the design and development stage served as the basis for selecting Hastelloy X as the cladding material. The selection of $\text{UO}_2\text{-BeO}$ was based on a series of capsule experiments performed to evaluate fuel stability, radiation effects, and fuel/cladding interactions. The 19 pins were supported by an upper spider and were free to expand thermally into but were guided radially by a lower spider. The bundle of 19 pins was surrounded by an inner and an outer liner; between these liners the thermal insulation and the burnable neutron poison were placed. The liner assembly served to insulate the reactor pressure tubes from the hot coolant gas and acted as an axial support member for the fuel element assembly.

The reactor core for the ML-1 was water-moderated and consisted of 61 fuel elements contained in a calandria-type pressure vessel. The reactor was assembled on a skid as shown in Figure IV-40. The calandria consisted of 61 type 321 stainless steel pressure tubes rolled and seal-welded into type 304 stainless steel tubesheets. The fuel elements were supported in the pressure tubes on machined seats in the upper tubesheet. The direction of gas flow was downward. The lower vessel cap was permanently welded to the vessel, and the

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Figure IV-39. ML-1-1 Fuel Element

Figure IV-40. ML-1 Reactor Skid

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top cap was removable for refueling. Ducting was provided in both caps to route the cooling gas into and out of the reactor.

The reactor control rods were semaphore-type with neutron absorbing blades that were inserted in the water-filled spaces between rows of calandria tubes. The neutron absorber was a silver-cadmium-indium alloy compounded specially to be effective in the ML-1 undermoderated neutron spectrum. The blades were actuated by a drive system housed in a dry cavity at the outer surface of the shield tank and connected to the blades by a drive shaft that penetrated the shield solution area. Six control rods were provided with five designated as shim-safety rods and one designed as a regulating rod.

The prototype ML-1 reactor skid and control cab were delivered to the National Reactor Testing Station (NRTS) for testing in the spring of 1961. Problems with the development of compact, high-performance heat exchangers and with the development of the gas turbine-compressor (t-c) unit delayed the completion of the PCS until mid-1962. Testing of the prototype proceeded intermittently from September 1962 until August 1965. The work was interrupted by minor (but difficult to repair) failures in the reactor, the requirement to design and install additional instrumentation and support systems (required to assure the safe operation of the developmental plant or to provide detailed data for the evaluation of the performance of plant components), and by sub-standard performance of the t-c unit (since no suitable spare unit existed, testing was interrupted for each t-c unit modification).

In October 1965, the decision was made by the USAEC to terminate the Army Gas-Cooled Reactor Systems Program. At the time of termination, the ML-1-I core loading had operated for approximately 2500 equivalent full-power hours. There were no indications of problems with the fuel. The results of the in-pile irradiations of a prototype ML-1 type test element indicated that the element was fully capable of operation to the 10,000-h goal.

c. Heat Pipe-Cooled Reactors

(1) SPR-4 and SPR-5

One of the earliest conceptual designs of a heat-pipe-cooled space reactor was developed at LLNL and was designated the SPR-4.^{IV-25} A schematic of the concept is shown in Figure IV-41.

The SPR-4 concept is for an integral reactor-boiler designed to operate at 2 MWt at 1500 K (2240°F). The concept uses UN fuel and W-25 Re as the structural material. The active core consists of 378 hexagonal-shaped fuel elements, 1.5 cm (0.6 in.) across the flats and each having a 0.9-cm- (0.36-in.-) diameter central hole in which a sodium heat pipe is placed. A unique feature of SPR-4 is that a potassium boiler region is located at each end of the core so the heat pipes are transferring heat toward both ends. The core bundle diameter is 34 cm (13 in.), and the active length is 42 cm (16.5 in.). A 6-cm- (2.4-in.-) diameter duct is located in the center of the core for potassium vapor transport.

Reactivity control is provided by dual control systems, each of which can provide the necessary control for the 20,000-h design life. One system employs movement of the molybdenum side reflector and the other uses twelve Li^6 filled control tubes in the core surrounding the potassium duct.

The SPR-5 was another heat-pipe-cooled reactor concept devised at LLNL. It is similar to the SPR-4 but relies on an out-of-core thermionic power conversion system rather than a potassium rankine cycle. It uses the same hexagonal fuel elements as SPR-4 with a heat pipe in the center.

Figure IV-41. LLNL SPR-4 2-MWt Uranium Nitride Fueled Integrated Reactor/Boiler

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Neither the SPR-4 nor the SPR-5 advanced much beyond the study phase although the SPR-5 may be considered a forerunner of the LANL SABRE design and the Rasor minipipe design.

(2) Space Arena Baseline Reactor (SABRE)

(a) General

The SABRE (formerly SPAR and SP-100) reactor incorporates heat pipes to transport the heat out of the core to the thermoelectric subsystem.^{IV-2} A fast-spectrum, highly enriched UO_2 fueled reactor was selected to meet the requirements of a long-life, compact core design. Using UO_2 as the fuel permits operation at high temperatures and keeps material interactions at an acceptable level. Heat pipes provide many heat removal paths without using pumps or compressors to transport heat from the reactor.

The reactor has a core region composed of fuel modules in which the fuel is arranged in layers between circumferential fins, which are attached to the heat pipes. The fins enhance heat transfer from the UO_2 to the heat pipes and reduce the temperatures in the UO_2 . These fuel modules are in rows around the central safety plug and comprise the core. Surrounding the core is a container that provides support to the fuel modules but is not a pressure vessel. The container also provides a noncompressive support for the multi-foil insulation. Multiple reflective insulation layers reduce the core heat loss to the reflector to an acceptable level. The reflector surrounds the core and reflects neutrons back into the fueled region. Located within the reflector are drums that are rotated by electromechanical actuators. On part of each drum is a neutron-absorption material, and the positions of this material are used to establish the reactor power.

(b) Reactor

Figure IV-42 is a cutaway view of the reactor. The reactor core is made up of heat pipe/fuel modules. The parts of the heat pipes within the core have circumferential fins made of the same metal as the heat pipes, a low rhenium content alloy of molybdenum. Fuel tiles are sandwiched between the fins. There is a central hole in the core to provide space for a boron carbide (B_4C) plug that, if the reactor were immersed in water with all the heat pipes broken, might maintain the core subcritical. This central plug is a safety feature used only during launch and is removed in orbit. Figure IV-43 shows the individual fuel modules. The fins are an integral part of the heat pipe; the fuel is fabricated as tiles and placed between the fins.

The fuel is fully enriched UO_2 . Distinct axial and radial power profiles exist in the core. Instead of varying the isotopic content of the fuel, the amount of fuel in each row of modules is varied so that an equal quantity of heat is generated per module. This means that each heat pipe transports approximately the same amount of power. Axial temperature flattening is naturally achieved by the heat pipe itself.

UO_2 fuel swelling is a function of both temperature and time. Preliminary data on fuel swelling in this fuel configuration are given in Figure IV-44. The design provides some void volume to accommodate fuel swelling. The stresses generated by fuel swelling must also be accommodated. This is taken care of in the axial direction by a fuel tile being periodically omitted to prevent an accumulation of axial stresses and core distortion. In the radial direction, void space is provided for the fuel to swell into.

A large void is shown in the center of the core. A B_4C plug will be inserted before the nuclear subsystem is installed in the shuttle and will remain there during launch and climb to orbit. This plug is designed to keep the reactor subcritical if the intact reactor were immersed in water with all heat pipes broken. Once in orbit, the B_4C plug will be removed.

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Figure IV-42. SABRE Heat Pipe Space Reactor

9467-42

Figure IV-43. SABRE Heat Pipe Fuel Module

9467-43

Figure IV-44. Preliminary Unconstrained UO_2 Fuel Swelling Data for SABRE

The reflector is a neutronic element to reduce neutron losses from the core. The side and section of the reflector not penetrated by heat pipes is made of beryllium. The reflector section that is penetrated by the heat pipes is made of beryllium oxide (BeO) and operates at a temperature of about 800 K (980°F). It is cooled by radiation to space from its outer surface, and multifoil thermal insulation is wrapped around the core to minimize heat leakage from the core to the reflector. Of most concern is swelling and loss of thermal conductance caused by neutron irradiation. The primary cause of the swelling in beryllium is the formation of helium by $(n,2n)$ and (n,α) reactions. At temperatures above 375 K (215°F), helium gas accumulates into bubbles, and swelling proceeds as the bubbles grow.

Within the reflector are 12 control drums. Each drum has an arc segment of neutron-absorbing material of B_4C on one side. Reactivity is increased or decreased by rotating the neutron-absorbing material away from or toward the core, respectively. Each drum will be rotated with a stepping-motor actuator, which will be located on the side of the radiation shield away from the core for protection from high nuclear and thermal radiation. The reactor has built-in (~5%) excess reactivity to account for control margin, fuel burnup, temperature coefficient, and uncertainties. The B_4C is enriched to 90% ^{10}B to provide a reactivity worth of 12.5%. It will increase in volume after 7 years by about 2%.

The reflector, which is designed to disassemble on atmospheric reentry to meet safety requirements, consists of segments held together by molybdenum retainer bands (Figure IV-45). These bands are designed to provide mechanical integrity to the reactor for launch and during operations and yet rapidly oxidize on atmosphere reentry.

Table IV-14 lists the characteristics of the nuclear reactor.

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Figure IV-45. SABRE Reflector Segmentation

TABLE IV-14
SABRE NUCLEAR REACTOR CHARACTERISTICS

Thermal power	1.47 MW
Heat pipe evaporator vapor temperature [K (°F)]	1500 (2240)
Maximum UO ₂ temperature [K (°F)]	1730 (2654)
BOL reactivity (K_{eff})	1.05
Fission during 7 years	8×10^{20} fis/cm ³
UO ₂ swelling after 7 years	TBD
Burnup	3.6%
Redeposition of Mo and UO ₂ on heat pipes [mm (in.)]	0.5 (0.02)
Core diameter [cm (in.)]	33.1 (13)
Core height [cm (in.)]	33.1 (13)
Be reflector average thickness [cm (in.)]	9 (3.5)
BeO reflector thickness [cm (in.)]	10 (4)
Reactor diameter [cm (in.)]	54.2 (21.3)
Reactor height [cm (in.)]	55.0 (21.7)
B ₄ C plug mass [kg (lb)]	3.1 (6.8)
Component mass [kg (lb)]	
Fuel and fin	185 (407)
Heat pipe mass to reactor exit	20 (44)
Inner and outer cans, multifoil	9 (20)
Fuel module support plates	5 (11)
Reflector	150 (330)
Actuator	42 (92)
Structure	29 (64)
Reactor total mass (excludes B ₄ C plug) [kg (lb)]	440 (968)

(c) Reactor Control

The current concept is to control the reactor by adjusting 12 rotating drums surrounding the core. The control drums are operated under closed-loop control by functionally redundant reactor controllers. Safety criteria dictate that, to prevent overheating of high-temperature elements, the reactor controller provide a power cutback capability. In addition, a shutdown control function is provided to scram the control drums should an emergency shutdown situation be detected.

For startup, temperature will be controlled based on the temperature of the core heat pipes. To avoid excessive power and temperature overshoots due to long-time constants in transfer functions, the current design startup takes several hours.

(3) Minipipe

Rasor Associates has studied an out-of-core thermionic design which they call the "Radiation Coupled Mini-Pipe" design.^{IV-26} In the minipipe design shown in Figure IV-46, a large number (500 to 5000) of small (approximately 10-mm- (0.4-in.-) diameter) heat pipes are used to remove heat from the reactor. These pipes extend out of the reactor core only 1 to 2 m (3 to 6 ft). These small heat pipes are then grouped around thermionic energy converters as shown in Figure IV-46. The high-temperature minipipes radiate to the hot outside electrodes of the energy converters. The cold inside electrodes are connected through a sheath insulator to heat rejection heat pipes that are coupled to the waste heat radiator.

The minipipe design has the advantage of mechanically decoupling the energy converters from the reactor, potentially eliminating high-temperature electrical insulators, and providing more uniform and redundant cooling in the

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Figure IV-46. Razor Radiation Coupled Minipipe Design

core. These features facilitate higher power operation. The design's primary disadvantages are the requirement that the energy converters must be located between the reactor and radiation shield. As a result, the shield must be larger, and in some instances this might result in a more massive system.

A reactor core using the minipipe design would consist of hexagonal fuel elements, with each fuel element containing 6-12 small heat pipes. No electrical insulator at the high temperatures is required since heat transfer occurs by radiation. This also simplifies the structural requirements for accommodating possible mismatches in thermal expansion.

The reactor core used in the minipipe design study is based on the SABRE concept described above.

d. Plasma Reactors

(1) Light Bulb

Reactor concepts that use uranium in a gaseous state (UF_6) as the fuel have been studied extensively. In these concepts, a transparent wall separates the fuel and propellant (working fluid) regions. Energy is transferred via thermal radiation, and hence the concept has been called the "light bulb" reactor. One such concept is illustrated in Figure IV-47, and Table IV-15 lists some of its characteristics. Seven cavities are used to increase the total radiating area of the fuel relative to that of a single cavity and to reduce critical mass.

Inside the cooled transparent tubes of fused silica, a neon vortex stabilizes the uranium plasma and keeps it from the walls. The neon is injected

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Figure IV-47. "Light Bulb" Reactor Rocket Engine Concept

TABLE IV-15
CHARACTERISTICS OF LIGHT-BULB REACTOR

Power (MWt)	4,600
Number of cavities	7
Total cavity volume [m^3 (ft^3)]	4.81 (170)
Volume per cavity [m^3 (ft^3)]	0.69 (24)
Cavity length [m (ft)]	1.83 (6)
Cavity diameter [m (ft)]	0.68 (2.23)
Vortex volume per cavity [m^3 (ft^3)]	0.34 (12)
Vortex (transparent tube) diameter [m (ft)]	0.49 (1.6)
Fuel region diameter [m (ft)]	0.41 (1.3)
System pressure [MPa (psi)]	50 (7,250)
Fuel material	^{233}U
Total fuel mass in cavities [kg (lb)]	14 (31)
Buffer vortex fluid	Neon
Moderator material	BeO
Effective fuel radiating temperature [K ($^{\circ}\text{F}$)]	8,300 (14,500)
Propellant exit temperature [K ($^{\circ}\text{F}$)]	6,700 (11,600)
Total flow rate of propellant [kg/s (lb/s)]	22.4 (49)
Cavity power density [MWt/ m^3 (MWt/ ft^3)]	960 (27)
Vortex region power density [MWt/ m^3 (MWt/ ft^3)]	1,920 (54)
Fuel region power density [MWt/ m^3 (MWt/ ft^3)]	2,660 (75)

tangentially to produce a vortex; it then flows axially toward the end walls and is removed by a central port in each end wall. The neon carries entrained fuel and fission products, which are removed by condensation and centrifugal separation. Liquid fuel is returned to the cavity, and the neon is also recycled after further cooling.

(2) Plasma

Reactors using a uranium plasma core have been studied for use as rocket engines. The plasma reactor concepts were designed to operate in the open-cycle mode with the hydrogen coolant being exhausted through the nozzle.

One concept of a cavity reactor, shown in Figure IV-48, consists of a porous cavity liner, an inner heavy-water moderator, a high-temperature beryllium reflector, an outer heavy-water reflector, and the pressure vessel. Hydrogen must be supplied at high pressures by a high-speed turbopump driven by hydrogen extracted from a suitable portion of the propellant circuit. In systems designed for high specific impulse, a radiator is required to reject to space neutron and gamma heat that cannot be absorbed by the flow of hydrogen.

4. Reactor Technology Assessment

a. Liquid-Metal-Cooled Thermal Reactors

The only nuclear reactor technology that could currently be considered qualified for space power application is the UZrH reactor subsystem. The SNAP-10A went through all the necessary steps required for space flight qualification and was successfully launched into earth orbit on 3 April 1965. The

(a) Conceptual Sketch

(b) Model for Nuclear Calculations

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Figure IV-48. Plasma Reactor Rocket Engine Concept

reactor had a thermal power capability of 50 kWt at a coolant outlet temperature of 810 K (1000°F). Its lifetime potential was at least 10,000 h, although it only operated 43 days due to a failure in an electrical circuit of the spacecraft.

The SNAP-8 reactors were ground tested at higher powers (1000 kWt) and at higher outlet temperatures of 980 K (1300°F). However, during both the S8ER and S8DR tests, extensive cracking of the fuel rod cladding occurred. Because of the necessity for hydrogen retention by the barrier on the cladding, this cracking was a life-limiting mechanism.

Subsequent UZrH programs were devoted to solving this problem in order to increase the power capability above that achieved by SNAP-10A. The 110-kWt reactor designed for use with a thermoelectric PCS to produce 5 kWe was the most recent attempt to design a higher-powered, longer-lived reactor than SNAP-10A. The 5-kWe reactor design life was 5 yr. Many of the causes for clad cracking in SNAP-8 were recognized, and the 5-kWe reactor design attempted to solve these problems. However, an actual reactor test was never made.

The limiting outlet temperature capability of a UZrH reactor for current technology is felt to be about 920 K (1200°F). Due to the complex phase structure of the fuel and the desire to remain in the delta phase throughout life, it is necessary to restrict the temperatures to this level. It is also necessary to restrict the maximum fuel rod linear power rating to about 6 kW/m (2 kW/ft). For these reasons, the UZrH reactor system has limited potential for far-term, high-power applications.

An assessment of the ZrH-moderated reactor technology is illustrated in Figure IV-49. A UZrH reactor in the 400 to 1000 kWt range with an outlet temperature of 920 K (1200°F) and a 5-yr life could be built and ground tested for 1 yr before the end of FY 1988. An improvement in the hydrogen barrier on the fuel elements would permit increasing the outlet temperature to 1000 K

Figure IV-49. Subsystem Technology Assessment – Liquid-Metal-Cooled Thermal Reactors

(1350°F), power to 1500 kWt, and lifetime to 7 yr. Development of a concept in which the ZrH moderator and fuel (UO_2) are separately cooled by two NaK loops would allow operation of the fuel at higher temperatures while maintaining the ZrH at a much lower temperature where hydrogen loss would be negligible. Separation of the fuel and moderator was common in early reactor concepts, so such a reactor does not present any feasibility questions. Such a reactor could achieve a 1100 K (1500°F) outlet temperature, and with reflector control, a power of 2000 kWt for a 7-yr life would be possible. Power levels higher than this would require development of in-core poison rod control.

The following issues are associated with liquid-metal-cooled thermal reactors (see Figure IV-49):

Technology Step 1

920 K (1200°F)
OUTLET
400-1000 kWt
5-year LIFE

Key Issue - Demonstrate an achievable 5-yr lifetime for a UZrH reactor subsystem operating at up to 1000 kWt with a coolant outlet temperature of 920 K (1200°F).

Resolution - Design and fabricate (elapsed time 3 years) and test for 2 years (2-yr test judged to be an acceptable risk versus 5-yr design lifetime based on extensive data base from SNAP-2, -8, and -10A programs); complete testing by end of FY 1988.

Technology Step 2

IMPROVED
 H_2 BARRIER
1000 K (1340°F)
OUTLET
1500 kWt
7-year LIFE

Key Issue - Develop an improved hydrogen barrier capable of operating at a coolant outlet temperature of 1000 K (approximately 1350°F) with acceptable hydrogen loss over a 7-yr design lifetime.

Resolution - Develop and test improved barriers (elapsed time 3 to 4 years); design and fabricate reactor subsystem (elapsed time 3 years with 1-yr overlap on barrier development activity judged to be acceptable risk); then perform 3-yr reactor subsystem test (3-yr test time versus 5-yr design lifetime judged to be an acceptable risk); complete total program by mid-1992.

Technology Step 3

Key Issue - Develop improved hydrogen barrier (see step 2 above) for ZrH moderator cladding operating temperature in the range of 1000 K (1340°F) and design and prove-out separate fuel element/moderator element cooling systems (operating at different temperatures) using a mixing chamber (subject to thermal fatigue due to mixing of different temperature coolant streams).

Resolution - Develop and test improved barriers (elapsed time 3 to 4 years as in step 2 above); develop and test in out-of-pile loops the mixing chamber (3 to 4 years elapsed time, overlapping barrier development by 2 years). Perform extensive design of two-coolant-loop system (3 to 4 years elapsed time, overlapping mixing chamber test by 1 year); fabricate reactor subsystem (3 years after design completion, no overlap with design effort); perform 4-yr test to verify operation at 2000 kWt for 5-yr lifetime (4-yr test versus 5-yr lifetime judged to be an acceptable risk); complete program by mid-1998.

b. Liquid-Metal-Cooled Fast Reactors

For future high-temperature reactors using liquid metal coolants, a fast reactor offers the greatest potential. The liquid metals that are appropriate for use in compact fast reactors are sodium, potassium, NaK eutectic, and lithium. The boiling point of each dictates its temperature range of applicability. Thermodynamic properties and material compatibility considerations also influence the selection of one liquid metal over the other. Ground-based breeder reactors used NaK and sodium in their early development phase before

finally selecting sodium. As a consequence, much experience has been accumulated in the use of these in a fast reactor. The MPRE considered a boiling potassium fast reactor, but a liquid-potassium-cooled reactor should also be possible. Because of its high-temperature boiling point and good heat transfer properties, lithium was selected early in the SNAP-50 program for use as the primary coolant. The use of lithium requires the use of refractory metals as fuel rod cladding and reactor structure.

To date, no lithium-cooled fast reactors have been built. Most of the effort has been devoted to materials development and design studies. Lithium is one of the most corrosive of the alkali metals, and much testing has been directed to solving the compatibility problems associated with the fuel, cladding, and coolant.

Many fast reactors have been operated using sodium as the coolant. There is less experience with NaK and none to date with potassium. These liquid metals are compatible with stainless steel, superalloys, and refractories.

The materials with which there is the most test experience as fast reactor fuels are UO_2 , UC, and UN. Of these, the experience with UO_2 far exceeds that with the other two. UO_2 has demonstrated high temperature and high burnup capability in a large number of fast reactors. Its low thermal conductivity somewhat restricts its high power potential. Both UC and UN have higher thermal conductivities than UO_2 and also higher densities. This permits more compact reactor designs with these fuels. The experience with UC and refractory alloys has not been good. As a consequence, SNAP-50 and other high-powered compact reactors have selected UN as fuel.

There are still unsolved problems associated with the use of UN because of its potential instability at high temperatures, which could result in free uranium being formed. This free uranium metal would vaporize and attack the cladding. If an overpressure of nitrogen could be maintained over the fuel,

then this problem would be eliminated. Additives to the fuel, such as Y_2O_3 , have also been suggested, but these require further examination.

Refractory metals that have received attention are tungsten, tantalum, molybdenum, or niobium based. The fuel pin irradiation tests that had the most success used Nb-1Zr as cladding for UN fuel. Some success has also been achieved with the tantalum-based alloy T-111.

The technology assessment of liquid-metal-cooled fast reactors is shown in Figure IV-50. A reactor using a stainless-steel-clad UO_2 fuel pin design could probably be built and ground tested by the end of FY 1991. Such a reactor would be based on LMFBR and MPRE technologies, for which irradiation testing has already been performed. A 1500-kWt reactor with a 920 K (1200°F) outlet temperature and 7-yr life would be the first design selection. Improved performance could be attained through the development of a superalloy-clad UO_2 fuel pin. An increase in reactor outlet temperature to 1100 K (1520°F) and power level to 2000 kWt would be possible with this improved fuel pin.

The earliest lithium-cooled reactor would probably employ a refractory-clad UO_2 fuel pin and could be ground tested by FY 1994. This early reactor would operate at an outlet temperature of 1350 K (1970°F) and at a power level of 2000 kWt. Because of the uncertainty associated with this fuel-cladding/coolant combination, the predicted life of the first version is two years. Development of refractory-clad UN fuel pins would permit a more compact reactor because of the improved density and thermal performance of UN over UO_2 . To advance the lithium-cooled fast reactor to even higher performance would require solutions to the UN stability problem and improved refractory alloys. If these developments were successful, then a 3000-kWt reactor with a 3-yr life would be possible with an outlet temperature of 1500 K (2240°F).

Figure IV-50. Subsystem Technology Assessment – Liquid-Metal-Cooled Fast Reactors

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The following technology issues are associated with liquid-metal-cooled fast reactors (see Figure IV-50):

For Na- or K-Cooled Fast Reactors

Technology Step 1

Key Issue - Demonstrate a 7-yr lifetime.

Resolution - Design and fabricate a stainless steel clad, UO₂-fueled reactor subsystem (elapsed time 3 years) and test for 4 years (4-yr test versus 7-yr lifetime judged to be an acceptable risk); complete by end of 1990.

Technology Step 2

Key Issue - Demonstrate superalloy performance at 1000 K (approximately 1350°F) for 7-yr design lifetime.

Resolution - Design, fabricate, and test an irradiation experiment in EBR-II or FFTF for approximately 10,000 h (satisfactory performance for 10,000 h judged acceptable risk to move forward with reactor subsystem design), elapsed time 5 to 6 years; based on satisfactory results, design, fabricate (with 2-yr overlap on irradiation test, judged to be acceptable risk), and test for 3 years the reactor subsystem (3-yr test versus 7-yr design lifetime judged to be an acceptable risk assuming irradiation test results are also acceptable), elapsed time 6 years; complete program by mid-1993.

For Li-Cooled Fast Reactors

Technology Step 1

Key Issue - Demonstrate refractory-clad UO_2 fuel performance at 1350 K (approximately 1970°F).

Resolution - The Mo-Re alloys are known to have good mechanical properties and to be producible, formable, and weldable. The behavior of such refractory alloys under long-term irradiation at elevated temperatures needs to be established. A logical approach would be an early materials coupon irradiation test in EBR-II or FFTF followed by a fuel pin test (or tests) confirming acceptable fuel/clad performance. The coupon test would require approximately 3 years and the fuel element test(s) another 4 years. Design and fabrication of the reactor subsystem test would overlap the fuel element test (judged to be an acceptable risk based on satisfactory results from the coupon test) and would be tested for the full 2-yr lifetime. Program completion by the end of 1993 appears reasonable.

Technology Step 2

Key Issue - Same as step 1 above with addition of demonstrated performance of refractory-clad UN fuel at 1400 K (approximately 2060°F).

Resolution - Same resolution steps as for step 1 above with addition of developing UN fuel production capabilities and verifying in-pile compatibility of UN fuel and refractory cladding. These two additional steps are estimated to require an additional 3 years, even with overlapping of activities. Program complete by end of 1996.

Technology Step 3

Key Issue - Same as for steps 1 and 2 above but with the addition of scoping development irradiation tests of doped UN fuel to verify performance characteristics and development of production capabilities for doped UN fuel.

Resolution - These two additional steps are estimated to add a total of 2 to 3 years beyond step 2 above; program complete by mid-1998.

c. Boiling-Liquid-Metal Fast Reactors

Of all the liquid-metal-cooled reactor concepts reviewed, only the MPRE developed at ORNL considered the use of boiling potassium as the primary coolant in a direct Rankine-cycle power plant. Other concepts considered potassium Rankine cycles for power conversion; but in these, such as SNAP-50, the primary coolant was another liquid metal, and the potassium was vaporized in a boiler exterior to the reactor. The use of a direct cycle was believed by ORNL to increase the reliability of the space power system through the elimination of the components in a separate primary loop.

An actual reactor was not tested on the MPRE program, but in-pile tests of fuel elements and several UO_2 critical experiments were performed. Materials tests that demonstrated the compatibility of the boiling potassium with the stainless steel cladding were also made. The fuel element tests demonstrated successful operation to 14,000 h. The critical experiments provided useful data relative to nuclear performance. The key technical feasibility question of the reactor is the control of free-liquid surfaces under zero-gravity conditions. Other feasibility questions exist in the remainder of the system. These are discussed in Section IV.D.

An assessment of the boiling-liquid-metal-cooled fast reactor technology is limited to potassium coolant and is illustrated in Figure IV-51. The technology developed on the MPRE and LMFBR programs would permit the development of a boiling-potassium-cooled fast reactor in the time period up through FY 1992. The reactor would use stainless-steel-clad UO_2 fuel pins based on LMFBR technology. The first ground test would be run at an outlet temperature of 1110 K (1540°F), like the MPRE, and could be designed for a 5000-kWt power level and 7-yr life. This test would provide solutions to many of the design problems of the boiling-potassium-cooled reactor. Improved cladding and fuel materials would be required to reach the 1400 K (2060°F) temperature level desired for an advanced boiling-potassium reactor. Potential candidates are UN and T-111.

The following technology issues are associated with boiling-liquid-metal-cooled reactors (see Figure IV-51):

Technology Step 1

Key Issue - Attaining a 5000-kWt power level for 3 years with stainless steel cladding operating at 1110 K (approximately 1540°F) in a boiling potassium, zero-g environment.

Resolution - Assume that reactor peripheral control can be used for the 3-yr/5000-kWt operation. It will be necessary to verify the satisfactory performance of stainless steel at >1500°F for 3 years by means of the subsystem test. Design, fabrication, and testing will require approximately 6 years. Another 2 years beyond this will be required to plan, design, fabricate, and test an in-orbit test of boiling fluid control (probably water) using the STS. Program completion by mid-1992.

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Figure IV-51. Subsystem Technology Assessment — Boiling-Liquid-Metal Fast Reactors

Technology Step 2

Key Issue - Demonstrate satisfactory performance of UN fuel and refractory-clad fuel elements. An in-pile irradiation test in EBR-II or FFTF (in-pile irradiation time approximately 3 years) would be the first step (total elapsed time approximately 6 years) followed by design, fabrication, and full 7-yr test of the reactor subsystem; completion by mid-1996 appears reasonable with some overlap between the irradiation test and the reactor subsystem design.

Resolution - The key issues to be resolved are those associated with step 2 for the Li-cooled fast reactor, and the program completion date (end of 1996) is the same. Note: Successful demonstration assumes that use of in-core control for a 5000-kWt power level for a 7-yr design lifetime is satisfactory. This is regarded as a medium- to high-risk judgment.

d. In-Core Thermionic Reactors

The in-core thermionic reactor design developed by GA, which employs the six-cell flashlight TFE, is the most highly developed of all the high-powered, direct-conversion concepts. The TOPAZ reactor, which is in operational use in the USSR space program, uses this concept, although there are differences in the cell geometry. The heat source and the PCS are combined in the compact TFE. There is flexibility in the power capability because any number of TFEs can be combined to obtain the desired power output. There is a practical range of 5 to 1000 kWe because of neutronic considerations. At the termination of the thermionic reactor program in 1973, the 120-kWe reactor had received the most attention.

Much of the effort was devoted to the development of the TFE. The irradiation environment in which it must operate extracts a toll in endurance capability. The swelling of the fuel caused by the accumulation of fission products can cause short circuits to occur in the TFE. The migration of elements at the high temperatures required for TFE performance may also degrade its performance. Radiation damage by fast neutrons to insulating materials may also limit TFE lifetime. The maximum life that had been achieved at project termination was 12,500 h for an in-pile test. No full-scale reactors were ever tested. Further discussion of the technology assessment of in-core thermionics is presented in Section IV.D.4.

e. Hydrogen-Cooled Reactors

Hydrogen-cooled reactors would be selected for propulsion systems or for semicontinuous high powers using an open-cycle turbine. The properties of hydrogen as a coolant and propellant make it ideally suited for this purpose. During the Rover program, the UC-graphite-fueled, hydrogen-cooled reactor for rocket engines was developed to the point that power runs of 1 to 2 h at power levels as high as 4000 MWt and outlet temperatures of 2200 K (3500°F) may be considered state of the art. The development of UC-ZrC fuels could improve the run times to 10 h. Different designs were tested at power levels from 100 to 4000 MWt. For early propulsion applications, designs derived from the SNRE concepts would be the logical choice.

For more compact systems, the metal-fueled fast reactors such as the GE-710 or MARRS would be considered. The latter offers the most potential for higher power levels and outlet temperatures. Neither concept has been tested, and both would require a sizeable development program.

An alternative to the solid core NERVA reactor would be the RBR, which would use an advanced TRISO particulate fuel with a ZrC coating. Considerable size reduction is projected for this concept because of the finely divided fuel, which greatly improves the heat transfer to the coolant. Development

requirements for the RBR would be substantially greater than for the other concepts. The problems associated with maintaining a fluidized bed in a space environment have not been adequately addressed. The behavior of TRISO fuel in high-temperature hydrogen has not been determined; however, ZrC coating shows some promise.

A nuclear rocket reactor such as SNRE could probably be built and tested before the end of FY 1990, as shown in the technology assessment of hydrogen-cooled reactors illustrated in Figure IV-52. Improvements in graphite and particulate fuel coatings such as ZrC would permit higher temperatures up to 2500 K (4040°F) and run times up to 10 h. Development of the RBR would permit even higher temperatures up to 3000 K (4940°F) but run times in space would be limited by the hydrogen inventory to about 2 h. An alternative to RBR development would be compact metallic or cermet fueled reactors such as the MARRS or GE-710 concepts. These offer about the same capability as the RBR with less complexity.

The following technology issues are associated with hydrogen-cooled reactors for propulsion or open-cycle Brayton use (see Figure IV-52):

Technology Step 1

Key Issue - The key issue for this reactor subsystem is similar to that for the UZrH liquid-metal-cooled thermal reactor; reestablish and demonstrate a workable subsystem based on previous work.

Resolution - Design and fabricate (elapsed time 3 years) a reactor subsystem. In parallel, design, construct, license, and check out a test facility. Elapsed time 6 years. Then install and test the reactor subsystem (elapsed time to final test approximately 1 year). Program complete by late 1990.

Figure IV-52. Subsystem Technology Assessment — Hydrogen-Cooled Reactors for Propulsion or Open-Cycle Use

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Technology Step 2

IMPROVED
ZrC COATING
500-4000 MWt
2500 K (4040°F)
OUTLET
10-h LIFE

Key Issue - Demonstrate ability of improved ZrC fuel to withstand 10-h firings.

Resolution - Perform a full-scale demonstration test. Reactor test facilities do not exist for testing fuels under these proposed conditions. It will be necessary to design, construct, and license a special test facility for this test, similar to that for step 1 above (probably at the same geographical location). This, coupled with development of the ZrC improved coating, is judged to result in program completion by the end of 1993.

Technology Step 3

REFRAC-
TORY METAL
CLAD CERAMIC
OR CERMET
FUELED REACTORS
4000 MWt
3000 K (4940°F)
OUTLET
2-h LIFE

RBR
4000 MWt
3000 K (4940°F)
OUTLET
2-h LIFE

Key Issue - Verify fluidized bed behavior in a zero-g environment and absence hydrogen embrittlement of refractory/metal-clad/cermet fuel.

Resolution - Full-scale testing, as in steps 1 and 2, will be required. In addition, behavior of the fluidized bed in a zero-g environment may well require a preliminary in-orbit test. If so, completion of the test program by the turn of the century appears to be a reasonable timeframe.

f. Inert-Gas-Cooled Reactors

In closed-cycle gas-cooled reactor systems, helium or helium-xenon mixtures are usually the selected coolant. Helium has excellent heat transfer properties, is chemically inert, and is commercially available. The addition of xenon improves the performance of the turbomachinery at lower power levels.

Helium-cooled reactors are considered almost exclusively for use with Brayton-cycle PCSs and usually in a direct cycle with the primary reactor coolant also serving as the working fluid. A slight disadvantage of xenon is its induced radioactivity, which causes an increase in primary coolant activity.

The reactor subsystems in this review that were specifically designed to use helium as coolant are the ML-1, GE-710, AVR, and the FBR. However, any of the gas-cooled reactors described in Section IV.A.3 could probably be adapted to use helium as the coolant. These latter reactors would, of course, have to be considered less developed in that case.

Another gas-cooled reactor that could be considered is the ML-1. Although this reactor was developed for a ground mobile application and used nitrogen as a coolant, the technology is applicable to space applications. It could also be readily adaptable to helium/xenon as a coolant. This reactor had an outlet temperature of 922 K (1200°F) and operated for an equivalent full-power time of 2500 h.

The GE-710 reactor uses tungsten-clad UO_2 and would require much more materials testing and design development.

The AVR, which uses coated-particle fuel formed into larger fuel balls, is an operating terrestrial reactor that has achieved outlet temperatures of 1220 K (1740°F). It is not a space reactor, so considerable modification and redesign would be required if this type of loose-particle fuel were to be used.

The FBR, on the other hand, uses the coated particles as is. The primary problem with this concept is probably in the development of a high-temperature frit to contain the fuel, and so the early version uses an Inconel frit. This reactor has no operating experience and only a little feature test experience.

An assessment of inert-gas-cooled reactor technology is shown in Figure IV-53. A helium-cooled space reactor based on HTGR technology could be designed and ground demonstrated by the end of FY 1990. This reactor would have a power up to 5000 kWt with a 1200 K (1700°F) outlet temperature and a 3-yr life. Development of a ZrC coating for particulate fuel would allow increasing outlet temperature to 1300 K (1880°F) and lifetime to 5 yr. Pursuit of the FBR concept would begin with a 1100 K (1520°F) outlet temperature up to a power level of 4000 kWt. It would employ an Inconel inner frit. The extension of this concept to a higher temperature of 1300 K (1880°F) is predicated on the ability to develop a frit that can withstand the high temperature and neutron irradiation environment for the required design life. An alternative to reactors based on particulate fuel is the ML-1 and GE-710 concept using Hastelloy-clad BeO-UO₂ initially and then tungsten-clad UO₂ fuel elements. This reactor would initially be designed for 1000 kWt at an outlet temperature of 900 K (1160°F); this would be advanced to 2000 kWt at a 1250 K (1790°F) outlet temperature with a 2-yr life. Development of fuels such as UN, UC, and cermets with improved thermal performance capabilities would permit increasing the power to 10,000 kWt at temperatures up to 1500 K (2240°F) for a 3-yr life.

The bi-modal reactor concept considers using the FBR reactor configuration with either hydrogen or helium as coolant depending on circumstances. For steady-state power, helium would be used as the reactor coolant. For pulse or semicontinuous power, the reactor coolant would be switched to hydrogen. Increases in power output of 4 to 5 times is possible in the hydrogen-cooled mode. To develop the FBR for hydrogen use would require further fuel development because high-temperature hydrogen is highly corrosive of graphite-based fuels. The bi-modal concept, therefore, would require several more

Figure IV-53. Subsystem Technology Assessment – Inert-Gas-Cooled Reactors

years beyond the proof test of the highest temperature FBR with helium coolant unless simultaneous development efforts were undertaken.

The following technology issues are associated with inert-gas-cooled reactors (see Figure IV-53):

Metal-Clad Inert-Gas-Cooled Fast Reactors

Technology Step 1

Key Issue - As with the UZrH liquid-metal-cooled thermal reactor and the hydrogen-cooled NERVA/SNRE reactor, reestablish reactor subsystem concept (based on ML-1 and GE-710 results).

Resolution - Design and fabricate (elapsed time 3 years) and test for 1 year the reactor subsystem. Program completion by FY 1988.

Technology Step 2

Key Issue - As with the Li-cooled fast reactor, the key issues are the performance of a refractory-clad UO₂ fuel element.

Resolution - As with the Li-cooled fast reactor, program completion is judged to be possible by the end of 1993.

Technology Step 3

Key Issue - Develop and prove performance of UC, UN, or cermet fuels, refractory clad, at power levels up to 10,000 kWt and coolant

outlet temperatures of 1500 K (approximately 2250°F) for design lifetimes of 3 years.

Resolution - The development steps are essentially similar to those for the Li-cooled refractory-clad UN fueled fast reactor. Program completion is judged possible in the same timeframe as that concept by the end of 1996. Note: It is important to recognize that it has been assumed that in-core control would not be required for the 10,000-kWt power at a 3-yr design lifetime. If this assumption proves incorrect, the development time would extend beyond the 1996 completion date.

Inert-Gas-Cooled HTGR-Type Reactor

Technology Step 1

Key Issue - Demonstrate a compact HTGR-type reactor subsystem.

Resolution - Design and fabricate (elapsed time 4 years) and test for 3 years. Program complete by end of 1990.

Technology Step 2

Key Issue - Performance of improved ZrC-coated fuel at 1300 K (approximately 1900°F) for 5-yr design lifetime.

Resolution - Design and fabricate (2-1/2-yr elapsed time) an EBR-II or FFTF irradiation experiment using improved ZrC coating fuel. Test for 2 years (acceptable risk versus 5-yr lifetime considering a reactor subsystem test will be conducted). Design and fabricate reactor subsystem (1-yr overlap with irradiation test). Test reactor subsystem for 3 to 4 years (acceptable risk versus 5-yr design lifetime). Program completion by late 1993. Note: It is assumed that in-core control is not required for 5-yr operation at 10,000 kWt. If in-core control is required, the development schedule will be extended.

Inert-Gas-Cooled Fixed Bed Reactors

Technology Step 1

Key Issue - The FBR is a conceptual reactor design. Extensive design and fabrication development will be required.

Resolution - It is estimated the design cycle will be of the order of 5 years, including fabrication development and out-of-pile testing of frits and other key technical features. Fabrication will require 2 to 3 years (no overlap with design due to novel nature of the concept). Reactor subsystem test will be full 3-yr design lifetime. Program completion by late 1993.

Technology Step 2

Key Issue - All of the above issues associated with step 1 for the FBR plus development of in-core control for 50,000 kWt/5-yr design lifetime.

Resolution - Development of in-core control (control rods, drives, seals, controllers, etc.) is considered a major technology step for compact reactor applications. It is estimated that the development effort would require 8 to 10 years to qualify such devices and that a full 5-yr reactor subsystem test would be required to prove out such a control scheme fully. This would allow program completion by mid-1998. Note: Development of a high-temperature frit is believed to be reasonably possible in the 8- to 10-yr timeframe required for development and qualification of the in-core control devices.

g. Air-Cooled Reactors

The air-cooled reactors developed during the ANP and Pluto programs are not appropriate for use in space power systems. But the abundance of information on materials development, fabrication techniques, operating experience, and design methods gathered can and should be used in future space reactor programs. The reactor concepts themselves could perhaps be modified for use in space with inert gas or hydrogen coolants, but the degree of modification would be significant. The fuel systems used in HTRE-3 and Tory IIC could probably best be used as a starting point for new design configurations specifically directed toward a space reactor. The actual test experience gained on these fuels would help shorten such a design process if an inert gas coolant were to be used because the performance relative to air should be better. High-temperature hydrogen-cooled systems have unique problems, and the usefulness of the available data would be limited.

h. Heat-Pipe-Cooled Reactors

The SPR-4 and SPR-5 were early conceptual designs of heat-pipe-cooled reactors that did not progress much beyond the study phase. Subsequent heat pipe reactor studies have rendered these concepts obsolete.

After cancellation of all space reactor programs in January 1973, the research and development on a space reactor system was concentrated at LANL. Work at a low funding level was being carried on at other locations, but the focus of attention was on the heat-pipe-cooled reactor that has come to be known as the SABRE.

An assessment of the SABRE technology would reveal that most of the development effort has been directed toward the heat pipe portion of the reactor. Heat pipe technology is discussed in Section IV.E.4.

The critical technology areas other than the core heat pipes are the performance of the UO_2 fuel wafers at the rated conditions, fuel-coolant compatibility, structural integrity of the reactor components during launch and over mission life, and stability and controllability of the reactor.

The subsystem technology assessment is shown in Figure IV-54. The initial version of the reactor uses superalloy (or possibly refractory alloy) heat pipes with sodium as the working fluid. The reactor would be limited to 1100 K (1520°F) to ease development, and the power would be limited to 1000 kWt because of the reduced capability of sodium heat pipes. Above this power, an excessive number of heat pipes might be required. The reactor would be developed and qualified in the FY 1992 time frame and could have a 2-yr life. Advancements of this subsystem technology would raise the temperature to 1500 K (2240°F) and use more heat pipes with lithium working fluid. This is the SABRE concept, and it would be capable of higher power (1500 kWt) because of the higher unit thermal power capability of the lithium heat pipe. Qualification tested by FY 1995, this reactor would still have a life of 2 yr based on burnup and heat pipe lifetime considerations. Further improvements in heat pipe technology (longer demonstrated life and higher temperature) and fuel technology (higher burnup and other fuels) could extend the lifetime to 4 yr or more.

The following technology issues are associated with the pipe-cooled reactors (see Figure IV-54):

Technology Step 1

Key Issue - Design and fabricate a multiple-heat-pipe reactor using annular UO_2 fuel wafers.

Figure IV-54. Subsystem Technology Assessment — Heat-Pipe-Cooled Reactors

Resolution - Design and fabricate a reactor subsystem (design time 2-1/2 years, fabrication assembly checkout 3-1/2 years), operate the reactor subsystem for the full 2-yr design lifetime to prove out the reactor design principles with particular emphasis on the subsystem dynamic behavior during startup(s) and simulated load changes; program completion by late 1991.

Technology Step 2

Key Issue - Design and perform development testing of curved Li/MoRe heat pipes in 8-m lengths; develop industrial capacity to produce the refractory heat pipe tubing and wicks and assemble into workable heat pipes; operate annular UO₂ fuel wafers for 2-yr lifetime at 1500 K (2240°F).

Resolution - Conduct heat pipe design and development program (elapsed time approximately 3 years); develop industry capability to produce refractory heat pipe tubing and wicks; and assemble hundreds of 8-m-long curved heat pipes (approximately 7 years); assemble reactor subsystem (elapsed time 2 years) and test for full 2-yr lifetime; program completion by early 1995.

Technology Step 3

Key Issue - Develop improved fuels with reduced swelling characteristics relative to UO₂, allowing longer operating lifetimes.

Resolution - Due to the fuel geometry associated with the heat pipe reactor, the improved fuels would originally be tested in the step 2 reactor subsystem. Assuming satisfactory operation in that subsystem test, the fuels would be fully tested in this technology step for the full 4-yr lifetime (design and fabrication having been done in parallel with the step 2 reactor subsystem test). The program completion would therefore be in 1999, 4 years (i.e., the reactor subsystem testing time) after the 1995 completion date of step 2.

i. Plasma Reactors

The known material limitations of solid core reactors led to consideration being given concepts that have the ability to convert the energy from fission to thrust without conductive heat transfer limitations. Gaseous and plasma cores were envisioned as having the capability to accomplish this by the direct transfer of energy to the working fluid by radiation. Surface temperatures of 10,000 K (17,500°F) with interior temperatures 10 to 20 times greater were felt possible. More than 15 years of research were devoted to investigating the feasibility of these concepts. Nuclear criticality, radiant heat transfer, fluid mechanics, plasma behavior, stability and control, wall transparency, and radiation effects were some of the areas toward which extensive research was directed. None of this research resulted in a workable reactor concept, but it did establish a technology on which future research into plasma reactors could be based. Relative to other concepts surveyed, it is felt that plasma reactors are the most distant. Since much more research is required before a demonstration reactor could be tested, the relative cost of a plasma reactor could be quite high.

B. SHIELDING SUBSYSTEM

1. Background

The use of nuclear reactor power for space applications necessitates the use of a radiation shielding subsystem to protect personnel and/or equipment from the damaging effects of radiation. The neutrons and gamma rays that accompany the fission process are a significant consideration in the design and application of nuclear reactor power systems.

The shield mass can vary greatly depending on such factors as overall configuration of the space vehicle, the size of the reactor and payload, the strength of the radiation source and its location relative to the vehicle, operational procedures, maintenance requirements, and permissible radiation doses to the payload.

2. Shield Considerations

Many of the problems associated with reactor shielding are largely independent of the application for which the reactor system is to be used. In space applications, however, some features require a different treatment, and some a different emphasis. The principal conditions encountered in space that tend to affect reactor shielding design are:

- 1) The absence of a surrounding gaseous medium (the system operates in vacuum).
- 2) The generally elongated shape of reactor power systems, with the reactor and payload located at opposite ends.
- 3) The likelihood that many of the potential missions will be unmanned and thus require less stringent dose criteria.
- 4) The absence of need, in some cases, to dispose of the reactor after the end of its useful life.
- 5) The necessity to reject heat by radiation to space.

The absence of a surrounding gaseous medium makes it possible to use a shadow shield and thus leave extensive surfaces of the reactor completely unshielded. This, in turn, makes it highly desirable to minimize the total solid angle of the shield subsystem and to place all PCS components and the payload within the shielded cone.

To minimize shield mass, most space nuclear reactor systems have an elongated shape, generally conical in the neighborhood of the reactor. Such shapes result in an inverse-square attenuation of radiation; also the shield must cover only a small solid angle to completely shadow the dose plane.

With unmanned missions, the dose criterion is established by the tolerance of the payload equipment rather than by human tolerance. Electronic equipment is more sensitive to neutrons than to gamma rays by a few orders of magnitude if compared on an equal energy basis. Consequently, the shielding emphasis is heavily on neutron shielding.

The absence of need, in most cases, for access to the spent reactor core eliminates the requirement of after-shutdown shielding and permits a greater degree of freedom in the selection of shield materials.

The necessity to reject heat to space by thermal radiation results in high reactor outlet temperatures to keep the radiator size and mass within reasonable limits. This necessitates using shield materials that can survive high temperatures, about 922 K (1200°F) or higher.

The mass of the shield would represent about 20% of the mass of the reactor power system for unmanned missions and about 50% for manned missions. Thus, the shield mass represents a large fraction of the total system mass. Since the allowable system mass is dependent on the capability of the launch vehicle, shields for space applications are usually optimized for minimum mass.

Five major steps are required in designing a nuclear reactor shield for space applications:

- 1) Determine the ground rules for the shield design (e.g., which geometry to use, what is to be shielded, what is the mission).
- 2) Establish the acceptable radiation levels to protect personnel and/or equipment.
- 3) Determine the various sources of radiation important to the shielding problem, their absolute energy spectra, and their spatial distribution. Secondary sources, such as gamma rays generated as a result of neutron interaction, in addition to scattered neutrons and gamma rays, should be analyzed in great detail.
- 4) Select shield materials on the basis of their shielding, physical, and chemical properties. Where a minimum mass shield is desired, cost may be of secondary importance if a definite improvement can be realized.
- 5) Perform shielding calculations and determine an optimum shield design.

A complete shield design involves following the above five steps not just once, but many times, each time using a different set of design parameters. Finally, a compromise design has to be established.

a. Dose Criteria

The establishment of acceptable radiation levels to protect personnel (for manned missions) and/or equipment (for unmanned missions) is one of the first steps required in the design of a nuclear reactor shield.

(1) Manned Missions

Any significant departure from the natural radiation environment in which man has evolved may introduce some risk. It is possible, however, to define a permissible dose that, in light of present knowledge, is not expected to cause appreciable bodily injury to a person at any time during his normal life span. The following is a summary of recommendations by the National Committee on Radiation Protection (NCRP) and by 10 CFR 20:

- 1) The maximum permissible dose (MPD) shall not exceed 3 rem/13 weeks (in addition, it shall occur at a slow rate rather than all 3 rem in a very short time).
- 2) The MPD shall not exceed 5 times the age minus 18, i.e., $5(\text{age} - 18)$ rem during a lifetime.
- 3) The MPD shall not exceed 200 rem during the normal life span.
- 4) Maximum accidental or emergency exposure occurring only once in a lifetime shall not exceed 25 rem.

Shield designs for terrestrial applications adhere rigidly to these standards of radiation tolerance. For space applications, however, to achieve such low radiation levels would require extremely high shield mass, so high as to perhaps make the mission infeasible. But because astronauts would only be on board the spacecraft for a short time, the dose criteria for manned space missions could be relaxed. For example, the dose criteria used by NASA in 1970 for a space station using space nuclear reactor power are presented in Table IV-16.

Limiting doses applicable to periods of occupancy of less than 2 years could be twice the annual limits given in Table IV-16, provided no exposure is received during the remainder of the 2-year period and that the quarterly and monthly limits are not exceeded. For example, 80 rem may be received by the marrow during a 6-month mission, provided no further exposure occurs during the ensuing 18 months.

TABLE IV-16
DOSE CRITERIA FOR ASTRONAUTS^a

Organ	Dose (rem)			
	Career/ 20 years	Annual	Quarter	Month
Skin	2400	240	150	-
Eye	1200	120	75	-
Testes	-	100	-	-
Marrow	400	40	-	25

^aTaken from Ref. IV-28

In addition to reactor radiation, there are space radiation sources. These consist of trapped protons and electrons, and Bremsstrahlung and high-energy solar flare radiation, all of which contribute to the total integrated dose. Furthermore, an additional dose could be received by the astronauts during rendezvous, accidental flyby, or reactor maintenance. Of the 80 rem/2-year limits to the marrow, 25 rem/6 months (or 5.72 mrem/h) from the reactor with less than 20% of the dose rate from neutrons was selected as the design dose criterion to the astronauts in the prior space station study.

(2) Unmanned Missions

When materials are exposed to a radiation environment, their properties are changed in ways that cannot be anticipated from previous experience with nonradiation environments. Thus, component performance will be altered when these materials are subjected to high levels of nuclear radiation. Two of the observed effects of radiation on materials are important to component life:

- 1) Rate Effects — This type of radiation effect is a function of the instantaneous radiation intensity and is usually associated with ionization. It affects the performance of sensitive circuits in electronic systems by producing noise.

- 2) Integrated Effects — This type of radiation effect is a function of the integrated radiation dose. It is usually the result of semipermanent lattice defects, which produce changes in practically all the physical and chemical properties of materials. The permanent changes that result, usually from prolonged radiation exposures, affect all types of components.

Radiation effects studies have shown that 10^6 nvt (n/cm^2) of fast neutrons ($E \approx 0.1$ MeV) and 1.0 rad of gamma rays cause about equivalent damage to semiconductor components. Figure IV-55 shows thresholds for radiation damage to typical electronic components that may be used for unmanned space applications. Up to a dose of 10^{11} nvt and 10^5 rads, off-the-shelf components can be used. The allowable dose may be increased to 10^{12} nvt and 10^6 rads by inspection hardening, i.e., by selecting components based on existing radiation sensitivity data. One more level may be achieved, 10^{13} nvt and 10^7 rads, at a cost about 5% higher than that of " 10^{12} nvt and 10^6 rad" components. The effect on cost of attempting to design circuitry and parts to survive 10^{14} nvt and 10^8 rads cannot be accurately determined at this time; however, estimates for the additional cost range from 25% upwards.

Some electronic devices are extremely sensitive to gamma rays. For example, commercial MOS devices may exhibit significant degradation for doses of only 10^4 rads. Thus, the selection of electronic components is critical in arriving at a minimum mass reactor shield.

The radiation-sensitive equipment on board a reactor-powered space vehicle can be divided into three parts:

- 1) Equipment to operate the reactor system (startup, continuous operation, shutdown, telemetry, etc.)

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Figure IV-55. Radiation Damage Thresholds

- 2) Vehicle housekeeping equipment (telemetry, vehicle orientation, etc.)
- 3) Mission equipment
 - a) Specific mission equipment (infrared detector, ultraviolet detector, TV camera, radar, surveillance, etc.)
 - b) Associated electronics (telemetry, amplifiers, etc.)

The above parts may be combined into two sections:

Section 1 – Parts 1, 2, and 3b

Section 2 – Part 3a

Components in section 1 are basic electronic components such as telemetry, amplifiers, digital and analog integrated circuits (ICs), etc. However, the components in section 2 (or part 3a) are very special pieces of equipment that may be very sensitive to high radiation fields. For section 1 components, a dose criterion of 10^{13} nvt and 10^7 rads can be used. However, the dose criterion for Section 2 will depend on the sensitivity of the experiment, i.e., its threshold of radiation damage. Usually, the dose criterion for section 2 equipment is lower than that for section 1 equipment. Thus, when a reactor shield is designed, three possibilities exist:

- 1) Design the reactor shield to the dose criterion of section 2 equipment.
- 2) Design the reactor shield to the dose criterion of section 1 equipment and place a local shield around the section 2 equipment to reduce the dose level.
- 3) Optimize the reactor shield and section 2 equipment local shield to yield a minimum weight.

b. Nuclear Properties

With a nuclear reactor heat source, only neutrons and gamma rays are considered in the amount of shielding required. Different types of materials are required to attenuate neutrons and gamma rays. Such materials must be picked to minimize system mass while being consistent with other system design requirements. Selection of a shield material cannot be made solely on the basis of its primary function, that of radiation attenuation. In addition to the nuclear characteristics of the shield material, such factors as its structural characteristics, physical and chemical properties, fabricability, availability, etc., must be considered. The ultimate choice of a shield material must be based on the following characteristics:

- 1) Effective radiation attenuating property
- 2) Minimum mass
- 3) Radiation resistance
- 4) High-temperature stability
- 5) Fabricability
- 6) Availability
- 7) Reasonable cost.

This section discusses the selection of shielding materials based on their nuclear properties.

(1) Neutron Shield

A neutron shield consists of low-atomic-number nuclei that will effectively slow down, i.e., reduce in energy, the high-energy neutrons (approximately 1 MeV) that are released during fission. In addition, after the neutrons have been slowed, the neutron shield material should be able to capture these neutrons without producing excessive gamma radiation. Hydrogen, the lowest atomic number element, is the most effective neutron thermalizer. Thus, shield materials with a high concentration of hydrogen atoms per unit

volume should be used. Use of these materials would minimize the thickness of the neutron shield; in addition, it would minimize the size (and hence the mass) of the gamma ray shield that may follow the neutron shield.

To minimize shield mass, the hydrogen must be combined with one or more other low-atomic-number elements to form a stable compound. That compound should have certain other specific characteristics, for example:

- 1) High hydrogen content
- 2) High melting temperature
- 3) Low hydrogen dissociation pressure
- 4) Low density.

Many different hydrogenous compounds have been considered as an attractive neutron shield on the basis of their high hydrogen content, high neutron removal cross-section-to-density ratio (this accentuates the performance of less dense materials), low density, and the specific characteristics given above.

Based on extensive research, LiH has been selected for the neutron shield material. Figure IV-56 shows a graph of neutron dose rate attenuation as a function of thickness (in g/cm^2) for various low-density neutron shield materials. LiH is an excellent shield material in comparison to the other materials and would be expected to result in a minimum mass shield. In addition, LiH has excellent physical and chemical properties in the presence of a high radiation field and high temperature. These characteristics are discussed in more detail in Section IV.B.2.e.

LiH has been selected for the neutron shield for many projects where neutron sources are present and where shield mass is very critical (e.g., the Aircraft Nuclear Propulsion (ANP) program, all the SNAP reactor power systems).

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Figure IV-56. Neutron Dose Rate Attenuation with Various Shield Materials

(2) Gamma Ray Shield

In shielding against gamma rays, high-atomic-number and high-density materials would be expected to result in a minimum-mass shield. Figure IV-57 shows as a function of gamma ray energy the shielding thickness of various materials that would be required to attenuate by a factor of ten.

The selected gamma ray shield will be a high-density metal (or alloy), and it should be placed as close to the reactor as possible in order to minimize the shield mass. However, as it is moved closer to the reactor core, it will be subjected to a higher neutron flux, and the resulting secondary neutron capture gamma rays generated within the shield will constitute an increasing contribution to the total dose. Thus, for manned space applications, where the gamma ray dose rate has to be reduced to very low values, the gamma ray shield might have to be split into two (or more) layers in order to minimize the shield mass.

Another consideration may be that the gamma ray shield around the reactor core should be a ductile material so that it would have the capability of an energy absorber to provide a high reliability for impact survival of the reactor.

Another consideration is excessive heat generation. For instance, the use of depleted uranium might result in a minimum mass shield; however, if it is used close to the reactor core, the heat generation due to fast neutron fission could be excessive. Thus, another material should be used that will result in a mass close to minimum.

The above shows that the selection of the gamma ray shield material could depend on many factors. In a study to determine the minimum mass system for a manned space application, the following high-density shield materials were considered: U-8Mo (depleted uranium with 8 wt. % Mo), Ta-10W (tungsten alloy,

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Figure IV-57. Thickness to Attenuate Narrow Beam
of Gamma Rays by a Factor of 10

10 wt. % W), lead, and stainless steel. The shield around the reactor core was required to be ductile so that it would have the capability to absorb energy and provide a high probability of reactor impact survival. Lead was rejected because of its low melting point, 600 K (620°F). Borated steel was rejected because of its excessive mass. U-8Mo was rejected because of high internal heating (due to fast fission) and its instability at high temperatures. Tungsten was rejected because it is not ductile. Ta-10W was selected because of its high density and because it possesses physical properties (for example, ductility) that meet the requirements.

For the gamma ray shield below the "gallery" region which housed the primary heat exchanger, U-8Mo was selected because it resulted in a minimum mass shield. Note that because the fast neutron flux is low below the gallery region, the heat generation due to fast fission and temperature of the U-8Mo are low. The above case was for a 25-kWe SNAP-8 type reactor system^{IV-29} using NaK for the primary coolant.

The above shows that the selection of the gamma ray shield will depend on the mission (manned or unmanned) and also on other factors, such as reactor thermal power, reactor coolant used, dose criteria, etc.

c. Scattering

To keep the shield mass to a minimum, shadow shields are used for space applications. Since in space the reactor will be in a vacuum, the shield will have to cover the dose plane entirely. For unmanned applications, the neutron shield generally attenuates the neutron flux from the reactor by a factor of at least 10^4 . Scattered neutron flux is about 1% of the direct flux. Thus, the flux from objects outside the shielded cone could be much higher than the shielded flux. Great care has to be taken to assure that no scattering sources exist, or that if they do, their contribution is very small.

If placed outside the conical shield, primary heat transport equipment will be a scattering source and must be analyzed. In the SNAP reactor program, a few approaches were investigated, by either having the primary coolant pipes penetrate the shield or by having the coolant pipes running along the outside of the shield or in grooves along the outside of the shield. In these last two cases, the angular distribution of the neutron leakage flux was used to determine the effect of scattering from the coolant pipes.

Another scattering source could be space modules that are located outside the shielded cone for either manned or unmanned systems. In these cases, the composition and location of the part outside the shielded cone should be determined so that the scattered flux to the dose plane can be determined.

d. Streaming

Streaming through penetrations in shields may be a major problem. Since the neutron shield usually attenuates the neutron source by a factor of at least 10^4 , neutrons that stream through penetrations in the shield could be a larger source than the attenuated neutron flux. These penetrations in the shields could be from:

- 1) Primary coolant piping or heat pipes
- 2) Reactor control actuator drives
- 3) Shield cooling penetrations (e.g., coolant pipes, heat pipes)
- 4) Electrical conduits for cables.

In addition, neutron streaming can take place through structures made of steel, especially if large neutron attenuations are expected. Thus, whenever there are penetrations in the neutron shield, the penalty due to these penetrations should be determined. The analysis of the effect of penetrations through thick neutron shields will contain large uncertainties. Thus, experiments should be performed to determine the effect of penetrations, especially if there are many of them (e.g., if heat pipes penetrate the shield for either shield cooling or transferring heat out of the reactor core).

In the ANP program, the shield for the direct gas cycle had ducts both in front and in back of the reactors for the air to get in and out of the engines. Shielding experiments showed that neutron streaming through these ducts was excessive.

e. Physical Properties

(1) Neutron Shielding Materials

As discussed above, LiH was selected as the primary SNAP neutron shield material after an extensive evaluation of its nuclear as well as physical properties. References IV-29 and IV-30 show the extent of the evaluation, which ranged from simple water to the more esoteric organic compounds, the saline hydrides of LiH, NaH, CaH_2 , etc., and the metallic hydrides of Zr, Y, and Ti, etc.

In the final analysis, only LiH possessed all the required properties:

- 1) A high hydrogen content and high neutron macroscopic removal cross section
- 2) A high melting temperature
- 3) A low dissociation pressure
- 4) A low density.

Another quality that added to its attractiveness was favorable raw materials availability, although the basic cost was high. Tonnage amounts are available from several commercial sources with an acceptable 97.5% LiH purity and excellent quality control.

Properties of specific interest are shown in Table IV-17 and Figures IV-58 through IV-61. (References IV-29 through IV-32 give more comprehensive discussions and additional properties.)

a. SOLID

b. MOLTEN

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Figure IV-58. Density of Solid and Molten LiH

Figure IV-59. Thermal Conductivity of LiH

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Figure IV-60. Thermal Expansion of LiH

Figure IV-61. Specific Heat of Solid and Molten LiH

TABLE IV-17
PROPERTIES OF LITHIUM HYDRIDE

Molecular weight	5.95
Density (g/ml at room temperature)	0.775
N_H ($\times 10^{22}$ atoms H/cm ³)	5.85
H content (wt. %)	12.68
Melting point [K (°F)]	959 (1266)
Σ_r/ρ (cm ² /g)	0.152
Lattice constant (Å)	4.083
Crystal structure	fcc
Molar volume (cm ³ /mole)	10.254
Heat of fusion (kcal/mole)	5.2 $\pm$ 0.3
Volume change on melting ^a (%)	+25 $\pm$ 2
Compressive strength, cold pressed (psi)	14000-24000

^aThis property is extremely important because it dictates that the LiH cannot be melted without the potential for harm to the shield vessel.

(2) Gamma Shielding Materials

Several high-density, high-melting-point materials (e.g., U, W, and Ta, various of their alloys, Pb) have been considered as gamma-shielding materials, as discussed above. Although the 90 W-7.5 Ni-2.5 Cu alloy was selected for gamma shielding for the GE-ANP reactor, other materials were considered for the SNAP-8 system (the SNAP-2 and -10A systems did not have a gamma ray shield). Table IV-18 lists the materials and some of their pertinent properties. Of these materials, serious consideration was given to only U-8Mo and Ta-10W. The Pb and Pb-Bi alloys were evaluated for use as liquid metal shields, and some promise was indicated for future reactor electric systems.

Some studies also were conducted on a combined neutron-gamma shield utilizing an aggregate of crushed or granular 98 W-2Mo or zirconium hydride mixed with LiH. This also showed promise in specific applications where the heat generation would not cause melting of the LiH.^{IV-29}

TABLE IV-18
PROPERTIES OF GAMMA-RAY SHIELDING MATERIALS

Property	Uranium ^a	Uranium - 8 wt. % Molybdenum ^a	Tantalum - 10 wt. % Tungsten	Lead	Lead - 55 wt. % Bismuth	Stainless Steel/Boron	Lead Oxide (PbO)
Density (g/cm ³)	19.0 at RT ^b	17.40 at RT	16.83 at RT	11.35 at 298 K	10.65 at 298 K	8.03 ^c	9.27
Thermal expansion (10 ⁻⁶ cm/cm·K)	12.2 at RT	14.4 at 673 K	6.3 at 373 K 6.9 at 773 K			18.9	26.5 at RT
Thermal conductivity (cal/s·cm·K)	0.060 at 298 K	0.32 at 773 K 0.69 at 773 K	0.182 at 298 K 0.166 at 773 K	0.082 at 298 K 0.070 at 573 K	0.022 at 373 K		0.0017 at RT
Specific heat (kcal/kg·K)	6.65 at 298 K 10.0 at 773 K	0.033 at 298 K 0.042 at 773 K	0.0358 at 298 K 0.0351 at 773 K	0.0305 at 298 K 0.034 at 573 K	0.0335 at 373 K	0.12	0.0505 at 298 K 0.053 at 473 K
Melting point (K)	1403	1403	-	600	398	-	1152-1163

^aRT = room temperature

^b2.0 wt. % boron

^cReference IV-33

f. Shield Radiation Damage

A significant number of irradiation tests have been conducted on cold-pressed and cast LiH samples, as indicated in Reference IV-32, and almost as many theories of radiation damage have been postulated. Without attempting to define the mechanisms of damage or nondamage, Figure IV-62 describes the observed behavior of the irradiated specimens of LiH. Unfortunately, the test observations do not all fall neatly into the patterns; however, many of the test results that did not agree were from tests that had been compromised by experimental error, defective containers, air contamination, etc.^{IV-31, IV-32} Figure IV-62 shows that for high neutron fluences and low LiH temperatures, the percent volume expansion could be as high as 25% for a neutron fluence of 2×10^{20} nvt. However, when the temperature of the LiH is increased to above 644 K (700°F), the percent volume expansion is less than 2%. These test data and the postulates leave unresolved the mechanism of damage and do not define what may occur in the broad temperature range in which space reactor electric systems might operate. It is thus recommended that until such definition is resolved, future shields should be designed to operate in the 644 K (700°F) to 922 K (1200°F) region, where no significant radiation damage has been observed in either cast or cold-pressed LiH.

Little or no irradiation testing was conducted on high-density gamma shielding material under the SNAP programs, except for U and several of its alloys. Based on irradiation testing in the materials testing reactor and on an extensive literature survey, it was recommended that U-8Mo be used for high-temperature (about 1144 K (1600°F)) applications and U-0.1 wt. % Cr for low-temperature (about 561 K (550°F)) applications. In both cases, to reduce the number of neutrons produced by the U^{235} fission, it was recommended that depleted U be used in the alloy.^{IV-33}

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Figure IV-62. Radiation-Induced Expansion in LiH (Natural)

g. Thermal

The thermal environment of reactor shields for space missions is an important consideration. When LiH melts, it expands about 25%. Thus, every effort should be made to keep the LiH below its melting temperature of 959 K (1266°F). The LiH heating is from two sources: the primary heat transport system and nuclear heating, i.e., from neutrons and gamma rays.

Reactors for space missions have coolant temperatures in excess of 810 K (1000°F). The method of primary heat transport from the reactor to the PCS through or around the LiH shield is an important consideration. In either case, insulation should be used around the piping to assure that the LiH does not locally melt.

In the case of SNAP 10A, which was placed in orbit, the reactor outlet temperature was about 810 K (1000°F), and the primary coolant pipes were placed outside the shield. Since the coolant temperature was below the melting temperature of LiH, no problems were encountered.

Nuclear heating is due to both neutron and gamma rays. In the case of LiH, the heating is mainly due to the reaction of neutrons with the Li^6 isotope, which constitutes 7.5% of natural lithium. In this reaction, a neutron is absorbed in Li^6 , and a H^3 atom is released. The energy released is 2.8 MeV per reaction. Since LiH has a low thermal conductivity, the temperature of the LiH could become important. In the case of the SNAP-10A reactor, the thermal power was only 43 kWt, and there was no heating problem. However, in the case of SNAP-8, which produced 600 kWt, neutron heating was a problem. In the latter case, the upper part of the shield was made of LiH with 99.99 wt. % Li^7 instead of natural lithium. Since the amount of Li^6 had been so greatly reduced, the heat generation was reduced to acceptable levels.

For gamma ray shields, since high-density materials are used to minimize the shield mass, heat generation may in some instances become a problem. U-8Mo shields have low mass when placed immediately around the reactor core. However, the internal heat generation could be excessive. This is due to the fast fissioning of the uranium even if depleted uranium is used (U^{235} less than 0.2 wt. %). In regions where the neutron flux is low, using U-8Mo will result in a minimum mass shield.

Cooling the shields is another problem that becomes important for reactor powers greater than 1 MWt. The LiH shield must be cooled either actively or passively. Active cooling would be by using either a coolant or heat pipes. In either case, the thermal analysis would be complex; in addition, neutron streaming could become a problem. Passive cooling of LiH may be by placing high-thermal-conductivity materials in the shield and thus conducting the heat radially. In this case, a mass penalty might result.

h. Mechanical Aspects

In addition to its shielding functions, the LiH shield acts as an integral part of the overall system structure. It is attached to the radiator structure at its aft end and supports the reactor at its forward end. Structural requirements are therefore an important aspect of the LiH shield design.

The LiH shield vessel serves the following functions: contains the LiH shield, provides a containment for the hydrogen overpressure, and serves as a structure to mate the reactor to the radiator. The vessel itself has to provide protection against meteoroids puncturing the vessel. In the zirconium hydride reactor program, for example, the design criterion for the shield regarding meteoroid impact was a 0.998 probability of not losing enough hydrogen to increase the dose rate at the dose plane by 10%. The shield vessel wall thickness was designed to meet this requirement.

In addition, the shield vessel (which is part of the structure) has to meet launch loads. The loads from the reactor and gamma ray shields will be transmitted through the LiH shield (vessel) to the radiator structure. This must be accomplished at the acceleration loads of the launch vehicle used. For high-power reactors (i.e., greater than 1 MWt), the long moment arm between the reactor center of gravity and the top of the shield may make the high transverse accelerations significant.

In some instances, providing an internal structure within the LiH shield to support the reactor may result in a lower mass than would increasing the thickness of the vessel wall. Thus, based on the launch loads and mass distribution, the vessel wall and structure must be optimized to minimize overall mass.

i. Unmanned/Manned Missions

The shield geometry will differ depending on whether the mission is unmanned or manned. The shield mass may comprise 20 to 50% of the total reactor electric system mass. Because launch costs are extremely sensitive to mass, the minimization of shield mass is a prime concern in the design of space nuclear reactor systems.

For unmanned reactor power systems, minimum mass is achieved by using a shadow shield. Since the reactor is generally smaller in size than the payload, it is obvious that the shield mass will be a direct function of the solid angle that the shield shadow must subtend. In addition, the shield geometry must be such that there are no paths or mechanisms by which radiation can bypass or be shunted around the shield (i.e., scattered by objects inside or outside the shielded cone); otherwise, the shield's effectiveness will be drastically reduced. As a result of this consideration, minimum mass shields require that the reactor have a minimum projected area and that all structural members be contained within the conical region defined by the dose plane and the extremities of the source.

Low-power reactor systems (less than 100 kWt) for unmanned missions generally require only a neutron shield. For higher power levels and/or for more stringent radiation dose criteria (i.e., less than 10^{13} nvt and 10^7 rads), a gamma ray shield is generally also required. For example, SNAP-10A had only a neutron shield. However, most past studies for higher power level systems had a gamma ray shield located between the reactor and the neutron shield.

The shield mass is a function of the distance between the reactor and the dose plane and of the diameter of the dose plane (Figure IV-63). As the separation distance is increased, the shield mass goes down for constant dose rate requirements. However, the mass of the (boom) structure holding the system goes up. There is a separation distance at which a minimum overall mass is achieved. Dose plane size complexly affects shield shaping and mass.

Reactor power systems for manned missions, as a result of the much lower radiation tolerance of man, require greater neutron and gamma ray attenuation. Neutron capture gamma rays become significant contributors to the total dose rate. In general, a manned system shield would be a laminated one with alternate layers of heavy metal followed by LiH. For example, a manned system shield for a reactor that uses a liquid metal coolant would generally consist of five regions:

- 1) A heavy metal gamma ray shield to reduce the core fission gamma ray dose rate to a level similar to that by any radioactive primary heat transport subsystem coolant in the "gallery" region (see Figure IV-64)
- 2) A LiH shield of sufficient attenuation to prevent neutron activation of any secondary heat transport equipment or working fluid in the "gallery" region and to keep the secondary gamma ray production in the second gamma ray shield to a minimum
- 3) A gallery region to house any primary heat transport equipment

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Figure IV-63. Unmanned Space Nuclear Reactor Shield
(Schematic)

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Figure IV-64. Manned Space Nuclear Reactor
Shield (Schematic)

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- 4) A heavy metal gamma ray shield to further attenuate the gamma ray dose rate (from core fission gamma rays, neutron capture gamma rays from the first metal shield, and primary heat transport subsystem gamma rays)
- 5) A final LiH shield to provide neutron attenuation.

A typical manned shield is shown in Figure IV-64. In some instances, the shield may be six regions instead of four to achieve a minimum mass.

The discussion above describes the basic shield geometries for unmanned and manned reactor electric systems. The dose criteria, reactor power, and type of PCS have a strong impact on shield design. Mission requirements also have an important bearing on shield design. For instance, in one study made for NASA, two dose criteria were imposed on the system.^{IV-34} The first was a dose plane (space station in this case) dose rate of 5 rem per 6 months (or 5.72 mrem/h); the second was a side dose rate of 72 rem/h at 30.48 m (100 ft). The side dose rate was 90° to the space station. The reason this value was set was to minimize the dose to the astronauts during both rendezvous and accidental flyby. Figure IV-65 presents a schematic of the shield for this study, and Figure IV-66 presents the radiation environment from the nuclear reactor around the space station. Both the rendezvous path and fly-by trajectory are shown.

The above discussion has shown that the shield design for a space nuclear reactor is strongly dependent on many factors, including reactor thermal power, dose criteria, PCS, and mission constraints.

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Figure IV-65. Schematic of a Manned Space Nuclear Reactor Shield with a Side Shield

Figure IV-66. Radiation Environment from a Nuclear Reactor
for a Manned Space Station

3. Shield Descriptions

a. Shields Built

(1) Early GE-ANP Shield

The attractive properties of LiH (e.g., high-hydrogen content, high-melting temperature, low-dissociation pressure, low density) (see Section IV.B.2.e) established the material as a unique neutron shield for mobile reactors. Extensive fabrication development was conducted at the General Electric Aircraft Nuclear Propulsion Department in the 1950s. Initial studies were based on the use of cold pressed and machined shapes of LiH for which a technology existed at the Union Carbide Nuclear Co. Y-12 plant.^{IV-35}

This technique was abandoned when difficulties arose in retaining the complicated shapes in the shield vessel in a manner that would accommodate the large differential thermal expansion between the stainless steel and the LiH and yet prevent shock and vibration loads from damaging either the vessel or the LiH.

Subsequently, melting and casting of the LiH directly into the shield vessel was adopted at GE-ANPD after exploratory studies at Union Carbide Nuclear Company Y-12 plant. GE established that zone freezing (from bottom to top) the LiH produced a sound, dense casting.^{IV-36,IV-37} Following the initial success on small-scale specimens, a more comprehensive program was established. That program eventually led to the casting fabrication of a 6-ft-long scale-model test specimen and several 9-ft-long prototype (Figure IV-67) shield sections.^{IV-38} The development work at GE-ANP on large LiH neutron shields ceased at the termination of the program in 1961.

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Figure IV-67. Nine-foot-Long LiH Casting
Produced at GE-ANPD

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Gamma shielding fabrication development at GE-ANP was limited to a cold pressed and sintered alloy of copper-nickel-tungsten (90% W, 7.5% Ni, 2.5% Cu), although development studies were conducted on other alloys, including an iron-nickel-tungsten alloy.^{IV-36} One massive gamma shield segment of Cu-Ni-W was produced for GE-ANP by the P. R. Mallory Co. to establish fabrication feasibility.

(2) SNAP-10A Shield

The SNAP-10A neutron shield utilized a single block of cold pressed and machined LiH produced at the Y-12 plant of Union Carbide. Extensive nuclear and nonnuclear testing was conducted on the cold pressed shields to establish the worthiness of the shield prior to the space launch of the first operating SNAP-10A reactor system from Vandenberg Space Center.^{IV-39}

Concurrently, a LiH casting development program was conducted at Atomics International, based on the GE-ANP casting technology; this new process presented the advantages of producing cheaper, more structurally reliable reactor shields and producing them faster. By mid-1962, a new casting facility had been completed, and over the life of the SNAP 10A program several shields and test specimens were fabricated at the facility.^{IV-40} Table IV-19 presents the characteristics of the various shields built. In general, the shields contained a large block of perforated stainless steel honeycomb (Figure IV-68) into which the molten hydride was cast. The cellular arrangement of the block was such that the shrinkage cracks that occur naturally in crystalline LiH as it is cooled were directed away from the longitudinal dimension of the shield, thus minimizing radiation streaming through the shield. A later test shield, the SNAP-10A DRM 1-C, also incorporated a low mass, internal rib structure to support the reactor; this shield demonstrated the casting process to be versatile enough to incorporate such features.

TABLE IV-19
CHARACTERISTICS OF CAST LiH SHIELDS AND TEST SPECIMENS

Shield/Specimen	Description	Total Volume [m ³ (ft ³)]	Void Volume [m ³ (ft ³)]	Percent Dense	LiH Specific Gravity
a. SNAP-10A test casting	Cylindrical vessel containing 91 kg (200 lb) LiH and two unmachined honeycomb blocks	0.119 (4.21)	0.0068 (0.24)	94.4	0.732
b. SNAP-10A cast shield	Conical vessel containing 91 kg (200 lb) LiH and a machined honeycomb	0.122 (4.32)	0.0082 (0.29)	93.3	0.722
c. SNAP-10A DRM-1C development shield	Conical SNAP-10A vessel containing 91 kg (200 lb) LiH, a machined honeycomb lock, and a low-mass internal rib structure	0.122 (4.32)	0.0062 (0.22)	95.0	0.737
d. SNAP-2 development shield	Conical vessel containing 136 kg (300 lb) LiH and heavy internal ribs on the cone to support reactor loads	0.178 (6.27)	0.0113 (0.40)	93.6	0.726
e. SNAP-2 DRM-1 shield	Conical vessel containing 136 kg (300 lb) LiH and heavy internal ribs on the cone to support reactor loads	0.178 (6.27)	0.0113 (0.40)	93.6	0.726
f. SNAP-8 thermal shield	Conical SNAP-10A vessel with hemispherical head containing 109 kg (240 lb) LiH and a machined honeycomb shape, instrumented with 15 internal thermocouples	0.146 (5.15)	0.0081 (0.29)	94.3	0.730
g. SNAP-8 two-region test casting	Cylindrical vessel containing 205 kg (450 lb) LiH	0.266 (9.41)	0.0190 (0.67)	92.9	0.719
h. SNAP-8 two-region DRM-1 shield	Conical vessel containing 327 kg (720 lb) LiH and a machined block of honeycomb in each region	0.423 (14.93)	0.0261 (0.92)	93.8	0.727
i. SNAP-8 experimental shield 1	Conical SNAP-10A vessel containing 91 kg (200 lb) LiH, a steel wool filler, and a low-mass internal tubular support for the reactor loads	0.122 (4.32)	0.0068 (0.24)	94.2	0.730
j. SNAP-8 experimental shield 2	Conical SNAP-10A vessel containing 91 kg (200 lb) LiH, a 91-kg (200-lb) layer of granulated tungsten alloy infiltrated with LiH, and two shield traversing pipes	0.120 (4.25)	0.0074 (0.26)	93.7	0.726
k. SNAP-8 DR shield	Same as SNAP-8 two-region DRM-1 shield, except converted to a single region shield	0.423 (14.93)	0.0252 (0.89)	94.2	0.730
l. SNAP-8 experimental shield 3	Annular shield with conical side, containing 55 kg (120 lb) LiH and a honeycomb	0.072 (2.57)	0.0051 (0.18)	93.1	0.721

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Figure IV-68. SNAP-10A Shield Vessel, Shaped Honeycomb Block, and Vessel Bottom

(3) SNAP-2 Shield

Whereas the SNAP-10A shield contained about 91 kg (200 lb) of LiH, the SNAP-2 shield contained about 50% more LiH, about 136 kg (300 lb), providing a larger diameter shadow of protection. Two SNAP-2 shields were fabricated by the casting process and showed identical characteristics, as indicated by d and e of Table IV-19, effectively demonstrating the repeatability of the process. Both shield vessels contained heavy internal rib structures for reactor support to minimize the radiation scatter that would occur off an external support structure.

(4) SNAP-8 Shield

Seven shields and test specimens were fabricated using the casting process under the SNAP-8 program, including the SNAP-8DR (Figure IV-69), which was tested in the SNAP-8 Reactor Test Facility. These components incorporated many innovations, including:

- 1) A two-region shield fabricated by the simultaneous casting of LiH from dual reservoirs
- 2) The use of an inexpensive steel wool filler to replace the costly machined block of honeycomb (Figure IV-70)
- 3) A low mass, internal, tubular support structure for the reactor
- 4) A cast-in-place layer of granulated tungsten alloy gamma shield
- 5) Twin shield-traversing pipes for the conductance of the liquid metal heat transfer fluid from the reactor to the PCS (Figure IV-70).

Also included in the program was the fabrication of a small annular shield to demonstrate the feasibility of producing a wrap-around shield by the casting process (Figure IV-71). Table IV-19 summarizes the characteristics of the seven SNAP-8 shields, and References IV-40 through IV-42 describe in detail the fabrication, testing, and examination of the many SNAP shadow shields.

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Figure IV-69. Instrumented SNAP-8DR Shield

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Figure IV-70. Experimental SNAP-8 Shield 2 —
View from Bottom During Assembly

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Figure IV-71. Annular Shield with Conical Side
(Experimental SNAP-8 Shield 3)

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(5) Advanced Manned SNAP Shield

The later emphasis on the manned exploration of space, and the resulting large electrical power requirements, directed the shield fabrication efforts toward the production of larger neutron shields beyond the size capability of the AI small casting facility. A new facility built at AI could produce shields up to 2.44 m (8 ft) in diameter and weighing up to 4545 kg (10,000 lb).

One large shield (experimental shield 4) was cast in the facility to check out the casting process and the new equipment. Figure IV-72 is a schematic of the large shield, shown inverted from its normal attitude to permit filling with the domed surface on top. The loading, melting, and freezing operations are illustrated in Figure IV-73. The large, pancake shape was 2.18 m (86 in.) in diameter and 0.33 m (13 in.) thick at the center and contained about 773 kg (1700 lb) of LiH (Figure IV-74).

Radiographic examination of the completed shield revealed the casting to be the soundest produced at AI, exhibiting fewer than average of the small cracks and fissures normally occurring between the LiH crystals. Following the nondestructive examination, the shield was transferred to a specially designed and built N₂-atmosphere examination room. An extensive destructive examination was conducted wherein the stainless steel dome was cut from the shield, and the LiH body sectioned and examined for defects, large cracks, and casting voids. The examination confirmed the soundness and uniformity of the large casting as indicated by the nondestructive testing. References IV-43 and IV-44 describe in detail the design, fabrication, casting, and destructive examination of experimental shield 4.

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Figure IV-72. Schematic of LiH Experimental Shield 4

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Figure IV-73. LiH Shield Filling and Casting Steps

Figure IV-74. Completed LiH Experimental Shield 4

b. Shield Designs

In addition to the shields actually built for space nuclear systems, many have been designed for various systems. This section describes some such shields.

(1) SNAP-50 Shield

The SNAP-50 shield design is shown in Figure IV-75. Only a shield of LiH is used. In this design, the LiH is 0.65 m (25.8 in.) thick, with a maximum diameter of 1.27 m (51.0 in.). The top 0.254 m (10 in.) of the LiH shield is made of Li^7H (i.e., lithium depleted of the Li^6 isotope) to reduce the heat generation.

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Figure IV-75. PWAR-20 (SNAP-50/SPUR) Reactor and Shield Concept

Interesting features of this design are the control drives penetrating the LiH shield, and the primary coolant pipes going outside the LiH shield, with the two "patch" shields to reduce neutron scattering off the two primary cooling pipes. Having the shield away from the reactor reduces the heat generation in the shield.^{IV-45}

(2) SABRE Shield

The SABRE shield concept is shown in Figure IV-76. LiH and tungsten shields are used. The control drive actuators penetrate the shield. The reactor uses heat pipes to transfer the heat to the PCS; the pipes are located

Figure IV-76. LANL SABRE Shield Concept

on the outside of the shield. Shield collars are placed above the heat pipes on the outer periphery of the shield to reduce neutron scattering off the heat pipes. Several problem areas are associated with this shield design: streaming through the control drives that penetrate the shield, streaming through the reactor heat pipes, shield cooling, keeping temperatures above 644 K (700°F) to minimize swelling, and scattering off the heat pipes around the periphery of the LiH shield.^{IV-46}

(3) Nuclear Rocket Program Shield

The nuclear rocket program used the liquid hydrogen propellant as a shield. As the hydrogen was used up, the reactor would shut down, and the residual radiation level on the payload was due to the reactor fission products, which decayed rapidly. This concept had a zirconium-hydride shield just below the reactor support plate. The purpose of this shield was primarily to reduce engine component heating rates by roughly a factor of ten. It also reduced the propellant tank heating by roughly a factor of two, and payload doses by a factor of nine. Figure IV-77 shows the LANL small nuclear rocket engine alpha concept.^{IV-47}

(4) Aircraft Nuclear Propulsion Shield

In the ANP program from late 1950 to about 1962, extensive research and development for shield materials were performed. One of the materials developed was LiH. Extensive shielding studies and experiments were performed at General Electric, Convair, Pratt and Whitney, and Oak Ridge National Laboratory (ORNL).

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Figure IV-77. Small Nuclear Rocket Concept, with Shield

North American Aviation Company (now Rockwell International) performed studies related to placing the Pratt and Whitney reactor in the B-70 in 1960.^{IV-48} In these studies, LiH and depleted uranium were used for the shield around the reactor.

4. Technology Assessment

Present space nuclear reactor shield technology is based on a reactor coolant maximum temperature of about 978 K (1300°F) and a reactor thermal power of less than 1 MWt. When either of these limits is exceeded, then new technology will have to be developed to meet these new conditions.

a. Neutron Shield

LiH has been selected for the neutron-shield material because of its high melting point of 959 K (1266°F) and low hydrogen dissociation pressure. For reactors with a thermal power in excess of 1 MWt and reactor coolant outlet temperature in excess of 978 K (1300°F), care must be taken to keep the LiH in the solid form since, when LiH melts, it expands 25 vol %. Thus, passive and active shield-cooling methods must be developed and tested. The shield-cooling methods discussed in Section IV.B.2.g (i.e., conduction, liquid coolant, and heat-pipe methods) should be investigated in detail. In addition, further development is required to verify the optimum shield-cooling method.

The effect of neutrons streaming through the LiH shield due to either reactor coolant piping/ducts, heat pipes, or shield-cooling tubes must be analyzed. In this case, shielding experiments would provide the most meaningful data. An unshielded compact reactor would be required for this experiment, such as the SNAP-10A reactor at ORNL.

Another area of development would be methods of transferring the heat from the reactor core to the PCS through or around the neutron shield. When the reactor coolant pipes are routed through the shield, shield temperature

and neutron streaming will both have to be considered. If the primary coolant pipes are routed around the shield, scattering will also have to be investigated in depth.

The effect of shutting the reactor down and restarting it is a major concern. During a startup/shutdown cycle, the shields will be heated and cooled. The LiH material can withstand high neutron fluxes at temperatures above 644 K (700°F). No experimental data exist at this time to determine what will happen to the LiH if it is exposed to high neutron fluxes and a temperature above 644 K (700°F) and then cooled down below this temperature with zero neutron flux. Thus, the effect of shutting down and restarting the reactor (to full power) is a subject for further development.

Since space nuclear reactors may require coolant temperatures above 978 K (1300°F), new neutron-shield materials should be investigated that can withstand temperatures above the melting point of LiH so that active cooling is not required. If such materials are found, their fabrication techniques will need to be developed.

Figure IV-78 schematically shows the technology status of the neutron shield as a function of time. In 1992, specific LiH neutron shields could be qualification tested with active cooling. A hypothetical molten shield could be through qualification in 1996. This shield material could be molten LiH or some new shield material that does not require active cooling.

The following technology issues are associated with the neutron shield (see Figure IV-78):

Figure IV-78. Subsystem Technology Assessment — Neutron Shield

Technology Step 1

LiH CAST
IN SS VESSEL
 $T_{MAX} \leq 930$ K
PASSIVE
COOLING

Key Issue - Reestablish LiH shield casting capability from SNAP-2, -8, and -10A programs.

Resolution - Refurbish the LiH shield casting facility and cast several shields to demonstrate the capability; program completion in 3-1/2 years (by mid 1987).

Technology Step 2

LiH SHIELD
 $T_{MAX} \leq 930$ K
ACTIVE
COOLING

Key Issue - Design and fabricate a cast LiH shield with internal cooling passages.

Resolution - Design and analyze shield to establish geometry and cooling passage circuits and layouts (elapsed time 2 years); fabricate developmental shields to above geometry with imbedded thermocouples (elapsed time 2 to 3 years); test shield by measuring temperature response to active heating and compare to analytical predictions, thereby validating thermal models and performance (elapsed time 3 years); evaluate and verify neutron/gamma streaming through cooling passages (in parallel with thermal testing program); program completion by 1992.

Technology Step 3

MOLTEN
NEUTRON
SHIELD

Key Issue - Develop ability to design and fabricate a solid shield with internal provisions to allow melting the solid shield without producing damaging distortions of the shield container; gain ability to freeze and rethaw the shield in orbit (during reactor subsystem dormancy periods).

Resolution - Perform extensive design and developmental fabrication of shield concepts to demonstrate continuous melting-freezing-thawing cycles without damaging the shield container; perform metallurgical tests to ensure that molten shield material is compatible with shield container material; perform analytical studies to determine control of molten shield material in zero-g environment; estimated elapsed time 9 to 13 years; program completion by early 1997.

b. Gamma-Ray Shield

Similar to the LiH shield, the gamma-ray shields present problems at high temperatures. Materials, such as U-8 Mo, that once were acceptable may now present problems of thermal ratchetting when they undergo thermal cycling. Thus, all candidate gamma-ray shield materials need to be reexamined at the higher temperatures. Fabrication techniques, particularly for the larger shield sizes, also need to be reexamined. The effect of neutrons on the shields is also an important consideration, since higher total fluences will occur.

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C. PRIMARY HEAT TRANSPORT SUBSYSTEM

1. Background

The function of the primary heat transport subsystem is to transport heat from the reactor to the power conversion subsystem (PCS) or to a second heat transport loop that leads to the PCS. Heat transport can occur by radiation or conduction, but in most cases of practical interest here, it occurs by mass transport of hot fluids. In most cases, the fluids are metals in either the liquid or vapor phase or are gases. In all cases of interest, the fluids are transported in thin-walled, metal alloy piping.

The general requirements of the primary heat transport subsystem (HTS) are as follows: (1) the subsystem must be hermetically sealed so that no fluid leakage occurs; (2) the HTS must have a minimum mass compatible with the power system mass budget; (3) the HTS must be sufficiently reliable; (4) the HTS must interface with other subsystems (such as the reactor, shield, and PCS) in a satisfactory manner; (5) the HTS must be able to start up and function in its prescribed way after launch and after a normal shutdown; (6) the HTS must remove all heat from the reactor and its components so that no reactor part exceeds its design temperature; (7) the fluid pumping power must be consistent with the total system mass and power requirements; (8) the HTS must accommodate all volumetric and dimensional changes caused by temperature variations; and (9) thermal losses must be minimized.

2. Primary Heat Transport Considerations

Heat transport subsystems can be divided into two general types: (1) pumped and (2) heat pipe. The pumped-loop type includes the all-liquid, the liquid-vapor, and the gaseous. The all-liquid type is strictly a heat transport loop, while the liquid-vapor and some gas-loop types contain the working fluid of the PCS. Liquids can be pumped by dc or ac electromagnetic pumps or by canned rotor mechanical pumps. Gases can be circulated by fans in

pure heat transport loops or by compressors if the fluid is the PCS working fluid. Two-phase fluids are pumped in the liquid phase.

Heat pipes provide their own circulation by means of the evaporation, condensation, and capillary processes.

a. Functional Arrangement

(1) Pumped

Pumped heat transport arrangements are shown in Figures IV-79a, b, and c. A fluid (liquid metal or gas) is circulated from the reactor to either a heat exchanger or the PCS and is then returned to the reactor. In some cases, a liquid metal travels to the reactor and returns to the PCS in the vapor phase, as shown in Figure IV-79c.

The loop consists of thin-walled metal piping. The material used can be stainless steel, super alloy, or refractory alloy – the selection depending largely on the temperature. The loop is all-welded and is hermetically sealed. Materials are discussed in Section IV.A.2.

In all-liquid-metal loops, volume accumulators must be used to accommodate the change in both coolant volume and loop volume as the operating temperatures vary (see Figure IV-79a). Also, expansion joints may be necessary to avoid excessive stresses in the piping and other liquid metal components. Loops that contain vapors or gases do not need volume accumulators but may require expansion joints.

The primary HTS is near the peak temperature of the entire power system. The coolant undergoes a temperature change as it cycles through the loop. A small temperature change implies high flow rates and high pumping power. On the other hand, a large temperature change can impose undesirable thermal gradients on the reactor and other components in the loop.

a. LIQUID METAL TO HEAT EXCHANGER

b. GAS TO HEAT EXCHANGER

c. GAS OR VAPOR TO PCS

d. HEAT PIPE COOLED REACTOR

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Figure IV-79. Various Primary Heat Transport Subsystem Arrangements

The reactor imposes several constraints on loop design. The coolant must adequately remove the reactor heat with a reasonable fuel temperature and coolant pressure drop. The nuclear properties of the coolant must be compatible with the criticality and control of the reactor, while the chemical properties must be compatible with the fuel cladding materials.

The reactor nuclear radiation shield also constrains the HTS design. Penetrations of piping through the shield must not compromise the effective shield radiation attenuation. Piping that goes outside the shield diameter must not significantly increase radiation scattering into the dose plane. Heat transfer from the high-temperature piping must not heat the shield materials beyond their specified limits. Finally, the induced radioactivity in the coolant must not be so high as to compromise the performance of the shield.

Often the HTS transports heat to a heat exchanger. As with the reactor, the heat exchanger imposes constraints on the coolant such as good heat transfer, low pressure drop, and materials compatibility. If the primary loop fluid is also the PCS working fluid, then the constraint of coolant compatibility extends throughout the PCS. In some cases, the fluid may also be the radiator working fluid.

Primary loop mass is another consideration. The primary loop piping is usually sized so that the combined mass of the piping, piping coolant, pump, and that portion of the power system assigned to generate the parasitic pump power is a minimum.

Gas loops usually use helium, which has good heat transfer properties, is chemically inert, has satisfactory nuclear properties, and may also be used as the working fluid in turboelectric equipment. In the latter case, the helium may be mixed with other noble gases (such as xenon) to obtain better PCS performance. Gas heat transport piping results in large voids around the reactor and shield. Thus, careful attention must be paid to neutron streaming and scattering.

In some designs, redundant components may be indicated, such as pumps, piping, and pump power supplies.

(2) Heat Pipe

A schematic of a heat pipe loop is shown in Figure IV-79d. A heat pipe is a hermetically sealed tube that contains a heat transport fluid and a wick. Heat is transferred to the tube at the evaporator end and travels to the condenser end of the tube by means of high speed vapor flow. The liquid is condensed and returns to the evaporator end as a liquid in the wick by means of capillary action. Each tube is self-contained – there may be as many heat pipes as required in a given power system.

The heat pipe can receive heat from the reactor directly or from an intermediate heat exchanger between it and the reactor. A heat pipe is a two-phase flow device. Metals such as sodium, potassium, lithium, cesium, and mercury can be considered for nuclear space power heat pipes.

Like pumped loops, heat pipes are in thin-walled tubing. The selection of the material depends on the temperature, the fluid, and to some extent the interface components. The heat pipe provides its own pumping action and requires no separate pump. Heat pipes are generally considered for heat transport only – the fluid does not communicate directly with the PCS cycle.

Since most of the heat pipe contains vapor, there is no need for a volume accumulator. However, since heat pipes are typically long, thin structures, they may need some form of accommodating support.

A heat pipe is very nearly an isothermal device and does not have the temperature swing of a pumped loop. However, significant local thermal gradients occur during startup.

If heat pipes are used, then several interface requirements must be met in the reactor. The voids introduced by the heat pipes must satisfy criticality requirements, and the stability of these voids must satisfy control requirements. Heat transfer from the reactor fuel to the heat pipe wall must lead to satisfactory fuel peak temperatures and temperature gradients. The power distribution within the reactor and the desirable heat flux through the pipe wall must be compatible. Conditions where one or more heat pipes fail must not cause system failure.

b. Operational Considerations

The primary HTS must be designed to withstand the static and dynamic loads during launch and any maneuvers in space. As the startup may not be attempted for some time, the loop must be designed so that either the coolant does not freeze or that startup can be safely performed if it does. This may require using auxiliary power to supply heat to the loop.

For pumped-looped subsystems, auxiliary power will also be required for pumps to circulate the working fluid for startup and to remove reactor decay heat if the reactor is shut down. The amount of decay heat that the primary heat transport loop will be required to remove from the reactor is a function of three parameters:

- 1) Power level at which the reactor was operated
- 2) Time the reactor was operated at power
- 3) Time after reactor shutdown.

The variation of these parameters is shown in Figure IV-80. The decay heat power (percent of full power) is presented as a function of time after shutdown. Two reactor operating times are shown — 30 days and infinite. At 100 s after shutdown, the reactor is generating 3.5% of the operating power for an infinite operating time and 2% for a 30-day operating time. This power decays rapidly and is a strong function of the operating time. Also shown on Figure IV-80 is the total energy released after shutdown has occurred.

Figure IV-80. Reactor Decay Heat Characteristics

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3. Heat Transport Description

a. Liquid Metal Components

(1) Pumps

(a) DC Electromagnetic Pumps

Considerable fabrication and test experience on dc electromagnetic pumps has been accumulated. This dates back to the SNAP-10A pump, which was designed to circulate NaK at $0.05 \text{ m}^3/\text{min}$ (13 gpm) and 0.033 MPa (1.2 psi) at a NaK temperature of 817 K (1010°F). This pump configuration is shown on Figure IV-81a. This pump operates on the Faraday principle, where a current from an external source is caused to flow in the liquid metal, which is confined in a permanent magnet field. The essential elements of the Faraday pump are shown in Figure IV-81a. A more advanced pump was under development for the 5-kWe program.^{IV-49} This pump was designed to pump two separate liquid metal loops. There are two magnet assemblies and a series power bus. The electrical current passes through one pump throat and then the other pump throat. Figure IV-81b shows this pump.

(b) AC Electromagnetic Pumps

The Faraday principle can also be used in a pump that uses alternating current. In this application, the stator winding is connected to a source of polyphase power; a traveling magnetic field is produced that induces currents in the liquid metal. The reaction between the induced currents and the traveling magnetic field causes the liquid metal to follow the traveling magnetic field. The action is the same as in a polyphase induction motor. A pump of recent design, an annular linear induction pump (ALIP) for lithium, is shown in Figure IV-82. This pump was developed by Westinghouse Hanford for group test operations at temperatures of up to 922 K (1200°F). It has been estimated that this type of pump could be developed to operate at up to 1100 K (1520°F) provided that adequate construction materials could be developed.

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a. SNAP-10A Pump (powered by thermoelectrics)

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b. 5-kWe Reactor System DC Conduction Pump (powered by thermoelectrics)

Figure IV-81. Liquid Metal DC Conduction Pumps

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DESIGN POINT:	2.84 m ³ /s at 414 kPa (GAUGE) (750 gpm at 60 psig)
INPUT POWER:	100 kW
OPERATING TEMPERATURE:	589 to 922 K (600 to 1200°F)
WEIGHT:	612.4 kg (1350 lb)
DIMENSIONS:	121.92 cm LONG, 45.72 cm DIAM (48 in. LONG, 18 in. DIAM)
	15.24 cm DUCT (6-in. DUCT)

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Figure IV-82. Westinghouse Hanford Annular Linear Induction Pump
(Ground-based experimental lithium pump)

(c) Mechanical Pumps

Hermetically sealed centrifugal pumps powered by canned electric motor drives are often the pumps of choice for liquid metal applications. A cross section of a SNAP-50 coolant pump is shown in Figure IV-83. This pump design is for $1.3 \text{ m}^3/\text{min}$ (345 gpm) of 1310 K (1900°F) lithium with a head rise of 48 m (158 ft). The canned rotor-type pump has a higher efficiency than the dc or ac electromagnetic pump. A columbium-lithium pump was operated for 10,000 h at 810 K (1000°F) at Pratt & Whitney.^{IV-50}

(2) Heat Pipes

The heat pipe provides an alternative to a pumped system for the transport of heat from the reactor to the PCS. The fundamental principles of the heat pipe are shown in Figure IV-84. The circulatory system transports heat from the evaporator section to the condenser section. Heat input causes the working fluid to evaporate from a wick structure, thereby increasing the vapor pressure at that end of the pipe. As a result, vapor travels down the core of the pipe toward the opposite end. Heat is removed from the pipe as the vapor condenses into the wick. The condensate returns to the evaporator by capillary action of the wick structure. Since the fluid evaporates and condenses at about the same temperature, the temperature along the heat pipe tends to be uniform (within several tens of degrees Kelvin in the worst case).

The performance of heat pipes is dependent on several limits. These limits are shown in general form on Figure IV-85, where axial heat flux is shown as a function of temperature. The limits are as follows: (1) sonic, (2) wicking where vapor is the dominant constituent, (3) wicking where liquid is the dominant constituent, and (4) boiling.

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Figure IV-83. SNAP-50 Reactor Lithium Pump

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Figure IV-84. Heat Pipe — Principle of Operation

Figure IV-85. Heat Pipe Performance Limits

The working fluid used in a heat pipe must be selected to fit the operating range and to provide maximum heat transport. The relative liquid transport capabilities of various liquid metal working fluids are shown in Figure IV-86. The capability of sodium and lithium in a specific heat pipe are compared in Figure IV-87. Since the wicking limit for lithium above 1400 K (2060°F) is far superior to that of sodium, it becomes a preferred working fluid at elevated temperatures up to 1800 K (2780°F).

Heat pipes have been extensively tested and operated. High-temperature heat pipes have been tested independently and in reactors. Selected tests are summarized in Table IV-20. Reactor tests are presented in Table IV-21.

A conceptual design of a reactor and PCS coupled by heat pipes has recently been published by LANL.^{IV-51} This concept, designated SABRE, includes a fast-spectrum UO_2 -fueled reactor with a liquid metal, heat pipe thermal power removal system; a conical radiation shadow shield; and a reactor controller. This is a radiative-coupled system, which means that the thermal power is transferred radiatively from the core heat pipes to the PCS. The PCS consists of TE converters. A major feature of the concept is the use of heat pipes to provide many heat removal paths without using pumps or compressors to transport heat from the reactor. The reactor core heat pipes include a closed containment pipe, a capillary wick structure, a heat transfer fluid, and a material for gettering impurities. The heat pipe characteristics are shown in Table IV-22.

Figure IV-88 illustrates the design of the PCS concept.

Several features of the application of heat pipes in this design still require experimental development to determine their suitability. These features include:

- 1) Sustained operation at the design temperatures
- 2) Fabrication in the desired length

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Figure IV-86. Heat Pipe Liquid Transport Factor vs Temperature for Various Working Fluids Normalized to Sodium at 810 K (1000°F)

Figure IV-87. Comparison of Sodium and Lithium
Working Fluids in One Specific Heat Pipe

TABLE IV-20
HEAT PIPE LIFE TEST DATA

Wall Material	Working Fluid	Heat Pipe Temperature [K (°F)]	Time of Operation (h)	Remarks
CVD-W	Li	1873 (2911)	1,000	No failure
W-26 Re	Li	1873 (2911)	10,000	No failure
W-26 Re	Li	2073 (3271)	740	Corrosion failure
TZM	Li	1773 (2731)	10,526	Weld leak
TZM	Li	1773 (2731)	10,400	Weld failure in end cap
TZM	Li	1773 (2731)	9,800	Weld failure in end cap
VAC-Mo	Li	1500 (2240)	20,000	Continuing
Nb-1Zr	Li	1773 (2731)	9,000	
Nb-1Zr	Na	1123 (1561)	16,000	
304SST	Na	1073 (1471)	12,760	No failure, terminated
316SST	Na	944 (1239)	11,860	Failed, pinhole in evaporator
304SST	Na	1005 (1349)	11,500	No failure, terminated
Ni	Na	1073 (1471)	13,755	No failure, terminated
Haynes 25	Na	1005 (1349)	12,000	No failure, terminated
Hastelloy-X	Na	988 (1318)	35,000	No failure, terminated
347SST	K	783 (950)	6,500	No failure, terminated
Ni	K	873 (1111)	41,000	Gas controlled, no failure
Ni	K	873 (1111)	10,000	Gas controlled, no failure
Nb-1Zr	Cs	1273 (1831)	1,000	Terminated, no failure
Nb-1Zr	Cs	1123 (1561)	5,000	
Ti	Cs	673 (751)	2,000	No failure, terminated
304 SS	Hg	673 (751)	41,900	No failure, terminated

TABLE IV-21
REACTOR TESTS WITH HEAT PIPES

Reactor Test Number	Wall Material	Fluid	Heat Pipe Temperature [K (°F)]	Hours of Operation	Fluence (n/cm ²) Energy > 0.1 MeV
OWR-I	347SST	Na	1023 (1381)	7780	1.6 x 10 ²¹
OWR-TLV	347SST	Na	1023 (1381)	3150	5.3 x 10 ²⁰
EBR-II LADI-1	347SST	Na	1023 (1381)	4166	2.7 x 10 ²¹
EBR-II K-2	347SST	Na	1023 (1381)	1162	4.5 x 10 ²¹
EBR-II K-3, K-4	304LSST	Na K	1023 (1381)	2431	1.5 to 6.0 x 10 ²¹
EBR-II K-5	347SST	Na	1003 (1345)	4080	1.45 x 10 ²²

TABLE IV-22
SABRE HEAT PIPE CHARACTERISTICS

Heat pipe OD (mm)	15.5
Wall thickness (mm)	0.75
Total heat pipe length (m)	9.08
Evaporator length (m)	0.33
Adiabatic length (m)	1.22
Condenser length (m)	7.53
Design power transmission (kW)	12.3
Evaporator radial power density at ID (W/cm ²)	85
Evaporator vapor temperature [K (°F)]	1500 (2240)
Evaporator wall outside temperature [K (°F)]	1507 (2253)
Minimum condenser wall temperature [K (°F)]	1480 (2204)
Number of arteries	4
Artery ID (mm)	2.1
Distribution screen thickness (mm)	0.3
Wick pore size	TBD
Working fluid	Lithium
Evaporator vapor pressure at 1500 K (2240°F) [kPa (psi)]	47 (7)
Getter material	Hafnium, zirconium
Container material	Mo - 13 wt. % Re

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Figure IV-88. SABRE Primary Heat Transport

- 3) Operation with bends to avoid interference with the radiation attenuation shield
- 4) Control of contaminants and impurities.

(3) Piping and Auxiliary Equipment

(a) Piping

Pumped liquid metal heat transport subsystems have been constructed for use with many working fluids, including sodium, potassium, lithium, mercury, and the eutectic mixture of sodium and potassium (NaK). The piping and containment for these liquid metals have been made of stainless steels for the lower temperature NaK systems, such as that used in the zirconium hydride reactors (800 to 980 K (1000 to 1300°F)). In addition, stainless steels have been well characterized and are well understood through the terrestrial liquid metal fast breeder reactor program, where they are used as sodium containment in a similar temperature range. Higher temperature primary heat transport piping will require the use of superalloys or refractory alloys. These have been used in ground-level demonstration and test systems with varying success. These alloys are partially characterized as discussed in Section IV.A.2. Some development, especially in the areas of fabrication, processing, and joining, is still required. However, the use of these alloys, particularly refractories, is paramount to achieving the ultimate performance of reactor electric systems.

(b) Volume Accumulators

A primary liquid metal piping loop must have a method to accommodate thermal expansion and contraction of the loop and the liquid metal inventory. This is necessary to maintain a void-free system at the appropriate pressure. The net volume change is the difference between the physical volume change of the loop and the volumetric change in the liquid metal. Volume accumulators

were developed for this purpose in the SNAP programs. The accumulator consists of a variable volume container (bellows assembly) that is designed to keep a volume of the liquid metal in compression by means of either a mechanical spring or a gas spring. The accumulator is usually located in a relatively low-temperature area and connected to the piping loop through small-diameter tubing.

More than 50 accumulators such as the one shown in Figure IV-89 were fabricated and tested for over 100,000 h at temperatures up to 670 K (750°F). Another type, shown in Figure IV-90, is a design in which system pressure is maintained by gas. This concept has a primary and secondary bellows. In addition, there are redundant gas volumes so that a single gas leak will not reduce liquid metal pressure to an unacceptable level.

b. Gas Components

(1) Gas Pumping Equipment (Fans and Compressors)

Fans, blowers, and compressors are used to circulate gases in the primary heat transport loop. Fans are used to move gases against small pressure differentials. The term fan is applied to those gas-handling pieces of equipment that do not increase the density of the gas by more than 7% as the gas passes through the device. Commercial fans are available with capacities in excess of 4000 kW and temperature limits of up to 922 K (1200°F). To date, however, specific designs for fans to be used on space power equipment have not appeared.

Compressors fall into two broad categories: displacement and rotodynamic machines. Displacement machines are available as commercial units with capacities ranging from 0.01 to 4 m³/s (20 to 9000 cfm). Reciprocating compressors, however, have not been proposed as elements of space nuclear electric systems.

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Figure IV-89. SNAP-10A Volume Accumulator

Figure IV-90. Liquid Metal Volume Accumulator

The rotodynamic type depends upon its high impeller velocity to impart sufficient momentum to the gas to effect the necessary rise in pressure. The centrifugal compressor dominates this category, with commercial sizes available from 75 to 15,000 kW, with volumes of up to $95 \text{ m}^3/\text{s}$ (200,000 acfm). For space nuclear power systems, the major application of centrifugal compressors is to provide the flow of working fluid in the gas turboelectric power conversion equipment.

The development of centrifugal compressors for use in space nuclear reactor systems has paralleled the development of Brayton-cycle gas turbines. Most such compressors share a common shaft with the gas turbine. Several of the critical components developed for these compressors would, therefore, need to be developed as part of a gas turbine development program. This is particularly true of the bearings, shafting, and housings. Other components, such as the impeller, inlet guide vanes, and diffuser, are specific to the compressor.

Five closed Brayton-cycle turboelectric subsystems were developed by the Garrett Turbine Engine Company for space applications.^{IV-53} Each contained a centrifugal compressor as an integral part of the combined turbine, alternator, and compressor rotating unit. For comparison purposes, the electrical output of these units ranged from 0.16 to 10.5 kWe. As far as the compressors in these are concerned, the basic result of these programs is that trouble-free, predictable compressors can be developed in a straightforward manner.

Larger compressors would be needed for larger Brayton-cycle equipment. Commercial compressors, however, are available for installations of virtually any size. The performance of these units also appears to be readily predictable and has achieved levels compatible with both closed and open Brayton-cycle requirements.^{IV-54} Compressors for commercial gas turbines are available on engines of over 100 MWe output.

The need for high temperature fans and/or compressors may develop to support primary cooling schemes for gas-cooled reactors. It is expected that advances in high-temperature fan/compressor technology will closely parallel the development of high-temperature turbine technology as discussed in Section IV.D.

(2) Piping and Auxiliary Equipment

The most commonly used method of transporting gases from one point to another is to force the fluid to flow through a piping system. Pipe of circular cross section is most frequently used because that shape offers not only a greater structural strength but also a greater cross-sectional area per unit of wall surface than any other shape. Pipes and tubes of virtually all common engineering alloys are commercially available in virtually any diameter required.

Refractory materials may be considered for gas piping for high temperatures. The development of these materials is expected to follow the development of high-temperature gas turbine and nuclear reactor components, as discussed in Sections IV.A and IV.D. The material specifications developed for gas turbine housings and flanges can be expected to be applicable to piping.

Refractory materials in gas piping may cause several special engineering problems. These include joining (welding) and brittle stress failure.

In addition to pipes, gas conveyance subsystems may contain various other fluid elements. These elements include orifices, valves, filters, flow straighteners, and flow splitting devices. Only a few special elements can be entirely evaluated by rational mathematical means; all others require evaluation methods that are based, at least in part, on empirical data. Primary sources of such data are contained in References IV-55 through IV-57.

In general, the primary method of preventing gas leakage is to use all-welded piping and equipment. Rigorous inspection procedures such as X ray are available to assure that welded pipe sections are sound and will not provide an opportunity for flow leakage. Refractory piping will require procedures to be redeveloped for most of the materials currently being investigated.

Valves represent a particular opportunity for gas leakage. Their application must, therefore, be given special consideration. To date, valves have been designed and tested for use in the gas management systems associated with the Brayton cycle equipment developed by Garrett for NASA. Other examples of valve development for space use can be found in the application of rocket engines to upper stage and planetary booster craft.

c. Heat Transport Working Fluids

(1) Liquid Metals

For reactors with high thermal fluxes operating at high temperatures, liquid metals have special interest as coolants. They have excellent thermal properties (e.g., high thermal conductivity, low vapor pressure), and metals of low atomic weight (e.g., lithium, sodium) also have relatively high specific heats and volumetric heat capacities. In addition, liquid metals are stable at high temperatures and in intense radiation fields. Their main disadvantage is their generally great chemical reactivity at high temperatures; they must be protected from oxidation, and suitable containing materials must be chosen to minimize corrosion. Nevertheless, liquid sodium and sodium-potassium alloys have been used successfully as coolants in both fast and thermal reactors.

The heat-transfer characteristics of several liquid metals are superior to those of water, and it is possible to achieve the same rate of heat removal with smaller areas of contact between fuel and coolant. On the other hand,

the volumetric heat capacities of liquid metals are less than that of water, and so a higher flow rate is necessary to remove the same amount of heat. The pumping power depends on several factors, and it may be greater or less than for water. For the metals of low mass number, it is generally somewhat less; but it is larger for elements of high mass number (e.g., bismuth, lead).

Table IV-23 lists the basic characteristics of the liquid metals most often considered as coolants for space nuclear reactors. Of these, NaK is often favored because of its low melting point and, therefore, simplified startup procedures. Sodium is often favored because of its large technology base from the liquid metal fast breeder reactor programs. Its use in very high-temperature systems, however, is limited by its boiling point. Lithium is currently favored for very high-temperature subsystems, primarily because of its high boiling point.

(2) Gases

On the basis of general radiation and thermal stability, ease of handling, and absence of hazardous conditions, there is much to recommend the use of a gas as coolant. Coolant passages in a gas-cooled reactor core must be larger than those for liquid metal cooled reactors. This imposes a penalty on the core fuel volume fraction that must be compensated for by the fuel enrichment and the reactor control scheme. Although the heat-transfer characteristics are not as good as those of water and liquid metals, it appears that, by suitable design, gas-cooled power reactors may be competitive with those using liquid coolants.

Hydrogen would be the best gaseous coolant, as far as both heat transfer and pumping power are concerned. Its containment, especially under pressure, would require the use of steels not subject to attack or embrittlement by hydrogen. Consequently, no proposal has been made for the use of hydrogen as coolant except in special cases, such as short-lived, semicontinuous power systems.

TABLE IV-23
PROPERTIES OF COMMONLY USED LIQUID METALS

	Lithium (natural)	Mercury	Potassium	Sodium	Na (22%) K (78%)
General physical properties					
Atomic number	3	80	19	11	
Atomic weight	6.94	200.61	39.100	22.997	
Density [kg/m^3 (lb/ft^3)]	479.0 (29.9)	13,231 (826)	714.4 (44.6)	823.3 (51.4)	741.7 (46.3)
Viscosity [$\text{MN}\cdot\text{s/m}^2$ ($\text{lb/h}\cdot\text{ft}$)]	339 (0.82)	1141 (2.76)	169 (0.41)	227 (0.55)	178 (0.43)
Surface tension [N/m (lb/ft)]	0.3472 (0.0238)	0.4482 (0.0307)	0.0834 (0.0057)	0.1471 (0.0101)	0.1138 (0.0078)
Electrical resistivity ($\mu\Omega/\text{cm}$)	36	108.3	48	29	71
Thermal properties					
Melting point [K ($^{\circ}\text{F}$)]	454 (357)	234.5 (-37.97)	337 (146)	371.2 (208.1)	262 (12)
Boiling point [K ($^{\circ}\text{F}$)]	1604 (2428)	631 (675)	1034 (1402)	1154 (1618)	1057 (1443)
Specific heat [$\text{J/g}\cdot\text{K}$ ($\text{Btu/lb}\cdot^{\circ}\text{F}$)]	1.286 (0.996)	0.0421 (0.0326)	0.235 (0.182)	0.389 (0.301)	0.270 (0.209)
Thermal conductivity [$\text{kJ/h}\cdot\text{m}\cdot\text{K}$ ($\text{Btu/h}\cdot\text{ft}\cdot^{\circ}\text{F}$)]	172 (27.6)	36.1 (5.8)	132 (21.2)	231 (37)	93.8 (15.05)
Heat of fusion [kJ/kg (Btu/lb)]	88.96 (185.9)	2.39 (5.0)	12.20 (25.5)	23.31 (48.7)	
Heat of vaporization [kJ/kg (Btu/lb)]	3995 (8349)	59.8 (125)	408.2 (853)	822.2 (1718)	
Prandtl number (c_p/κ)	0.0295	0.0154	0.0035	0.0044	0.0060
Change of volume on fusion (% solid volume)	1.5	3.6	2.41	2.5	2.5
Nuclear properties					
Fast activation cross section (mb)	0.03	89.0	0.59	0.87	0.64
Nonelastic scattering cross section at 2 MeV (barns)	<1	2	4 (3.7 MeV)	<1	NA
n,γ daughter half-life	None	48 days	12.4 h	15 h	See K, Na
n,γ daughter activity (MeV)	None	0.28	0.32, 1.51	2.775, 1.368	NA
ξ , average log energy decrement	0.2643	0.0100	0.0507	0.0852	NA
General properties					
Container material	Cb-Zr	Ferrous metals	304 SS	304 SS	304 SS
Reaction with uranium	Slight	Soluble	Slight (i)	Slight (i)	Slight
Reaction with plutonium	Slight	Soluble	Slight (i)	Slight (i)	Slight
Toxicity	Moderate	High	High	High	High
Fire or explosion hazard	High	None	High	High	High

From several points of view, helium has much to recommend it as a gaseous coolant. Although its thermal properties are not quite as good as those of hydrogen, it has a negligible cross section for neutron capture and is not hazardous. It is also stable to heat and to the action of radiation. In addition, it is inert chemically and has no harmful effects on the moderator or on structural and containing materials. For this reason, helium is the preferred coolant for graphite-moderated reactors operating at temperatures above 873 K (1100°F). High gas pressures are used to minimize pumping power and to decrease core size.

The properties of hydrogen and helium are compared in Table IV-24.

Mixtures of helium and xenon have been proposed for use in direct-cycle applications. The advantages of this mixture result from a tradeoff of the higher heat transfer rates (thermal conductivities) attainable with low-molecular-weight gases against the higher cycle efficiency attainable with the higher-molecular-weight gases. This efficiency effect results because the Reynolds number, for the rotating machinery of the sizes being considered for space power, is in the range where it considerably affects blade efficiency. The efficiency improves as the Reynolds number increases, and the Reynolds number increases as the molecular weight increases.

4. Technology Assessment

The primary heat transport technology assessment is subdivided into three main categories: (1) pumped liquid metal from reactor to power conversion subsystem, (2) gas or vapor circulated from the reactor to the power conversion heat exchanger, and (3) a heat-pipe-cooled reactor. Figure IV-79 shows these various arrangements.

TABLE IV-24
GASES: THERMAL AND NUCLEAR PROPERTIES

	Hydrogen	Helium
General physical properties		
Atomic number (or formula)	H ₂	2
Molecular weight	2.016	4.0
Density [kg/m ³ (lb/ft ³)]	0.030 (0.00185)	0.060 (0.00375)
Viscosity [N·s/m ² (lb/h·ft)]	1.74 x 10 ⁻⁵ (0.042)	3.8 x 10 ⁻⁵ (0.094)
Thermal properties		
Heat capacity [J/g·K (Btu/lb·°F)]	4.68 (3.625)	1.61 (1.245)
Thermal conductivity [kJ/h·m·K (Btu/h·ft·°F)]	1.40 (0.224)	0.99 (0.159)
Prandtl number	0.660	0.735
Nuclear properties		
σ _a , microscopic absorption cross section (barns)	0.332	0.0070
n,γ daughter half-life	None	None
n,γ daughter activity (MeV)	None	None
ξ, average log energy decrement	1.0	0.4281
General properties		
Cost (\$/100 ft ³)	0.85	1.90
Container material (24)	Carbon steel	Carbon steel
Reactions with U (43)	Reacts	None
Reactions with Pu (43)	Reacts	None
Toxicity	None	None
Fire or explosion hazard	High	None

a. Liquid Metal

The technology development for liquid metal heat transport components, including piping and auxiliary equipment, is currently at a level that a stainless steel subsystem with a reactor outlet temperature of 922 K (1200°F) is achievable in the near term. The progression of the technology is shown in Figure IV-91. A stainless steel subsystem would have an upper temperature limit of 978 K (1300°F); however, the next step in the technology progression is to superalloys, where the temperature range is 978 to 1145 K (1300 to 1600°F). A superalloy primary heat transport subsystem is achievable by 1990; the reactor outlet temperature could be 1145 K (1600°F) with sodium or lithium being used as the coolant with an electromagnetic pump.

The progression to reactor outlet temperatures above 1145 K (1600°F) will require the development of refractory metals. This step in technology could be achieved by 1994 where reactor outlet temperatures are in the range of 1420 K (2100°F). Lithium would be the coolant with an electromagnetic pump. The development of the primary heat transport subsystem accepting high coolant at temperatures in the 1145 to 1420 K (1600 to 2100°F) range is paced by materials technology.

The requirement for startup (or restart) in space presents technology hurdles that can be overcome, but they must be addressed. Since liquid metals have a positive coefficient of expansion at phase change, expansion of the liquid metal will occur as it passes from a solid to a liquid state. This expansion must somehow be accommodated in the heat transport subsystem design.

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Figure IV-91. Subsystem Technology Assessment — Liquid-Metal Primary Heat Transport Equipment

The following technology issues are associated with liquid metal heat transport systems (see Figure IV-91):

Technology Step 1

Key Issue and Resolution - No technological barriers; availability of the heat transport system by mid 1986 is paced only by lead times for materials and components.

Technology Step 2

Key Issue - Demonstration of superalloy compatibility with Li at 1145 K (1600°F) for 2-3 years.

Resolution - Design of subsystem (1 year), materials procurement and compatibility testing (2-3 years), and fabrication and qualification testing of individual components and entire subsystem (2-3 years); program completion by mid 1990.

Technology Step 3

Key Issue - Compatibility of refractory materials with lithium at temperatures of approximately 1420 K (2100°F) will require close control of coolant oxygen content.

Resolution - Development of refractory cleaning techniques and storage conditions to exclude oxides (2 years) and appropriate techniques for gettering oxygen from the Li coolant during system loading (3 years, overlapping the previous step), fabrication of components and subsystem (3-4 years), and qualification testing of components and system (3 years); program completion by mid 1994.

b. Gas

Gas primary heat transport subsystems, as mentioned in Section IV.C.3, can transfer reactor heat to a heat exchanger as in Figure IV-79b or directly to the PCS as in Figure IV-79c. The heat exchanger case requires a fan or blower to move the gas, while the PCS has the gas compressor. Otherwise, the two arrangements are very similar.

If the temperature requirements are moderate, such as 978 K (1300°F), then the HTS could be flight qualified in one to two years. If temperatures of 1145 K (1600°F) are required, then the HTS could be flight qualified by about 1986, as shown in Figure IV-92. The higher temperatures will require the use of superalloys as discussed in Section IV.A.2.f. Blowers have been made commercially at these two temperatures but would need to be developed. The typical head rise would be 5.0%.

At still higher temperatures, such as 1310 K (1900°F), refractory metals may be necessary and flight qualification would likely be delayed to about 1989, as shown in Figure IV-92. If temperatures of 1480 K (2200°F) are needed, then development will extend to 1992. In general, the needed materials are available, but development will be required.

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Figure IV-92. Subsystem Technology Assessment – Gas Primary Heat Transport Subsystem

The following technology issues are associated with inert gas heat transport systems (see Figure IV-92):

Key Issue and Resolution - Materials compatibilities are not judged to be an issue in inert gas heat transport subsystems. Availability of the subsystems is limited only by materials/components availability and qualification test cycle times.

Key Issue and Resolution - As with Steps 1 and 2 above, materials compatibilities are not seen as issues in inert gas heat transport subsystems. Availability of the subsystems is paced by materials and components availability in refractory materials and the qualification test times required or the operating temperatures involved.

c. Heat Pipes

The technology progression of heat pipes is shown in Figure IV-93. Material technology is the pacing item in the development of the pipes for high temperatures and heat rates. A 1100 K (1520°F) stainless steel heat pipe using sodium could be achievable within 4 to 6 years. The development of reactor core heat pipes that can operate in the 1500 K (2240°F) temperature regime is still 8 to 10 years in the future, and considerable development will be required. The advance to still higher temperatures of 1900 K (2960°F) appears feasible, but it would require additional development time. This heat pipe technology is about 11 to 13 years in the future.

The feasibility of a single, 8-m-long heat pipe appears questionable; however, there appear to be conceptual techniques to join the three basic segments (evaporator, adiabatic, and condenser) together to form a high-temperature heat pipe with bends. The advancement of a wick design will be required to achieve the performance of high temperature heat pipes.

Several technology issues are associated with the development of the reactor heat pipe heat transport subsystem (see Figure IV-93). These issues and their resolutions are discussed in the reactor subsystem section. Heat pipe development must precede reactor subsystem testing on the scale shown in Figure IV-93 relative to Figure IV-54. The proven heat pipes must be available approximately 1 year before the completion of the reactor subsystem assembly. This need date is consistent with those shown in Figures IV-54 and IV-93.

Figure IV-93. Subsystem Technology Assessment — Heat Pipe Primary Heat Transport Subsystem

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D. POWER CONVERSION AND PROCESSING SUBSYSTEM

1. Background

Nuclear energy in a space power reactor is converted to heat energy by the fission process. This heat energy must be converted to electrical energy by means of a power conversion subsystem (PCS). Exceptions to this are when heat energy is used directly to provide propulsion, life support, or refrigeration. Only the conversion of heat to electricity is considered in this section.

Many possible PCSs can be considered. They can be categorized as dynamic or static, depending on whether moving machinery is incorporated. Dynamic systems can be divided into rotating and reciprocating equipment. Rotating PCSs use Rankine or Brayton cycles, while reciprocating PCSs use the Stirling cycle.

Static PCSs include thermoelectric (TE) and thermionic types. Thermionic power conversion is accomplished in-core (in the nuclear reactor) or out-of-core, while TE conversion is only out-of-core.

PCSs can also be classified as steady state or semicontinuous. Steady-state subsystems would operate for several years, while semicontinuous devices would operate for only a few hours. Most available information on power conversion is for steady-state subsystems.

PCSs also differ as to whether ac or dc electric power is produced. The conversion to electric power using dynamic PCS cycles like the Brayton, Rankine, and Stirling uses an alternator. The TE and thermionic cycles directly convert heat to dc power.

Power processing as it applies to the reactor electric system has as its major function maintaining the bulk electrical power within required voltage and frequency limits (for ac). This must be accomplished over the required range of payload demand and the range of heat source and power converter degradation. Regulation is ordinarily done on static PCSs by using parasitic loads (shunt regulators) across the full power converter, by tapping off a variable fraction of the power converters, or by a combination of the two. In dynamic (alternator) PCSs, voltage and frequency regulation is accomplished by a combination of variable parasitic loads and varying the alternator parameters, where the alternator design allows.

Depending on the PCS concept and power level, longer-term power balancing may require adjusting the reactor thermal output. Compensation for shorter term payload changes are ordinarily handled by the power processing equipment. Power dissipated in parasitic loads must be radiated, adding to spacecraft mass and complicating thermal designs.

Since the techniques available for power regulation will differ for different types of PCSs, some approaches will require larger parasitic loads. The power processing needs may influence the choice of PCS for a given electrical power output rating and mission.

Depending on the interface between the space reactor electric system and the payload, the power processing unit may also include batteries, battery charge and discharge controls, voltage conversion equipment, switchgear, cabling, and the auxiliary equipment (coolers, radiators, controllers) needed to keep these components operating properly. See Figure IV-94 for a block diagram of a typical power processing unit.

As a base data point, the only space reactor electric system flown by the free world was the 500 We SNAP-10A, which used SiGe TE power conversion. This was a low-temperature (833 K (1040°F)) system. The Soviet Union has launched about 20 reactor electric systems. Many radioisotope thermoelectric generator

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Figure IV-94. Schematic of a Typical Power Processing Unit

(RTG) powered spacecraft have been successfully employed in space. These have used TE power conversion and have attained lifetimes of 10 to 15 years. These RTGs operate at peak temperatures of 800 to 1275 K (1000 to 1835°F).

Although only static power conversion has been used in space, a great deal of development has been performed on dynamic converters. Dynamic equipment may offer several advantages. It inherently has a higher device efficiency and may lead to overall power systems that are smaller and lighter. At low power levels (1 kWe), this may not be as significant as at higher power levels (>100 kWe), where dynamic systems become strong competitors to static converters.

Each type of PCS — whether dynamic or static — seems to have several advantages and several disadvantages. Although each one has been strongly advocated at one time or another, it is by no means clear that any one type of PCS is uniquely best suited to space reactor electric systems — especially given the range of powers that will be required. The "best" PCS depends on basic program requirements such as (1) power level, (2) lifetime, (3) mass, (4) reliability, (5) cost, and (6) schedule.

2. Power Conversion and Processing Considerations

Key considerations in the selection and design of PCSs are (1) efficiency, (2) mass, (3) reliability, (4) peak temperatures, (5) heat rejection temperatures, (6) working fluid properties, (7) lifetime, (8) vibration and torque, (9) power range, (10) modularity, (11) startup and shutdown, and (12) power processing. Of these, efficiency, mass, and reliability are considered to be the most important.

a. Efficiency

PCS efficiencies can range from 1.8% (as for the SNAP-10A thermoelectric converters) to over 40% (in certain dynamic converters). A better insight to efficiency can be gained by considering it in two parts: (1) the Carnot efficiency and (2) the fraction of Carnot efficiency that is actually achieved. This separates the purely thermodynamic losses from the subsystem or "device" losses. Carnot efficiency is the efficiency of an ideal heat engine and is given by

$$\eta_c = \frac{T_H - T_C}{T_H} ,$$

where

η_c = Carnot efficiency

T_H = absolute temperature at which the heat engine accepts heat

T_C = absolute temperature at which the heat engine rejects heat.

A plot of this function is given in Figure IV-95.

Device efficiency includes losses in the PCS other than the inherent Carnot loss. Such losses could be due to mechanical losses, pressure losses, electrical losses, and many other irreversible effects. The PCS efficiency can be expressed as the product of the Carnot and device efficiencies:

$$\eta_p = \eta_c \cdot \eta_D ,$$

where

η_p = PCS efficiency

η_D = device efficiency.

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Figure IV-95. Device Efficiency Is a Convenience for Comparing Types of Power Conversion Subsystems, Where

$$\eta_{\text{PCS}} = \eta_{\text{Carnot}} \cdot \eta_{\text{Device}}$$

In the sections that follow, ranges of device efficiency are presented for the different PCSs. This gives a measure of the "internal" efficiency of various PCSs without confusing the issue with the "external," purely thermodynamic, Carnot efficiency. Caution is advised in interpreting "device efficiency" as a measure of the desirability of one PCS relative to another. For example, some relatively high device efficiency subsystems such as the Brayton or Stirling may require large radiators because of their low heat rejection temperatures. This emphasizes the fact that PCS tradeoffs must be made at the system level. However, care must be taken not to obtain high efficiency by penalizing the equipment through highly stressed devices, high cycle temperatures, etc., since this could lead to lower reliability, shorter life, etc.

b. Mass

A given PCS not only has an appreciable mass in itself but also influences the mass (and size) of the system. In most cases, it is the efficiency of the PCS that most strongly affects the mass of a given system. Efficiency is often the major determinant of the masses of the reactor, shield, heat transport, and heat rejection subsystems. However, maximum PCS or system efficiency does not necessarily go hand in hand with minimum system mass.

c. Reliability

PCS reliability depends on such things as the number of moving parts, number of welds, cycle pressures, cycle temperatures, tolerances, wear modes, control design, and many others. Static PCSs may have a reliability advantage over dynamic subsystems. However, if static devices are operated at very high temperatures or have close tolerances, for example, they may lose their advantage.

Reliability also depends on the trades between redundancy and failure rate. At first look, the use of redundant components and subsystems appears proper to achieve high reliability, as shown in Figure IV-96. However, there are many potential drawbacks to redundancy, including added complexity, more mass, higher cost, unknown dependencies (among redundant pieces of equipment), common cause failures, and synergistic failures (one failure enhances another).

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Figure IV-96. Reliability versus Redundancy and Failure Rate

d. Peak Temperatures

PCS cycle temperatures are usually increased to obtain a better Carnot efficiency to a point where further increases would cause a significant drop in reliability and life. High temperatures permit selecting heat rejection temperatures that can lead to smaller heat rejection equipment. Historically, however, extracting performance by means of extreme cycle temperatures has given rise to more technical and programmatic problems than any other design approach. It may be more reasonable to initially select conservative cycle temperatures in PCSs that have the potential to evolve to higher temperatures and higher performance in later years.

e. Heat Rejection Temperatures

The heat rejection temperature of a PCS is usually selected on the basis of power system mass and radiator area. At low rejection temperatures, the Carnot efficiency is high and the benefits of high efficiency mentioned above can be obtained. High radiator temperatures, however, lead to smaller and lighter radiators because of the T^4 effect. There is usually some intermediate temperature that leads to an optimum overall power system. This temperature depends mainly on the power system requirements and the type of PCS selected. Sometimes a high heat rejection temperature appears to be optimum because the peak cycle temperatures have been forced to unreasonably high levels or the heat rejection temperature to unreasonably low levels.

f. Working Fluid Properties

Working fluids (not to be confused with heat transport fluids) are used only in dynamic systems and are fluids like Dowtherm "A," mercury, potassium, helium, xenon, and hydrogen. Organic fluids are subject to thermal decomposition and nuclear radiation effects, are temperature limited to 670 K (750°F), and have only fair heat transfer properties. Mercury is corrosive, has wetting problems, and is a mediocre lubricant. Potassium is used at high temperatures like 1420 K (2100°F), is corrosive, and consequently has strict purity requirements. Helium and xenon, being noble gases, are not corrosive and in the usual mixtures (molecular weights of 20 to 40) have reasonable heat transfer properties. These gases are typically used at a pressure of many atmospheres, and require suitable containment. Hydrogen is a candidate for the semicontinuous, open-cycle dynamic PCSs. For these short-lived applications, its potential problems of corrosion, embrittlement, and leakage can be overcome.

g. Lifetime

A long PCS lifetime is desirable since maintenance on a nuclear system in space is unlikely for the foreseeable future, and the total cost of any nuclear

powered spacecraft is bound to be high. RTGs in space are currently projected to have a total life of at least 16 years.

It is not clear whether a long PCS lifetime can be assured for systems needed in the late 1980s or early 1990s, unless such lifetime has already been demonstrated. It may be as cost effective to first require modest (1 to 2 year) lifetime goals. Early failures should be expected and accepted. Thus, modest early goals should be set if adequate growth potential exists to meet long-term goals.

h. Vibration and Torque

Vibration in the power system might interfere with antennas and sensors in the spacecraft. This problem would not exist in static PCSs and would probably not be significant in rotating PCSs, but it might be significant in reciprocating PCSs. Various solutions to this problem are possible to eliminate or dampen the vibrations.

Torques can be generated in both rotating equipment and in circulating fluids. The usual solution is to use multiple PCS equipment with symmetry to neutralize every torque with an opposing torque. This places restrictions on the PCS design.

i. Power Range

Early space reactor electric systems can be expected to have lower electric powers than later systems. In addition, requirements for semicontinuous power may evolve. Historically, power systems tend to grow by several decades in power level. Although there is no requirement that PCSs for the total foreseeable power range be generically similar, it would be of great programmatic benefit if one general type of PCS could be used for all missions for the next 20 or 30 years or so.

Power level selection tends to influence the voltage levels at which it is desirable to operate spacecraft power equipment. In the kilowatt range, previous missions have generally used 28 Vdc. In the multikilowatt range, studies — particularly on fuel cells and batteries — relating to the selection of a proper voltage have been made on a variety of cases for the range of 100 to 400 Vdc, which would correspond to 500 to 125 A for a 50-kWe system. The lower voltage leads to higher currents, larger cables, and problems with switching; the higher voltage leads to plasma losses and arcing. As the demands develop for increased power, both steady-state and semicontinuous, higher voltages must be provided to minimize the cost, mass, volume, and parasitic loss per kilowatt.

j. Modularity

A modular PCS design would have two or more PCS units per power system that could operate independently of one another. Modularity can provide part power should one PCS fail. It can also provide full power by bringing a standby PCS on line should an active PCS fail. Modularity in the PCS also makes available to the designer a greater variety of power system arrangements and power levels.

k. Startup and Shutdown

The PCS will be designed to be started in space, and in most cases there will be a positive shutdown mode. It is probable that a typical power system will not only be started and shut down several times during its lifetime but also that it will be run at various power levels. Thus, the PCS should be able to accommodate the thermal and electrical transients associated with these various operating modes. One possible requirement is that the PCS may be inactive in space for months or years before it is brought up to rated power. Many coolants and working fluids can freeze in the environment of

space, creating startup problems. For semicontinuous operation, not only will the power levels be high, but the startup and shutdown may be very rapid.

1. Power Processing

The merit of power processing equipment must be measured in terms of how well it accomplishes the following:

- 1) Maintaining bulk output voltage and frequency (if ac) within required limits for both transient and long-term changes in required power, and compensating for degradation of the reactor and PCSs in conjunction with reactor control
- 2) Requiring minimum mass and volume including auxiliary cooling equipment and radiators
- 3) Providing voltages optimized for payload requirements
- 4) Minimizing power lost in regulation and distribution
- 5) Providing power for startup and restart
- 6) Meeting reliability requirements.

The major decisions that must be made are the generation and distribution voltage levels, the type of voltage and frequency regulation, and the storage types and capacities. Other design decisions will be based on the above choices.

For static power conversion (thermoelectric, thermionic, thermophotovoltaic), the PCS minimum dc output voltage will be dictated by the lowest voltage at the payload in standard use, 28 V nominal, and the mass and power loss penalty that results from high current. The maximum voltage will be set by the potential for arcing and plasma losses. For dynamic power conversion, the same general considerations apply, with standard minimum voltages being

above 120 V. The choice of frequency will be based on a compromise among the competing goals of minimizing mass, volume, and power losses.

Perhaps the most important decision will relate to the choice of a voltage and frequency regulation concept. Requirements may include those in which the payload power varies (in extreme cases) from 10 to 100%. Also, reactor heat sources and PCSs may suffer long-term performance degradation, and some form of regulation in conjunction with automatic or ground-programmed control is required to maintain voltage and frequency within required limits. This must be accomplished for both transient load changes (load dropping) and longer term load changes.

In the past, both for reactor power sources and solar cell arrays, shunt regulators have been used for static converters. Output voltage is sensed, and the shunt current (in parallel to the load current) is caused to vary. This causes the source voltage drop to vary and maintains a constant load voltage. Before payload load reduction, the shunt current represents a parasitic heat load that must be radiated at the cost of added mass. The added mass is not significant in small systems (500 to 1000 W) but becomes important for larger systems. Thus, other approaches need to be investigated. These will include combinations of shunt regulation and automatic voltage tap selection combined with automatic or ground-controlled variation in reactor power. Voltage tap selection refers to bringing out leads from partial groups of converters such that the portion of the converter output applied to the shunt regulation can be varied stepwise in response to load change or source degradation. A smaller shunt regulator must be retained for finer control and small transients.^{IV-146} The degree to which load tapping can be applied without simultaneous reactor power adjustment will depend on the degree to which the electrical load is thermally coupled to the reactor. For the lower efficiency static PCSs, the coupling is minimal.

For dynamic PCSs, previous designs have predominantly used parasitic load to regulate frequency (or speed) under varying load demand. In this scheme, the PCS is operated at essentially constant total load. Changes in load cause the alternator frequency to deviate from a reference. These frequency deviations are sensed, and the parasitic load is varied as necessary to maintain frequency. Due to the relatively high efficiencies of dynamic converters, the mass penalties incurred by parasitic load regulation may be significant, and alternative approaches that can potentially reduce these penalties should be developed. These include other methods of dumping excess power (e.g., controlled bypass of a turbine, load-following operation, combinations of parasitic loading and limited load-following). Regulation of alternator output voltage has typically been by control of the excitation current to the alternator field windings, although other methods (described below) have been used.

Additional power processing equipment that may be part of the power processing unit are batteries, battery charge/discharge controls, voltage conversion equipment, inverters, cabling, etc. This equipment is extremely dependent on the specific application and is not considered in detail here. This equipment is also not expected to be greatly influenced by use with a reactor power source except by the possible requirement for additional radiation hardening and increased power capability.

With regard to radiation hardening, electronic components of the PPS must be designed to operate through their entire lives in a radiation environment. The susceptibility of electronic devices to radiation has been studied extensively over the last 20 to 30 years. Although much of this work has been directed to the short, high-intensity pulses from nuclear weapons, there has been a concurrent effort to model the basic mechanisms so that low-level, long-term effects can be predicted. Although electronic circuitry is composed of various active and passive components with varying methods of construction, the interest for radiation effects is normally centered on the semiconductor devices. Initially, only bipolar discrete transistors had to be addressed,

but the field has been broadened by the introduction of field effect transistors, both JFETs and MOSFETs. The widespread use of integrated circuits introduces a wide range of variables to the problem as the geometries of the different regions of the device are manipulated.

In Section IV.B, the failure thresholds for typical commonly used circuit elements is discussed. The tolerance to given radiation doses of overall subsystems using these circuit elements will depend on the success of the designer in making the subsystem circuitry tolerant to radiation.

3. Power Conversion Descriptions

a. Turboelectric

Five turboelectric PCSs have received sufficient study and development to permit their characterization as potential candidates for space power applications. These are the (1) mercury Rankine, (2) organic Rankine, (3) potassium Rankine, (4) closed-cycle Brayton, and (5) open-cycle Brayton subsystems.

(1) Mercury Rankine Power Conversion

This cycle was originally selected because its temperature-pressure characteristics closely matched the capabilities and temperature limits of the hydride reactor. The cycle provides a reasonably high subsystem efficiency coupled with a small radiator area requirement.

In the power cycle of the basic mercury Rankine (MR) subsystem, a reactor is used to heat NaK-78 fluid recirculating in a sealed loop. The hot NaK is circulated through a boiler to evaporate and superheat mercury. The Hg vapor is then expanded within a turbine which drives an alternator and a Hg boiler feed pump. The alternator generates ac electrical power. The exhaust Hg vapor from the turbine is condensed, subcooled, and then pumped back to the boiler by means of a high-pressure boiler feed pump.

Significant features of the MR subsystem include:

- 1) A high heat-rejection temperature, which favors small radiator area and low mass.
- 2) Reasonable simplicity with a minimum of operating auxiliaries.
- 3) A reasonably high turbine inlet temperature that yields good cycle efficiencies without being so high as to incur serious reliability penalties. (40% of Carnot efficiency appears to be achievable.)
- 4) A simple single-shaft rotating machine concept yielding high reliability.

From 1960 to 1970, there were two programs to develop mercury Rankine PCSs for space power use. One of these was the 3- to 5-kWe SNAP-2, the other was the 35-kWe SNAP-8 (later studied for powers up to 120 kWe). These PCSs are described in Refs. IV-56 through IV-62.

(a) SNAP-2 Mercury Rankine

The basic SNAP-2 subsystem schematic is shown in Figure IV-97. The heart of the SNAP-2 PCS was the combined rotating unit (CRU), which consisted of the alternator, turbine, and mercury pump, mounted on a single shaft supported by mercury-lubricated bearings. The CRU turboalternator was designed to meet the following criteria:

- 1) Net power output of 3 to 5 kWe

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Figure IV-97. SNAP-2 Mercury Rankine PCS

- 2) Isothermal, hermetically sealed housing
- 3) Hg-lubricated bearings
- 4) Permanent-magnet (PM) alternator
- 5) Single-shaft assembly
- 6) Vapor cooling of alternator.

In the SNAP-2 design, mercury vapor entering at 895 K (1150°F) and 0.79 MPa (115 psia), expands through the two-stage axial-flow turbine that drives the 3-phase, 6-pole PM alternator. The shaft of the unit rotates at 36,000 rpm. The Hg vapor exhausts to a pressure of 0.062 MPa (9 psia) and cools the alternator as it flows through the unit and over the finned stator housing. The shaft is supported by two Hg-lubricated journal bearings for which lubricant is supplied by the onshaft centrifugal Hg pump. The enclosure housing is hermetically sealed, as is the alternator stator. The frequency of the alternator output is regulated using a parasitic load speed controller. A block diagram of the SNAP-2 speed controller, which was implemented using magnetic amplifiers, is presented in Figure IV-98. The speed controller is designed to present a variable-controlled load in parallel with the payload so as to stabilize the PCS for various loads with a speed drop characteristic. As the alternator is excited by a permanent magnet rotor, active control of field excitation is not necessary.

The design point and electrical-output characteristics of the combined rotating unit (CRU) are shown in Table IV-25. This unit was designated CRU-V and represents a design in which there was significant testing. The design prior to CRU-V was designated the CRU-IVM. Table IV-26 lists the chief CRU test results.

Tests on the CRU showed several problems. Table IV-27 lists the problems found in CRU-V and the corrective action taken.

Figure IV-98. SNAP-2 Mercury Rankine PCS Control Block Diagram

TABLE IV-25
SNAP-2 CRU-V DESIGN POINT AND ELECTRICAL CHARACTERISTICS

Net power output (kWe)	5
Voltage	110
Frequency (Hz)	1800
Turbine inlet pressure [MPa (psia)]	0.79 (115)
Turbine inlet temperature [K (°F)]	895 (1150)
Turbine back pressure [MPa (psia)]	0.062 (9)
Turbine flow rate [kg/s (lb/min)]	0.152 (20)
RC exit temperature [K (°F)]	424 (303)

Two turbine configurations with different aerodynamic designs were used in CRU testing. One was designed for optimum power output and one for a lower power range. Because they were essentially interchangeable, the CRU could be tailored to a broad range of power applications. CRU testing verified a turbine efficiency in the range of 50 to 75% for a gross power output of 2 to 6 kWe.

The CRU Hg pump performed satisfactorily in both the CRU-IVM and CRU-V. A jet inducer was used to provide the pump with sufficient NPSH to minimize cavitation problems. The Hg pump was able to produce a discharge pressure of approximately 3.45 MPa (500-psia) at flow rates up to approximately 0.45 kg/s (60 lb/min). The pump efficiency determined in component tests was approximately 36%. With the inducer, the pump was able to operate at low Hg inlet pressure.

Many thousands of hours of Hg boiler testing were accumulated. The majority of this test time was obtained on prototype boilers. In most of these tests, boiler data acquisition, although not the main reason for the test, was a valuable byproduct of the operation of CRUs or complete subsystems. Other, more specific tests were made in an attempt to understand more fully the detailed boiling-heat-transfer characteristics and anomalies in performance observed during some of the boiler tests.

TABLE IV-26
SUMMARY OF MAJOR SNAP-2 CRU-IVM AND -V TEST RESULTS

	CRU-IVM	CRU-V
Total test time (h)	8,680	21,196
Longest test time — one unit (h)	6,637	7,183
Longest continuous operation (h)	1,060	4,186
Number units tested 2500 h or more	1	3
Number normal injection startups	218	98
Most startups — one unit	77	37
Number shock-vibration tests	0	2
Total accumulated test time (h)		29,876

TABLE IV-27
SNAP-2 CRU-V SUMMARY OF DESIGN CORRECTIONS
AND/OR IMPROVEMENTS

Deficiency	Correction
Interference between bearing mount and turbine housing	Revised assembly dimensions
First-stage turbine wheel fracture	Revised mounting configuration
Alternator stator short	Evacuated stator operation introduced
Strap failure	Redesigned strap footings
Thermal distortion of insulation	Changed insulation material and revised assembly design
Alternator bearing seal rub	Enlarged running clearance
Chevron seal fracture	Changed from an interim fin to O-rings
Lube O-ring seal leakage	Incorporated new bayonet-sealed feed lines
Scored thrust bearing	Changed design to hydrostatic-type bearing
Foreign particles	Filters installed at inlet ports and adjusted some running clearances

The boiler served to transfer the heat generated in the reactor to the Hg working fluid. This heat was transported from the reactor to the boiler by means of a NaK primary forced-circulation loop. The Hg leaving the Hg pump had to be preheated, vaporized, and superheated before entering the turbine. This was most easily achievable in a once-through boiler rather than in three separate components. The boiler used a counterflow design to avoid the complexity of having more than one Hg pass.

Based on the operation of all boilers, both prototype and experimental, the following conclusions were reached:

- 1) Compact, low mass NaK-to-Hg boilers are applicable to MR systems.
- 2) By far the most critical requirement for conditioned boiler performance is surface wetting by the Hg.
- 3) Loss of nucleation sites, as a mechanism inducing slug flow, is not the primary cause of deconditioning.
- 4) Although heat transfer improves slightly by providing flow breakup (vortex flow), this mechanism alone, without surface wetting, does not provide conditioned operation.
- 5) Oil contamination results in deconditioned (nonwetting) boiler performance and was the most probable cause of all prolonged deconditioning problems encountered in the program.
- 6) Final boiler designs were based on the empirical dropwise boiling equations obtained early in the program. With a deconditioned or partially deconditioned operation, the heat transfer was significantly poorer than predicted by the design equations.
- 7) Turbine performance under partially conditioned boiler operation was generally adequate.

(b) SNAP-8 Mercury Rankine

In its original concept, the SNAP-8 power PCS was a two-loop Rankine--cycle providing 30 kWe of 1000 Hz electrical power for 10,000 h when supplied with 300 kW of thermal power from a nuclear reactor heat source. Two PCSs were to be used for a total net electrical output of 60 kW. A basic change in subsystem concept from the original two-loop unit to a multiple-loop unit, incorporating a compact mercury condenser and liquid metal heat-rejection loop, was accompanied by a change to 35 kW of 400-Hz electrical power. (See Figure IV-99.) As the figure shows, an organic loop is included to provide cooling and bearing lubrication. Table IV-28 lists the requirements for the 35-kWe SNAP-8 subsystem.

The SNAP-8 35-kWe subsystem must start and shut down automatically in gravitational fields ranging from zero gravity to 1 g under differing initial conditions. The subsystem must undergo an initial startup when the fluids and components are cold. During this startup, the entire mercury inventory is injected into the Rankine-cycle loop. Two types of shutdown are required. The first, normal shutdown, is a gradual, controlled shutdown sequence required by predetermined test planning or in the event of minor operating difficulties. The second, emergency shutdown, is a rapid, controlled shutdown sequence resulting from indicated potential major operating difficulties.

An endurance test was conducted on a single set of components. The objective was to have a specific PCS operate as long as possible without replacing any component. This approach allowed observing any life-related system deficiencies or unexpected operating characteristics. The test met and exceeded expectations. Originally, the objective was to operate 2500 h with the single set of components. The endurance test time was extended such that

TABLE IV-28
MAJOR REQUIREMENTS — SNAP-8 MERCURY RANKINE POWER CONVERSION SUBSYSTEM

Net electrical output (kWe min)	35
Voltage (rms, line to line) (Vac)	208
Frequency (Hz)	400
Operating life (continuous) (h)	10,000
Input power (kWt max)	500
Input temperature (steady-state range) [K (°F)]	965 to 995 (1280 to 1330)
Environment:	
Gravitational field (operating) (g)	0 to 1
Radiation	
PCS (integrated dose for 10^4 h)	
Fast neutrons (0.1 MeV or greater) (nvt)	5×10^{12}
Gamma rays (rads)	5×10^7
Solid-state electronics (integrated dose for 10^4 h)	
Fast neutrons (0.1 MeV or greater) (nvt)	10^{11}
Gamma rays (rads)	10^6
Acceleration (system not operating)	
Longitudinal axis (g)	6
Transverse axis (g)	2
Restart capability	
Number of automatic restarts without servicing	20
Gravitational field (g)	0 to 1

Figure IV-99. SNAP-8 Mercury Rankine PCS

there was a minimum of 7300 h on each component. The primary reason for terminating the test was to inspect the turbine, which had accumulated more than 10,000 test hours. The operating times for each of the components at the time of test termination are as follows.

Turbine-alternator	10,800 h
Primary loop NaK pump	7,600
Heat rejection loop NaK pump	7,900
Mercury pump	7,300
Lubricant-coolant pump	12,700
Boiler	8,700
Condenser	13,200
Electrical controls	11,300

Endurance testing provided a quantitative measure of the reliability of the SNAP-8 PCS. The components and system as a whole demonstrated a clear potential for at least a 10,000-h life. Throughout endurance testing, the subsystem was stable and gave no evidence of any life-limiting characteristics. The long-term transients that were observed were within the range for which compensation and allowance had been provided.

Testing generated performance data for all of the components in a subsystem configuration. Specific phenomena observed were:

- 1) Quantity of unvaporized liquid droplets in the mercury vapor stream
- 2) Effects of deconditioned boiler performance
- 3) Pressure fluctuations generated by the boiler and the effect on subsystem output
- 4) Effects of partial loss of mercury inventory
- 5) Pressure transients generated within the boiler during startup
- 6) Degradation of turbine efficiency
- 7) Effects of mass-transfer buildups on turbine performance
- 8) Effects on the system of various types of component failures
- 9) Effects of the presence of noncondensable gas in the subsystem
- 10) Condenser pressure instabilities
- 11) Condenser choked flow phenomenon
- 12) Effects of rapid electrical load variations
- 13) Speed control subsystem perturbations
- 14) Effects of mass transfer and gas accumulations on pump performance

The turbine configuration used in the CRU was a four-stage impulse type. There was dual-path partial admission in the first two stages and full admission in the last two. The turbine was overhung (cantilevered) supported on two angular-contact ball bearings. The design operating parameters for the turbine are as follows:

Inlet temperature [K (°F)]	950 (1250)
Inlet pressure [MPa (psia)]	1.9×10^7 (249)
Mercury vapor flow [kg/s (lb/h)]	1.72 (11,800)
Exit pressure [MPa (psia)]	0.10 (14.5)
Speed (rpm)	12,000

The alternator was developed to meet the SNAP-8 PCS requirements, which included the ability to start and generate 60 kW of electrical power unattended in space for 10,000 h. The alternator configuration used is a solid rotor, brushless, homopolar inductor type operating at 12,000 rpm and producing 120/208-V, 3-phase ac power, with an output frequency of 400 Hz. The alternator is lubricated and cooled by a polyphenyl ether organic fluid. The reliability goal was 97.35% for 10,000 h of continuous operation.

The mercury pump circulates mercury through the loop and acts as the boiler feed pump. It is a single-stage centrifugal pump with a jet pump driven by bypass mercury flow. Suction conditions for the overall pump are established by the condenser pressure and by the degree of subcooling provided by the condenser. The centrifugal pump is driven by an electric motor at a speed of approximately 7800 rpm. The pump shaft is supported by ball bearings. The bearings are lubricated and cooled by an organic polyphenyl ether. The same fluid is used to cool the motor and space seal.

In the SNAP-8 subsystem, the NaK-to-mercury heat exchanger, or boiler, is the interface between the heating fluid (NaK) and the working fluid (Hg). During the evolution of the SNAP-8 boilers, three major design improvements separated the initial design from the final boiler concept. The first improvement was a change from a tube-in-shell, cross-counterflow design to a tube-in-tube with pure counterflow. As a result, heat transfer improved with the increased NaK and mercury flow. The change also provided accurate diagnostic data in terms of boiler tube length. Along with the change from tube-in-shell to tube-in-tube was an increase in the number of tubes from four to seven, which decreased the mercury flow area. The second major change was in

the mercury containment material from 9 Cr-1 Mo steel (9M) and Haynes 25 steel to the refractory metal tantalum.

The SNAP-8 PCS utilizes an organic fluid to lubricate bearings and to cool subsystem components. This fluid, a radiation resistant polyphenyl ether, is circulated through the system by a centrifugal pump driven by an electric motor. The entire pump-motor unit is hermetically sealed. The motor is lubricated and cooled by a small flow of the pumped fluid through the motor and down the hollow shaft, which returns the flow to the inlet of the main pump.

The SNAP-8 condenser is a counterflow, tube-in-shell heat exchanger with the condensing fluid flowing through the shell. The mercury inlet and outlet are concentric and parallel to the tube bundle, while the NaK entry and exit flow directions are perpendicular to the tube bundle. There are 73 tapered mercury-condensing tubes rolled and welded to the fixed headers in a triangular pitch array.

The mercury flow control valve is located in the Rankine-cycle loop between the mercury pump discharge and boiler inlet. This valve meters the flow of mercury to the boiler during system startup and shutdown cycles. During startup and shutdown, flow is metered to the boiler according to prescribed rates of increase or decrease. The rate can be varied by changing the voltage supplied to the valve drive motor.

The SNAP-8 electrical controller provides the control and regulating functions necessary for startup, steady-state operation, and shutdown of the subsystem. These functions include:

- Regulation of the turbine-alternator speed to provide constant output power frequency to the load
- Regulation of the alternator output voltage
- Control of the startup and shutdown sequencing

- Providing startup and shutdown power
- Protection of the power subsystem and connected load

The components that performed these functions are shown in Figure IV-100. The major control components are listed below:

- 1) Speed Controller — This includes the speed control module, saturable reactor, and parasitic load resistor. The speed of the turbine-alternator is regulated by maintaining a load on the PCS equal to the subsystem load capability. This is accomplished as follows: the speed control module senses alternator frequency and controls power flow through the saturable reactor to the parasitic load resistor to minimize the error between the sensed frequency and the reference frequency. Thus, for constant speed operation, the parasitic load resistor is required to absorb any power in excess of the load requirements.
- 2) Voltage Regulator-Exciter — The alternator output voltage was controlled by manipulating the excitation current to the alternator field. The voltage regulator/exciter supplies excitation to the alternator as required to maintain output within the specified tolerance.
- 3) Programmer — This controls all functions of the PCS, including startup and shutdown sequencing, and provides fault protection in conjunction with the protective equipment.
- 4) Protective Equipment — The protective equipment detects internal and external faults in the electrical equipment and initiates the necessary corrective action.
- 5) Inverter — This provides variable-frequency, variable-voltage power to operate the pump motors during startup and shutdown.
- 6) DC Power Supply — This provides power for startup and shutdown of the PCS and includes means of recharging and maintaining the batteries during PCS operation.

Figure IV-100. Electrical Control System Block Diagram — SNAP-8 Mercury Rankine PCS

- 7) Vehicle Load Breaker — This connects the vehicle load to the PCS.
- 8) Motor Transfer Contactors — These transfer power for the pump motors between inverter power during startup and shutdown and alternator power during steady-state operation.

To improve alternator efficiency, reliability, and power output, capacitors using polyimide film dielectric were employed for power factor correction.

(2) Organic Rankine Power Conversion

In the 1960s and early 1970s, several companies performed R&D on organic Rankine PCSs for nuclear space power, including Aerojet-General, Rocketdyne, Sundstrand, and TRW. Of these, only Sundstrand has remained active in the field. In addition, about six companies are pursuing the organic Rankine cycle for for terrestrial power, including Barber-Nichols Company; EG&G Idaho, Inc.; Mechanical Technology, Inc.; Sundstrand; Thermo Electron Corporation; and Whiting Corporation. IV-63

A basic reason for interest in organic Rankine power conversion technology is the potential of these devices to produce power with very high device efficiency from low-temperature heat sources. Such sources provide low Carnot efficiency but when combined with a high device efficiency can produce a respectable overall efficiency. Organic Rankine device efficiency can be expected to reach values on the order of 50% with full development. It is this high efficiency that can permit subsystems with source temperatures of 644 K (700°F) to achieve overall efficiencies on the order of 15 to 20%.

The bulk of the space nuclear technology appears to reside at Sundstrand. Originally, Sundstrand worked on organic Rankine PCSs adapted to reactor power such as their 6-kWe development program, their 66-kWe study, and 10-, 25-, 50-, and 75-kWe studies. IV-64 through IV-66

(a) 6-kWe Organic Rankine

The Sundstrand 6-kWe organic Rankine cycle (ORC) subsystem was operated with both a surface condenser and a jet condenser at 617 to 645 K (650 to 700°F) turbine inlet temperatures. Performance measurements indicate that both subsystems would produce an overall efficiency of over 17%.

The 645 K (700°F) turbine inlet/jet condenser version was operated later in the program for over 3000 h. All rotating equipment was in excellent condition after the endurance test. The bearings actually operated 3170 h with a total of 152 start/stop cycles without damage or significant wear.

The CRU was a turbo-alternator and pump mounted on a single shaft with working-fluid-lubricated bearings. The objective of the effort was to design, build, and test a CRU that, with only very minor changes to the unit, would be able to produce between 2 and 10 kWe of useful power. Using an existing turbine wheel forging and existing turbine blade tooling, a wheel was designed with the capability of producing 6 kWe at a speed of 24,000 rpm. The turbine inlet temperature was a conservative 617 K (650°F), and the back pressure was 3450 Pa (0.5 psi). The turbine diameter was 18 cm (7.1 in.). The turbine had 135 impulse blades and a design point efficiency of 72%. The alternator pole diameter was 9.4 cm (3.7 in.). The poles for the eddy current brake, which was used for speed control, were also 9.4 cm (3.7 in.) in diameter. There were six poles on the alternator and six on the eddy current brake. The alternator design efficiency was 91.9%. The stators were cooled by passing the Dowtherm flow over the back iron before the flow was delivered to the regenerator inlet.

The bearings utilized working-fluid-lubricated tilting pads. Each radial bearing had three tilting pads; the thrust loads were accommodated by two pairs of gimbal-mounted tilting pads. The turbine-driven feed pump impeller was typical of a low specific speed pump.

The regenerator module was a low-mass, compact unit. It had a plate fin design with one vapor pass and eight crossflow liquid passes arranged in an overall counterflow configuration. The module had a mass of 5.7 kg (12.5 lb) dry, had a design effectiveness of 0.9 with half the subsystem flow, and had a pressure drop of 350 Pa (0.05 psi) on the vapor side. Two such units were required for the 6-kWe power level.

The vaporizer consisted of a tube-and-shell heat exchanger in which the tube-side fluid was the heated Dowtherm fluid and the shell-side was high-pressure Dowtherm vapor that condensed on the boiler tube. The hot, high-pressure Dowtherm vapor was supplied by a commercial gas-fired vaporizer. The boiler tube consisted of tubing arranged in six passes inside a standard-size steel pipe.

In the jet condenser, subcooled liquid Dowtherm was injected through a central liquid nozzle coaxially with the vapor coming from the regenerator. The vapor condensed on the liquid jets, and a portion of the momentum of the injected liquid was recovered in the diffuser section of the jet condenser. The jet condenser was capable of stable operation (no floodout) with subcooled liquid outlet pressures up to 62% of the injector inlet liquid pressure. The pump used for operation with the jet condenser was significantly oversized for both the 617 K (650°F) and 645 K (700°F) turbine inlet jet condenser operation and was used due to its availability. This fact is important when assessing the measured performance of the subsystem with a jet condenser.

The complete PCS was run for a total of 3003 h, and the longest continuous run was 367 h. During the last 500 h, it was operating virtually continuously.

(b) Dynamic Isotope Power System Organic Rankine

Sundstrand has performed significant development on an organic Rankine PCS specifically for use in space with an isotope heat source. This was called the dynamic isotope power system (DIPS). DIPS was only a 1.3-kWe system, although many of its features closely resembled those of higher powered reactor systems. Power conversion data for DIPS are reported in Refs. IV-67 and IV-68.

Figure IV-101 shows the DIPS PCS arrangement, the temperature/entropy diagram, and the eight-cycle state points. The key features of the design were an axial impulse turbine, homopolar inductor alternator, centrifugal pump, working fluid (Dowtherm A) lubricated film bearings, once-through boiler, jet condenser, and liquid working fluid pumped radiator. The baseline PCS was designed for 1.3 kWe, 28 Vdc; had a mass of less than 210 kg (450 lb); and required 10.8 m^2 (116 ft^2) of radiator area. Figure IV-102 shows a pictorial cutaway view of the PCS.

Referring to Figure IV-101, the liquid outlet from the pump was split into two loops. The power loop passed liquid through the alternator (for cooling) and then to the regenerator to extract energy from the turbine exhaust vapor. Flow from the regenerator then passed to the boiler, where heat was transferred to the working fluid. The vaporized fluid then drove the turbo-alternator pump. The cooling loop from the pump passed liquid flow through the radiator to reject waste heat and then to the jet condenser to condense the turbine exhaust vapor emanating from the regenerator. The combined liquid was returned through the accumulator to the pump.

For the DIPS application, Dowtherm A was selected after extensive trade-offs with other organic fluids. Dowtherm A exhibits the characteristic positive slope of the saturated vapor line, allowing 100% quality vapor expansion

	PRESSURE		TEMPERATURE	
	MPa	psia	K	(°F)
①	0.665	96	373.8	213
②	0.596	86	348.2	167
③	0.506	73	380.1	225
④	0.475	69	520.1	477
⑤	0.393	57	630.2	675
⑥	0.0009	0.137	558.7	546
⑦	0.0007	0.105	389.1	241
⑧	0.218	31	373.4	213

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Figure IV-101. DIPS Organic Rankine PCS

9467-93

Figure IV-102. Cutaway View of DIPS
Organic Rankine PCS

through the turbine and the use of a regenerator for increased subsystem device efficiency.

The CRU was integrated with the regenerator and jet condenser to minimize volume and mass. The complete package was supported at three points located in the regenerator exit plane. All housings were made from stainless steel, and all support struts were made from aluminum alloy.

The Rankine-cycle PCS had a flow control device that maintained turbine inlet conditions at the selected state point to ensure that the turbine would always be supplied with superheated vapor. In the orbital operating mode, speed was controlled by a speed control parasitic load bank, and the flow control valve only corrected for minor turbine inlet temperature fluctuations and isotope heat source decay.

Changes in radiator environment during the mission could have caused variations in the heat rejection capability. Average liquid temperature in the radiator could be as much as 22 K (40°F) cooler with a cold sink environment than with a hot sink environment. This subcooling of the working fluid would reduce the power output capacity because the turbine temperature flow control valve would decrease the flow slightly to compensate for the lower temperature fluid entering the boiler. Heat rejection control was required to assure that the power subsystem could deliver rated load regardless of the radiator environment. This was accomplished by the controlled bypassing of radiator inlet fluid.

Independent control of alternator speed and PCS output voltage were provided by a solid-state controller. Figure IV-103 shows a block diagram of the controller. Its circuitry fell into three functional groups: the speed regulator, voltage regulator, and power conditioner. The speed regulator controlled the alternator field current to provide constant torque on the shaft and constant power output. Since the output voltage was regulated to 28 V,

9467-195

Figure IV-103. DIPS Organic Rankine PCS Controller Block Diagram

the generator acted essentially as a current source, and the current was divided between the payload and the parasitic load resistors. The voltage regulator maintained a constant 28 Vdc by shunting the appropriate current to the parasitic load resistors. Using this scheme, the payload could be changed from 0 to 100% of PCS capacity with resulting speed changes of less than 60 rpm. A full-wave bridge circuit was used for rectifying the alternator output. An L-C filter was used to provide the required 0.1% voltage ripple in the output power.

The regenerator was composed of an all-aluminum finned tube wound spirally around the electrical generator housing. This configuration minimized the length, volume, and mass of the power generating subsystem, and simplified the vapor flow passage while keeping the axis of rotation collinear with the jet condenser axis. As the vapor left the turbine diffuser, it underwent a sudden expansion at the entrance of the regenerator. The vapor then flowed over the outside surface of the finned tube, giving up the superheat to the internally flowing liquid, before entering the annular transition duct, which accelerated the flow to the jet condenser.

The Dowtherm A organic working fluid underwent a slight pyrolytic decomposition under long-term thermal cycling at high temperatures, forming gases and high-molecular-weight substances (liquid droplets) at the boiler outlet conditions. In addition, other subsystem components might evolve gases at a low rate during operation. To assure the reliable performance of the subsystem during the mission in the presence of these factors, a noncondensable gas separator was used to remove the gaseous degradation products, and a centrifugal separator was used to remove the high-molecularweight portion of the degradation products and to remove liquid carryover from the boiler.

The jet condenser consisted of three basic components: the vapor funnel, the liquid injector, and the diffuser. Low-pressure vapor was ducted into a funnel coaxially with liquid jets of subcooled working fluid. The liquid jets were injected into the vapor at approximately 30 m/s (100 ft/s) and aimed at

the throat of the vapor funnel. The vapor condensed on the subcooled liquid jets, and the combined liquid jet passed through a throat and into a diffuser, where it underwent a sudden expansion to fill the entire cross section. In the sudden expansion, there was a loss of total pressure but a gain in the static pressure of the liquid stream. The behavior of the liquid jet in the diffuser was identical to that which occurred in the diffuser of a cavitating venturi. Testing showed that a 50% recovery was achievable by the DIPS condenser.

An organic Rankine ground demonstration system (GDS) was built and tested. The PCS in this test program was run under conditions close to those of space operation. To minimize air in-leakage and to simulate heat losses, the GDS was operated in a vacuum chamber. In Phase I of this test, over 1100 h of operation was achieved, with more than 1600 h on the components. An efficiency of 15.1% was realized in Phase I, whereas 18.1% had been the goal.

Phase II, a technology verification phase, was initiated to improve the performance of the PCS. In Phase II, the PCS efficiency was increased from 15.1% to 16.6%, or about half way to the goal. Table IV-29 lists the component performance achieved at the end of the Phase II. Table IV-30 lists the PCS performance achieved.

Several improvements have been proposed to advance DIPS performance beyond Phase II. Table IV-31 lists the components where improvements could be made, the nature of the improvement, and the effect on PCS efficiency.

Table IV-32 lists the test hours accumulated in the GDS program. A total of 11,146 h of testing was accumulated in the GDS program. The endurance test had been scheduled to run 5,000 h but was intentionally terminated at 2,002 h. Before the endurance test, after 4,298 h of GDS operation, the unit was torn

TABLE IV-29
DIPS ORGANIC RANKINE SUBSYSTEM COMPONENT PERFORMANCE

Turbine efficiency (%)	68
Regenerator effectiveness (%)	97
Radiator pressure drop [MPa (psia)]	0.069 (10)
Pump efficiency (%)	71
Alternator efficiency (%)	93.5
Controller efficiency (%)	90.6
Bearing & seal loss (W)	95
Jet condenser vapor pressure [MPa (psia)]	8.7×10^{-4} (0.126)
Jet condenser inlet temperature [K (°F)]	343 (157)
Rotational speed (rpm)	34,610

TABLE IV-30
DIPS ORGANIC RANKINE SUBSYSTEM PERFORMANCE

Turbine inlet temperature [K (°F)]	645 (700)
Electrical input power (W)	7,515
dc output power (W)	1,239
dc efficiency (%)	16.56
ac output power (W)	1,387
ac efficiency (%)	18.5
Turbine flow rate [kg/s (lb/h)]	0.014 (111.6)
Rotational speed (rpm)	34,610

down for inspection. All of the hardware was found to be in excellent condition except the radial bearing, which was slightly burnished. It was concluded that the PCS was inherently capable of 60,000 h of operation.

During and following the 2,002-h endurance run, several problems were noticed. One problem was a series of jet condenser floodouts. When the causes of the floodouts had been removed, a problem of noncondensable gas formation arose, leading to a gradual increase in the low-pressure regions of the cycle. Reworking the noncondensable gas separator led to stable pressures, indicating satisfactory removal of these gases.

TABLE IV-31
DIPS ORGANIC RANKINE COMPONENT PERFORMANCE IMPROVEMENT POTENTIAL

Component	Current Performance	Projected Performance	Subsystem Improvement	Improvement Method
Turbine efficiency (%)	68	69	0.23	Blade height selection
Alternator efficiency (%)	93.5	95	0.27	Samarium cobalt design
Controller efficiency (%)	90.6	96.3	1.14	Reduced quality output Higher voltage
Turbine inlet temperature [K (°F)]	645 (700)	660 (725)	0.36	Further fluid testing required

TABLE IV-32
DIPS ORGANIC RANKINE POWER CONVERSION SUBSYSTEM OPERATIONAL HOURS SUMMARY

Phase	Build	Period	Hours
I	Development unit	December 1977 - March 1978	540
I	Mod 1	April 1978 - August 1978	<u>1,124</u>
		Subtotal	1,664
II	Build A	September 1978 - April 1979	3,182
II	Build B	May 1979 - May 1980	4,298
II	Endurance	August 1980 - December 1980	<u>2,002</u>
		Subtotal	<u>9,482</u>
		Total	11,146

(3) Potassium Rankine Power Conversion

In the 1960s, a large amount of work was performed on Rankine PCSs for space power that used potassium, cesium, or rubidium for the working fluid. Ref. IV-69 lists 12 companies that did significant R&D work. By 1970, the level of work had been greatly reduced. In terms of high-temperature test hours, the largest contributors to boiling potassium research and development were AiResearch, General Electric, ORNL, and Pratt & Whitney. Much less work was done on cesium and rubidium fluids.

Table IV-33 lists these organizations and summarizes the hours of operation on various systems and components for both potassium and cesium. Subtotals for the boiling corrosion subsystems, component test boiling subsystems, simulated Rankine cycle power plant subsystems, and the total operating time for all boiling subsystems are presented. The dimensions of all the components in the simulated subsystems were proportioned to provide operating characteristics and responses similar to those expected in the operation of a Rankine cycle space subsystem, and this is the chief distinguishing feature between the simulated and the component test boiling subsystems. The total operating time for each component operated in these subsystems is also provided. Note that the total amount of operating experience at each installation is smaller than the sum of the operating time concurrently for all of the components because two or more components were operated in each subsystem. Similar information is provided for the all-liquid, bimetallic corrosion subsystems operated at Allison and United Nuclear. A considerable amount of reflux capsule corrosion test operation data not presented in Table IV-33 has also been accumulated.

TABLE IV-33

SUMMARY OF OPERATING TIME WITH POTASSIUM AND CESIUM POWER
CONVERSION SUBSYSTEMS AND COMPONENTS AT TEMPERATURES ABOVE 810 K (1000°F)
(h)

	Potassium Systems										Cesium Systems		
	AGN ^a	AiResearch ^b	Allison ^c	GE ^d	JPL ^e	NASA ^f	ORNL ^g	P&WA ^h	Rocketdyne ⁱ	UNC ^j	BNL ^k	AGN ^l	HA ^m
Corrosion Test Systems													
Boiling systems													
Thermal convection	--	--	--	10,500	--	--	43,600	12,000	--	--	--	--	420
Forced convection													
One loop	--	1,300	--	--	--	--	14,200	--	--	--	1,100	2,000	--
Two loops	--	--	--	5,000	--	--	5,000	--	--	--	--	--	--
Subtotal	--	1,300	--	15,500	--	--	62,800	12,000	--	--	1,100	2,000	420
All liquid systems													
Thermal convection	--	--	--	--	--	--	--	--	--	100,000	--	--	--
Forced convection	--	--	53,000	--	--	--	--	--	--	3,000	--	--	--
Component Test Systems (Boiling)													
One loop	100	5,000	--	11,900	--	--	2,800	3,600	200	--	--	2,500	--
Two loops	--	900	--	7,700	--	400	--	1,300	--	--	--	--	--
Three loops	--	--	--	--	--	600	--	--	--	--	--	--	--
Subtotal	100	5,900	--	19,600	--	1,000	2,800	4,900	200	--	--	2,500	--
Simulated Power Plant Systems													
One loop	--	--	--	--	--	--	10,200	--	--	--	--	--	--
Two loops	--	--	--	--	1,000	--	--	--	--	--	--	--	--
Subtotal	--	--	--	--	1,000	--	10,200	--	--	--	--	--	--
Total for all boiling systems	100	7,200	--	35,100	1,000	1,000	75,800	16,900	200	--	1,100	4,500	420
Component - Power Plant													
Boilers													
<35 kW	--	1,300	--	15,500	1,000	--	71,400	16,900	--	--	--	--	--
>35 kW	100	5,900	--	19,600	--	1,000	4,400	--	200	--	1,100	4,500	420
Subtotal	100	7,200	--	35,100	1,000	1,000	75,800	16,900	200	--	1,100	4,500	420
Turbines													
With K lubricating bearings, <10 kW	--	3,000 ⁿ	--	--	--	--	5,000 ⁿ	--	100	--	--	--	--
With K lubricating bearings, >10 kW	--	50 ⁿ	--	--	--	--	--	--	--	--	--	--	--
With oil lubricating bearings, >10 kW	--	--	--	5,100 ⁿ	--	--	--	--	--	--	--	--	--
Boiler feed pumps													
Electromagnetic	100	7,200	--	24,600	3,600	1,000	26,900	4,900	200	3,000	1,100	4,500	--
Turbine-driven centrifugal	--	--	--	--	--	--	5,000 ⁿ	--	--	--	--	--	--
Radiator	--	--	--	--	--	600	--	--	--	--	--	--	--
Condenser - liquid metal loop	--	--	--	2,500	--	600	5,000	--	--	--	--	--	--
Condenser - air cooled	--	5,900	--	16,100	--	--	43,600	4,900	200	--	--	--	--
Condenser - radiator (combined)	100	1,300	--	16,500	1,000	400	27,200	12,000	--	--	1,100	4,500	420
Subtotal	100	7,200	--	35,100	1,000	1,600	75,800	16,900	200	--	1,100	4,500	420
Electrical generator													
Operation in potassium vapor	--	--	--	--	--	--	--	--	--	--	--	--	--
External load device	--	50	--	--	--	--	--	--	--	--	--	--	--
Pumps in all-liquid systems	--	5,900	53,000	18,600	3,600	1,600	6,100	1,300	1,600	3,000	--	--	--
Potassium seals	--	3,050 ⁿ	--	5,100 ⁿ	--	--	--	--	--	--	--	--	--
Seal test	--	--	--	300	--	--	--	--	--	--	--	--	--
Bearing test rig													
Instrumented	--	3,000 ⁿ	--	--	--	--	--	--	1,600	--	--	--	--
Single bearing endurance test	--	300	--	--	--	--	4,500	--	--	--	--	--	--

^aAerojet-General Nucleonics, San Ramon, California

^bAiResearch Manufacturing Company, Phoenix and Los Angeles

^cAllison Division of General Motors, Indianapolis

^dGeneral Electric Company, Cincinnati, Ohio

^eJet Propulsion Laboratory, Pasadena, California

^fNational Aeronautics and Space Administration, Cleveland, Ohio

^gOak Ridge National Laboratory, Oak Ridge, Tennessee

^hPratt and Whitney Aircraft (CANEL), Middletown, Connecticut

ⁱRocketdyne, Canoga Park, California

^jUnited Nuclear, White Plains, New York

^kBrookhaven National Laboratory, Upton, Long Island, New York

^lAerojet-General Nucleonics, San Ramon, California

^mWestinghouse Astronuclear, Large, Pennsylvania

ⁿThe time shown for boiler feed pump, potassium seal, and test bearing was accumulated during operation of the turbine indicated in that column

The major attraction of the potassium Rankine PCS is that it is compatible with higher source temperatures than is the mercury Rankine PCS. This potential provides for higher Carnot efficiency and, therefore, higher overall efficiency since the device efficiency can be expected to be approximately equal to the potential efficiency of mercury Rankine devices (i.e., about 40% of Carnot).

These programs stopped short of performing any detailed studies on power processing; however, at the increased power levels, the approach to voltage and frequency regulation of the alternator output would be expected to be similar to that used for the SNAP-8 mercury Rankine PCS.

(a) NASA Potassium Rankine

A typical boiling potassium PCS was a 300-kWe power subsystem proposed by NASA with support from several contractors (Ref. IV-70). The specific subsystem was based on the use of a 2 Mwt fast nuclear reactor and was to produce a nominal 300 kWe. The power conversion loop was to use potassium vapor from the boiler to produce electrical power in the turbo-alternator.

In this design, superheated potassium vapor at 1400 K (2100°F) is supplied to the multistage turbine. The turbine drives an alternator in which 1600-Hz electric power is generated. The spent turbine exhaust vapor is condensed and subcooled in four parallel tube-and-shell condensers. The liquid is pumped back to the boiler inlet by an EM boiler feed pump.

Potassium liquid is used to cool and lubricate the turbine and alternator bearings, the alternator rotor and stator, and the EM pump stators. An

organic fluid cools the electronic equipment. The turbine bearings are lubricated and cooled separately from the alternator and the pump stators. A temperature level of 644 K (700°F) is used for cooling the alternator rotor because of strength limits imposed by H-11 rotor magnetic material at the higher temperature. The low temperature level used for the alternator stator coolant reduces heat transferred to the alternator rotor and thereby improves its mechanical integrity.

The subsystem has the feature that as the instantaneous mission load varies, the excess power generated is passively rejected to space through the use of a high-temperature parasitic load resistor. Long-term variation in power requirements are handled by modulating the reactor power.

Additional features of the working fluid loop include the moisture separators and the regenerative boiler feed heater. If no moisture removal devices are used, the turbine exit moisture would approach 20%. With this high moisture, the erosion damage problem in the latter stages of the turbine could be substantial. By using moisture removal devices, the moisture levels at critical areas in the turbine can be reduced and the potential for erosion damage lessened. The analyses available thus far indicate that no erosion problems should be expected.

Trade studies indicated the advantage of preheating the boiler feed flow using the liquid and vapor removed from the turbine. The boiler feed flow temperature would be increased by over 111 K (200°F), thus reducing the size of the boiler and improving subsystem efficiency. These gains would more than offset the penalty of the additional component.

Table IV-34 shows the power output at various stages of generation for the nominal 300-kWe Rankine subsystem. The 2 MW of net reactor thermal input is converted to 375 kWe of net electrical power at 1600 Hz, using turbine and alternator efficiencies of 81.3% and 94.5%, respectively. This yields a net

TABLE IV-34
POWER DISTRIBUTION
ADVANCED POTASSIUM RANKINE POWER CONVERSION SUBSYSTEM

Net thermal power input (MW)	2.0
Net shaft power (kW)	498
Alternator output power (kWe)	428
Internal engine requirements (kWe)	53
Net unconditioned power to user (kWe)	375

efficiency of nearly 19%. About 12.4% (53 kWe) of the gross alternator output of 428 kWe is used for internal engine power requirements.

A significant variable is the choice of turbine exhaust pressure. As the exhaust pressure (and, hence, condensing pressure and temperature) is reduced, the subsystem efficiency increases but at the expense of a larger and heavier radiator.

The selection of containment materials is based largely on the inertness to lithium and potassium at operating temperatures exceeding 1367 K (2000°F). The corrosion resistance of tantalum, molybdenum, columbium, and tungsten is excellent based strictly on the very low solubility of the pure elements in these alkali metals. However, corrosive attack is drastically accelerated when oxygen is present, requiring the addition of zirconium or hafnium as a getter to form stable and insoluble oxides. A sample of tantalum alloy T-111, selected for use as the prime high-temperature containment material for the system, was exposed to potassium at 1589 K (2400°F) for 2000 h with negligible corrosion. By way of comparison, an ungettered tantalum alloy, Ta-10W, suffered catastrophic attack at 1256 K (1800°F) after only 128 h. The tantalum oxides, in the latter case, were preferentially attacked and removed by the potassium, leaving voids in the parent alloy.

The gettering element can, however, become saturated with oxygen. It has been established that oxygen pickup during all phases of processing must be

closely controlled, including limiting the oxygen content in the alkali metals and welding atmospheres to less than 10 ppm. In general, refractory alloys containing hafnium or zirconium display negligible corrosion in contact with the alkali metals for times up to 10,000 h and at temperatures of up to 1478 K (2200°F) when established procedures of limiting oxygen content are followed.

The boiler must produce relatively dry, superheated potassium vapor at a temperature of 1422 K (2100°F) and perform stably during system startup and at operating conditions. In addition, the boiler must be structurally resilient to withstand environmental stresses over the power plant life. Compactness is still another major requirement. Preliminary studies have identified a boiler design that might achieve the above stringent requirements. An artist's sketch of this boiler design is shown in Figure IV-104. Potassium flows in a single pass (once-through) inside the tubes, and the lithium heating fluid flows in the space between the tubes and the shell in a direction counter-current to the potassium. The construction material used throughout the boiler is the refractory-metal alloy T-111. The temperature of the lithium and potassium fluids as a function of length along the boiler is shown in Figure IV-105. In addition, the heat transfer regions generally encountered in potassium boilers are shown crosshatched in this figure.

Figure IV-104. 2-MWt Once-through Potassium Boiler,
 NASA Rankine PCS
 ESG-DCE-13398

Figure IV-105. Temperature Distribution in Boiler
for Potassium Rankine PCS

The vapor exiting the turbine is liquefied and subcooled inside convectively cooled condensers. These condensers consist of tubes contained inside a shell. The potassium vapor flows within the tubes, and the radiator fluid, liquid NaK, flows in the shell space countercurrent to the potassium. Previous experimental investigations conducted with potassium condensing inside single tubes have resulted in correlations of condensing heat-transfer coefficients and pressure differences. These correlations make possible the steady-state thermal design of potassium condensers. However, zero-gravity operation, condensate management within the tubes, and flow stability are areas of uncertainty for condenser design. Minimizing gravitational effects has been suggested and needs to be investigated. Orifices at the outlet end of the tubes are believed to be required to prevent tube-to-tube instabilities. The pressure drop across these flow control devices constitutes the major pressure loss across the condenser.

The turbo-alternator in Figure IV-106 consists of a ten-stage axial flow turbine flexibly coupled to a 10-pole homopolar alternator. The turbine and alternator rotors are each supported on a pair of potassium-lubricated pivoted-pad thrust bearings in each component. The turbo-alternator rotates at a design speed of 19,200 rpm and produces a net electrical power of 428 kWe at the alternator terminals at the design cycle state points.

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Figure IV-106. Potassium Turboalternator, Preliminary Design
for NASA Rankine PCS

The turbine aerodynamic design has been optimized to achieve an efficiency of 81.3% within the prescribed design cycle state points. However, in the direct expansion of the two-phase potassium vapor to its exhaust pressure, the moisture level at the turbine exit is an excessive 20%. Unless steps are taken to reduce this moisture fraction and the corresponding blade tip velocities, blade damage may occur from moisture impact erosion. This source of damage has been anticipated from similar problems encountered in steam turbines.

The mechanical design of the turbine is characterized by the use of curvic couplings to join both the rotor discs and the nozzle diaphragms. All of the blading, rotor discs, and tie-bolts are fabricated from TZM. T-111 is used for the high-temperature static parts, while Cb-1 Zr alloy is used for the lower-temperature parts. Turbine stages six through nine use shrouded blades, inserted in the rotor discs, to improve overall efficiency and also to act as slingers where moisture removal is required.

The alternator is a potassium-cooled, brushless, homopolar inductor generator designed by Westinghouse. Its design rating is 450 kWe at a power factor of 0.75 lagging, 480/832 V, 3 phase, 1600 Hz. The unit has a calculated electrical efficiency of 0.945 including rotor windage.

To provide adequate strength and magnetic properties, the ten-pole alternator rotor is a one-piece H-11 forging. The potassium coolant temperature was reduced to 644 K (700°F) to provide adequate rotor strength. The stator consists of two magnetic lamination assemblies separated by a toroidal field core. The laminations are made of Hiperco 27. The stator and field coils are mounted in a solid magnetic frame made of a Hiperco 27 forging. A circumferential manifold on the stator frame and a manifold around the field coil comprise the potassium cooling configuration, which is designed to limit the hot-spot temperature to 811 K (1000°F).

(b) Medium Power Reactor Experiment Potassium Rankine

ORNL performed studies and development on boiling potassium power conversion for space. Two versions of the PCS were identified in this program: (1) one that was intended to operate at a peak potassium vapor temperature of 1110 K (1540°F) with stainless steel construction and (2) a more advanced PCS using 1365 K (2000°F) vapor with refractory metal construction. The ORNL program was named the Medium Power Reactor Experiment (MPRE). In this report, MPRE will connote the 1110 K (1540°F) PCS and advanced MPRE will refer to the 1365 K (2000°F) PCS (see next subsection).

MPRE was designed as a reactor experiment intended to investigate the feasibility of a boiling-potassium reactor subsystem that was conceived as a major simplification of multiloop designs. The primary cooling of a liquid-cooled reactor, with its pump and boiler, was to be replaced by a single reactor assembly with an integral vapor separator.

In this design, potassium vapor is expanded through the main turbine to generate electricity and then is condensed in the radiator. A small side stream of vapor is permitted to expand through a free-turbine-driven feed pump that returns the condensate to the boiler (reactor core). Jet pumps driven by a portion of the feed-pump discharge flow are also installed near the condenser outlet manifolds to facilitate the removal of condensate from the condenser under zero-gravity conditions. A regenerative feed heater reduces the size of the radiator, increases the cycle efficiency, and may reduce thermal stresses in the boiler recirculation loop.

The centrifugal feed pump is not an integral part of the turbine-generator unit; it is driven by a free turbine. This arrangement made it

possible to separate the development problems and proceed with tests on the MPRE subsystem without a turbine-generator unit. The turbine-generator was simulated by an orifice and heat exchanger with similar flow characteristics.

The separation of the pump from the turbine-generator unit may have several additional advantages. It simplifies the design of the bearing and shaft for the turbine and generator units. More important, however, it eases the problem of matching the characteristics of the various components of the subsystem through a pressure-balance line. The liquid potassium for the generator heat removal loop would be circulated by a small, low-pressure-rise centrifugal pump impeller mounted on the generator shaft.

The major accomplishments in the MPRE program include:

- 1) The compatibility of stainless steel with boiling potassium was demonstrated in over 160,000 h of component and subsystem tests, including 9000 h of testing reduced-scale mockups of the MPRE. One of these mockups operated for 2500 h with no interruptions except those caused by troubles with the electrical equipment used to heat the boiler. These tests were in the range of 1100 to 1150 K (1500 to 1600°F).
- 2) Sufficient data on the heat transfer and burnout limitations for boiling potassium were obtained to validate the feasibility of the boiling-potassium reactor concept.
- 3) Good boiling-flow stability and good flow distribution were demonstrated in electrically heated core mockups.
- 4) Explosive boiling of potassium was found to be a major problem if nucleation sites were not deliberately introduced, and nucleation rings were developed to give a satisfactory solution to the problem of initiating and maintaining smooth nucleate boiling of potassium.

- 5) A low-mass, tapered-tube direct-condensing radiator designed for space operation was developed and shown to give a uniform flow distribution with good flow stability in ground tests.
- 6) A potassium-lubricated tungsten carbide bearing was operated in a bearing test rig for 4000 h with 350 starts and stops under load with no loss in performance or detectable surface damage.
- 7) A potassium-vapor turbine was operated, and over 4000 h of operating experience was accumulated with potassium-vapor turbines running on potassium-lubricated bearings.
- 8) The performance of typical jet pumps in the cavitating regime was established and the pump characteristics for zero-gravity operation of a Rankine cycle were demonstrated.
- 10) A single-loop Rankine cycle subsystem was devised in which the only electronic control equipment is that required to couple the reactor to the generator output and that required to regulate the generator voltage. Studies that included analog simulation and subsystem tests showed stability and control characteristics.

Although a significant development program has occurred for the MPPE potassium-Rankine subsystem, there are two areas that need additional development. The first of these is the control of free liquid surfaces under zero-g operation. Resolution of this problem will most likely require zero-g development testing. Fortunately, such testing could initially be performed with water rather than potassium to scope the investigation prior to liquid metal tests. The second problem that will require additional development is a potassium vapor bore seal for the alternator.

(c) Advanced MPRE Potassium Rankine

In the summary report on the ORNL boiling-potassium program,^{IV-73} the emphasis was on the Advanced MPRE design [1365 K (2000°F)], whereas most of the development had been performed on the moderate temperature MPRE [1110 K (1540°F)]. A schematic of the Advanced MPRE PCS is shown in Figure IV-107. Potassium vapor enters the turbine at 1365 K (2000°F) and discharges from the turbine at 1025 K (1385°F). The turbine inlet pressure is 1.05 MPa (152 psia), while the discharge pressure is 0.10 MPa (14 psia).

The schematic in Figure IV-107 is essentially the same for the MPRE and the Advanced MPRE. The latter is at a higher temperature and pressure and would use a refractory alloy such as niobium-1% zirconium. The reasons for considering designs at higher temperature and pressure are to achieve either a higher subsystem efficiency for a given radiator area or a much reduced radiator area for a given efficiency.

The development of this concept was rather limited. As of October 1966, 15,000 h of thermal convection corrosion tests had been completed on niobium up to 1365 K (2000°F), and 3300 h of forced convection corrosion tests had been completed at similar temperatures. Ref. IV-74 through IV-83 are representative of similar R&D performed on components for PCSs using potassium, cesium, and rubidium in the 1960 to 1966 time period.

(4) Closed-Cycle Brayton Power Conversion

The closed-cycle gas turbine Brayton PCS is adaptable to a wide range of applications, environments, and power levels. Practical competitive subsystems have been designed and/or built for space, terrestrial, marine, and underwater environments with power levels ranging from 500 W to more than

Figure IV-107. Advanced MPRE Boiling Potassium Rankine PCS

500 MW. Input thermal energy can be from solar, nuclear (either reactor or isotope), fossil fuel, or chemical combustion with waste heat rejected to the environment by a wide variety of methods.

A major reason for the interest in the closed Brayton cycle is its high demonstrated device efficiency. Brayton machines have been constructed with device efficiencies of 40-50%. Even with the relatively low source temperatures that have been currently demonstrated, overall subsystem efficiencies of 20 to 35% have been obtained. The basic closed gas turbine cycle, shown in Figure IV-108, is applicable to a wide variety of applications.

Selection of the compressor inlet temperature for a space application is complex. As a starting point, minimum specific radiator area has in the past been achieved at a cycle temperature ratio (ratio of compressor inlet temperature to turbine inlet temperature) of about 0.35.

Selection of the turbine inlet temperature is primarily dictated by the heat source characteristics and the temperature limits of materials available for use within the high temperature portion of the subsystem. Current demonstrated technology for commercially viable combustion heater designs incorporates a turbine inlet temperature of 1089 K (1500°F). The next significant component temperature limit is reached at a turbine inlet temperature (T_6) of 1228 to 1256 K (1750 to 1800°F), where the turbine materials must be changed from superalloy to refractory metals such as molybdenum or columbium. Refractory metals in the turbine require either inert gas working fluids or suitable coatings on the refractory metal components in an air cycle machine. Refractory metal turbine components will increase the maximum T_6 to approximately 1500 K (2250°F), above which the use of ceramic turbine components is required. Note that the above is based on uncooled turbine components. Higher turbine inlet temperatures can be achieved with turbine cooling; however, detailed tradeoff studies must be conducted to determine if the higher temperatures will result in competitive designs when the efficiency penalties and complexities due to cooling are included. ^{IV-86}

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Figure IV-108. Closed-Cycle Brayton Power Conversion Subsystem

Selection of the recuperator effectiveness is second only to selection of the turbine inlet temperature in importance when optimizing a closed Brayton engine. The choices are a function of the recuperator configuration, subsystem packaging, and the application. The recuperator is frequently the most massive single component; therefore a tradeoff of efficiency versus mass, volume, and/or cost almost invariably is required.

For a constant fluid molecular weight and cycle pressure ratio, the compressor and turbine tip speed is essentially constant. Therefore, increasing the shaft speed is beneficial (reduced rotor mass) to the point where one of the following factors limits the benefit:

- 1) The compressor and turbine diameters must decrease to maintain the tip speed, and acceptable clearances will decrease with diameter. Below a certain diameter, the control of blade angles and blade thicknesses, necessary for high component efficiencies, cannot be maintained.
- 2) For shaft-mounted alternator drives, the alternator tip speed will exceed the capability of current technology. High windage losses will be incurred that will reduce the cycle efficiency.

Gas bearing rotor supports are used for space power subsystems. The use of gas bearings to support the rotor has led to designs with the potential for unlimited bearing life. Both solid geometry and compliant geometry designs have been used. The compliant designs are preferred because of their tolerance to thermal gradients, misalignment, vibration, and shock. The compliant (foil) bearing designs are particularly well adapted to relatively low-mass rotor designs, in which pure hydrodynamic starting and stopping can be used. The foil-bearing designs have been applied to a wide variety of open and closed cycle rotating designs. The DC-10 and A-200 aircraft air conditioning unit has accumulated over 5×10^7 h of operation without a bearing failure.

Three Brayton PCSs are described below.

(a) NASA B Engine Brayton

Four Brayton cycle "B" engines using Brayton rotating units (BRUs) were developed under contract to NASA and were sent to NASA-Lewis Research Center for performance and endurance testing. The purpose of these tests was to demonstrate the technology for space application of the closed Brayton cycle engines. Two of these engines have together accumulated over 41,000 h of successful operation, while achieving cycle efficiencies of 30 to 33%. Both units met or exceeded performance objectives with no detectable performance degradation during endurance testing. The engine tested for 38,000 h showed erosion of the turbine stator, probably resulting from dust in the subsystem. The other two units were used in early development testing, refurbished, and held in standby during the endurance testing.

The NASA B engine produced 2.25 to 10.5 kWe of 1200-Hz power and incorporated solid geometry gas journal and thrust bearings. In 1972, an upgraded version of the BRU was delivered to the NASA. Designated BRU-F, this machine was designed to develop 15 kWe and to utilize hydrodynamic, compliant foil gas bearings. This BRU was never tested.

(b) Turboalternator-Compressor (TAC) Brayton

The B-engine program led to the initiation of a development program for a 100-kWe rotating unit designated the Turboalternator-Compressor (TAC). Component development activities on this program for the NASA-Lewis Research Center were directed towards:

- 1) Fabrication of a 0.172-m (0.566-ft-) diameter bimetallic Rice alternator rotor
- 2) Design and experimental verification of a compliant foil gas bearing suspension for the rotor.

These development activities were successful. Continued development of this system was stopped, however, due to a change in emphasis from high-power reactor systems to low-power radioisotope systems. The engine was never operated.

(c) Mini-BRU Brayton (BIPS)

A Brayton turboalternator-compressor assembly in the 400- to 2100-W power range was developed for use in the Brayton Isotope Power System (BIPS). The rotating assembly, commonly referred to as the Mini-Brayton Rotating Unit (Mini-BRU), consisted of a single-stage centrifugal compressor, a Rice alternator rotor, and a radial turbine mounted on a common shaft. The shaft was supported on gas-lubricated foil journal and thrust bearings with self-acting (hydrodynamic) start-stop capability. The journal bearings were located in the alternator secondary gap to minimize shaft length; no shaft seals were used. The bearings were cooled by conduction and orifice controlled gas flows. The external envelope of the Mini-BRU was 22.8 cm (9.0 in.) in diameter by 30.5 cm (12 in.) long, and the unit had a mass without heat exchanger of 16.3 kg (36 lb). Design speed of the rotating group was 52,000 rpm when a xenon-helium (Xe-He) gas mixture (molecular weight of 83.8) was used as the cycle working fluid. The working fluid also served as the bearing lubricating and cooling medium and the alternator coolant. The compressor discharge gas is ducted through an axial heat exchanger wrapped around the outside of the alternator and bearings.

The Mini-BRU, including a recuperator and the necessary heat input and rejection equipment, was fabricated and tested as part of the BIPS program. The engine gas bearings failed after about 50 h of testing at design conditions. Failure analyses indicated the potential causes as (1) stackup of clearances and tolerances and (2) incorrect Teflon coating on the bearing foil. The unit was reworked and subsequently was successfully retested for 1100 h.

(5) Open-Cycle Brayton Power Conversion

Nuclear open-Brayton cycles have been suggested as possibilities for applications requiring limited duty at multimegawatt electrical power levels. Because the working fluid would be dumped overboard in the open-cycle application, the subsystem would not require a means of heat rejection to the environment. In addition, since the working fluid would be stored on board the spacecraft, high performance of the turbomachinery would be the primary design driver and hence fluid use rates would be minimized.

For minimum system specific mass, the enthalpy (or heat capacity) in terms of working-fluid mass should be as high as possible. Because the heat capacity of hydrogen is at least a factor of three higher than that of any other substance, hydrogen would be the ideal working fluid, even though it would have to be stored in space as a cryogenic liquid.

The turbines appropriate for use in open-cycle Brayton PCSs are similar to those used in closed-cycle subsystems so that the discussions in the prior section apply in this case as well. The optimization of the open-cycle subsystem, however, would lead to significantly different design criteria. In the open cycle, the working fluid is compressed (usually as a liquid in the case of H_2) and heated (usually as a supercritical fluid) either by passing

through a nuclear reactor or by combustion; the gases are then expanded through a turbine. The turbine drives the fluid pump and an electrical generator. Since there is no heat rejection subsystem, the optimization reduces to a trade between turbine efficiency and system mass. Since system mass includes the mass of the stored working fluid, the mass of the turbomachinery is not significant in the optimization. It is, therefore, expected that high efficiency will be required for open Brayton cycle turbomachinery for space power use.

The most significant parameter in determining turbine efficiency is turbine inlet temperature. In the case of the closed-cycle turbine, improving turbine inlet temperatures tends to depend on improving materials technology, rather than using actively cooled turbines. Open-cycle systems, on the other hand, are not particularly sensitive to pressure loss in the turbine and would, therefore, be good candidates for actively cooled turbine designs.

To date, no specific hardware configurations exist for specific space applications. Relatively detailed designs and subsystems have, however, evolved for high-power airborne power supplies. These devices consist of liquid propellant gas generators that drive a low-mass turbine and alternator. The turbines used in these applications derive directly from the turbopumps used on large liquid-fueled rocket engines. Table IV-35 summarizes the gas turbines that have been considered for use in high-power airborne power supplies.^{IV-87}

Ref. IV-88 indicates that such turbines are suitable for use in the 10- to 50-MWe power range. However, the adaptation of these machines to space power use will require carefully evaluating several of the characteristics that were developed for the airborne application. The chief characteristics that are of concern are the type of bearings used, the use of gearboxes to match power output to generators, and the use of relatively low gas turbine inlet temperatures.

TABLE IV-35
OPEN CYCLE TURBINE GENEALOGY AND EXPERIENCE

Designation	MARK-15F	4-MW APU	MARK 15-E0	MK-15-E0	MK-15-E1	MK-15-E2	MK-15-E3-1	MK-15-E3-1	MK-15-E3-2	MKK-15-E3-2
Design	Production	Experimental	Experimental	Experimental	Experimental	Experimental	Experimental prototype	Experimental prototype	Experimental prototype	Experimental prototype
Application	J-2 rocket	APU system	FST transient	Facility C/O	Performance	E3 performance predictions	Deliverable	Deliverable	Deliverable	Deliverable
Drive gas	48.5% O ₂ 51.5% H ₂	85% N ₂ H ₄ 15% H ₂ O	N ₂ H ₄	N ₂ H ₄	N ₂ H ₄	Air	GN ₂	N ₂ H ₄	GN ₂	N ₂ H ₄
Power (kW)	6524	4474	ND	3186	4391	853	74	7645-6229	ND	4643
Speed (rpm)	28,266	26,000	23,000	6,308-24,318	6,232-29,785	5,600-14,200	1,438-5,591	0 ^a -29,840	4,350-6,090	26,129
Inlet temperature [K (°F)]	976 (1297)	950 (1250)	1167 (1640)	1122 (1560)	1221 (1739)	818 (1012)	294 (69)	1080 (1484)	ND	1137 (1587)
Inlet pressure [MPa (psi)]	5.05 (732)	4.25 (616)	3.43 (497)	3.05 (442)	5.05 (732)	1.92 (278)	0.70 (101)	4.98 (722)	ND	5.03 (730)
Flow rate [kg/s (lb/s)]	3.45 (7.6)	5.62 (12.4)	4.49 (10)	4.08 (9)	3.60 (8)	3.37 (7.4)	No data	3.91 (8.6)	ND	3.80 (8.4)
Pressure ratio	7.45	12.0	13.8-13.9	14.3-33.7	32.6-40.1	41.7-49.6	1.8-6.5	40.3-42.8	ND	39.2-43.1
Type of test	DNA		Fast-start transients	Dynamometer checkout	Steady-state performance	Steady-state performance	Instrumentation calibrations	Acceptance integrity/performance	Instrumentation calibrations	Performance mapping
Number of tests	-		8	17	23	11	13	7 ^a	10	9
Total run time (s)	-		6.18	87.5	183.5	735.0	143.1	33.8	7.0	37.4
Load device	LH ² pump		Flywheel	Dynamometer	Dynamometer	Dynamometer	Dynamometer	Dynamometer	Dynamometer	Dynamometer

^aIncludes two stall torque tests

DNA - Does not apply
ND - Not determined

b. Reciprocating

Reciprocating engines are commonly used in nearly every type of ground-based power generation application. For space applications, only the free piston Stirling engine has been seriously proposed. The basic attraction of the Stirling engine is its capability for developing high cycle efficiency while using relatively low temperature heat sources.

(1) Free Piston Stirling Power Conversion

The free piston Stirling engine (FPSE) is a thermally driven mechanical oscillator operating on a Stirling engine cycle, deriving power from heat flow between a source and a sink.^{IV-89, IV-90} The desired displacer motion in the FPSE is produced, unlike in most commonly known Stirling engines, by gas forces rather than kinematic linkages. This engine operates at the highest overall subsystem device efficiency of all known heat engines. In addition, the FPSE is uniquely suited to driving direct coupled reciprocating loads, such as linear generators, in a hermetically sealed configuration and without the need for high pressure shaft seals or contaminating lubricants.

The engine consists of three basic components: a heavy piston (power piston), a low-mass piston (displacer), and a sealed cylinder. The displacer rod passes through the power piston and is in communication with the bounce space, which is essentially a constant pressure chamber. Figure IV-109 is a schematic of the FPSE and the pressure-volume relationship.

To start the cycle, it is assumed that the engine is at rest, the pressure is equal throughout, and the temperature is ambient. Heat is then applied to the expansion space, causing an increase in pressure that acts on both the power piston and the displacer, moving them down the cylinder. The forces and acceleration on the pistons are functions of the exposed piston, displacer, and rod areas and their respective masses. If the ratio of the mass of the piston to the mass of the displacer is large, then the displacer

- 1-2 PISTON EXPANDS WORKING GAS; DISPLACER ON PISTON.
- 2-3 PRESSURE IN BOUNCE SPACE GREATER THAN PRESSURE IN WORKING SPACE, FORCING DISPLACER TOWARD HOT SPACE, WORKING GAS MOVED INTO COOLED SPACE, PRESSURE DROPS.
- 3-4 PISTON DRIVEN INTO WORKING SPACE BY HIGHER BOUNCE SPACE PRESSURE.
- 4-1 DISPLACER DRIVEN TOWARD COLD SPACE BY WORKING SPACE PRESSURE HIGHER THAN BOUNCE SPACE PRESSURE.

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Figure IV-109. Free Piston Stirling Engine Operation

will accelerate rapidly, moving the cold fluid from the cold compression space to the hot expansion space. The displacer and the piston will eventually come into contact with each other, after which they will move together and as a result the flow of working fluid will stop.

The pressure will now begin to decrease in the engine as the expansion continues. This expansion will continue until the pressure of the working fluid drops to the pressure of the bounce space. Power piston inertia will, however, be sufficient to continue the expansion so that the working pressure will fall below the bounce space pressure, which will subject the piston and displacer to decelerating forces. Due to its low mass, the displacer will respond first and will separate from the piston. The working fluid will start to flow from the expansion space through the regenerator, giving up its heat while the working pressure falls sharply, causing the displacer to move up rapidly toward the cylinder top.

Piston movement downward is now stopped, and the pressure differential will cause the piston to move upward and compress the working fluid until the pressure is again above the bounce space pressure. The displacer will then start to move downward again until it contacts the piston, starting the cycle again.

There are only two moving parts in the FPSE engine-alternator unit: (1) the displacer and (2) the power-piston assembly. The power-piston assembly consists of the power piston directly connected to the alternator armature, thereby eliminating the need for converting reciprocating motion to rotary motion and the need for lubricating such a device. This assembly is supported by two hydrostatic (pressurized) helium gas bearings, the power piston itself being one bearing; the second bearing is located at the ID of the alternator armature. Pressurized helium for these two bearings is supplied by a small positive-displacement compressor built into one end of the alternator armature. Because of the small P required from the compressor, no piston rings are required, and the piston will reciprocate without any contact with

the cylinder wall. Compressor inlet and discharge flow is controlled by means of fixed ports, rather than valves, to eliminate any possibility of valve failure. Therefore, inherent long-term reliability is incorporated in the design.

A significant advantage of the single-cylinder displacer/piston engine arrangement is that the power piston and piston cylinder operate at low temperature. Both the displacer and the power-piston gas bearings, therefore, operate at low temperatures and are not subject to differential thermal expansion or thermal distortion problems. The alternator bearing likewise operates in a cool environment.

At the initiation of engine startup, no pressure is available for the hydrostatic bearings. However, gas bearing material technology is well developed for this type of application, and several material combinations are available that can withstand tens of thousands of sliding starts without performance deterioration. The FPSE engines currently being tested do not use any intentional lubrication. Side loadings are low on the displacer and power piston, and conventional surface treatments or coatings appear to be sufficient for obtaining reliable bearing operation.

A matrix-type regenerator is placed between the heat source and the heat sink. Heat is stored in the regenerator as the gas moves toward the cold (compression) space and is then released as the working fluid returns from the cold space to the hot (expansion) space. In this way, the regenerator reduces the loss of heat energy during the cycle and greatly improves engine efficiency. The regenerator can be stacked layers of wire screen or wrapped foil.

The FPSE conversion machine, operating as a single engine alternator unit, will not produce any steady-state friction or reaction torques. However, the reciprocating motion of a single engine will cause vibratory forces to be transmitted to the vehicle unless some means is provided to attenuate or

essentially eliminate these transmitted forces. One approach is to use identical engines in pairs, mounted such that the reciprocating members are driven in phase opposition.

The electrodynamic linear alternator design is based on the well-known principles of electromagnetic induction. The uniqueness of the alternator lies in the design configuration used to implement the electromagnetic induction principles. A dc flux loop is established by the dc field coil. As the armature moves back and forth in the stator, the dc flux links the ac coils and induces an ac voltage. Figure IV-110 shows a layout of a typical free piston Stirling engine using such a linear alternator.

(a) 10 to 50 kWe Space Free Piston Stirling

In 1975, Mechanical Technology, Inc. (MTI), studied two free-piston Stirling engines for reactor space power application. The two engines were for powers of 10 and 50 kWe net and, therefore, had gross powers of 10.5 and 51.0 kWe, respectively. The 51-kWe engine was a four-cylinder opposed configuration, with hot-end coupled pairs of pistons running in opposition to cancel shaking forces. There were four alternators; these constituted nearly 35% of the total engine mass. The 10.5-kW engine used two opposed cylinders, also hot-end coupled, with two attached alternators. Both engines operated at 6.0 MPa (870 psi) with a pressure ratio of about 1.3.

Type 316 stainless steel was assumed for all hot parts, and titanium for the alternator pressure enclosure. The piston was to be made of aluminum with carbon sleeves and the displacer of titanium. Dimensions were chosen consistent with the required life in the cyclic stress region imposed by gas forces.

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Figure IV-110. Free Piston Stirling Engine –
Alternator Conceptual Design

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The engine was a counterflow-tube design sized to an appropriate tradeoff between friction and temperature drop losses. The heater was a tube bundle with parallel flow of NaK. The heater tubes linked the regenerator of one cylinder to the expansion space of the opposed cylinder. This was to ensure proper opposed motion of the piston pairs. Regenerators were of the wound-foil type; this type has near-optimum effectiveness versus pressure drop. Tables IV-36 and IV-37 summarize the sizing study results as arrived at by computer simulation of the geometry described.

(b) Terrestrial Free Piston Stirling

Free piston Stirling engine development has been performed during the past decade by Sunpower, General Electric, NASA-Lewis, MTI, and others. MTI has built and operated several FPSEs up to 4.0 kWe. The overall efficiency has gradually been increased from 22% to 41%. The device efficiency has risen from about 35% to 65%, and 70% may be attainable. About 2000 h of subsystem test time has been accumulated, plus about 16,000 h of component test time. MTI is planning a 1000-h endurance test on a 4.0-kWe FPSE in 1983. At present, 12 kWe is considered to be the maximum desirable power for one FPSE because of heat transfer limitations. However, multiple units up to at least 100 kWe total power are considered feasible.

As compared to conventional rotary alternators, the linear alternator has little operating history, is limited to relatively low frequencies (~60 Hz), and has higher harmonic content in the output waveform. Of these limitations, low frequency is the most significant for space applications because the mass of an electrical component generally increases as frequency decreases.

Refs. IV-92 through IV-103 are a representative selection of reports and papers on Stirling engines.

TABLE IV-36

FOUR THERMODYNAMIC PERFORMANCE POINTS FOR TWO STIRLING ENGINE CONCEPTS FOR SPACE USE

Engine Gross Power (kWe)	Frequency (Hz)	NaK Tempera- ture [K (°F)]	Coolant Tempera- ture [K (°F)]	Carnot Efficiency (%)	Engine Gross Efficiency (%)	Device Efficiency (% of Carnot)	Coolant	
							Flow Rate [kg/s (lb/s)]	Pressure Drop [kPa (psi)]
51	46	978 (1300)	478 (400)	51	34	67	17.8 (39.2)	76 (11)
	46		422 (300)	57	38	67	16.0 (35.2)	
	46		367 (200)	63	42	67	14.5 (32.0)	
	46		311 (100)	68	46	67	13.3 (29.2)	
	45	978 (1300)	478 (400)	51	36	70	4.7 (10.3)	30 (4.4)
	45		422 (300)	57	40	70	4.2 (9.3)	
	45		367 (220)	63	44	70	3.9 (8.5)	
	45		311 (100)	68	48	70	3.5 (7.8)	
10.5	46	978 (1300)	478 (400)	51	34	67	17.8 (39.2)	76 (11)
	46		422 (300)	57	38	67	16.0 (35.2)	
	46		367 (200)	63	42	67	14.5 (32.0)	
	46		311 (100)	68	46	67	13.3 (29.2)	
	45	978 (1300)	478 (400)	51	36	70	4.7 (10.3)	30 (4.4)
	45		422 (300)	57	40	70	4.2 (9.3)	
	45		367 (220)	63	44	70	3.9 (8.5)	
	45		311 (100)	68	48	70	3.5 (7.8)	

TABLE IV-37
FPSE DIMENSIONS – SPACE APPLICATION STUDY

Power Output (kWe)	Number of Cylinders	Bore [cm (in.)]	Stroke [cm (in.)]	Mass [kg (lb)]	Length [m (in.)]	Width [m (in.)]	Height [m (in.)]
51	4	9.3 (3.66)	3.2 (1.26)	400 (880)	1.76 (69.27)	0.546 (21.49)	0.273 (10.75)
10.5	2	6.18 (2.43)	3.2 (1.26)	130 (286)	1.53 (60.2)	0.236 (9.3)	0.236 (9.3)

c. Static

Three static PCSs have received sufficient study and development to permit their characterization as potential candidates for space power applications: (1) thermoelectric (TE) converters, (2) thermionic converters, and (3) thermophotovoltaic converters. TE and thermionic subsystems have been the object of many years of research and development effort. The thermophotovoltaic converter is a relatively new, but interesting concept.

(1) Thermoelectric Power Conversion

TE devices were a development of the space power programs in the early 1960s. Their use as a reliable source of electricity for space vehicles has been consistently demonstrated in the RTG applications. SNAP-10A used a TE subsystem. TE devices use solid-state conductor technology to directly convert heat energy into electricity using the Seebeck effect. Figure IV-111 illustrates the basic operating principles of a TE device.^{IV-104} An individual converter consists of two semiconductor legs bonded to two heat transfer surfaces known as the hot and cold shoes. One of the semiconductor legs is doped with a small amount of boron to form a p-type material, while the other leg is doped with phosphorus to form an n-type material. One end of the TE couple is maintained by the bonded hot shoe at a high temperature (T_H), while heat is removed from the cold shoe at the other end of the couple at a lower temperature (T_C). Thermally driven transport of the electrons through these n- and p-type materials in each TE leg creates a voltage potential across the two cold shoe plates. By connecting an external circuit across the two cold shoe plates, a current is generated through the external electrical load. The power in the external circuit is maximized when the load resistance is matched to the internal resistance of the converter. These TE uncouples can be connected in an external series-parallel circuit to provide redundancy against open circuits. The efficiency of a TE converter couple is a product of the theoretical Carnot efficiency and the device efficiency.

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Figure IV-111. Thermoelectric Converter Operating Principles

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$$\underbrace{\text{Thermoelectric Efficiency}}_{\eta} = \underbrace{\text{Carnot Efficiency, } \eta_c}_{\frac{T_H - T_C}{T_H}} \times \underbrace{\text{Device Efficiency, } \eta_D}_{\frac{(1 + Z\bar{T})^{1/2} - 1}{(1 + Z\bar{T})^{1/2} + (T_C/T_H)}},$$

where

T_H = hot shoe temperature (K)

T_C = cold shoe temperature (K)

$\bar{T}$ = average temperature in each TE couple leg (K)

Z = combined figure-of-merit for the n- and p-type materials (K^{-1}).

The figure-of-merit (Z) is a material property only and is a function of the Seebeck coefficient, the electrical conductivity, and the thermal conductivity of the materials comprising the couple's legs:

$$Z = \alpha^2 \frac{K_e}{K_{th}},$$

where

α = Seebeck coefficient of the material (V/K)

K_e = electrical conductivity of the material ($\Omega^{-1} \cdot \text{cm}^{-1}$)

K_{th} = thermal conductivity of the material (W/cm·K).

Figure IV-112 illustrates the relationship of these material properties with a material's figure-of-merit.^{IV-104} Two general classes of semiconductor materials, binary-doped silicon-germanium and lead-telluride, have found application in the space nuclear power programs of the past. There are different classes within these materials as discussed below. There are also newer materials in development.

Figure IV-112. Typical Thermoelectric Material Performance Parameters

(a) SNAP-10A Binary-Doped Silicon-Germanium TE Converter

The SNAP-10A reactor electric system used binary-doped silicon-germanium TE material. The material was formed into individual elements and was directly bonded to the flat side of a "D"-shaped tube, as shown in Figure IV-113. This figure also shows the insulator stack that was developed for use between the tube and the element. This TE converter is called "direct-coupled" because this contact coupling is used.

The elements, manufactured by RCA, operated at a hot junction temperature of 772 K (930°F) and a cold junction temperature of 597 K (615°F). The couple efficiency was 2.5% at these temperatures. About 5600 elements (2800 couples) were used on each SNAP-10A system in a series-parallel arrangement to provide approximately 70-V open circuit at 20 A. The flight system operated 43 days in space at power and temperature. The ground-test system operated 10,000 h at power and temperature. The elements exhibited a slow decrease in efficiency due to the normal TE degradation processes. In addition, the PCS (flight and ground test) exhibited an increasing parasitic power loss. This loss resulted from current shunting to ground, where the shunt path was a low resistance surface layer that was developing on the element-to-D tube insulator stack. A modification of the stack geometry was shown to eliminate deposition of the shunt layer. The SNAP-10A TE converters were a successful application of static conversion technology. However, they represent an early state of TE technology.

For voltage control, a simple shunt regulator was used with a maximum dissipation of approximately 70% of nominal payload power. No automatic control

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a. Direct-Coupled Thermoelectric Converter

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b. Converter Construction

Figure IV-113. SNAP-10A Thermoelectric Power Conversion Subsystem

of reactor power was provided (see Figure IV-114). In orbital flight, the equipment performed as expected. The voltage regulator output voltage maintained a steady 28.7 V for the duration of the flight test. A failure occurred, however, in the regulator that was the initiator of reactor shutdown.

(b) RTG Binary-Doped SiGe TE Converter

Binary-doped SiGe converters have been extensively used in space in RTGs. Multi-hundred-watt (MHW) RTGs have seen service in the Voyager and LES 8/9 missions and have demonstrated high reliability for many years. The latest use of these TE converters is in the general-purpose heat source (GPHS) RTG. The state-of-the-art in binary-doped SiGe TE converters for RTGs is shown in Figures IV-115 and IV-116.^{IV-105} This is a "radiatively-coupled" device (as opposed to a direct-coupled device); heat is transported by radiation from the isotope heat source to the TE uncouple hot shoe. An illustration of a typical power processor for an RTG PCS is shown in Figure IV-117. This one is for the Galileo spacecraft.^{IV-147} The dc bus regulator consists of a 600-W sequential shunt regulator and a discharge controller operating as a complementary stage capable of supplying 3.5 W s of stored capacitor energy to the bus for transient energy support. RTG output power in excess of spacecraft demand is dissipated in the retro propulsion module shunt heaters as usable heater power (a viable option to a shunt radiator due to the large thermal time constant of the module).

This converter is produced by the General Electric Company's Advanced Energy Program Department. It is capable of producing 0.5 W of power output at 0.2 Vdc when operated between a hot-shoe temperature of 1273 K (1832°F) and a cold-shoe temperature of 573 K (572°F). The weight of the uncouple is approximately 10 g. The beginning-of-life (BOL) efficiency of the uncouple

Figure IV-114. SNAP-10A Electrical Schematic

Figure IV-115. General Electric RTG Thermoelectric Unicouple (binary-doped SiGe)

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Figure IV-116. General Electric RTG SiGe Unicouple

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Figure IV-117. Galileo RTG Power Processor Block Diagram

is 6.6%. The BCL figure-of-merit (Z) for the uncouple material is $0.7 \times 10^{-3} \text{K}^{-1}$. The power output from SiGe thermoelectric uncouples degrades over an extended power duty cycle. Four percent degradation occurs in the first year due to the supersaturation of the n-type material dopant (i.e., essentially transport of the dopant material within the grain boundaries of the SiGe host material). Following this initial degradation, the power output degrades an additional 1% with each year of continuous duty. Thus, the degradation amounts to about 10 to 12% over a 10-year duty cycle at the reference temperatures. This uncouple will operate either in vacuum or in air atmosphere.

For these converters, the hot-shoe temperature is limited to 1273 K (1832°F) because of material sublimation problems that shorten the operating lifetime. Silicon nitride is used to coat the SiGe surfaces and suppress sublimation of the SiGe material. Two methods^{IV-104} have been proposed to raise this operating temperature limit: (1) using a chemical-vapor-deposition (CVD) process to provide a silicon carbide or oxide coating to the SiGe material or (2) using a high-temperature glass to replace the present silicon nitride coating. It is also possible to increase the silicon content in the TE uncouple to raise its operating temperature by approximately 100 K (180°F). Both methods await laboratory tests to verify increased converter life expectancy.

The present production capacity for SiGe TE converters is about 100 uncouples per week (i.e., about 50 We per week), which is sufficient to fulfill current RTG program requirements. An increase in production rate, however, can be reasonably obtained. This increase could be possible by increasing the number of current processing machines and introducing new high-volume production equipment that is available within the semiconductor industry. No technological breakthrough would be required to meet the increased production capacity. Thus, with appropriate investment and time, substantially increased production rates could be achieved.

(c) RTG Ternary-Doped SiGe TE Converter

A natural extension of the SiGe TE element for RTGs would use a ternary-doped SiGe material (the additive is GaP). The initial results of laboratory tests with this material operating at a hot-shoe temperature of 1273°K (1832°F) indicated a combined figure-of-merit Z of $0.85 \times 10^{-3} \text{K}^{-1}$ (i.e., approximately a 20% improvement in RTG TE device efficiency for the same Carnot efficiency^{IV-104}). By further material optimization, it has been estimated that device efficiencies could be improved from 25 to 35% over current binary-doped SiGe materials. There is little laboratory experience with this material, and additional research efforts and funding are necessary to develop an RTG TE converter using this material. Schedules for this development are discussed in Section IV.D.4.

(d) High Performance TE Converter

Advanced concepts for TE converters may use rare earth sulfides (specifically, lanthanum, cerium, or samarium sulfides), boron-carbon compounds, or other materials.^{IV-104} Preliminary results of materials investigations on lanthanum sulfide indicate a projected figure-of-merit Z of 1.0 to $1.5 \times 10^{-3} \text{K}^{-1}$ for an optimized material at a temperature of 1573 K (2370°F). These investigations, however, must be considered only as a basic materials research effort at the present time. The potentials of these advanced materials are shown in Figure IV-118. Improvements in temperature and performance would be expected. No converters have been built using these materials.

FIGURE OF MERIT-Z

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Figure IV-118. Potential Improvements for Thermoelectric Materials

(e) SABRE TE Converters

The converters proposed for use in the SABRE heat pipe reactor TE system operate at a hot junction temperature of 1355 K (1980°F) and employ a material with a figure of merit of $1.0 \times 10^{-3} \text{ K}^{-1}$. At the present, the material needed to achieve the projected specific power and performance and to operate for sustained times at the design temperature is yet to be developed. As discussed above, it could be ternary-doped SiGe or an advanced material.

A series-parallel connection of the couples was proposed with approximately 1 kW per block. The proposed voltage regulation scheme involved the automatic shunting out of blocks of couples to maintain the desired voltage with load changes and PCS degradation.

(f) SNAP Lead Telluride Compact TE Converter

For low-temperature sources of heat energy (922 K, 1160°F), TE materials research in the early 1960s indicated that PbTe would be a prime candidate for TE converters. PbTe showed the potential for a high combined material figure-of-merit Z of $1.5 \times 10^{-3} \text{ K}^{-1}$ at an operating temperature of 600 K (620°F). Figures IV-119 and IV-120 compare several TE materials and show the advantages of PbTe. Converter development was sponsored under the SNAP program. Since the coolant outlet temperature for the zirconium-hydride class of reactors was limited to 922 K (1200°F), the PbTe TE converter was well matched in temperature to the reactor. A program was established at the Westinghouse Astro-nuclear Laboratory (WANL) to develop a modular PbTe TE tubular converter for advanced UZrH systems.

Figure IV-119. Figure-of-Merit for Typical n-Type Thermoelectric Materials

Figure IV- 120. Figure-of-Merit for Typical p-Type Thermoelectric Materials

The tubular TE module is shown in Figure IV-121. The module is cylindrical, with annular disks of PbTe material located around a central flow tube (the hot junction). An annular passage surrounds the module and acts as the cold junction. The module requires a separate heat rejection NaK loop. The module had a projected efficiency in the 6 to 8% range.^{IV-106}

The tubular PbTe TE material is encapsulated between its inner and outer claddings in a completely compact void-free design that is structurally rugged. Heat flows radially from the inner to the outer surface. The n and p elements are connected electrically in series by conductor rings at the hot and cold junctions. The alternating p- and n-type washers are separated by thin, insulating washers of natural mica. The circuit is electrically isolated from the structural claddings by thin sleeves of boron nitride that provide low thermal resistance between the rings and the claddings.

Table IV-38 summarizes the test experience accumulated on these modules. A total of 225 tubular modules were tested for nearly 1,000,000 module test hours at temperatures from 811 to 894 K (1000 to 1150°F). Some modules were tested at higher temperatures. The TEM-X-S/N-3 series of test modules obtained a 5% energy conversion efficiency (16% device efficiency) at average hot- and cold-junction temperatures of 858 and 537 K (1084 and 506°F), respectively.

Program funding limitations prevented using a ternary TE material and module design modifications to increase the module efficiency to 6.65% (kWe/kWt). The projected characteristics of this ternary TE material, 95% PbTe - 5% GeTe (n) and 95% PbTe - 5% PbTe (p), in the tubular module design are shown in Figure IV-121.^{IV-107} No tubular modules were developed and tested using this material.

*Power Output (We)	364
*Efficiency (%)	6.65
Device Efficiency	17.3
Heat Input (kWt)	5.48
Average Hot Cladding Temperature [K (°F)]	866 (1100)
Average Cold Cladding Temperature [K (°F)]	533 (500)
Approximate Mass [kg (lb)]	5.5 (12.0)
Inner Diameter [cm (in.)]	1.9 (0.75)
Outer Diameter [cm (in.)]	3.8 (1.50)
Active Length [cm (in.)]	38 (15.0)
Total Length [cm (in.)]	48.3 (19.0)

*5% efficiency was demonstrated. These figures represent projected performance based on utilization of a ternary thermoelectric material [95% PbTe - 5% GeTe(n) and 95% PbTe - 5% PbSe(p)]

9467-127

Figure IV-121. Schematic and Characteristics of a Typical PbTe Thermoelectric Tubular Module

TABLE IV-38
TUBULAR PbTe THERMOELECTRIC MODULE DEVELOPMENT EXPERIENCE

Total number of modules tested	225
Total TE couple test (h)	40,000,000
Total module test (h)	860,000
Module test times in temperature range (h)	
811 to 839 K (1000 to 1050°F)	410,000
839 to 894 K (1050 to 1150°F)	415,000
Above 894 K (1150°F)	35,000
Longest module test run (h)	52,000

(g) RTG TAGS TE Converter

The TE material TAGS was developed during the mid 1960s. Its major constituents are tellurium, antimony, germanium, and silver (hence "TAGS"). TAGS is an undoped inherent p-type material. When combined with a PbTe 2N-doped leg to form the TE uncouple, a stable TE-generating device is formed in uncouple sizes up to 1 We. It has the same basic operating temperatures as PbTe — 811 K (1000°F) hot shoe, 422 K (300°F) cold shoe.^{IV-108} TAGS TE technology was used in the SNAP 19 RTG packages for the Pioneer 10 and 11 missions and on the Viking-Mars landing missions. In actual performance on board Pioneer 10 and 11, the SNAP 19 RTG generators showed less than a 15% degradation in energy conversion efficiency of the TE converters in over 10 years of continuous duty in space. The TAGS uncouple technology has a BOL combined figure-of-merit Z of $1.22 \times 10^{-3} \text{K}^{-1}$ and has demonstrated greater stability than the PbTe technology at comparable hot- and cold-shoe temperatures. The TAGS subsystem does require using an inert atmosphere (Ar-He) at 20-MPa (2-atm) pressure to limit uncouple degradation over long power duty cycles.

Application of the TAGS technology to space reactor PCSs would most likely require a compact tubular module with a design similar to that of the previously discussed PbTe module. Figure IV-122 illustrates the comparative performance of these TE technologies in RTG applications.^{IV-109} The total TAGS RTG uncouple experience includes over 1,000,000 operating hours in RTGs launched into space. In the present design, the TAGS RTG PCS is approximately 85% heavier than the PbTe module previously discussed, but it has an overall device efficiency that is 22% greater. A 328-W electric power module is estimated to have a mass of approximately 12 kg (26 lb) for a PCS using the TAGS RTG TE converter technology.^{IV-108} Section IV.D.4 discusses the application of the PbTe and TAGS TE technologies to potential future space reactor power missions.

(2) Thermionic Power Conversion

The active development of thermionic converters for space nuclear power applications was pursued in the United States from 1958 to 1972 and in the USSR from the early 1960s to the present. Thermionic converters are static power conversion devices that directly convert heat energy into electricity by a process of boiling electrons from a hot emitter surface (~ 1800 K) across a small interelectrode gap (~ 0.5 mm) to a cold (~ 1000 K) collector plate. Figure IV-123 illustrates the operating principles, typical operating parameters, and component materials of a practical thermionic converter.^{IV-110} The vacuum tube diode is an example of thermionic technology used to control the voltage in an electric circuit. In a thermionic converter, the voltage potential established across the hot emitter and the cold collector surfaces is used to drive a current through an externally connected electrical load. The interelectrode gap in vacuum is too small for practical converters and low pressure 130 Pa (approximately 1 mm of Hg) cesium vapor was introduced into the interelectrode gap to build the first practical thermionic converters.

Figure IV-122. SNAP-19 Performance Improvements and Projections

TYPICAL OPERATING REGIME

EMITTER TEMPERATURE: 1600-2000 K (2420-3140°F)
 COLLECTOR TEMPERATURE: 800-1100 K (980-1520°F)
 ELECTRODE EFFICIENCY: UP TO 20%
 POWER DENSITY: 1-10 W/cm²

MATERIALS

EMITTER MATERIALS: W, Re, Mo
 COLLECTOR MATERIALS: Nb, Mo
 INSULATOR MATERIALS: Al₂O₃, Al₂O₃/Nb CERMET
 ELECTRODE ATMOSPHERE: Cs AT 1 Torr

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Figure IV-123. Thermionic Converter Schematic

These devices are high-current-density, low-voltage generators. Figure IV-124 presents the current-voltage characteristics for a thermionic converter.^{IV-111} Their performance is limited by the peak voltage potential that can be maintained across the interelectrode gap. All thermionic converters can be rated in their energy conversion efficiency performance by an inverse figure-of-merit known as the barrier index, V_B . In 1960, the barrier index was about 2.9 eV, and this yielded a device efficiency of only 10% at 2073 K. By 1970, the barrier index had been reduced to about 2.1 eV, corresponding to a device efficiency of 35% at 2073 K.

The barrier index may be considered as the difference between a practical thermionic converter's developed voltage at the external circuit leads (~ 0.5 V) and the theoretical voltage possible (2.5 V for the Boltzmann line in Figure IV-125) for the device with a collector plate temperature at 0 K. For thermionic converter designs used within a nuclear reactor core, the collector plate material has generally been niobium operating at 900 to 1000 K. Approximately 1.35 to 1.5 eV of the barrier index was not recoverable. This nonrecoverable voltage potential is more commonly known as the collector work function for the thermionic converter. Another 0.5 eV of the barrier index is made up of electron losses in the cesium plasma arc drop across the interelectrode gap. This lost voltage potential can be thought of as the fraction of electrons leaving the emitter surface that never arrives at the thermionic converter's collector surface. This results in the present state-of-the-art in practical thermionic converters with a barrier index $V_B = 2.0$ eV. Figure IV-125 illustrates the comparative performance of the thermionic converters with TE converters. With a lower barrier index, the thermionic converter can produce more power at the same emitter-collector temperatures or operate at lower electrode temperatures for the same power level, as illustrated in Figure IV-126.^{IV-104}

a.

b.

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Figure IV-124. Voltage-Current Characteristics for a Thermionic Converter

Figure IV-125. Performance Comparison between Thermionic and Thermoelectric Converters

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Figure IV-126. Characteristic of Thermionic Converters
versus Barrier Index, V_B

(a) GA Technologies In-Core Thermionic Converter

The first application of thermionic converters to nuclear reactor heat sources was made in the U.S. thermionic reactor research and development program. The primary objective of this program was to demonstrate the feasibility of an in-core thermionic reactor design. This program resulted in the development of a successful in-core thermionic converter design but was terminated in 1973 due to funding limitations. At the time of termination, a total of about 65,000 h of in-core testing of thermionic converters had been compiled by General Atomics. Table IV-39 (taken from Ref. IV-112) presents typical results of the in-core TFE tests performed by GA. A total of 38 carbide and oxide fueled thermionic devices were life-tested in-core from 1965 to 1972. They included (1) 15 MK6 experimental converters containing emitters 1.58 cm (5/8 in.) in diameter and 2.54 cm (1 in.) long; (2) six prototype converters, three (MK7A) containing emitters 1.58 cm (5/8 in.) in diameter and 5.08 cm (2 in.) long, and three (MK7B) containing emitters 2.8 cm (1.1 in.) in diameter and 5.08 cm (2 in.) long; (3) three partial length thermionic fuel elements containing MK7A converters; (4) seven partial length thermionic fuel elements containing MK7B converters; (5) five full-length thermionic fuel elements containing six MK7B converters each. Figures IV-127 through IV-129 illustrate a GA-designed converter, a typical large-diameter TFE design, and a thermionic "flashlight" reactor design developed during this program.

Of the in-pile tests, the 2E series (the Mark 7A series as designated by GA), with a 1.58-cm- (5/8-in.-) diameter, 5.1-cm- (2-in.-) long emitter were the most successful. These TFEs used UO_2 fuel, a CVD-constructed tungsten cup, cesium working fluid in the interelectrode gap at a 600 K reservoir temperature, a niobium collector, a Al_2O_3 (Lucalox) insulator, and a Nb-1% Zr clad. The Mark 7A series tests indicated a 12,000-h in-core lifetime (see Figure IV-130).

TABLE IV-39
SUMMARY OF THERMIONIC FUEL ELEMENT TESTS

Type of Test	Device Designation	Fuel	Emitter Temperature (K)	Average Electrode Power (W/cm ²)	Testing Time (h)	Burnup (10 ¹⁹ fission Δ/cm ³)	Fast Neutron Fluence (10 ²⁰ nvt) (>0.1 MeV)	Failure Mode of Converter
Out-of-pile unfueled	LC-9	--	1970	8.0	46647	--	--	Still operable
Out-of-pile fueled	LC-11	90UC-10ZrC-4W	1870	6.5	18569	--	--	Still operable
	LC-3	W-UO ₂ cermet	2000	8.9	10406	--	--	Electrode short by Ti getter ring
Capsule Irradiation	V-2C	90UC-10ZrC-4W	1873	--	11000	30	--	--
	FC-3	UO ₂	1910	--	10244	12	9	--
In-pile MK6 experimental converter	I-4	UO ₂	1840	6.0	8754	31	4	Envelope leak
In-pile MK7B prototype converter	C11	90UC-10ZrC	1840	3.4	7881	6.8	3	Emitter crack
In-pile MK7A partial fuel element	2E-1	UO ₂	1820	4.0	12534	24	9	Emitter swelling
	2E-2	UO ₂	1820	5.6	11084	22	8	Still operable
In-pile MK7B partial length fuel element	1F-1	UO ₂	1780	4.0	8560	6.6	4	Emitter swelling
In-pile MK7B full length fuel element	6F-2	90UC-10ZrC-4W	1780-1950	2.3	7685	6.6	4-8	Emitter cracking
	6F-3	UO ₂	1740-1820	3.0	8062	7.3	2.7-5.9	Still operable

Taken from Ref. IV-112

COMPONENT	OPERATING TEMPERATURE, K
DUPLEX EMITTER	1881 (AVERAGE)
TRILAYER	983 (COLLECTOR AVERAGE)
TANTALUM-TUNGSTEN DIFFUSION BOND	1262 -
CERAMIC-TO-METAL SEAL	1078 (CERAMIC MAXIMUM)
EMITTER CAP	1213 (CENTER)
TRANSITION PIECE	1153 (AVERAGE)
SHEATH TUBE	973
ALIGNMENT SPRING	1773 (MAXIMUM)

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Figure IV-127. Arrangement of Components and Operating Temperatures for a Thermionic Converter (GA Technologies F Series Cell)

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Figure IV-128. Typical Thermionic Fuel Element (TFE)

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Figure IV-129. Liquid Metal Cooled "Flashlight"
Thermionic Reactor Design

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Figure IV-130. Life Test Record of TFE-2E2 In Core

Life-limiting features in the design were UO_2 fuel swelling with consequential emitter swelling, and Al_2O_3 insulator swelling due to in-pile fast neutron fluence. The insulator swelling problem was subsequently identified as a 20,000-h life-limiting process.^{IV-113} Two new insulator materials were identified, Y_2O_3 and $\text{Al}_2\text{O}_3/\text{Nb}$ cermet, that showed less swelling than Al_2O_3 in fast neutron irradiation tests. However, both of these materials may suffer from electrical leakage problems under combined irradiation and electrical fields over long exposure periods. The program was terminated before these materials could be extensively tested to determine their suitability as a substitute for the Al_2O_3 (Lucalox) insulator material.

GA Technologies proposed using ZrH to moderate the reactor and reduce the fast neutron flux by a factor of two to three, and thus increase the insulator lifetime by the same amount. However, a moderated core design resulted in a large penalty in U^{235} inventory due to the use of tungsten, tantalum, and rhenium in the TFEs. The moderated or partially moderated core design indicated that molybdenum thermionic emitters would be the preferred choice. In any event, such a design should minimize the total amount of tungsten required within the core.

(b) General Electric In-Core Thermionic Converter

The General Electric in-core thermionic converter research and development effort was substantially smaller than the GA Technologies effort. The GE program had a limited objective of eliminating converter failure due to UO_2 fuel swelling effects. The GE-designed converter used UO_2 fuel disks with molybdenum foils interspaced between the disks in a 1-cm-diameter molybdenum emitter capsule with no vent for the fission gases. The UO_2 fuel swelling problem, primarily due to fission gas and volatile solids (Cs) vaporizing in the fuel, was resolved when these tests on an emitter designed to contain the fission gas failed. The emitter cracked and released the

fission gases and vapors Xe, Kr, and Cs into the interelectrode gap at 3300 h into the test. The performance of the converter temporarily deteriorated until the operating parameters of the converter were readjusted. The converter then performed up to 6900 h at 10% less power output until the program was terminated due to funding termination. This test indicated that venting the fission gases directly into the interelectrode gap would not significantly degrade converter performance.

The thermionic reactor program was a technical success. It was terminated for lack of a mission and lack of general funding for space nuclear power research. The thermionic reactor program through 1972 in the United States is reviewed in Ref. IV-107.

(c) USSR TOPAZ In-Core Thermionic Converter

After the funding cutoff in the United States, in-core thermionic reactor research shifted to the USSR with the development of the TOPAZ series of reactors. Ref. IV-107 describes this effort in as much detail as was known in 1974. A short discussion follows.

As far as can be determined, the Soviet thermionic effort began following the visit of some Soviet academicians to LANL in 1959; prior to that time, U.S. thermionic reactor development had been classified. For several years, the Soviets pursued the initial LANL approach of using unclad UC-ZrC emitters. Presumably they encountered the same difficulties as LANL with this approach and shifted to UO_2 -fueled molybdenum emitters. Thereafter, they used molybdenum- UO_2 fueled emitter configurations in a small-diameter fuel rod and venting of fission gases through the interelectrode gaps, moderating with ZrH_x rods of a size comparable to the TFE size, and moved quickly to the construction of an experimental reactor without concern about the lifetime of the first core (1000 h or 10,000 h) and without insistence that the full potential output power be achieved on the first try.

A major milestone was achieved by the Soviets in early 1971 when they tested the world's first thermionic reactor called TOPAZ (see Figure IV-131). Since that time, two additional thermionic reactor cores have been tested, each more successful than the last with one core attaining a lifetime approaching 5000 h. The emitter construction of the second core, which ran at full design electrical power of 10 kWe for 1200 h and half power for approximately 4000 h, consisted of alternate wafers of UO_2 and molybdenum in a molybdenum cup 10 mm in diameter. The emitter power flux was held at the low value of 1 We/cm^2 . The main thermionic elements were connected into six parallel circuits, each containing ten power channels. The output voltage was in the range of 20 to 25 V.

9467-103

Figure IV-131. USSR "TOPAZ" Thermionic Reactor:
15 to 20 kWe

(d) Out-of-Core Thermionic Converter

Thermionic development efforts in the United States continued at a low level with terrestrial applications of thermionic converters. Preliminary

results from the GA technologies series of tests on out-of-pile thermionic converter performance and life tests had indicated a life expectancy of 2 to 5 years. Figure IV-132 presents extended life performance test results for the LC-9 converter.^{IV-113} Subsequent tests since 1973 on thermionic converters for terrestrial applications have confirmed the potential for 5-year life of out-of-core thermionic converters. Development work on thermionic converter performance for terrestrial applications has significantly improved converter efficiency, as is evident from the 50% improvement in converter efficiency shown in Figure IV-124. This improvement was the result of introducing WO_2 onto the surface of the collector element. Improvements in thermionic converter performance (i.e., reduction in the barrier index V_B) resulted in a larger interelectrode gap spacing and a lower emitter temperature for the same conversion efficiency. This research has concentrated on lowering the collector work function by using different collector materials or different coatings for the collector material. Steady progress has been made in lowering the barrier index V_B in the last few years. Figure IV-133 illustrates the progress made in this area.^{IV-111} The effort in reducing the collector work function has been taken to the point shown by Figure IV-134 with a thermionic converter barrier index V_B of 1.87.^{IV-111} This is an advanced converter design and its long-term performance characteristics have not yet been proven by laboratory tests.

This technology has no direct bearing on in-core thermionic converters because of the choice (for nuclear reasons) of niobium for the collector material. The barrier index V_B will remain at 2.0 eV for the initial in-core thermionic converters. The advanced technology in terrestrial thermionic converters has a direct application to space nuclear electric power subsystems when combined with a high-temperature reactor that uses heat pipes to remove core thermal power. A proposed design for an out-of-core thermionic converter/core heat pipe power conversion device is shown in Figure IV-135.^{IV-114}

CONVERTER: LC-9
INPUT POWER: 1 kWt
HEAT SOURCE: ELECTRICAL
FAILURE: NONE

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Figure IV-132. Extended Life Performance Tests for a Thermionic Converter

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Figure IV-133. Progress in Barrier Index Reduction for Thermionic Converters
(Performance Improvement)

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Figure IV-134. An Advanced Design of a Terrestrial Thermionic Converter

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Figure IV-135. Out-of-Core Thermionic Converter Concept

This concept separates the thermionic converter design from the restrictions imposed on any converter located within the environment of a nuclear reactor core. It frees up both the converter and reactor designers to use the materials and designs that produce the best overall design for their individual components. More importantly, the primary life-shortening processes that were present in the in-core thermionic converter tests are not present in the out-of-core concept. The concept of driving a thermionic converter using high-temperature heat pipes has recently been demonstrated in the laboratory.^{IV-104} Figure IV-136 illustrates the laboratory equipment used in a dual heat-pipe thermionic converter design for the out-of-core thermionic converter concept. On a laboratory basis, the out-of-core thermionic reactor concept seems plausible using a high-temperature heat pipe reactor concept. Ref. IV-114 provides a systems analysis of this concept.

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Figure IV-136. Detail of the Out-of-Core Thermionic Converter Laboratory Test Apparatus

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(3) Thermophotovoltaic Power Conversion

Thermophotovoltaic (TPV) subsystems offer a viable approach to increasing the efficiency of thermal-to-electric conversion. A few analyses of solar thermal-to-electric TPV subsystems^{IV-104, IV-105} have been reported in the literature. The general conclusion of these analyses is that device efficiencies on the order of 51% are theoretically possible with silicon cells. Basically, the solar TPV system consists of a cavity having a highly reflective inner surface. Over this reflective surface are placed photovoltaic cells. Suspended near the center of the cavity is an absorber/radiator converter. This converter absorbs the concentrated solar energy and heats up to an equilibrium temperature of 2000 to 2300 K (3140 to 3680°F). It then radiates energy to the photovoltaic cells. An elliptical cavity solar TPV design has been proposed^{IV-104} to eliminate the curvature problem associated with mounting a flat plate cell structure to a focusing surface. This concept is shown in Figure IV-137.

A recent study has indicated that tungsten emitters operating at 2000 K (3200°F) would produce an overall improvement in TPV performance compared to the idealized black body radiator/emitter. This effect results from the relative amount of energy emitted at the longer wavelengths ($>1.0 \mu\text{m}$) being significantly lower for the tungsten emitter and the relative amount of energy emitted in the spectral range useful to silicon cells ($\leq 1.0 \mu\text{m}$) being higher. This potentially would allow using a very high temperature tungsten heat pipe reactor as the emitter or as a driver for the emitter in large TPV designs.

d. PCS Summary

Key information about dynamic and static PCSs is presented in Tables IV-40 through IV-42.

9467-114

Figure IV-137. Elliptical Cavity Optics for
Solar Thermophotovoltaics

TABLE IV-40
SUMMARY OF DYNAMIC POWER CONVERSION TECHNOLOGY

Description	Peak Temperature [K (°F)]	Pejection Temperature [K (°F)]	Working Fluid	Fluid Pressure [MPa (psi)]	Reference Power (kW)	Subsystem Efficiency (%)	Device Efficiency (%)	Mass/Power [kg/kW (lb/kW)]	Flow Rate [kg/s (lb/s)]	Component Test Time (h)	Subsystem Test Time (h)	Alternator	Bearing Lube	Technology Rating
SNAP-2 mercury Rankine	895 (1150)	590 (600)	Hg	0.79 (115)	5.0	6.3	18.5	38 (84)	0.18 (0.40)	35,000	21,000	Permanent magnet 1800 Hz	Hg	8.0
SNAP-8 mercury Rankine	980 (1306)	510 (456)	Hg	1.73 (251)	35	10.3	21.5	33 (73)	1.45 (3.19)	420,500	12,200	Homopolar	Polyphenyl ether	8.0
6-kWe organic Rankine	617 (650)	411 (280)	Dow. "A"	0.61 (89)	6.0	17.6	52.7	--	0.071 (0.156)	28,000	3,000	Homopolar	Dow. "A"	8.0
Dynamic Isotope Power System organic Rankine	645 (700)	350 (168)	Dow. "A"	0.55 (80)	1.3	18.8	41.1	52 (115)	0.011 (0.025)	20,000	11,000	Homopolar	Dow. "A"	8.0
NASA potassium Rankine ^a	1420 (2100)	935 (1220)	K	1.66 (240)	375	18.7	54.8	9.58 (21.1)	1.48 (3.25)	19,200	0	Homopolar 1600 Hz	K	4.5
Medium Power Reactor Experiment potassium Rankine	1110 (1540)	778 (940)	K	0.21 (30)	367	16.7	55.8	5 (11)	6 (4)	160,000		--	K	4.7
Advanced MPRE potassium Rankine ^a	1365 (2000)	1003 (1345)	K	1.55 (225)	376	17.1	64.5	5.08 (11.2)	6.14 (13.5)	18,300	0	--	K	2.8
NASA B engine Brayton	1145 (1600)	300 (80)	He-Xe M = 83.8	0.283 (41)	10.5	30	40.7	26.7 (58.7)	1.96 (4.32)	41,000	41,000	Modified Lundell 1200 Hz	He-Xe	8.0
Turboalternator-compressor (TAC) Brayton	1145 (1600)	390 (240)	He-Xe M = 39.9	0.88 (128)	105	30.9	46.9	3.6 (8)	3.07 (6.75)			Rice 2-pole		3.3
Closed cycle Brayton (Mini-BRU BIPS)	1025 (1385)	267 (20)	He-Xe M = 83.8	0.434 (63)	1.35	27.0	36.5	59.0 (130)	0.105 (0.23)	1,100	1,100	Rice 2-pole	He-Xe	7.1
Space free piston Stirling 10- to 50-kWe	978 (1300)	478 (400)	He	6.0 (900)	51 (4 units)	36	67	7.84 (17.3)	--	0	0	Linear 46 Hz	He	1.0
Terrestrial free piston Stirling	922 (1200)	303 (86)	He	7.0 (1000)	4.0	38.0	56.6	-	--	16,000	2,000	Linear 60 Hz	He	7.2

^aNot run as a subsystem

TABLE IV-41
SUMMARY OF THERMOELECTRIC CONVERTER TECHNOLOGY

Description (Reference)	Hot Junction Temperature (K)	Cold Junction Temperature (K)	In-Use System		Efficiency (BOL) (%)		Device Efficiency (%)	Indicated Figure-of- Merit Z (10^{-3}K^{-1})	Converter Specific Mass (kg/kWe)	Test Time (10^3h)	Technology Rating
			RTG	Reactor	System	Unicouple					
SNAP-10A SiGe (IV-145)	772	597		X	1.5	2.5	10.9	0.7	120 ^a	10	10
RTG SiGe (IV-105)	1273	573	X		6.6	8.9	16.3	0.7	19.9	13	10
RTG ternary doped SiGe lab ^b (IV-104)	1273	573			NA	10	18.8	0.85	16.4	<0.002	4
SABRE converter (IV-115)	1355	840	X	X	6.8	8.2	21.7	1.0	2.18	NA	2.5
SNAP lead telluride compact converter	858	537		X	5	5.9	15.7	1.0	20	8	8
RTG TAGS (IV-109)	822	469	X		6.3	7.6	17.6	1.22	37	10	10

NA = Not applicable

^a Converter/radiator specific mass

^b Laboratory tests only

TABLE IV-42
SUMMARY OF THERMIONIC CONVERTER TECHNOLOGY

Description (Reference)	Hot Junction Temperature (K)	Cold Junction Temperature (K)	Working Fluid and Pressure	In-Use System		Efficiency (%)	Power Density (W/cm ²)	Device Efficiency (%)	Barrier Index (V_B) (eV)	Converter Specific Mass (kg/kWe)	Test Time (10^3h)	Technology Rating
				RTG	Reactor							
GA in-core (IV-112)	1820	983	Cs ~3 Torr		X	10	5	22	2.1	5.1 ^a	0.235 ^b	5
GE in-core	1820 (est)	980 (est)	Cs ~3 Torr (est)		X	Unk	Unk	18-20 est	Unk	Unk	0.01	4
USSR TOPAZ in- core (IV-116, IV-117)	1873	~825	Cs ~10 Torr		X	7	1.2	~12.5	>2.3	~11 ^c	>70	8
Out-of-core (IV-104)	1800	880	Cs <1 Torr		X	13	6	25	<2.0	2.9	<0.066	2

^a Fueled emitters in TFE

^b 65,000 total test hours on all in-core modules

^c Estimated with fueled emitters in power channel

4. Technology Assessment

Using the information acquired in preparing Section IV.D.3, a technology assessment of various PCSs was performed. This assessment was chiefly to determine when the PCS will be through development test and flight-qualified, and what the subsystem characteristics might be. It was assumed that the work would start in FY 1984 and that reasonable and adequate funding would be available.

Generic types of PCSs were used for this assessment. For each type, logical technology steps are shown pictorially (see Section III.A). Each step is defined by the limits that should be achievable on temperature, device efficiency (η_D) range, life, and electric power output.

a. Turboelectric

(1) Mercury Rankine Turboelectric PCS

Figure IV-138 presents the technology steps versus time. A mercury Rankine unit, which is like the basic NASA Aerojet SNAP-8, might be flight qualified by FY 1988 or shortly thereafter. This PCS would be up to a 35-kWe design with a device efficiency of 22 to 25%. In the SNAP-8 final report,^{IV-124} it was shown that a 90-kWe design having a device efficiency of 28 to 32% could be derived from SNAP-8. Also, with some development, a 120-kWe design having a device efficiency of 36 to 40% could be expected.

The turbine efficiency of the 35-kWe SNAP-8 was only 57%, whereas an improved version (Mark 70 turbine) had an efficiency of 78%. In addition, the turbine back pressure, which was 0.1 MPa (14 psia) in the SNAP-8 configuration, was reduced to about 0.014 MPa (2 psia) in later versions. Together

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Figure IV-138. Subsystem Technology Assessment – Mercury Rankine Turboelectric PCS

with normal developments in other components and overall PCS optimization, devices efficiencies of 28 to 32% could be expected by FY 1992 and 36 to 40% in a 120-kWe engine by FY 1996.

Mercury Rankine PCSs have the advantage of producing fairly high performance at moderate cycle temperatures along with small radiator areas. These advantages, perceived as long ago as 1956, are still valid. However, mercury is a particularly corrosive fluid and boiler wetting is a problem. About \$350 million (1983 dollars) were spent on mercury Rankine PCSs in an extensive development program; many real problems were identified and some of them resolved.

The following technology issues are associated with the mercury Rankine turboelectric PCS (see Figure IV-138):

Technology Step 1

$T = 980\text{ K}$
(1300°F)
 $\eta_D = 22\text{-}25\%$
LIFE = 1 YEAR
 $P_e = 35\text{ kWe}$

Key Issues - The technology base from the SNAP-8 mercury Rankine PCS needs to be reestablished.

Resolution - Use the NASA Aerojet SNAP-8 unit design to build and qualification test the unit. Estimated time required is 3-1/2 years to resurrect the design, update it, and build a 35-kWe test unit. With a 1-yr test period, the program could be completed by mid 1988.

Technology Step 2

$T = 980\text{ K}$
(1300°F)
 $\eta_D = 28\text{-}32\%$
LIFE = 2 YEARS
 $P_e = 90\text{ kWe}$

Key Issues - Turbine efficiency needs to be improved through improved design, larger unit power, and extended (2-yr) design lifetime. Boiler conditioning problems must receive a final successful resolution.

Resolution - Design improved efficiency turbine (90-kWe unit), fabricate the unit, and test with its boiler for 2-yr design lifetime. Using technology step 1 as a springboard, approximately 4 more years beyond mid 1988 are estimated to be required. Program completion is estimated for mid 1992.

Technology Step 3

Key Issues - Continued improvement is needed toward higher turbine efficiency and lower exhaust pressure. Unit rating needs to be increased to 120 kWe with an extension in design lifetime to 3 years.

Resolution - Follow same approach as technology step 2, with approximately a 1- to 2-yr overlap with step 2. Testing for the full 5-yr design lifetime is required, giving a program completion by mid 1996.

(2) Organic Rankine Turboelectric PCS

The organic Rankine PCS could be flight qualified in about 4 to 5 years. As shown in Figure IV-139, this would be a fairly conservative design, initially having a temperature of 630 K (675°F), a device efficiency of 42 to 45%, a life of 1 year, and a power of 20 kWe. The DIPS development program tends to readily support such a forecast. By an orderly progression over the ensuing 10 years (FY 1989 to FY 1999), it is forecast that temperatures could be increased to 670 K (745°F), the device efficiency could be raised to 56 to 58%, the life could be extended to 4 years, and power levels of at least 200 kWe could be achieved.

The organic Rankine cycle possesses a major advantage in that the cycle temperatures are low. This will tend to reduce costs, reduce risk, and shorten schedules. On the other hand, it is a two-phase cycle, so that it is possible that startup, shutdown, and other transient operations could produce some problems.

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Figure IV-139. Subsystem Technology Assessment – Organic Rankine Turboelectric PCS

As noted in Figure IV-139, the expected increase in organic Rankine cycle performance will result from increases in cycle temperatures, improvements in component efficiencies, and the benefits of going to larger sizes. However, organic fluids are limited in their temperature growth potential. Dowtherm A, for example, probably should not be used above 672 K (750°F).

The following technology issues are associated with the organic Rankine turboelectric PCS (see Figure IV-139):

Technology Step 1

$T = 630 \text{ K}$
(675°F)
 $\eta_D = 42-45\%$
LIFE = 1 YEAR
 $P_e = 20 \text{ kWe}$

Key Issues and Resolution - The organic Rankine subsystem received attention in the 1960s and 1970s; this step is basically a reestablishment of that technology. This should be possible within 5-1/2 years, given a 4-1/2-yr design and fabrication cycle and a full 1-yr test, enabling completion by mid 1989.

Technology Steps 2, 3, and 4

$T = 640 \text{ K}$
(690°F)
 $\eta_D = 47-51\%$
LIFE = 2 YEARS
 $P_e = 50 \text{ kWe}$

$T = 660 \text{ K}$
(725°F)
 $\eta_D = 52-55\%$
LIFE = 3 YEARS
 $P_e = 100 \text{ kWe}$

$T = 670 \text{ K}$
(745°F)
 $\eta_D = 56-58\%$
LIFE = 4 YEARS
 $P_e = 200 \text{ kWe}$

Key Issues - The maximum operating temperature of the organic Rankine PCS is limited to 670 K by thermal decomposition of the organic fluid and by bearing technology.

Resolution - Each technology step builds on the successes and experience of its predecessor. Gradual increases in device efficiency, lifetime, and unit power can be achieved from step to step. Only moderate increases in operating temperature are foreseen. The timeframe for each step is estimated to be 3 years, requiring some overlap between design of each successive unit relative to the testing of the preceding unit. Program completion should be possible by mid 1998 following this logical development progression. (NOTE: Bigger development steps could be taken if higher risk is assumed to be acceptable.)

(3) Potassium Rankine Turboelectric PCS

The technology progression is shown in Figure IV-140. The potassium Rankine cycle has very high device efficiency and characteristics that can lead to small radiator sizes and compact power systems. Potassium appears to avoid the corrosion and boiler wetting problems encountered in the mercury Rankine cycle. On the other hand, the full potential of the potassium PCS is only realized at higher temperatures ($>1365\text{ K}$ (2000°F)), where the cost and risk would probably be high.

The MPRE potassium cycle at 1110 K (1540°F) and a device efficiency of 55% could probably have a lifetime of 1 year at a power of 25 kWe by 1990. Higher temperatures could be obtained later at some initial loss in device efficiency. By the mid to late 1990s, PCSs such as the ORNL advanced MPRE and the NASA design (at 1365 K (2000°F) and 1420 K (2100°F), respectively) could be achieved. However, those PCSs would have only a short to moderate lifetime, chiefly because of the very high cycle temperatures.

When the gains achieved by going from 1110 K (1540°F) to 1365 K (2000°F) are considered in the light of the probable greater risk, it can only be concluded that a moderate temperature cycle is the most reasonable first step. Note also that potassium freezes at 336 K (145°F), which could cause some operational problems.

The following technology issues are associated with the potassium Rankine turboelectric PCS (see Figure IV-140):

Technology Step 1

Key Issues and Resolution - This step basically involves the MPRE data base. Resurrecting that design and fabricating and testing the unit should be possible by mid 1989.

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Figure IV-140. Subsystem Technology Assessment – Potassium Rankine Turboelectric PCS

Technology Step 2

$T = 1280 \text{ K}$
(1845°F)
 $\eta_D = 45\text{-}50\%$
LIFE = 1 year
 $P_e = 20\text{-}30 \text{ kWe}$

Key Issues - To realize the full potential of the potassium Rankine cycle, high operating temperatures must be achieved. This is the first step in that realization and will require the use of refractory materials in the system.

Resolution - This step is basically springboarded from the advanced MPRE subsystem. It will be necessary to design and fabricate the refractory components and conduct a full term test. It is estimated that this step could be accomplished within approximately 3 years of the completion of step 1 (i.e., program complete by late 1994).

Technology Step 3

$T = 1330 \text{ K}$
(1935°F)
 $\eta_D = 52\text{-}55\%$
LIFE = 2 years
 $P_e = 50 \text{ kWe}$

Key Issues - Building on the results of step 2, components will be scaled up, efficiencies increased through use of improved component designs, and temperatures (relative to step 2) raised somewhat.

Resolution - This step is a logical progression from step 2 and is estimated to require an additional 3 years for development and a full 2-yr test time; program completion is expected by mid 1977.

Technology Step 4

$T = 1380 \text{ K}$
(2025°F)
 $\eta_D = 56\text{-}58\%$
LIFE = 3 years
 $P_e = 100 \text{ kWe}$

Key Issues - Issues are the same as for step 3, with components to be further scaled up (i.e., higher power output units), temperatures increased, efficiency improved through improved component designs based on step 3 experience and increased lifetime.

Resolution - This is a logical progression of the technology from step 3 and is estimated to require an additional 3 years (2-yr test time judged to be a reasonable risk based on the extensive data base built on steps 1, 2, and 3) to finish; program completion is expected by mid 2000.

(4) Closed-Cycle Brayton

Ref. IV-125 indicates that the long-term development/test effort on the Garrett/NASA Brayton PCS for space power constitutes substantial verification of Brayton turbine technology in the 0.5- to 10-MWe size range. In addition, commercial gas turbine experience indicates that there would be no technological impediment in scaling to a 100-MWe size in a single machine. Subsystem size, therefore, does not appear to be a limiting factor in development.

Cycle performance will be limited by turbine inlet temperatures. In the case of the closed Brayton cycle, it does not appear that using actively cooled turbine concepts will be compatible with the low pressure loss requirements. Turbine performance is, therefore, limited by the development of materials to permit operation at higher turbine inlet temperatures.

Figure IV-141 shows projected improvements for the closed-cycle Brayton PCS. By FY 1988, a conservative design at 1100 K (1520°F), 38 to 42% device efficiency, 2-year life, and 30-kWe power output could be flight qualified. With the improvements noted, the PCS could reach 1400 K (2060°F), 53 to 56% device efficiency, 4-year life, and 200 kWe by about FY 2000. At lower operating temperatures, a greater subsystem life would be expected.

Minor improvements in turbine, compressor, and heat exchanger/recuperator efficiency can result from relatively extensive development. Improvements in turbine inlet temperature improve performance by increasing Carnot efficiency, while improvements in turbine engine component efficiency result in improvements in overall device efficiency. Current turbine designs have achieved

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Figure IV-141. Subsystem Technology Assessment – Closed-Cycle Brayton Turboelectric PCS

expansion efficiencies close to 92%. Improvements of from 1.0 to 1.5% may be achievable with extensive development of a particular design. Such development could be achieved in 3 to 5 years.

The effectiveness of heat exchangers/recuperators in space application of the closed Brayton cycle has usually optimized at approximately 95%. This limit usually results from mass and pressure loss considerations. Improvements in design, fabrication, and cooling techniques could add as much as 3 percentage points to this value without a significant penalty in mass or pressure loss. These gains could be realized in 5 to 7 years with an aggressive development program.

Compressors can currently achieve adiabatic efficiencies of as high as 88%. Improvements in diffuser design, impeller geometry, and duct/inducer aerodynamic matching could result in gains of from 1.5 to 2.0%. These could be realized in 3 to 5 years with the aggressive development of a particular design.

The following technology issues are associated with the closed cycle Brayton turboelectric PCS (see Figure IV-141):

Technology Step 1

Key Issues - This technology step is basically an extension of prior work done on the NASA B engine, TAC unit, and Mini-BRU (BIPS).

Resolution - Despite the bearing problems encountered with the BIPS unit, it is believed that the Brayton technology exists to allow design, fabrication, and testing of a 30-kWe unit for 2 years at 1100 K by mid 1988 (4-1/2 years elapsed time).

Key Issues - These steps can be achieved by using ever improving materials (niobium alloys-to-refractories-to-ceramics), allowing an increase in operating temperatures. This will also result in some increase in device efficiency. Other increases in efficiencies will result from improvements in component designs (e.g., compressors and recuperators). Design lifetimes will be improved as will the benefits of scale (i.e., larger power units) in each step.

Resolution - Steps 2, 3, and 4 are logical progressions of Brayton development, each step building on the experience from prior steps. Materials changes are believed possible based on the time frame between steps (i.e., 4 years). Some overlap between design of a succeeding unit with testing of the preceding unit is seen as an acceptable risk. The overall program should be completed by the year 2000.

(5) Open-Cycle Brayton

An assessment of the open cycle Brayton technology is given in Figure IV-142. This assessment relies solely on the status of the technology of commercial gas turbines and airborne high-power turbine drive systems. Both technologies are well verified but only partly applicable to space power systems. Primary elements that will require careful development are shaft support bearings, effects of mechanical and thermal transients, power transmission, and methods of cooling surfaces to allow higher inlet temperatures. It

Figure IV-142. Subsystem Technology Assessment – Open-Cycle Brayton Turboelectric PCS

is estimated that the initial time required to develop these aspects of the turbomachine will be approximately 3 years. Subsequent development should be compatible with the schedules indicated for the closed Brayton cycle machines. There appears to be no technological impediment to the development of turbomachines for very high outputs.

Development can proceed along the same lines as discussed for the closed-cycle Brayton system except that actively cooled turbine elements could be developed in parallel with ceramic turbines. Active cooling is more attractive in the open-cycle case, since the cycle is not as sensitive to pressure loss as is the closed cycle.

The following technology issues are associated with the open cycle Brayton turboelectric PCS (see Figure IV-142):

Technology Step 1

Key Issues - Development of this open cycle Brayton will borrow heavily from aircraft APU technology and bearing development previously performed on the closed cycle Brayton.

Resolution - Design, fabrication, and testing of this unit will occur slightly later in time than the corresponding first step for the closed cycle Brayton since more adaptation (from aircraft technology) will be required. Completion of this first step by mid 1990 should be possible.

Technology Steps 2, 3, and 4

Key Issues - The issues associated with technology development steps 2, 3, and 4 for the open cycle Brayton are essentially the same as for the closed cycle Brayton and follow the same time frame. The single exception is the possible development of actively cooled turbines for step 4 of the open cycle Brayton development. This option could be developed either in parallel with the ceramic turbine or instead of it.

Resolution - Same as the closed cycle Brayton, with approximately the same completion time(s).

b. Reciprocating

(1) Stirling

The technology assessment is shown schematically in Figure IV-143. About 6 years would be needed to flight qualify a free piston Stirling engine. By FY 1990, such an engine would have a cycle temperature of 970 K (1285°F), a device efficiency of 60 to 64%, a life of 1.0 year, and a power level of 16 kWe. This engine could consist of four existing 4.0-kWe engine modules. By FY 1996, cycle temperatures could be raised to 1050 K (1430°F), device efficiency to 70 to 75%, life to 7.0 years, and the power level to 100 kWe. This would be accomplished by using higher cycle temperatures, by advancing component technology, and by reaping the benefits of scaling.

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Figure IV-143. Subsystem Technology Assessment – Stirling Cycle Reciprocating PCS

Stirling engines can produce high device efficiencies while requiring only moderate peak cycle temperatures. This engine, like the Brayton engine, has a single-phase fluid (helium) that is noncorrosive and will not freeze. The Stirling engine has only two moving parts and uses gas bearings. Long life is inherently possible.

The chief disadvantage of this engine is that its specific development for space use has been limited. Being a reciprocating engine, it will tend to have some vibration. MTI is now designing a new engine that has a more balanced action. Stirling engines are now thought to be limited to 10 to 15 kWe in size. However, they can be modularized to achieve powers of 100 kWe or more. Recent advances have reduced the engine mass, particularly in the linear alternator.

The following technology issues are associated with the Stirling cycle reciprocating PCS (see Figure IV-143):

Technology Step 1

Key Issues - The Stirling engine is still in the development stage. Engines of 4 kWe have been laboratory tested with device efficiencies increased to the 60% level.

Resolution - Qualification of a 4-kWe unit (four units used for a total power of 16 kWe), even though ~2000 h of laboratory testing have been accumulated, will require approximately 6-1/2 years (3 years for design, 2-1/2 years for fabrication and checkout, and a full 1 year for test); completion is estimated by mid 1990.

Technology Step 2

Key Issues - Operating temperature needs to be increased, the linear alternator verified in the 12-kWe power range, and lifetime extended to 4 years.

Resolution - Based on experience gained in designing, fabricating, and testing the 4-kWe unit, build and test a 12-kWe unit for the full 4-yr lifetime. This would overlap the step 1 testing and is considered a risk worth taking because of the high efficiencies and potential long lifetime of the Stirling engine. Completion is estimated for mid 1993.

Technology Step 3

Key Issues - These are the same as for step 2, with advancement to higher operating temperature, longer lifetime, higher efficiencies, and larger units (up to 25 kWe each).

Resolution - This is the same as for step 2, with testing to 5 years (considered nominal risk given success with steps 1 and 2). Completion is estimated by mid 1996.

c. Static

(1) Thermoelectric PCS

The technology assessment for high temperature TE materials is shown in Figure IV-144. Binary-doped SiGe unicouples (using SiGe with a figure-of-merit $Z = 0.7 \times 10^{-3} K^{-1}$) are the state of the art in RTG technology. They have a life expectancy of 7 to 10 years. They have been qualified and flown in space for more than 7 years. The uncouple design will require modifications for a space reactor PCS. High-temperature TE converters use radiative coupling to the heat source and a direct conducting radiator for heat rejection. Any modification of the design of the RTG uncouple to adapt it to space reactor electric system technology is primarily an engineering effort of 2 to 3 years in design and ground engineering tests, with 1 to 2 years of

Figure IV-144. Subsystem Technology Assessment – High-Temperature Thermoelectric PCS

qualification testing. It is expected that development times for direct coupled TE converters will be somewhat longer than for radiative coupled converters. These differences, however, are covered in the range of development time estimates.

A much more critical problem with SiGe technology is the necessity of dramatically expanding production capacity in the United States. The present rate of manufacture of SiGe converters is 100 uncouples (i.e., 50 We) per week. The manufacturer believes this production rate could be increased to 250 uncouples (125 We) per week with the present facilities. With some automation of the slicing operation, this figure could be improved to 500 uncouples (250 We) per week. If the facilities were duplicated, and there were absolutely no changes in the material or configuration, then a productivity of 2000 uncouples (1000 We) per week might be realized in a 2- to 3-year effort. This time is required to purchase the equipment, set up the specifications, check out the production line, and train the additional personnel to operate the equipment. The SNAP-10A experience indicated that it takes approximately 4-1/2 times the number of TE converters to qualify a space reactor electric system as will be used in it. Therefore, the maximum gross electrical output for an initial space nuclear reactor/TE system in the near term may be approximately 40 kWe (i.e., 2 to 3 years for production setup plus 2 to 3 years of uncouple production at 52 kWe/year) if the effort could start now. Taking system thermal inefficiencies into account, the more realistic number is slightly less than 32 kWe for an initial system. This limitation is simply due to the limited production capacity of SiGe TE uncouples and the time needed to improve it.

Certain improvements could be made to the SiGe material within the present uncouple design. These include increasing the amount of silicon in the uncouple and adding higher temperature SiO_2 or SiC CVD sublimation barrier coatings to raise operating temperatures by 100 K (i.e., hot shoe operating temperature of 1350 K). However, the initial life of such a converter might be 2 years with additional development effort, improving the converter life to

a 7- to 10-year goal. The critical technologies for this advancement include work in sublimation barriers; compatibility of materials; low resistance contacts; and, especially, bonding materials properties at these higher temperatures. This advancement would require a 2- to 3-year effort to develop and life-test the SiGe converters at these higher operating temperatures.

An increase in TE converter performance could also be achieved by substituting a more efficient (i.e., higher figure-of-merit) material for the SiGe in the present unicouples. If ternary-doped SiGe (GaP) can be developed for the TE converter, it could improve performance by 20% ($Z = 0.85 \times 10^{-3} \text{K}^{-1}$) and allow higher operating temperatures. This would add an estimated 3 years to the SiGe schedule. The additional time and money would be required to perform extensive laboratory tests, optimize the materials in the converter, and perform life tests on the new material with higher operating temperatures.

The use of rare earth sulfides (i.e., candidate materials such as La_xS_y) or boron carbons as potential TE materials for advanced converters is much farther in the future. A gross schedule estimate indicates that 5 to 8 years would be required beyond the SiGe (GaP) development. The estimated converter performance is a hot-shoe operating temperature of 1550 K and a figure-of-merit of $Z \geq 1.0 \times 10^{-3} \text{K}^{-1}$ (~22% device efficiency). The investigation of the material properties of La_xS_y is only beginning and no TE converters have yet been constructed from this material.

The following technology issues are associated with high temperature thermoelectric PCSs (see Figure IV-144):

Technology Step 1

Key Issues and Resolution - SiGe is a mature PCS technology, but adaptation to a reactor-compatible configuration will be required.

The estimated time to design, fabricate, and test the reconfigured converter is 3 years. Production capacity would need to be greatly expanded (by nearly a factor of 20) to produce unicouples for a 25-kWe space reactor electric system by 1990.

Key Issues - SiGe technology could be advanced in two ways. First, by qualifying the converter for operation at higher temperatures (1350 K) for a 2-yr lifetime with subsequent qualification for a 7-yr lifetime. Second, by substituting SiGe with GaP (with an associated increase in device efficiency to 17 to 18%) and qualifying at 1250 K, first for a 2-yr lifetime and then for a 7-yr lifetime.

Resolution - The design and development effort would include work on sublimation barriers, materials compatibilities, and bonding materials properties. It is estimated that this effort would require an additional 3 years beyond technology step 1 for the 2-yr lifetime subsystem qualification and another 3 years beyond that for qualification of the 7-yr design lifetime subsystem. As with technology step 1, it is assumed that the substantial increase in production capacity could be accomplished.

Key Issues - This advancement in SiGe technology involves adding GaP and increasing operating temperature to 1350 K. The technology issue is materials optimization to improve the figure of merit (and therefore the device efficiency).

Resolution - Design and develop the SiGe (GaP) uncouple by building and testing various materials combinations to optimize converter performance. This would be a natural progression from technology step 2 and would involve increasing the operating temperature by 100 K (to 1350 K). It is estimated this step would take an

additional 3 years beyond the step 2 timeframe to qualify a 2-yr lifetime converter. Extending the design lifetime to 7 years would require another 3 years of testing time.

Technology Step 4

Key Issues - The converter materials for this advanced TE subsystem need to be developed, and this work is still in its infancy.

Resolution - Problems with design, materials development and optimization, device efficiency determinations, and operating temperature limits will need to be resolved by a design and development program similar in nature to the SiGe TE development program. The total elapsed time for the development of technology step 4 should be comparable. It is estimated that the program could be completed with qualified subsystems available by the end of 1998 (i.e., a 14-yr design, development, and qualification program).

Lead telluride was developed under the SNAP Advanced Thermoelectric Technology Program in the late 1960s into the compact tubular module. This compact converter module had demonstrated a 5-year life expectancy in ground tests with a device efficiency of 15%. The development costs for these modules and their life tests were approximately 5.1 million dollars (1967\$) in a 4-year schedule. A 3-year production startup schedule for this type of lower temperature TE technology appears to be reasonable with a capacity of 80 kWe/year after the fourth year (i.e., with a 1-year program startup effort). TE manufacturing limitations would initially limit the space vehicle output to 50 kWe for 5-year missions. There would be no reason to add additional R&D effort to improve the performance of PbTe technology since the TAGS TE technology already exists in RTGs and represents an increase in performance.

The technology assessment for low-temperature materials is shown in Figure IV-145.

Figure IV-145. Subsystem Technology Assessment — Low-Temperature Thermoelectric PCS

The TAGS TE technology is very similar to the PbTe. TAGS RTG technology would have to be adapted to a space reactor configuration similar to the PbTe module noted above. This effort would involve a 3-year design, fabrication, and test schedule. A 3-year production startup schedule was given as the best estimate with a capacity of 80 kWe per year after the fifth year (assuming a 1-year overlap of the two schedules). TE manufacturing limitations would limit the gross electrical output of the space vehicle to 65 kWe for a 5-year or longer mission. TAGS is a mature TE technology, and no concepts have been put forth to date to improve the technology beyond the effort described above.

The following technology issues are associated with low temperature thermoelectric PCSs (see Figure IV-145):

Technology Step 1

PbTe
 $T_H = 858 \text{ K (1085}^\circ\text{F)}$
5-year LIFE
 $\eta_D = 15\text{-}16\%$

Key Issues and Resolution - PbTe tubular converter modules are a mature technology, with a 5-yr life expectancy and $\geq 15\%$ device efficiency. A 1-1/2-yr program startup followed by a 3-yr production run would make these converters available by mid 1998.

Technology Step 2

TAGS
 $T_H = 800\text{-}820 \text{ K (980-1015}^\circ\text{F)}$
7-year LIFE
 $\eta_D = 17\text{-}18\%$

Key Issues and Resolution - The TAGS TE converter subsystem is a mature technology and has been used in the SNAP-19 RTG with lifetimes of over 7 years at device efficiencies of 17 to 18%. A design effort would be required to adapt the technology to a reactor-compatible subsystem. The design effort and associated production and test of this design would require approximately 3 years, and the subsystem production run would require another 3 years; program completion is estimated for the end of 1989.

(2) Thermionic PCS

The technology assessment for in-core thermionics is shown in Figure IV-146.

Two TFE concepts evolved during the development of the in-core thermionic converters. Performance parameters for these two concepts are presented in Table IV-43. The first concept is that developed by GA Technologies and has by far the largest amount of development experience. The large-diameter TFE uses a W emitter and UO_2 fuel, and the fission product gases are vented to a separate gas plenum. This concept was developed for high-power in-core thermionic fast reactor applications. For power levels below about 100 kWe, driver fuel is required for criticality consideration. Thus, overall reactor/converter efficiency would be degraded by an amount proportional to the thermal power generated in the driver fuel. The second concept uses a Mo emitter and UO_2 -Mo foil fuel, and the fission gas is vented to the interelectrode gap and swept to a common cesium reservoir. Because Mo is used as the emitter, moderator rather than a driver fuel can be used at the lower power levels without appreciable loss in reactor/converter efficiency. This second concept was developed by General Electric (and reported to be used in the Russian TOPAZ reactor).

The in-core thermionic converter test data developed by General Atomic (now GA Technologies) and General Electric in the late 1960s through 1972 indicate an initial power duty cycle lifetime of approximately 1 year. The lifetime of current-technology converters is limited by UO_2 fuel swelling and Al_2O_3 insulator swelling. These data, along with the subsequent successful operation of the Russian TOPAZ thermionic reactor, would indicate a reasonably achievable goal of about a 1-year life expectancy for an initial in-core thermionic converter. Device efficiencies for these initial converter designs should be 14 to 20%.

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Figure IV-146. Subsystem Technology Assessment – In-Core Thermionic PCS

TABLE IV-43
IN-CORE THERMIONIC CONVERTER PERFORMANCE PARAMETERS

	Concept 1	Concept 2
<u>Materials</u>		
Emitter	W	Mo
Collector	Nb	Nb
Insulators	Al ₂ O ₃	Al ₂ O ₃
Fuel	UO ₂	UO ₂ -Mo foils
<u>Temperatures [K(°F)]</u>		
Emitter	1850 (2870)	1800 (2780)
Collector	950 (1250)	850 (1070)
Insulators	1000 (1340)	950 (1250)
Cs reservoir	590 (602)	590 (602)
<u>Performance</u>		
Device efficiency (%)	18-20	14-16
Power density (We/cm ²)	5-10	1-2
TFE power (kWe)	~1	~0.12
Barrier index (eV)	2.0	2.0
<u>Size</u>		
TFE diameter [cm (in.)]	3.3 (1.3)	1 (0.4)
<u>Fuel venting</u>	To fission gas plenum	To interelectrode gap

Performance and lifetime can be increased by improving these technologies. For performance improvement, laboratory tests have shown that introducing WO₂ onto the collector's surface (oxygenating the converter) is effective in increasing the converter's device efficiency (to 20 to 24%). Using UO₂-molybdenum foil fueled emitters with a simplified fission gas venting system to the converter's interelectrode gap should increase the life expectancy to 2 years. This improvement is anticipated to take 2-1/2 to 3 years of additional R&D effort.

An advanced converter concept using a W(110) emitter and a sublimed Mo collector should increase the device efficiency to 25 to 30%. For converter

lifetimes beyond 2 years, a new insulator material and low-swelling UO_2 -Mo cermet or coated particle fuels are required. Y_2O_3 and $\text{Al}_2\text{O}_3/\text{Nb}$ cermet have been proposed as candidates for a longer-life insulating material. An initial improvement in converter device efficiency to 25 to 27% could be expected with the advanced converter design. Converter life expectancy should increase to approximately 4 years with the use of a UO_2 -Mo cermet fuel with an additional 3 years of R&D effort. With the development of UC-ZrC coated particle fuel, converter life expectancy should increase to approximately 7 years. Increasing the advanced converter's device efficiency to 28 to 30% should be reasonably achievable for a second-generation converter. This increase in converter performance and life should be available within an additional 3 years.

The following technology issues are associated with in-core thermionic PCSs (see Figure IV-146):

Technology Step 1

Key Issues - The technology base established by the work done at GA and GE needs to be reestablished.

Resolution - The design and fabrication of this in-core thermionic subsystem could proceed immediately based on the GA and GE work performed in the 1960s and early 1970s. A 4- to 5-yr design/fabrication time cycle appears reasonable followed by a full 1-yr reactor subsystem test (the converter and reactor subsystems are inseparable for the in-core thermionic PCS concept). Program completion is estimated by mid 1989.

Technology Step 2

Key Issues - Oxygenated (UO_2) collector surface converters using UO_2 -Mo foil fuel need to be developed.

Resolution - Develop a process for oxygenating the collector surface and a process to produce UO_2 -Mo foil fuel. Successful development of these processes would take approximately 3 years. Then fabricate the reactor subsystem and test for the full 2-yr lifetime (elapsed time for fabrication and 2-yr test is estimated to be 5 years). Program completion is estimated by mid 1992.

Technology Step 3

Key Issues - Sublimed Mo collector, improved insulators (Y_2O_3 or $\text{Al}_2\text{O}_3/\text{Nb}$ cermet), and UO_2 -Mo cermet or coated particle fuels need to be developed.

Resolution - A materials development program would require approximately 4 years; design and fabrication of the reactor subsystem would take approximately 3 years, followed by a full 4-yr test phase; program completion is estimated by mid 1995.

Technology Step 4

Key Issues - These are the same as for step 3 but with lifetime increased to 7 years.

Resolution - This is the same as for step 3 but with a full 7-yr test phase; program completion is estimated by mid 1998.

The technology assessment for out-of-core thermionics is shown in Figure IV-147. Out-of-core thermionic development has progressed for terrestrial applications and has been proposed for space application to eliminate the fuel swelling and insulator damage problems. Such a concept would require the development of heat pipes and a heat pipe reactor to operate at temperatures

Figure IV-147. Subsystem Technology Assessment – Out-of-Core Thermionic PCS

of 1800 to 1900 K (2780 to 2960°F). Laboratory tests have been conducted on an out-of-core thermionic converter design using a W/Li heat pipe for heating the emitter surface and Nb-Zr/K heat pipes for cooling the converter's collector surface. Table IV-44 presents the current performance parameters expected for out-of-core thermionic converters.

TABLE IV-44
EXPECTED OUT-OF-CORE THERMIONIC CONVERTER PERFORMANCE

Materials	
Emitter Heat Pipe	W-Re/Li
Emitter	W, Mo, Re
Collector	Nb, Mo, W-W-WO ₃
Insulators	Al ₂ O ₃
Temperatures [K (°F)]	
Emitter heat pipe	1850 (2870)
Emitter	1800 (2780)
Collector	880 (1124)
Insulator	1000 (1340)
Cs Reservoir	580 (584)
Performance	
Device Efficiency (%)	25-27
Power density (We/cm ²)	4-8
Power per converter (kWe)	~1
Barrier index (eV)	2.0

Thermionic converter device efficiency can be substantially improved to 28 to 34% or the emitter/heat pipe temperature could be lowered by 200 K by optimizing the material selections within the converter. Life expectancy could be increased from 2 to 5 years since there would be no fast neutron life-limiting environments. The core heat pipe would most likely be radiative coupled to electrically isolate the emitter surface of the converter and allow for independent development of the reactor and heat pipes.

The schedule for an out-of-core thermionic converter PCS is difficult to estimate because of the uncertainty associated with the development schedule of a high-temperature core heat pipe space reactor design. Assuming these

components follow a parallel development path, an out-of-core thermionic converter could be developed and flight qualified within a 9- to 10-year time period. Further improvements in this technology could be expected with an additional 5-year R&D effort as shown in Figure IV-147.

The following technology issues are associated with out-of-core thermionic PCSs (see Figure IV-147):

Technology Step 1

OXYGENATED
CONVERTERS
 $T_E = 1800 \text{ K}$
2-YEAR LIFE
 $\eta_D = 25-27\%$

Key Issues - The development of this out-of-core thermionic PCS would be paced by the development of a high-temperature heat pipe reactor to provide the energy for driving the thermionics. The development of oxygenated converters by mid 1992 is judged to be reasonable (see in-core thermionics technology step 2) based on IPL and GA work performed to date.

Resolution - The operation of oxygenated converters at 1800 K would proceed naturally from the in-core thermionics step 2. Testing at this temperature should be reasonably demonstrated by mid 1993.

Technology Step 2

OPTIMIZED
MATERIALS
IN CONVERTER
 $T_E = 1600-1800 \text{ K}$
5-YEAR LIFE
 $\eta_D = 28-34\%$

Key Issues - As yet undefined improved/optimized converter materials capable of a 5-yr lifetime with device efficiencies in the 28 to 34% range need to be developed.

Resolution - An extensive materials search and development program would be required. It is judged that such a program would easily require a timeframe approaching the end of the century before the PCS could be qualified.

(3) Thermophotovoltaic

Recent studies on thermophotovoltaic subsystems for space power missions^{IV-104, IV-105} have concluded that (1) TPV subsystems with device efficiencies of 30% and greater are practical and (2) TPV subsystems are technically feasible and could be made with current technology, materials, and components.

Thermophotovoltaic power subsystems application to future space nuclear reactor power missions would appear to be limited, however, due to the high emitter/radiator temperatures (2000 K) required for TPV subsystems and the sensitivity of silicon photovoltaic cells to nuclear radiation fields. The most probable applications in the space nuclear field could be in employing a high-temperature radioisotope fuel capsule as the emitter/radiator within a large TPV design.

d. Power Processing

The technology assessment for power processing is shown in Figure IV-148. What distinguishes the power processing equipment requirements for the PCs just assessed from those of existing spacecraft is the possibly much higher electric power output and perhaps more severe radiation environment. Those components with designs not influenced by higher power levels should be available now or should become available as needed as a result of our ongoing space development activities. Some radiation hardening above existing levels may prove desirable on a system mass basis.

The components that would be expected to be most influenced by the large electrical power outputs are those associated with accommodating large and rapid changes in payload power consumption. The thrust of needed development would be in the direction of (1) increasing load handling capacity from a few kWe to up to 10 kWe and (2) reducing dependence on parasitic loads to accomplish voltage and frequency regulation.

Figure IV-148. Subsystem Technology Assessment – Power Processing Equipment

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For static converters, this would involve the development of high-current solid-state switching equipment to couple more or less of the converter output to the load. Some development may be required on modestly sized shunt regulators to accompany the switching equipment. For dynamic converters, development would involve (in addition to minimizing parasitic loads) updating the design of flight-rated speed and voltage controls using semiconductor devices that have become available since effort on such subsystems was curtailed. It would also involve investigating new approaches to controlling voltage and frequency that use the control power and flexibility of currently available microprocessor hardware.

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E. HEAT REJECTION

1. Background

The basic requirement of any thermodynamic cycle is the rejection of waste heat. In a space environment, this heat must be rejected by radiation. Radiators for heat rejection tend to dominate in the selection of operating conditions for space power systems because the heat rejection rate is proportional to the fourth power of the radiator surface temperature. This can be observed by examining the basic Stefan-Boltzmann equation that governs radiant heat transfer from a surface:

$$Q_R = \epsilon \sigma A \left(T_C^4 - T_S^4 \right), \quad (\text{IV-1})$$

where

Q_R = heat radiated

ϵ = surface emissivity for thermal radiation

σ = Stefan-Boltzmann constant

A = area of radiating surface

T_C = absolute temperature of radiating surface

T_S = absolute temperature of radiative sink.

Therefore, the surface area and mass of the space radiator are very sensitive to heat rejection temperature. Higher rejection temperatures for thermodynamic cycles thus mean lower radiator mass. On the other hand, the thermodynamic cycle efficiency is also sensitive to the heat rejection temperature.

The overall PCS efficiency can be expressed as:

$$\eta_P = \eta_D \eta_C, \quad (\text{IV-2})$$

where

η_P = power conversion subsystem efficiency

η_C = Carnot efficiency

η_D = device efficiency.

The heat input, Q , from the reactor to the PCS can be expressed as

$$Q = \frac{P}{\eta_P} = \frac{P}{\eta_D \eta_C}, \quad (\text{IV-3})$$

where P is the gross electrical power produced by the PCS. The heat rejected by the radiator, Q_R , is

$$Q_R = Q - P = P \left(\frac{1}{\eta_D \eta_C} - 1 \right). \quad (\text{IV-4})$$

The Carnot efficiency is defined by

$$\eta_C = \frac{T_H - T_C}{T_H}, \quad (\text{IV-5})$$

where

T_C = PCS rejection temperature

T_H = PCS input temperature.

Combining Eqs. IV-1, 4, and 5 yields (neglecting T_S , which is ~ 255 K for space)

$$\frac{P}{A} = \frac{\sigma \epsilon T_H^4 (1 - \eta_C)^4}{\left(\frac{1}{\eta_D \eta_C} - 1 \right)}. \quad (\text{IV-6})$$

The maximum PCS electric power per unit radiator area or the minimum radiator area for a given electric power at a fixed T_H can be determined by differentiating Eq. IV-6 and solving for η_C . This results in the following:

$$\eta_C \bigg|_{\text{min area}} = \frac{2}{5 \pm \sqrt{25 - 16\eta_D}} . \quad (\text{IV-7})$$

This equation has only a minor dependence on device efficiency, as shown in Figure IV-149. In the range of device efficiencies for the PCSs considered (10 to 75%), the Carnot efficiency for minimum radiator area varies from only 21 to 23%. This is a very important observation.

Figure IV-149. Optimum Carnot Efficiency for Minimum Radiator Area as a Function of Conversion Device Efficiency

Because of the high energy content of a space nuclear reactor, PCS designs generally tend toward a range around this minimum-area Carnot efficiency. It is obviously not true that minimum system mass occurs at minimum radiator area. The physical size of the space shuttle capability and the potential structural awkwardness of the radiator are major considerations. Higher Carnot efficiencies are often selected based on heat source limitations, PCS characteristics, or the desire to maximize electric power.

The generalized relationship between radiator area and PCS Carnot efficiency is shown in Figure IV-150. For the PCS device efficiency range of interest (10 to 75%), there is little change in this relationship. Figure IV-150 also shows points for various space reactor electric systems or concepts. Most systems tend to have Carnot efficiencies above the value for minimum radiator area.

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Figure IV-150. Radiator Area versus Carnot Efficiency

The net result of all this is that space reactor electric systems have much lower efficiencies than are generally used on the ground. This lower efficiency places a high premium on heat source temperature to minimize radiator area and weight.

2. Heat Rejection Considerations

a. Configuration

A wide variation of radiator configurations have been proposed for space reactor electric systems. These radiators can be generally classed into two groups – fixed and deployable. Several examples in each of these groups are presented in Figure IV-151. The cylindrical/conical configuration (either completely conical or a combination of a cone and cylinder) is the most popular because of integration considerations with the reactor/shield and payload, structural integrity, and compatibility with the launch vehicle. The flat plate and cruciform are configurations that have been proposed for various applications. The launch vehicle characteristic is by far the most important factor in selecting a radiator configuration. When radiator size requirements exceed the volume limits of the launch vehicle, then deployable concepts must be used. Figure IV-151 shows two examples of deployable configurations: a nested cylindrical and a folded flat plate. Radiator concepts that do not fall in the groups discussed above have also been studied. These include the moving-belt and liquid-droplet radiators.

Figure IV-152 shows an example of a conical radiator configuration. This photograph shows the SNAP-10A reactor electric system that was launched into space. The individual radiator panels are actually bonded to the cold junction of a TE module. Figure IV-153 shows an example of a deployable radiator configuration. This is an artist's concept of a nested cylindrical design used in the study of a 10- to 75-kWe space reactor electric system.

b. Functional Arrangement

(1) Heat Pipe

A heat pipe used to reject heat to space is shown schematically in Figure IV-154. Heat is received at the evaporator section of the heat pipe

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Figure IV-151. Radiator Configuration Examples

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Figure IV-152. SNAP-10A Conical Radiator

Figure IV-153. Nested Cylindrical Radiator Concept

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Figure IV-154. Radiator Heat Pipe Schematic

and transported along the heat pipe, where heat is radiated to space at essentially an isothermal temperature. The heat pipe can have a high emissivity coating or be attached to a high-emissivity fin that provides radiator surface. The heat pipe arrangement can be one of several configurations. The radiator schematic shown in Figure IV-155 is in the shape of a frustum of a right circular cone. The radiator in this case would use two sets of heat pipes in series. The axial stringer heat pipes, arranged on the cone slant surface, uniformly transfer heat to the radiating surface made up of many heat pipes laid side by side at right angles to the stringers around the radiator circumference. Heat is conducted from the stringer heat pipes to the cross heat pipes and then radiated to space.

The use of heat pipes for a radiator was conceived as an alternate for the SNAP-50 as shown in Figure IV-156. This is a concept where potassium is cooled by convection/conduction to the evaporator section of sodium heat pipes. This concept consisted of 150 separate heat pipe elements designed to dissipate 50 kWt at 1044 K (1419°F).

Figure IV-155. Conceptual Heat Pipe Radiator

Figure IV-156. 50-kWt Heat Pipe Radiator Concept for SNAP-50

(2) Pumped

Figure IV-157 is a schematic of a pumped heat rejection loop. The components in the loop consist of a waste heat exchanger, a radiating surface, a volume accumulator, and a pump. The schematic shows an EM pump, which indicates that it is a liquid metal loop. Organic liquids have also been considered in heat rejection loops, in which case a rotary pump would be used. The volume expansion has two functions in a closed loop. The first is to compensate for both the fluid expansion and the component containment expansion in heating from shutdown temperatures to operating temperatures. The second function is to maintain the fluid at a predetermined pressure. If the loop fluid is gaseous, a volume accumulator would not be required.

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Figure IV-157. Typical Pumped Heat Rejection Loop

With a Rankine cycle PCS, the heat exchanger would be a condenser, as was used with the SNAP-8 mercury Rankine equipment. For a TE PCS, the heat exchanger would be the cold junction of the TE converters. For a Brayton PCS,

the heat exchanger would be gas to liquid metal or gas to organic. With in-core thermionics, the heat exchanger would actually be the thermionic collector in the reactor. In general, the heat exchanger shown in Figure IV-157 is actually a part of the PCS. However, this component is discussed in this section because of its importance to the requirements of a pumped heat rejection loop.

One characteristic of a pumped heat rejection loop is the temperature loss in the loop. The radiator operates below the PCS rejection temperature. This penalizes radiator area. To overcome this, the SNAP-2 mercury Rankine system and the MPRE potassium Rankine system, for example, used the radiator directly as a condenser. The radiator temperature was then essentially constant at the cycle heat rejection temperature.

c. Operational Requirements

(1) Material Considerations

The operating temperature range of radiators for space reactor electric systems is significantly below that required for the reactor, primary heat transport, and power conversion subsystems. Heat rejection temperatures range from a low of 400 to 600 K (200 to 620°F) to a high of 950 to 1050 K (1250 to 1430°F). At such temperatures, there is a wide choice of radiator materials to be considered. In addition, at the operating temperature, the radiator and supporting structure are required to have little strength since they will be in a zero-g environment. Strength requirements are important only at low temperatures for withstanding the launch environment. Because of the size (area) of most radiators, low mass per unit area is an important consideration.

Several key material areas must be considered in designing space radiators:

- High strength-to-mass ratio
- High thermal conductivity
- Resistance to meteoroid impact damage
- Compatibility with heat transport fluid
- High emissivity
- Structural integrity.

A high strength-to-mass ratio is important to minimize radiator mass. As mentioned above, this requirement is at launch condition temperatures. Low-temperature ductility is also important for the structural portion of the radiator. Because of these considerations, the SNAP-10A system used a separate, low-mass titanium structure to support the radiator.

High thermal conductivity is an important consideration in fin materials for minimizing mass. Aluminum was used as the fin material in SNAP-10A, and Be is widely considered for higher temperature applications. Carbon/carbon composite is also promising as a radiator material.

In designing for resistance to meteoroid impact damage, Al and Be are often selected as armor material to protect other portions of the heat rejection subsystem.

The heat transport fluid (if used) also influences material selection. Long-term material compatibility is a must between the fluid and the containment barrier. Fluids considered for radiator working fluids include NaK, organics, gas, and potassium. In some cases, an inner containment material for the fluid is bonded to a fin/armor material. Such an arrangement allows independent selection of containment and armor/fin materials. In addition, such a bonded tube geometry offers a significant advantage in reducing armor requirements.

Efficient radiators must have a high surface emissivity to minimize area. High emissivity is usually obtained by applying a coating (or layers of coatings) to the radiating surface. Considerations in coatings include compatibility with the base material, thermal expansion coefficient, emissivity, and solar absorbtivity (for lower temperature radiators). Coatings, however, are subject to meteoroid erosion and to degradation over time. Such considerations in the future may require the selection of a base material with an inherently high emissivity. The carbon/carbon composites may have such a feature.

The structural integrity of the radiator is a consideration that is often overlooked in conceptual analyses. Thermal stresses established by expansions between the large radiating surfaces and other radiator components can often be a design-limiting consideration.

(2) Micrometeoroid Protection

Meteoroids of varying sizes, densities, and velocities are one of the hazards encountered by space vehicles. Whenever a puncture from meteoroid impact would lessen or destroy the integrity of a particular component, suitable precautions must be taken.

Some form of protection against damage from impacting meteoroids must be provided for radiators. Two general concepts of protection are usually considered. The first approach, the armor concept, uses a solid mass of material to stop an impacting particle before any serious damage can occur. In the second approach, the bumper concept, a relatively thin shield of material, displaced from the vulnerable surface, is used to fragment an impacting particle and spread its energy over a greater area.

Many geometric forms for finned-tube radiators are possible, as illustrated in Figure IV-158. The configurations on the left are based primarily on the armor concept, whereas those on the right embody the bumper concept.

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Figure IV-156. Examples of Meteoroid Armor Placement
on Finned-Tube Radiators

The three principal mechanisms by which meteoroids can cause the failure of a space radiator are: (1) puncture of fluid-carrying passages, with the resultant loss of either working fluid or coolant; (2) spalling, in which sizable pieces of metal are expelled from the inside of the fluid-carrying passage and find their way to equipment with catastrophic results; and (3) erosion of emissivity surface coatings. Puncture and spalling are two distinctly different phenomena. The amount of armor required to protect against damage is sensitive, of course, to the damage mechanism, and to the nature and distribution of meteoroids in space.

By far the greatest amount of information on meteoroids is based on earth observations. Both optical and radar methods have been used to determine relationships between the frequency of the particles and the magnitude of luminosity produced as the meteors traverse the atmosphere. Reducing this

data to estimates of meteoritic mass has been difficult because of the uncertainties of the particle drag characteristics and the luminous efficiency of the process.

The space vehicle meteoroid design discussion is divided into two portions: (1) selection of the meteoroid environment model to be used and (2) the selection of the appropriate analytical methods to be used to assess meteoroid damage.

(a) Meteoroid Environment Model

NASA has established the recommended meteoroid environment model for engineering application to space missions in a near-earth orbit. The following meteoroid flux-mass model and the associated particle density and velocity values are taken from Ref. IV-127.

The average total meteoroid (average sporadic plus a derived average stream) environment is to be used for preliminary design and for mission periods that cannot be specified. This environment model encompasses only particles of cometary origin and is composed of sporadic meteoroids in the mass range between 10^{-12} and 1 g, and stream meteoroids in the mass range of 10^{-6} to 1 g. The characteristics of the average total meteoroid mass-flux model are as follows:

- 1) Particle density: The mass density is 0.5 g/cm^3 (31 lb/ft^3) for all meteoroid sizes.
- 2) Particle velocity: The average meteoroid velocity is 20 km/s (65,000 ft/s).
- 3) Flux-mass model: The average cumulative meteoroid flux-mass is described mathematically as:

$$\log_{10} N_t = -14.37 - 1.123 \log_{10} m \quad (\text{IV-8})$$

$$\text{for } 10^{-6} \leq m \leq 10^0$$

and

$$\log_{10} N_t = 14.339 - 1.584 \log_{10} m - 0.063 (\log_{10} m) \quad (\text{IV-9})$$

$$\text{for } 10^{-12} \leq m \leq 10^{-6} ,$$

where

N_t = particle flux of mass m or greater ($\text{meters}^2 \cdot \text{s}$)⁻¹
 m = particle mass (g).

Eqs. IV-8 and IV-9 should be used for radiator design. Particles smaller than 10^{-6} g will not penetrate aluminum sheets with a thickness greater than about 0.08 mm (0.020 in.).

(b) Meteoroid Armor Equation

The reference meteoroid armor equation that is used to establish the amount of armor required to protect the fin-tube radiator assembly coolant tube from meteoroid impact was developed by Haller and Lieblien:^{IV-128}

$$\delta = \gamma_{Ra} \left(\frac{\rho_p}{\rho_a} \right)^{1/2} \left(\frac{v_p}{c_a} \right)^{2/3} \left(\frac{6}{\pi \rho_p} \right)^{1/3} \left(\frac{E \alpha A_v t}{- \ln p_0} \right)^{1/3\beta} \left(\frac{2}{3n\theta\beta+2} \right)^{1/3\beta} \left(\frac{T}{T_R} \right)^{1/6} , \quad (\text{IV-10})$$

where

δ = armor thickness (cm)

γ_R = room temperature cratering coefficient
 a = rear surface damage thickness factor
 ρ_p = meteoroid average density (0.5 g/cm^3)
 ρ_a = armor density (g/cm^3)
 V_p = meteoroid average velocity (20 km/s)
 C_a = sonic velocity in armor (km/s)
 E = armor earth shielding factor = 0.7631
 α = meteoroid flux constant ($10^{-14.37} \text{ g}^\beta/\text{m}^2 \cdot \text{s}$)
 A_v = vulnerable area (cm^2)
 t = mission time (s)
 P_0 = design probability of no critical damage
 n = damage factor for oblique impact = 1.0
 θ = penetration constant = 0.667
 β = meteoroid flux constant = 1.213
 T = armor temperature (K)
 T_R = room temperature (K).

The meteoroid flux constants used in this equation are^{IV-127}

$$\alpha = 10^{-14.37} \text{ g}^\beta/\text{m}^2 \cdot \text{s}$$

$$\beta = 1.213.$$

The values of V_p and ρ_p are^{IV-127}

$$V_p = 20 \text{ km/s}$$

$$\rho_p = 0.5 \text{ g/cm}^3.$$

The values of n and θ used in the armor equation are^{IV-127}

$$n = 1.0$$

$$\theta = 2/3.$$

The oblique impact term in the armor equation, $[2/(3n\theta\beta + 2)]^{-1/3\beta}$, then reduces to $[1/(1 + \beta)]^{-1/3\beta}$.

The earth shielding factor E used in the armor equation can be calculated as a function of orbital altitude and is defined as the ratio of the shielded to the unshielded flux (Figure IV-159):^{IV-127}

$$E = \frac{1 + \cos \theta}{2} = \frac{1 + 0.5242}{2} = 0.7621 \quad , \quad (IV-11)$$

where

$$\sin \theta = \frac{R}{R + H} = \frac{3444 \text{ n.mi}}{3444 + 600 \text{ (n.mi)}} = 0.8516 \quad (IV-12)$$

R = radius of shielding body

H = altitude above surface

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Figure IV-159. Earth Body Shielding Factor for Randomly Oriented Spacecraft

The γ_R values at elevated temperatures have been correlated to room temperature values by the $(T/T_R)^{1/6}$ term in the armor equation.

The cratering coefficient (γ_R) and the rear surface damage factor (a) in the meteoroid armor equation vary for different armor materials and as a function of temperature. Values for γ_R taken from Ref. IV-130 for various materials are given in Table IV-45.

TABLE IV-45
CRATERING COEFFICIENT VALUES FOR
SELECTED MATERIALS

Target Material	Material Cratering Coefficient, γ_R
356-T51 aluminum	2.15
7075-T6 aluminum	2.00
2024-T6 aluminum	1.70
Columbium - 1% zirconium	1.81
Type 316 stainless steel	2.19
A-286	1.77
Inconel-718	1.85
Cobalt L-605	2.00
Vanadium	1.71
Tantalum	1.77
TZM-molybdenum	2.00

The rear surface damage thickness factors (a) for incipient dimple, spall, and perforation for various materials are given in Ref. IV-131 (Table IV-46) at room temperature. The three modes of damage by meteoroid impact are defined as follows:

- 1) Dimple — The impacted surface is physically dented, but the integrity of the rear surface is not disrupted.
- 2) Spall — The impacted surface may have been partially penetrated, and spallation may have occurred from the rear surface; however, the complete thickness of the material has not been perforated.

- 3) Perforation — The complete thickness of the impacted material has been physically perforated.

The rear surface damage thickness factors given in Table IV-46 were experimentally determined using flat plates as the target material. Impact studies, using unlined aluminum 2024-T6 tubes, have shown that the factors given in Table IV-46 for aluminum 2024-T6 are usable for single-material (unlined) radiator tubes.

TABLE IV-46
DAMAGE THICKNESS FACTORS FOR INCIPIENT
DIMPLE, SPALL, AND PERFORATION

Material	Rear Surface Damage Thickness Factor, a		
	Dimple	Spall	Perforation
Aluminum 2024-T6	2.5	2.3	1.7
Type 316 stainless steel	2.4	1.9	1.4
A286	2.4	1.9	1.4
Columbium - 1% zirconium	4.5	4.0	1.7
Inconel-718	3.0	2.5	-
Cobalt L-605	2.5	2.1	1.7

The presence of an inner liner was found to improve the damage resistance of radiator tubes. An inner liner, although dimpled on the inside on impact, prevented metal spalling into the tube.^{IV-129} In Ref. IV-130, the statement is made that, in general, the inner surface integrity is not lost, even at complete closure of the tube by the liner. Values of the rear surface damage thickness factor as a function of dimple height produced in the liner are given in Figure IV-160 for cast aluminum armor over 0.028-in. stainless steel liners of various IDs.^{IV-130} The solid line in Figure IV-160 is an average of the experimental data for the three ID tube sizes used in the study. The dashed line has been drawn to encompass all of the experimental data. The dashed line is recommended for use in engineering design. It will provide an armor thickness that is somewhat conservative for a given dimple ratio.

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Figure IV-160. Variation of Rear Surface Damage Thickness Factor with Tube Dimple Height

In using Figure IV-160, however, it is implied that the rear surface damage thickness factor is not dependent on the type of bond that exists between the tube liner and the armor material. Data on lined aluminum tubes with various types of bonds indicate that the type of bond between liner and armor has very little effect on liner damage.^{IV-130}

Figure IV-161 shows the manner in which meteoroid impact to the outer surface of one side of a radiator can damage the coolant tubes of the other side of the radiator or other internal components. A primary meteoroid strikes and penetrates the outer surface of the radiator. On impact, the primary particle is fragmented and dispersed over a large cone angle inside the radiator structure. The fragments of the primary particle, plus spallation from the meteoroid puncture, in turn strike other interior surfaces. The primary impacted surface serves as a bumper for the interior components. The recommended bumper thickness and minimum spacing are also shown in Figure IV-161 (δ is the armor thickness calculated from Eq. IV-10).

(3) Emissivity Coatings

Spacecraft radiators achieve maximum effectiveness by using thermal control coatings that exhibit a high thermal emittance (ϵ) and a low solar absorptance (α_s). Generally, it may be stated that the radiator surface area and weight are proportional to the thermal emittance property of the surface coating; thus, judicious coating selection is of major importance to the radiator design.

The selection of thermal control coatings for use on spacecraft radiators is primarily predicated on the optical properties of the coating and the stability and integrity of the coating in the space environment at the design operating temperature of the radiator.

Emissivity coating candidates tend to fall into three operating temperature regimes: temperatures to 422 K (300°F), between 422 K (300°F) and 644 K (700°F), and between 644 K (700°F) and 811 K (1000°F).

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Figure IV-161. Bumper Effect Upon Primary Meteoroid Impacts

(a) Coatings Used Below 422 K (300°F)

For temperatures below 422 K (300°F), one radiator coating stands out because of its excellent optical properties and proven reliable performance. This coating, designated Z93, is a zinc oxide pigment contained in a potassium silicate binder. In the temperature region up to 422 K (300°F), the emittance of Z93 is 0.90, and the solar absorptance is equal to 0.20 maximum.

The coating has undergone stringent testing, particularly from the aspect of stability in a particulate and electromagnetic particle environment. Studies of long-term (8000-h) UV irradiation of Z93 have shown that the solar absorption property of the coating remains almost totally stable. Low-energy proton irradiation, similar to the solar wind spectrum, shows that the coating remains stable at an integrated flux of 10^{14} particles/cm².

Applying the coating to typical radiator substrates is relatively straightforward. An aqueous solution of the coating is sprayed on a substrate, and curing is performed at 413 K (283°F). As with all spacecraft coatings, care must be exercised in handling the finished product to prevent surface damage.

(b) Coatings Used for 422 to 644 K (300 to 700°F)

A radiator coating development program was performed in support of the SNAP-2 and -10A programs. The initial design requirement for the radiator coating was $\epsilon = 0.85$ and $\alpha_s = 0.30$, operating at a temperature of 589 K (600°F). Since the coating would operate in high neutron and gamma environments, it was decided not to investigate paints or enamels because of their known instabilities in nuclear radiation fields. Primary emphasis was placed on using two common inorganic binding agents, aluminum phosphate and potassium silicate.

Included in the several hundred coatings investigated during these programs were many of the basic metal oxides with particular interest devoted to the white oxides. Screening tests were begun with an evaluation of the spectral reflectance and total normal emittance of candidate metallic oxides, including the oxides of Ba, Ce, Sb, Si, Sn, Ti, Zn, and Zr.

From this study many important factors were determined; the most pertinent of these are

- 1) The thermal emittance at 589 K (600°F) for coatings with the aluminum phosphate binder was consistently 2 to 5% higher than coating with the potassium silicate binder (Figure IV-162).
- 2) The solar absorptance for coatings with the aluminum phosphate binder was consistently 10 to 20% higher than for coatings with the potassium silicate binder.
- 3) The white pigment that exhibited the highest thermal emittance at 589 K (600°F) was stannic oxide.

Because the optimum thermal effectiveness of the coating in the 422 to 644 K (300 to 700°F) temperature range is determined primarily by the thermal emittance, further experimentation emphasized use of the stannic oxide pigment and an aluminum phosphate binder. Measurements of the total hemispherical emittance of the stannic oxide coating system yielded a value of $\epsilon = 0.88$ at 589 K (600°F).

Many black coatings have a high thermal emittance in the short wavelength IR region (1 to 4 μm). To further increase the bulk emittance, a composite coating was devised in an effort to maximize the spectral emittance throughout the IR region. This was done by using a high thermal emittance black subcoat and the stannic oxide as a topcoat.

The AI-93 coating used a bilayer, consisting of a subcoat containing a chrome-cobalt-nickel oxide mixture and a top coat containing stannic oxide.

Both layers use aluminum phosphate as the binding agent. Between 422 and 644 K (300 and 700°F), the total hemispherical emittance of the AI-93 is 0.90 minimum, and the solar absorptance is 0.35 maximum. As in the case of Z-93, AI-93 has undergone complete qualification in a simulated space environment; in addition, it has attained in excess of 1000 h of successful operation in the space environment at a temperature exceeding 644 K (700°F).

For radiators operating between 422 and 644 K (300 and 700°F), the thermal emittance property is the dominant factor; however, a low solar absorptance does contribute a measurable bonus to the net heat rejection capability of the radiator. Figure IV-163 illustrates the effect of temperature on emittance for white coatings now in use or under development.

(c) Coatings Used above 644 K (700°F)

Above 644 K (700°F), the solar absorptance property has little effect on the thermal effectiveness of a coating; thus, coating development in this temperature region was directed primarily toward achieving maximum thermal emittance. Coating materials were evaluated at elevated temperatures and high vacuum for their emittance stability, adherence, and compatibility when applied to AISI-310 stainless steel or columbium - 1% zirconium.

Best results were obtained with iron titanate and calcium titanate coatings. An iron titanate coating on AISI-310 stainless steel has been tested at 1006 K (1350°F) in a vacuum of $\sim 2 \times 10^{-8}$ Torr for 5300 h. An emittance of 0.88 or better was obtained throughout this period. A second iron titanate coating on columbium - 1% zirconium has been tested at 1200 K (1700°F) for 6250 h, with an emittance of 0.85 or better throughout the test. A coating of calcium titanate on AISI-300 stainless steel has been tested for 6300 h at 2005 K (3150°F), with an emittance of 0.90 throughout the tests. Adherence tests have been conducted for both of the coatings, and the bond strengths were found to be excellent.

Figure IV-162. Thermal Emittance of Candidate Radiator Coatings vs Temperature

Figure IV-163. Total Hemispherical Emittance of White Coatings As a Function of Temperature

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The work conducted to date indicates that the calcium titanate and iron titanate are suitable materials for high-temperature space radiator applications. The specimens tested with these coatings have revealed the following properties:

- 1) Calcium titanate on AISI-300 stainless steel
 - a) Total hemispherical emittance of 0.90 for 6300 h at 1006 K (1350°F) in a vacuum of 10^{-8} Torr. In the temperature region of 644 to 811 K (700 to 1000°F), the emittance was 0.87 to 0.88.
 - b) Ability to withstand thermal cycling from 1006 K (1350°F) to ambient.
 - c) Ability to maintain good adherence when subjected to fatigue testing for 10^7 cycles at stress levels up to 379 MPa (55,000 psi).
- 2) Iron titanate on AISI-300 stainless steel
 - a) Total hemispherical emittance of 0.88 for 5300 h at 1006 K (1350°F) in a vacuum of 10^{-8} Torr. In the temperature region of 644 to 811 K (700 to 1000°F), the emittance was 0.85.
 - b) Freedom from adverse effects when subjected to thermal cycling from 1006 K (1350°F) to ambient.
 - c) Ability to maintain good adherence when subjected to fatigue testing for 10^7 cycles at 120 Hz at stress levels up to 379 MPa (55,000 psi).

Other coatings have been suggested for high-temperature use. The most promising of these uses the CVD of tungsten and silicon carbide on molybdenum and tungsten alloys. These coatings, however, still require in-vacuum testing for verification of their emissive and absorptive properties.

(4) Emissivity Enhancement

Adding fins to a smooth radiator surface is a technique that can be used to enhance the effective surface emissivity. Adding these fins has the effect of creating a cavity that allows multiple reflections within a heated pseudo-isothermal wall. These reflections act to augment the emissivity relative to that of a planar surface at identical temperature and emittance. Figure IV-164 shows the effect of such enhancement.^{IV-132} The additional weight of the fins has to be traded off against the enhancement. As shown in Figure IV-164, radiators with high-emissivity coatings do not show much improvement when fins are added.

3. Heat Rejection Subsystem Descriptions

a. Direct Conducting Subsystems

(1) SNAP-10A Radiator

The SNAP-10A radiator represents a heat rejection subsystem in its simplest form. It used only conduction for transporting heat. Aluminum sheets were bonded to the cold junction of TE elements as shown in Figure IV-165. The complete SNAP-10A radiator, shown in Figure IV-152, contained 2600 individual panels. It used a separate titanium substructure to support the thermoelectrics and radiator. The radiator was a light color due to the AI-93 coating. This radiator was completely flight-qualified and operated in space.

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Figure IV-164. Apparent Emittance Results for Reflecting Rectangular Groove Cavities

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Figure IV-165. SNAP-10A Radiator

(2) SABRE Radiator

The SABRE concept system shown in Figure IV-166 uses TE converters to convert thermal power to electric power.^{IV-133} Rejected heat from the TE converters is conducted to radiator panels through the TE cold shoe and an insulator. The radiator panel material currently suggested is carbon/carbon composite, with an emissivity of 0.85. The average operating temperature of the panels is 800 K (980°F). These panels are arranged to form a frustrum of a cone about 7 m long which begins at the end of the shield and grows in diameter until reaching a diameter of 4.3 m (14 ft) where the panels then extend back to the full radiator length in a cylindrical shape. The radiator is also configured to act as an integral structure.

b. Pumped Liquid Subsystems

(1) Condensing

(a) SNAP-2 Radiator

The radiator condenser design selected for SNAP-2 is shown in Figure IV-167. The heat rejecting condenser was a direct, once-through type where the working fluid, Hg, would be condensed and subcooled with a total heat rejection of 51.0 kWt. The radiator condensor shown in Figure IV-161 also served as the main structural member. This aluminum-fin steel-tube semi-monocoque concept, shown in Figure IV-167, was fabricated after the development of a manufacturing process for brazing titanium-coated steel tubes directly to the aluminum fin. This process was accomplished by a vacuum-pressure-temperature technique. A multiple-layer emittance coating was

Figure IV-166. SABRE Radiator Concept

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TYPICAL CROSS SECTION AT TOP OF CONICAL SECTION

TYPICAL CROSS SECTION AT CYLINDRICAL SECTION

DIMENSIONS IN cm (in.)

- 24 LINEARLY TAPERED TUBES
- SKIN THICKNESS IS CONSTANT AT 0.16 cm (0.063 in.)
- 12.4 m² (134 ft²) AREA
- TUBES AND ARMOR L-605
- AI FRAMES 6061-T6

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Figure IV-167. SNAP-2 Radiator Condenser (Hg working fluid)

selected; it consisted of a subcoat of chromium-cobalt-nickel spinel and a top coat of stannic oxide. The total coating thickness was about 0.0075 cm (0.003 in.).

The prototype unit, shown in Figure IV-168, was fabricated and underwent a series of tests that verified the basic feasibility of the design. This included a full series of structural, environmental, and thermal tests. The thermal design objectives of the SNAP-2 radiator condenser were the following: Hg flow rate of 9.1 kg/min (20 lb/min), total heat rejection of 51 kW, inlet Hg vapor temperature of 603 K (625°F) at 97% quality, and a maximum exit temperature of 462 K (371°F).

(b) ORNL Condensing Potassium Radiator

The potassium Rankine PCS schematic is shown on Figure IV-169. This subsystem used a single loop with a direct condensing radiator. The biggest concern with condensing radiators was with zero-gravity effects. To evaluate this problem, a small water model of a tapered tube condensing radiator was run in the NASA-LeRC drop tower and operated successfully for 3 s in the zero gravity environment. A full-size 850-kWt tapered tube condensing potassium radiator was built and run at 811 K (1000°F) for 400 h at ORNL. The radiator was designed as a flat surface and operated almost horizontally in an attempt to simulate zero-g effects. The critical technology for condensing radiators is the control of the free liquid surface (i.e., inventory control) under zero-g operating conditions. Additional zero-g tests would most likely be required to resolve this technology issue. Water would be an appropriate fluid to conduct early tests; however, tests using potassium in a zero-g environment would eventually be required.

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Figure IV-168. SNAP-2 Aluminum-Steel Radiator-Condenser Prototype Unit

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Figure IV-169. Potassium Rankine PCS Schematic

(c) ORNL Reflected Tube Radiator

As part of the potassium Rankine PCS activities at ORNL, work was performed on a novel radiator called the reflected tube radiator. A schematic of this concept is shown in Figure IV-170. Condensing potassium flows inside the tubes, and heat is radiated in the normal manner to space on the front side of the fins. The back side of the fins and the rear fin radiate to a surface that reflects the heat away from the fins to space. The reflecting surface is circular and is designed such that any ray from a point on the radiator will leave the reflector such that it will not intercept the radiator surface. Shutters were provided to maintain high temperatures and, consequently, higher potassium vapor pressures for part-load operation. Design studies indicated that the reflected tube radiator could reduce radiator mass by 45% over a conventional connected fin radiator. A single electrically heated tube model was built and tested in a vacuum.

(2) Noncondensing

(a) SNAP-8 Radiator

The SNAP-8 heat rejection NaK loop removes heat from the PCS mercury condenser and transports this to the radiator for rejection to space. The conical radiator design is made up of a fin-tube configuration, as shown in Figure IV-171 (for the 35-kWe version of the SNAP-8 PCS). The characteristics of the overall radiator assembly are as follows: total area of 151.4 m^2 (1630 ft^2), total mass of 792 kg (1745 lb), and a total height of 13.6 m (44.6 ft). The radiator is composed of the heat rejection loop radiator; a

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Figure IV-170. Reflected Tube Radiator Configuration

high-temperature, lubricant-cooled radiator; and a low-temperature, lubricant-cooled radiator.

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Figure IV-171. SNAP-8 Radiator Fin-Tube Configuration

(b) 5-kWe Reactor Thermoelectric System Radiator

Figure IV-172 shows the configuration of the reference radiator/structure. The overall height is 5.85 m (19.2 ft). The 1.83-m- (72-in.-) diameter base of the radiator is designed to mate with the reactor/shield assembly. The radiator area is 24.8 m^2 (266.8 ft^2). All PCS and other heat rejection subsystem components are located within the radiator. The radiator used Lockalloy (beryllium-aluminum alloy) fins and stainless steel tubes, stringers, and rings. There are 48 equally spaced NaK coolant tubes. Two supply lines deliver NaK from the manifolds of the TE converter module to the header at the top of the cone. The NaK flows down through the coolant tubes and is gathered in a header at the base of the cylinder. Two lines then return the cooled NaK to TE converters.

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a. 5-kWe Reactor Thermoelectric
System Radiator

Figure IV-172. Noncondensing Pumped Liquid Radiators

b. Reference Lockalloy-38 Radiator for the
5-kWe Reactor Thermoelectric System
Figure IV-172. Noncondensing Pumped Liquid Radiators

Figure IV-172 also shows fin/tube cross sections at two locations. The fins are of uniform thickness and vary in width from 4.94 cm (1.944 in.) at the top of the core to 11.97 cm (4.712 in.) on the cylinder. The 0.152-cm (0.060-in.) Lockalloy fins are attached to D-shaped stainless steel tubes. An armor thickness of 0.190 cm (0.075 in.) is required to provide a meteoroid noncritical damage probability of 0.99 for 5 years.

The radiator/structure characteristics are shown in Table IV-47.

TABLE IV-47
5-kWe REACTOR THERMOELECTRIC SYSTEM
RADIATOR CHARACTERISTICS

Thermal power rejected (kW)	94.9
NaK temperature [K (°F)]	
Inlet	595 (611)
Outlet	515 (467)
Emissivity	0.90
Solar absorptivity	0.50
Average fin effectiveness	0.835
Coolant flow rate, total [kg/s (lb/s)]	1.32 (2.9)
Radiator area [m ² (ft ²)]	24.8 (266.8)
Total mass [kg (lb)]	208.7 (460)

(c) SNAP-50 Radiator

In the SNAP-50 reactor electric system, a three-loop approach is used. The primary heat transport loop used Li as the working fluid. The PCS used a potassium-Rankine cycle with waste heat rejected in a compact potassium condenser. Heat is removed from the condenser by a NaK-cooled heat-rejection

loop. NaK enters a finned-tube radiator at 948 K (1246°F) and exits at 864 K (1096°F). NaK is pumped around the heat rejection loop using an electrically driven centrifugal pump. The configuration of the SNAP-50 concept is shown in Figure IV-173. Only conceptual design of the heat rejection subsystem exists as no development activities were performed.

c. Gas-Cooled Subsystems

Radiator designs for gas-cooled systems have been conceptually considered for heat rejection. These systems would be used for closed-cycle Brayton PCSs. No definitive designs have been developed to date.

d. Heat Pipe Subsystems

Heat pipe radiator subsystems have several attractive features. The heat pipe transports rejected heat at essentially constant temperature from the PCS to the radiating surface. This provides a minimum temperature drop between subsystems. The heat pipe offers redundancy, since each heat pipe is self-contained. A large number of finned heat pipes eliminate the need for heavy meteoroid protection since the radiator could be sized to accommodate a nominal loss of a few heat pipes due to meteoroid penetration.

Many innovative system designs for heat pipe radiators can be envisioned. Figure IV-174 shows a concept for rejecting heat from thermionic converters using heat pipes.^{IV-134} The Space Constructible Radiator (SCR) is a very low temperature subsystem currently being developed by Grumman Aerospace Corporation for NASA-JSC. The SCR consists of individual heat pipe radiator elements that are attached to a heat transport loop through a dry interface. Central to the SCR is a monogroove high-capacity heat pipe that forms the core of each radiator element. Aluminum heat pipes are used in this application and have a multilegged evaporator joined by a header to a 14.6-m- (48-ft-) long condenser section that is imbedded in a 30.5-cm- (12-in.-) wide, double-sided, aluminum radiator fin. The monogroove heat pipe uses ammonia as the working fluid.

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Figure IV-173. SNAP-50 Radiator Configuration

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Figure IV-174. Heat Pipe Radiator Concept
for a Thermionic Converter

The radiator heat rejection at a temperature of 300 K (80°F) is 431 W/m^2 (40 W/ft^2). At higher temperatures, the radiator performance will obviously improve. For heat rejection at 300 K (80°F), the SCR weighs approximately 0.017 kg/W (0.038 lb/W). A 400-kW subsystem, therefore, would require 333 heat pipes and 528 m^2 ($10,000 \text{ ft}^2$) of radiator area and would weigh about 6804 kg (15,000 lb).

e. Advanced Radiator Subsystems

(1) Liquid Droplet Radiator

The liquid droplet radiator concept for space applications is illustrated in Figure IV-175.^{IV-135} The liquid working fluid absorbs heat rejected by the PCS and is projected through space as a thin converging sheet of droplets. The droplets are collected in a droplet collector. The sheet of droplets radiates its heat to space during its passage from the droplet generator to the collector. One example of generator and collector designs is shown in Figure IV-176. The generator is a vibrating header with nozzle holes in the 50 to 100 μm range. This technology is based on conventional ink-jet printers.^{IV-135}

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Figure IV-175. Two-Pass Liquid Droplet
Radiator Concept

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Figure IV-176. Concepts for Liquid Droplet
Generator and Collector

The droplets must be a low-vapor-pressure liquid (e.g., silicone oils: 250 to 350 K; liquid metal eutectics: 370 to 650 K; liquid tin: 550 to 1000 K) to avoid loss of working fluid inventory. The subsystem has the potential for large heat rejection at low mass. The size of a stowed liquid droplet radiator has been reported as small compared to a heat pipe radiator of comparable heat rejection capability.

The efficiency of the radiator depends on the flow rate, the overall effectiveness of the heat exchanger, and the emissivity of the heat transport liquid. The emissivity of the liquid strongly affects efficiency. For example, lithium has a base metal emissivity of 0.1; however, the sheet may have an effective emissivity of 0.54 when it is optically thick. Since oils have emissivities of about 0.6, the sheet can have an emissivity of 0.8 or higher.

(2) Moving Belt Radiators

One concept for a moving belt radiator consists of a moving heat transfer surface which has a section in continuous contact with a rejection heat exchanger.^{IV-136} An example of this is shown in Figure IV-177. The belt does not contain a cooling fluid; hence its penetration by meteoroids would not significantly impair its operation. The chief advantage of the moving belt radiator is its low mass as compared to a pumped liquid subsystem.

Another similar concept employs direct contact between the PCS working fluid and the moving belt.^{IV-137} This is shown in Figure IV-178. In this configuration, the contact resistance to the heat exchanger is eliminated; however, a seal is required where the belt enters and exits the working fluid chamber.

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Figure IV-177. Direct-Contact Moving Belt Radiator Concept

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Figure IV-178. Convection Heated Moving Belt Radiator Concept

(3) Electrostatic Thermal Radiator

The electrostatic thermal radiator concept^{IV-138} offers the potential of an extremely low mass. The concept is shown schematically in Figure IV-179. The working media is a liquid metal such as tin that is formed into small (50- μ m-diameter) droplets and charged in the range of 10^{-13} to 10^{-15} C. The droplets are formed into a closely spaced train and ejected into space. An applied charge on the spacecraft that is opposite the droplet charge causes the droplets to have an elliptical orbit; hence, the collector for the particles can be located on the spacecraft. During the orbit, each droplet radiates heat to space.

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Figure IV-179. Schematic of the Electrostatic Thermal Radiator Concept

4. Technology Assessment

The technology assessment and issues for direct conducting, pumped liquid, and gas-cooled radiators are shown in Figure IV-180. The issues relate almost entirely (with the exception of the carbon/carbon composite radiator) to design and qualification testing of the concepts. Materials development is not an issue. Even with the carbon/carbon composite radiator, development and qualification is judged to be possible by the early 1990s, which supports the development schedules for other subsystems (i.e., reactor, power conversion, heat transport, shield subsystems).

a. Direct Conducting Subsystems

The direct conducting radiator is the simplest of all radiator concepts and has only found application with TE PCSs. The SNAP-10A used an aluminum conduction fin and a titanium substructure. The technology base is well established for this type of radiator. The selection of materials would be dictated by the radiator operating temperature. For advanced materials, such as carbon/carbon composite, a development effort would be required. For direct conducting radiator concepts, the conducting fin could also serve as the main radiator structure and must survive the launch environment. Although the carbon/carbon composites are state-of-the-art materials, process and fabrication procedures would have to be developed. It is estimated it would take 6 to 8 years to flight qualify a composite radiator. This development time phasing is shown in Figure IV-180.

b. Pumped Liquid Subsystems

The subsystem technology assessment is shown in Figure IV-180.

Figure IV-180. Subsystem Technology Assessment – Direct Conducting, Pumped Liquid, and Gas-Cooled Radiators

(1) Noncondensing

Noncondensing heat rejection subsystems have been proposed using both organic and liquid metal (NaK) as the working fluid. This type of subsystem has no pacing material development or feasibility technologies that need to be resolved. With NaK or organic as the working fluid, freezing is a concern; however, with the low freezing points of these fluids, heat shields could be used for initial startup. If restart is needed and/or the fluid is allowed to freeze, significant development would be required. It is estimated that development and qualification times would increase by 2 to 4 years for such a requirement.

(2) Condensing

Direct condensing radiators have been proposed using both potassium and organics. Much developmental work has been conducted on condensing heat rejection subsystems, particularly for potassium. Consequently, the technology for these subsystems has been advanced to some degree. There does remain the problem of liquid/vapor interface control, which will require additional development. Initial zero-g testing could be done with water, but eventually testing under sustained zero-g conditions with potassium and/or organics would be required. As with noncondensing radiators, similar schedule increases would occur if restart from freezing becomes a requirement.

c. Gas-Cooled Subsystems

The design technology for gas-cooled space radiators appears to be well established, as there are no outstanding material problems. Design, fabrication, vacuum ground testing, and flight qualification of a moderate-sized radiator using standard materials would require about 2 to 4 years. For large-sized radiators requiring deployment, 4 to 6 years would be required. This subsystem technology assessment is also shown in Figure IV-180.

d. Heat Pipe and Advanced Radiator Subsystems

Heat pipes have been proposed for a variety of radiator applications. These range from using mercury as the working fluid at 700 K (800°F) to potassium at 1100 K (1520°F). The preferred containment material for these heat pipes is usually considered to be stainless steel, although some consideration has been given to Nb-Ir for potassium heat pipes. It is estimated that the development time for the lower temperature range would be 4 to 6 years, while the higher temperature range would be 5 to 7 years as shown in Figure IV-181. For these development times, it was assumed the heat pipes would be self-starting after solidification.

Advanced radiator subsystems include the liquid droplet, moving belt, electrostatic, and others. Since these concepts are in their infancy and will require extensive in-orbit testing, it is difficult to quantify development times. It is anticipated, however, that it would take more than 10 years for development and qualification.

The technology issues for heat pipe and advanced radiators are shown in Figure IV-181. Heat pipe development (for stainless steel heat pipes) is discussed in the reactor subsystem section, and the development schedule for such heat pipes shown in Figure IV-181 is consistent with the need for such heat pipes in the reactor subsystem shown in Figure IV-54.

The technology issues for advanced radiator concepts (liquid droplet, moving belt, electrostatic) are numerous at this point. Behavior of fluid droplets in zero-gravity environments, control and aiming of electrostatic fields, etc., are — at best — difficult to estimate from a development standpoint. Such concepts are judged to be qualifiable in the late 1990s at best.

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Figure IV-181. Subsystem Technology Assessment – Heat Pipe and Advanced Radiators

V. SUBSYSTEM DEVELOPMENT PROJECTIONS

A. METHODOLOGY

Determining subsystem development costs and schedules was perhaps the most difficult but important part of this assessment. A methodology for doing this was established, based on our experience at ESG with space reactor electric system and subsystem costs and schedules. This methodology is presented in Figure V-1.

A generic development program was first formulated for the subsystems, including all testing from critical technology development to flight demonstration. Next, actual costs were collected for the SNAP-10A system. This is the only space reactor electric system to have undergone a complete life cycle, including a flight demonstration. Costs from the SNAP-2 and -8 programs were also used to establish historical trends. Escalation factors were established using U.S. Department of Labor data. These actual development costs were then escalated to mid-1983 dollars. Cost factors were estimated for each generic type of subsystem technology. These cost factors were derived from various considerations including the technology's past development experience, vintage of past development, current status of the technology infrastructure, and current status of supporting technologies. Development activities — from critical technology through flight demonstration — were then cost and schedule estimated for each subsystem technology. This section presents the details for each of these methodology steps.

1. Step 1 — Generic Development Program

A generic development program for a reactor electric system was formulated. This program, which included subsystem development activities, is shown in Figure V-2. It consists of three phases:

Figure V-1. Estimating Subsystem Development Costs and Schedules

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Figure V-2. Generic Development Schedule

- 1) Development
- 2) Qualification
- 3) Acceptance

The development phase starts with the critical technology development tests. During this phase, sufficient scoping tests are performed, and critical technology issues are resolved. Sufficient information is then available for subsystem designs. Component tests are performed in parallel with subsystem design. Next, subsystems are built, and ground demonstration tests are performed. These tests provide confirmation that the selected subsystems are sufficiently developed to allow proceeding to a system level test.

The next phase, qualification, generally overlaps the previous phase. This overlap provides a smooth program and minimizes the total development time. Flight qualification tests are performed on components and subsystems. A system ground demonstration test is also performed. In this system test, components are in a flight configuration. Subsystem hardware will be required for each of the following complete and partial systems to be tested in the qualification testing phase:

GDS-1	Ground demonstration system
FSM-1	Partial system for thermal/hydraulic qualification testing
FSM-2	Partial system for electrical qualification testing
FSM-3	Partial system for structural qualification testing.

The last three partial systems are flight system mockups (FSMs) used for qualification.

The final phase of the generic development program phase is acceptance. In this phase, the components, the subsystems, and the ground demonstration system are reconfigured into the final flight system, and this system is acceptance tested. This acceptance testing ensures that the system hardware

meets flight qualification standards. Two flight system (FS) tests are performed, one on the ground and one in space. The in-space test can be started in parallel with or slightly after the ground test. Subsystem hardware will be required for each of the following complete flight systems for this phase:

FS-1	Flight system for ground test
FS-2	Flight system for space test
FS-3	Flight system backup

2. Step 2 – Actual Life Cycle Costs

The costs incurred by Atomics International for the SNAP-2, -8, and -10A programs are reviewed in Table V-1. These are the costs for only the development testing phase and the ground demonstration system test. The costs do not include subsystems used in the FSMs or FSs. The costs are actual dollars expended in the GFY shown.

The SNAP-2 system was to use a 65-kWt, 922 K (1200°F) outlet temperature zirconium hydride reactor subsystem and a 3-kWe mercury Rankine power conversion subsystem (PCS). In the actual SNAP-2 program, the flight test was canceled in November 1963, and the program reverted to a technology development effort. The program was completely phased out in October 1966. To determine overall costs, the costs beyond FY 1963 are estimates through flight demonstration taken from program documents. The cost estimates were success oriented and assumed that no major development problems would occur. The cost breakdown data for the SNAP-2 program was available for the categories shown – reactor/shield subsystems, PCSs, and system integration. The system integration costs included not only overall system activities but also primary heat transport, heat rejection, and power processing subsystem (PPS) costs.

The SNAP-8 program encompassed only the development of the reactor and shield subsystems. Consequently, no system-level costs are shown in Table V-1. The reactor was a 600-kWt zirconium hydride unit with a 973 K (1300°F) NaK

TABLE V-1
SNAP PROGRAM COST SUMMARY
DEVELOPMENT AND QUALIFICATION COSTS
(Actual \$ Millions)

Program	GFY														Total
	56	57	58	59	60	61	62	63	64	65	66	67	68	69	
<u>SNAP-2</u>															
Reactor/shield subsystem	0.02	0.5	1.2	1.8	1.9	1.3	2.5	2.7	6.4	5.4	3.4	1.0			28.1
Power conversion subsystem			0.3	1.1	2.8	3.9	4.4	6.4	8.3	7.2	3.9	0.5			38.8
System integration and other subsystems		0.2	0.2	1.6	3.1	4.8	6.5	7.4	11.3	13.3	8.4	2.5			59.3
															127.2
<u>SNAP-8 (reactor/shield subsystems only)</u>															
Engineering and analysis					0.3	0.3	0.9	0.9	1.4	2.1	2.0	1.9	1.4	1.0	12.2
Component development and qualification					0.1	0.7	2.0	3.5	5.2	9.1	8.0	7.0	5.2	1.6	42.2
Subsystem development and qualification						0.4	2.6	3.7	5.2	5.9	5.2	3.8	2.5	2.2	31.5
															86.1
<u>SNAP-10A</u>															
Reactor/shield subsystems		1.0	1.5	1.7	1.9	2.0	2.0	3.6	2.0	0.9	1.0				17.6
Power conversion subsystem					0.5	1.0	3.0	1.6	1.0	0.6	0				7.7
System integration and other subsystems						0.9	8.3	10.0	7.2	7.0	2.4				35.8
															61.1

outlet temperature. The reactor shield was for unmanned applications. The SNAP-8 program was redirected to a technology development effort in January 1966. Consequently, the costs shown beyond this time are estimates for a success-oriented program. As is typical in any development program, there were technical barriers that had to be overcome. Such technical barriers affected the costs and schedule for the program.

The SNAP-10A system used a 43-kWt zirconium hydride reactor with a NaK outlet temperature of 811 K (1000°F). SiGe thermoelectric (TE) power converters produced 0.5 kW of electric power. SNAP-10A is the only system to have completed the full program cycle. One SNAP-10A flight system was launched in space and operated for 1000 h. An identical flight system was ground tested in a simulated space environment for 10,000 h. Note, however, that no SNAP-10A ground demonstration system test was performed before a flight system was ground tested. This was a programmatic decision based on schedule and risk assessments. The risk was considered acceptably low because the reactor used in SNAP-10A was a derated SNAP-2 reactor, and two of these had been successfully tested at higher powers and temperatures. The costs shown in Table V-1 are actual costs for this program. Again, these costs do not include the three FSMs or the three flight systems (estimated at \$1 million and \$2 million each in actual 1965 dollars). The cost of the SNAP-10A safety program is also not included in the costs given in Table V-1. The costs for this separate program are estimated to have been between \$35 and \$40 million (1963-1966\$).

3. Step 3 — Inflation Factors

The final cost information presented in this report is expressed in mid-1983 dollars. The cost information presented in Table V-1 was converted to 1983 dollars using escalation factors derived from Department of Labor data. For material costs, data for metals and metal products were used. For labor costs, data for aircraft workers (including aircraft research and development) were used. Figure V-3 shows the escalation rates from 1955 to the present

Figure V-3. Escalation Rates

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(extrapolated to mid-1983) for labor and materials. The escalation factor from mid-1963 to mid-1983 is 4.0 for labor and 3.37 for material. For the cost estimates presented, it was assumed that costs were comprised of 80% labor and 20% material. In actuality, these factors probably underestimate the cost escalation slightly due to the very high technology level of space reactor activities.

4. Step 4 – Subsystem Costs in 1983\$

The actual and estimated costs from Table V-1 were converted to mid-1983 costs on a year-by-year basis using the rates shown in Figure V-3. These mid-1983 costs are shown in Figures V-4 through V-6 for the SNAP-2, -8, and -10A systems. All of the curves have the same general trend. The shallow initial portion is the development testing phase; the steep portion is the qualification testing phase. Acceptance testing is not shown on the curves.

The costs given in Figures V-4 through V-6 and summarized in Table V-2 lead to several interesting observations. The first concerns the difference between the costs of the SNAP-2, -8, and -10A reactors. The SNAP-10A reactor was relatively low technology with modest temperatures and thermal power. The SNAP-2 reactor had the same thermal power level but a temperature 111 K (200°F) higher, which increased the cost by about 60%. The SNAP-8 reactor had a temperature 166 K (300°F) higher and a thermal power level more than a factor of 10 greater than SNAP-10A. The cost for the SNAP-8 reactor/shield subsystem is 4.5 to 6 times that for SNAP-10A. At the time the SNAP-8 reactor was developed, it could be considered an improved technology reactor, as it had benefited from the technology developed in the SNAP-2 and -10A programs.

Another interesting point from the Table V-2 data is the difference in development costs for thermoelectric (SNAP-10A) versus dynamic (SNAP-2) PCSs. This occurs since development on TEs can be conducted at the couple or module level while that on dynamics must eventually be conducted at full size. A factor of about 5 for dynamic versus static subsystem development costs should be expected.

Figure V-4. Historical Costs — SNAP-2
Reactor Electric System
(design, development, and qualification)

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Figure V-5. Historical Costs — SNAP-8 Reactor and Shield Subsystems (design, development, and qualification)

Figure V-6. Historical Costs — SNAP-10A Reactor Electric System
(design, development, and qualification)

TABLE V-2
SUMMARY OF ESTIMATED DEVELOPMENT AND QUALIFICATION COSTS

	Cost (\$ millions, mid-1983)		
	SNAP-2	SNAP-8 ^a	SNAP-10A
Reactor/shield subsystems	110	315 (410) ^b	70
Power conversion subsystem	150		30
System integration and other subsystems	<u>220</u>		<u>135</u>
Total development and qualification testing	480	315 (410) ^b	235
Unit cost of a flight system (FS)	9.5	7	7.5

^aReactor/shield subsystems only

^bEstimate with revised fuel swelling model

Based on these cost data, projections were made to account for power level (thermal and electrical) and level of technology (temperature, complexity, etc.). These estimates are presented in Figures V-7 and V-8 for the reactor/shield and dynamic PCSs. For a 1000-kWt reactor, the development and qualification testing costs range from about \$150 million for current technology up to about \$600 million for an advanced technology subsystem. Dynamic power conversion costs at 100 kWe for the same technology range are \$75 to \$500 million.

Three technology levels are shown in Figures V-7 and V-8: current, improved, and advanced. Current technology, as it implies, is based on well established designs and has a substantial background of hardware and test experience. An improved technology subsystem requires extrapolations from the current technology. These extrapolations may be significant increases in power, temperature, flow rate, etc., or they may be material changes. However, there must be little or no basic technology development required. An advanced technology is one that requires significant extrapolation of current technology. This may be large increases in power, temperature, etc., or the use of new materials.

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Figure V-7. Reactor and Shield Subsystem Development and Qualification Costs

Figure V-8. Dynamic Power Conversion Subsystem
Development and Qualification Costs

For static PCSs (thermoelectric and thermionic), development and qualification testing costs are essentially independent of power level. This cost independence occurs because activities can be performed on a modular (laboratory) basis. Because of this modularity, development and qualification testing costs will be significantly lower than for dynamic PCSs. Estimates of development and qualification costs were made for static devices based on information derived from companies working in the technology. These costs are as follows (mid-1983 dollars):

1)	Current technology	\$50 million
2)	Improved technology	\$50-\$80 million
3)	Advanced technology	\$100-\$165 million

5. Step 5 – Generic Subsystem Cost Factors

Each generic type of subsystem described in Section IV.D was assigned a technology status rating. This rating reflects many factors, including the maturity of the technology, the vintage of past development, the status of the technology's infrastructure, and the current status of supporting technology bases. This status rating is the extension of the descriptive rating (see Section III) to the present. It can be related to subsystem schedule and cost as shown in Figure V-9. This figure presents the fraction of total cost as a function of subsystem technology status rating (and also time). A generic type of subsystem with a technology status rating of five represents a fractional cost expenditure of less than 10%. The subsystem technology status ratings and associated fractional cost data in Figure V-9 were used as adjustment factors on the development and qualification testing costs from Figures V-7 and V-8. Adjustment factors were also estimated for the costs to reestablish a technology and its infrastructure, and for the cost reductions caused by parallel, supportive activities.

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Figure V-9. Subsystem Development and Qualification Testing Cost Versus Technology Status Rating

6. Step 6 – Subsystem Cost and Schedule Estimates

Using data and factors described above, costs and schedules were estimated for each of the five subsystems. In these cost estimates, certain cost elements were included and certain ones were excluded. The following cost elements were included:

- Subsystem scoping and development tests
- Subsystem design and engineering
- Subsystem fabrication and ground demonstration testing
- Subsystem component qualification testing
- Fabrication and delivery (to a system prime contractor) of a subsystem for incorporation into a ground demonstration system for qualification testing
- Fabrication and delivery (to a system prime contractor) of three flight subsystem mockups for system qualification testing
- Fabrication and delivery (to a system prime contractor) of three acceptance-tested flight subsystems
- All ancillary and support costs for the above including program management, administration, quality assurance, etc.

The following cost elements were excluded:

- Prime system contractor activities
- System design, and qualification and acceptance testing
- Test facilities and major manufacturing facilities
- Aerospace safety program activities
- Launch equipment and support activities
- Ground demonstration system testing
- Flight system ground demonstration testing.

The cost and schedule estimates for each generic type of subsystem for each of the five subsystems are given in Sections V.B through V.F.

B. REACTOR SUBSYSTEM

Tables V-3 and V-4 summarize the development, qualification, and acceptance-tested hardware costs for reactor subsystems. Table V-3 presents costs for a reactor power level of 500 kWt, which approximately represents that needed for an approximately 100-kWe system using a typical dynamic PCS (i.e., an overall system efficiency of about 20%). Table V-4 presents costs for a reactor power level of 1500 kWt, which represents that needed for an approximately 100-kWe system using a typical static PCS (i.e., an overall system efficiency of about 7%).

Reactor subsystem unit flight hardware costs were estimated from Figure V-10. These are the costs needed to produce the subsystem hardware used in the FSMs and FSs. The costs for reactor temperatures less than 980 K (1300°F) were based on actual SNAP-2, SNAP-8, and SNAP-10A reactor costs. The next temperature range of 980 to 1140 K (1300 to 1600°F) would require the use of superalloys, and a doubling of hardware costs was assumed. The upper temperature range would require the use of refractory alloys, and another doubling of hardware costs was assumed.

The total subsystem costs presented in Tables V-3 and V-4 are the sum of the development and qualification cost and the total flight hardware cost. The total flight hardware cost assumes subsystem hardware delivery for use in three complete FSs and three partial FSMs. The subsystem hardware for each FSM was estimated to cost one-half the amount of that for a FS. Thus, the flight hardware cost included in the total subsystem cost is four and one-half times the unit flight hardware cost.

The costs shown for the in-core thermionic reactor do not include the costs for the thermionic converter modules. These costs are shown separately under the PCS cost section (V.E). These PCS costs must be added to the reactor costs. In a similar manner for the heat pipe reactor, the costs for heat pipe development and flight hardware are treated as a primary heat transport subsystem (see Section V.D).

TABLE V-3
REACTOR SUBSYSTEM COSTS
500-kWt POWER LEVEL
(cost in millions of 1983 dollars)

Reactor Concept	Key Operating Conditions	Development and Qualification Costs	Unit Flight Hardware and Acceptance Test Costs	Total Subsystem Costs ^a	Development, Qualification, and Acceptance Time (years)
1. UZrH moderated, NaK cooled	920 K outlet, 5-year life	90-140	6-8	120-180	4-6
Improved H ₂ barrier	1000 K outlet, 7-year life	320-430	14-18	380-510	8-10
Separate fuel/mod. cooling systems	1100 K outlet, 7-year life	550-720	14-18	610-800	14-16
2. Lithium cooled, fast, refractory clad	1350 K outlet, 2-year life	540-700	30-40	670-880	9-11
UN fueled	1400 K outlet, 2-year life	580-750	30-40	710-930	12-14
UN additives	1500 K outlet, 3-year life	670-850	30-40	800-1030	14-16
3. Na or K cooled, fast, UO ₂ fuel, SST clad	920 K outlet, 7-year life	140-200	6-8	170-240	6-8
Na or K cooled, fast, UO ₂ fuel, superalloy clad	1100 K outlet, 7-year life	325-440	13-15	380-510	9-11
Boiling-potassium cooled, fast, SS clad, UO ₂ fuel	1110 K outlet, 3-year life	330-450	15-60	400-550	8-10
Boiling-potassium cooled	1365 K outlet, 7-year life	650-820	30-40	780-1000	12-14
4. In-core thermionics, UO ₂ fueled, current technology converter	1800 K, 1-year life	180-240	6-8	210-280	5-7
Oxygenated converter, UO ₂ -moly foil fuel	1800 K, 2-year life	340-460	14-18	400-540	8-10
Advanced converter, UO ₂ -moly cermet fuel, Mo collector	1800 K, 5-year life	450-550	14-18	510-630	11-13
Zr-C coated particle fuel, sublimed Mo collector	1800 K, 7-year life	550-650	14-18	610-730	14-16
6. He cooled, HTGR, graphite block fuel	1200 K outlet, 3-year life	100-140	14-18	160-220	6-8
Zr-C coated fuel particles in graphite block fuel	1300 K outlet, 5-year life	340-450	14-18	400-530	9-11
Helium cooled, particle fuel (FBR) Inconel frit	1100 K outlet, 3-year life	330-450	12-15	380-520	8-10
Particle fuel in high-temperature frit (FBR)	1300 K outlet, 5-year life	610-780	14-18	670-860	14-16
7. He cooled, metal clad Hast.-X, BeO/UO ₂ fuel	920 K outlet, 1-year life	120-170	6-8	150-210	4-6
Refractory clad, UO ₂ fueled, fast	1300 K outlet, 2-year life	550-720	30-40	590-900	9-11
Improved UC, UN, or cermet	1500 K outlet, 3-year life	590-760	30-40	730-940	12-14
8. Heat-pipe cooled, fast, Na heat pipes, UO ₂ fuel	1100 K outlet, 2-year life	320-440	12-15	370-510	7-9
Core heat-pipe cooled, UO ₂ fueled, fast, Mo-Re/Li heat pipes	1500 K outlet, 2-year life	550-710	30-40	690-890	11-13
Improved UO ₂ fuel, Mo-Re/Li heat pipes	1500 K outlet, 4-year life	580-750	30-40	720-930	14-16
UC-ZrC particle fuel, W-Re/Li heat pipes	1500 K outlet, 4-year life	650-820	30-40	790-1000	15-17

^aIncludes costs for four and one-half flight hardware units

TABLE V-4
REACTOR SUBSYSTEM COSTS
1500-kWt POWER LEVEL
(cost in millions of 1983 dollars)

Reactor Concept	Key Operating Conditions	Development and Qualification Costs	Unit Flight Hardware and Acceptance Test Costs	Total Subsystem Costs ^a	Development, Qualification, and Acceptance Time (years)
1. UZrH moderated, NaK cooled	920 K outlet, 5-year life	110-160	8-12	150-210	4-6
Improved H ₂ barrier	1000 K outlet, 7-year life	350-460	20-25	440-570	8-10
Separate fuel/mod. cooling systems	1100 K outlet, 7-year life	600-750	20-25	690-860	14-16
2. Lithium cooled, fast, refractory clad	1350 K outlet, 2-year life	580-730	50-60	800-1000	9-11
UN fueled	1400 K outlet, 2-year life	620-770	50-60	840-1040	12-14
UN additives	1500 K outlet, 3-year life	720-870	50-60	950-1140	14-16
3. Na or K cooled, fast, UO ₂ fuel, SST clad	920 K outlet, 7-year life	160-240	8-12	200-290	6-8
Na or K cooled, fast, UO ₂ fuel, superalloy clad	1100 K outlet, 7-year life	365-480	22-25	460-600	9-11
Boiling-potassium cooled, fast, SS clad UO ₂ fuel	1110 K outlet, 3-year life	370-485	25-30	485-620	8-10
Boiling-potassium cooled	1365 K outlet, 7-year life	700-850	50-60	920-1120	12-14
4. In-core thermionics, UO ₂ fueled, current technology converter	1800 K, 1-year life	200-280	8-12	240-330	5-7
Oxygenated converter, UO ₂ -Moly foil fuel	1800 K, 2-year life	380-500	20-25	470-610	8-10
Advanced converter, UO ₂ -Moly cermet fuel, Mo collector	1800 K, 5-year life	500-600	20-25	580-710	11-13
Zr-C coated particle fuel sublimed Mo collector	1800 K, 7-year life	600-700	20-25	690-810	14-16
6. He cooled, HTGR, graphite block fuel	1200 K outlet, 3-year life	110-100	20-25	200-270	6-8
Zr-C coated fuel particles in graphite block fuel	1300 K outlet, 5-year life	370-480	20-25	460-590	9-11
Helium cooled, particle fuel (FBR), Inconel frit	1100 K outlet, 3-year life	375-495	20-25	470-610	8-10
Particle fuel in high-temperature frit (FBR)	1300 K outlet, 5-year life	660-810	20-25	750-920	14-16
7. He cooled, metal clad Hast.-X, BeO/UO ₂ fuel	920 K outlet, 1-year life	140-200	8-12	180-250	4-6
Refractory clad, UO ₂ fueled, fast	1300 K outlet, 2-year life	600-750	50-60	820-1020	9-11
Improved UC, UN, or cermet	1500 K outlet, 3-year life	630-790	50-60	855-1060	12-14
8. Heat-pipe cooled, fast, Na heat pipes, UO ₂ fuel	1100 K outlet, 2-year life	360-480	22-25	460-590	7-9
Core heat-pipe cooled, UO ₂ fueled, fast, Mo-Re/Li heat pipes	1500 K outlet, 2-year life	590-740	50-60	810-1010	11-13
Improved UO ₂ fuel, Mo-Re/Li heat pipes	1500 K outlet, 4-year life	620-770	50-60	850-1040	14-16
UC-ZrC particle fuel, W-Re/Li heat pipes	1500 K outlet, 4-year life	700-850	50-60	920-1120	15-17

^aIncludes costs for four and one-half flight hardware units

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Figure V-10. Reactor/Shield Flight Hardware Costs

C. SHIELDING SUBSYSTEM

Table V-5 summarizes the development, qualification, and acceptance-tested hardware cost estimates for reactor shield subsystems. These estimates are based on the extrapolation of shield cost data accumulated during the SNAP programs. Shields were cast up to 6 ft (1.82 m) in diameter before the termination of the program in 1973. Facilities for manufacturing these shields are in a relatively good state, and the estimated cost for reactivating this facility is indicated to be less than \$0.5 million (1983 dollars). The lithium hydride technology up to shield temperatures of 820 K (1000°F) is well understood, and applying this technology to space reactor electric systems would represent little more than reactivating the manufacturing and testing facilities and then manufacturing and qualifying the shields for space flight.

Extending the lithium hydride technology into a higher operating temperature regime does not appear to be as economical a choice as using better heat management technology to thermally isolate and dissipate heat from the shield. Redesign of the shield with external insulation and a passive internal heat dissipation structure would be the most successful approach to the application of the lithium hydride shields in high heat flux environments. For higher temperature levels where heat could not be dissipated by a passive heat rejection structure, pumped-loop or heat-pipe heat removal could be employed for the shield. Ultimately, a molten LiH shield may be required for advanced technology applications.

These shield cost estimates are provided to indicate the relatively small amount that would be incurred in developing, manufacturing, and qualifying shielding subsystems compared to costs for the reactor.

TABLE V-5
SHIELDING SUBSYSTEM COSTS
(cost in millions of 1983 dollars)

Shield Concept	Key Operating Conditions	Development and Qualification Costs	Unit Flight Hardware and Acceptance Test Costs	Total Subsystem Costs ^a	Development, Qualification, and Acceptance Time (years)
Passively cooled with static radiator	$T_{LiH} < 800 \text{ K}$	8-9	0.3-0.4	9-11	4-6
Actively cooled by pumped loop or heat pipes with radiator	$T_{LiH} < 800 \text{ K}$	20-25	0.5-1.0	25-30	8-10
Molten LiH	$T_{LiH} > 960 \text{ K}$	50-80	2.0-4.0	60-100	13-15

Note: Costs are for unmanned shadow shields

^aIncludes costs for four and one-half flight hardware units

D. PRIMARY HEAT TRANSPORT SUBSYSTEM

Table V-6 summarizes the development, qualification, and acceptance-tested hardware cost estimates for the active components in primary heat transport subsystems. Other components such as heat exchangers and piping are not included in this table, since these components would be developed for specific reactor/power conversion configurations. These costs were estimated based on projections of actual costs incurred by Atomics International during the SNAP programs.

The cost estimates shown in Table V-6 are based on the removal of 1500 kW of thermal power from a reactor core; however, the costs are expected to be insensitive to power level. Three general concepts of primary heat transport are shown in the table. The first three items (i.e., electromagnetic pumps, pump power modules, and volume accumulators) are for liquid metal pumped-loop subsystems. The fourth item reflects the recent development work in high-temperature (>1300 K) reactor core heat pipes as a means of primary heat transport. The fifth item (gas recirculators) is included for gas-cooled reactor subsystem concepts which make use of a heat exchanger to separate the primary coolant from the PCS working fluid.

The cost estimates given in Table V-6 indicate that the primary heat transport subsystem costs will be a small fraction of the overall system development, qualification, and hardware costs. Although the items listed under primary heat transport subsystems are important to the successful development and deployment of a space reactor electric system, they represent only a small portion of the development effort and funding commitments required.

The costs and schedules presented in Table V-6 are based on heat transport subsystems in which the working fluid remains in the liquid form. If stop and restart are required, maintaining the working fluid above freezing (with higher freezing point fluids) may not be practical. Techniques for designing heat transport loops to transition from the solid to the liquid

TABLE V-6
PRIMARY HEAT TRANSPORT SUBSYSTEM COSTS
(cost in millions of 1983 dollars)

Concept	Key Operating Conditions	Development and Qualification Costs	Unit Flight Hardware and Acceptance Test Costs	Total Subsystem Costs ^a	Development, Qualification, and Acceptance Time (years)
<u>EM pumps</u>					
SST body 1 or 2 throat	$T_{IN} \leq 922$ K Na, NaK	15-20	0.1-0.2	15-20	3-4
Superalloy body 1 or 2 throat	$T_{IN} < 1145$ K Na, Li	40-50	0.4-0.6	40-60	6-8
Refractory metal body 1 or 2 throat	$T_{IN} < 1420$ K	120-140	1-2	125-150	10-12
<u>Pump power modules</u>					
Thermoelectric compact converter	PbTe, $T_H < 860$ K $T_C = 530$ K, 5-year life	20-25	1-2	25-35	3-4
Thermoelectric (SiGe)	$T_H < 1250$ K, $T_C = 700$ K, 7-year life	45-55	2-3	55-65	6-8
Thermoelectric SiGe (GaP) high-temperature coatings	$T_H < 1350$ K $T_C = 800$ K 5-year life	75-90	2-3	85-100	13-15
<u>Liquid metal volume accumulators</u>					
SST, gas or spring actuated	$T \approx 650$ K NaK, Na	10-15	0.1-0.2	10-15	3-4
SST, gas actuated	$T \approx 950$ K NaK, Na	25-30	0.2-0.3	25-30	5-6
Superalloy, gas actuated	$T \approx 1145$ K Na, Li	50-60	0.5-1	50-65	7-9
<u>Heat pipes</u>					
Sodium SST pipe	$T_E \approx 1000$ K 2 ± 2.5 kWt/cm ² 2-year life	20-25	8-10	50-70	4-6
Lithium Mo 13 Re core heat pipe	$T_E < 1700$ K $13-15$ kWt/cm ² 2-year life	50-60	2-4	60-75	8-10
Lithium W 26 Re core heat pipe	$T_E < 1900$ K $20-22$ kWt/cm ² 2- to 5-year life	80-90	2-3	90-100	11-13
<u>Gas recirculator</u>					
Superalloy body	$T_{IN} \leq 1145$ K noble gas	20-25	0.2-0.3	20-25	3-4
Refractory metal body	$T_{IN} \leq 1310$ K noble gas	35-45	0.5-1	40-50	5-7

^aIncludes costs for four and one-half flight hardware units

state have not been well established. In addition, requirements for restart and dormancy periods have not been firmly established. Consequently, costs for heat transport subsystems in which freezing occurs are not included in Table V-6. It is anticipated that to incorporate a restart capability would significantly increase development costs and schedules.

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E. POWER CONVERSION SUBSYSTEM

Development, qualification, and acceptance-tested hardware cost estimates for the PCS are presented in Tables V-7 and V-8. The two generic types of PCSs (static and dynamic) are presented separately due to the unique features associated with the development, qualification, and hardware costs of each type. No cost estimates are provided for the expansion of production facilities for the static or dynamic hardware.

Figure V-11 presents the production hardware cost estimates developed during the technology assessment. These costs are based on information provided by qualified organizations, SNAP program actual production costs, and previously published and unpublished reports. No production cost estimates are shown in Figure V-11 for in-core or out-of-core thermionic converters. The primary reason for their absence is the lack of production cost data for these converters. No mass production lines ever existed for this converter technology. A "best estimate" for the thermionic converter production costs was obtained during the discussions with qualified organizations. The range for dynamic power conversion hardware costs in Figure V-11 covers all of the various concepts considered. The organic Rankine hardware costs are represented near the bottom of the band. Stirling cycle subsystem hardware cost estimates are represented from the middle to lower bound of the band. Brayton cycle technology production cost estimates are represented by the middle to upper bound region. Liquid metal Rankine cycle technology production cost estimates are represented at or near the upper bound.

Table V-7 summarizes the cost estimates for static PCSs. The cost estimates are based on a 100-kWe unit size. These cost estimates show the dominance of production costs over development and qualification costs for static PCSs. The total cost estimates are essentially the same for the improved and advanced converter concepts in a particular technology. This occurs since the development and qualification for these concepts result in a more efficient

TABLE V-7
STATIC POWER CONVERSION SUBSYSTEM COSTS
100 kWe
(Cost in millions of 1983 dollars)

Converter Concept	Key Operating Conditions ^a	Development and Qualification Costs	Unit Flight Hardware and Acceptance Test Costs	Total Subsystem Costs ^b	Development, Qualification, and Acceptance Time (years)
High-Temperature Thermoelectrics					
Binary doped SiGe	1250 K hotshoe 7-year life 15-16% efficiency	25-35	55-60	270-305	6-7
Binary doped SiGe	1350 K hotshoe 2-year life 16-17%	40-50	45-50	240-280	9-10
Ternary doped SiGe (GaP) (alternate to above)	1250 K hotshoe 2-year life 17-18% efficiency	40-50	45-50	220-280	9-10
Binary doped SiGe	1350 K hotshoe 7-year life 16-17% efficiency	60-70	45-50	260-290	12-13
Ternary doped SiGe (GaP) (alternate to above)	1250 K hotshoe 7-year life 17-18%	60-70	45-50	260-300	12-13
Ternary doped SiGe (GaP)	1350 K hotshoe 2-year life 18-19% efficiency	65-75	40-50	240-300	12-13
Ternary doped SiGe	1350 K hotshoe 7-year life 18-19%	70-90	40-50	250-320	15-16
Rare earth sulfides or boron carbons	1550 K hotshoe 2-year life 20-22% efficiency	150-170	30-40	280-350	15-16
Low-Temperature Thermoelectrics					
Lead telluride compact converter	858 K hotshoe 5-year life 15-16% efficiency	30-40	15-25	100-150	4-6
TAGS converter	800-820 K hotshoe 7-year life 17-18% efficiency	40-50	10-20	90-140	5-7
Thermionics					
In-core, UO ₂ fueled current technology	14-20% efficiency 1-year life	50-60	50-60	270-330	6-8
In-core, UO ₂ -Mo foil fueled, oxygenated converter with vent to gap	20-24% efficiency 2-year life	70-80	40-50	250-300	9-10
In-core, UO ₂ -Mo cermet fuel, Mo collector, improved insulator	25-27% efficiency 2- to 4-year life	90-100	30-40	220-280	11-12
In-core, particle fuel sublimed Mo collector, improved insulator	28-30% efficiency 4- to 7-year life	150-170	25-35	260-330	14-16
Out-of-core, oxygenated converter	T _E = 1800 K 2-year life 25-27% efficiency	60-70	20-25	150-180	9-11
Out-of-core, optimized materials in converter	T _E = 1600 to 1800 K 28-34% efficiency 2- to 5-year life	80-100	15-20	150-190	14-16

^aEfficiency is "device" efficiency and must be multiplied by the Carnot efficiency to obtain overall PCS efficiency.

^bIncludes four and one-half unit hardware costs

TABLE V-8
DYNAMIC POWER CONVERSION SUBSYSTEM COSTS
100-kWe POWER LEVEL SUBSYSTEMS WITH A FACTOR OF TWO IN
COMPONENT REDUNDANCY (200 kWe TOTAL)
(cost in millions of 1983 dollars)

Subsystem Concept	Key Operating Conditions ^a	Development and Qualification Costs	Unit Flight Hardware and Acceptance Test Costs	Total Subsystem Costs ^b	Development, Qualification, and Acceptance Time (years)
Organic Rankine	T _{inlet} = 630 K 42-45% efficiency 1-year life 20-kWe units	40-70	3-3.5	50-85	4-6
Organic Rankine	T _{inlet} = 640 K 47-51% efficiency 2-year life 50-kWe units	100-130	1.5-2.5	110-140	10-12
Organic Rankine	T _{inlet} = 670 K 56-58% efficiency 4-year life 200 kWe units	240-290	1.5-2.5	250-300	15-16
Mercury Rankine	T _{inlet} = 980 K 22-25% efficiency 1-year life 35 kWe units	90-150	6-10	120-200	5-7
Mercury Rankine	T _{inlet} = 980 K 28-32% efficiency 2-year life 90-kWe units	250-300	6-10	280-350	8-9
Mercury Rankine	T _{inlet} = 1000 K 36-40% efficiency 3-year life 120 kWe units	400-500	6-10	430-550	12-14
Potassium Rankine	T _{inlet} = 1110 K 55-56% efficiency 1-year life 20-30-kWe units	110-130	14-18	170-210	6-8
Potassium Rankine	T _{inlet} = 1280 K 45-50% efficiency 1-year life 20-30-kWe units	200-300	14-18	260-380	8-10
Potassium Rankine	T _{inlet} = 1330 K 52-55% efficiency 2-year life 50-kWe units	350-450	8-12	390-500	11-13
Potassium Rankine	T _{inlet} = 1380 K 56-58% efficiency 3-year life 100-kWe units	500-600	8-12	540-690	15-17
Closed-cycle Brayton	T _{inlet} = 1100 K 38-42% efficiency 2-year life 30-kWe units	70-80	12-16	120-150	4-6
Closed-cycle Brayton	T _{inlet} = 1200 K 43-45% efficiency 3-year life 60-kWe units	190-230	6-10	220-260	6-9
Closed-cycle Brayton	T _{inlet} = 1300 K 48-52% efficiency 4-year life 100 kWe units	400-510	6-10	430-550	12-13
Closed-cycle Brayton	T _{inlet} = 1400 K 53-56% efficiency 4-year life 200 kWe units	520-640	8-12	560-690	16-18
Free-piston Stirling	T _H = 970 K 60-64% efficiency 1-year life 16 kWe units	150-200	12-16	200-270	6-7
Free-piston Stirling	T _H = 1010 K 60-70% efficiency 4-year life 50-kWe units	250-300	4-6	260-330	9-11
Free-piston Stirling	T _H = 1050 K 70-75% efficiency 7-year life 100 kWe units	400-500	8-12	430-550	12-14
Open-cycle Brayton	T _{inlet} = 1100 K 81-82% efficiency 10-20-h life	120-140	6-8	150-180	6-8
Open-cycle Brayton	T _{inlet} = 1200 K 82-83% efficiency 20-40-h life	260-320	6-8	290-360	8-10
Open-cycle Brayton	T _{inlet} = 1300 K 82-83% efficiency 30-60-h life	530-630	6-8	560-670	13-15

^aEfficiency is "device" efficiency and must be multiplied by Carnot efficiency to obtain overall PCS efficiency.

^bIncludes four and one-half unit hardware costs

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Figure V-11. Power Conversion Subsystem Flight Hardware Costs

static converter, and fewer production units are required to produce 100 kWe. This method of cost projection assumes that the basic materials costs do not significantly affect the cost of producing the converter units. For high-technology, high-quality controlled items, this assumption has been found to be accurate.

The results in Table V-7 indicate that the total costs for the out-of-core thermionic PCSs will be less than that of the in-core subsystems' costs. These cost differences are primarily due to the restrictions and complications added to the in-core thermionic converter design to operate within the reactor core's environment.

The lead telluride compact converter and TAGS converter production cost estimates are approximately one-half of the current silicon germanium converter production cost estimates. This occurs because a more expensive, high-temperature (>1300 K) process is required for the manufacture of the SiGe power converter versus the lower <950 K process involved in manufacturing the lead telluride and TAGS converter designs.

The estimated costs for in-core thermionics and SiGe thermoelectrics for a 100-kWe system are approximately the same. On a per converter basis, a thermionic converter is estimated to cost approximately 250 times one TE converter. However, only 800 thermionic converters are required for a 100-kWe system, whereas 200,000 TE converters are required.

The cost estimates for dynamic PCSs are summarized in Table V-8. These estimates are also based on a 100-kWe power level subsystem; however, a factor of two redundancy in the total number of power units producing the 100 kWe was used as a first approximation to providing a reasonable overall PCS reliability. Therefore, the unit hardware costs are for a total of 200 kWe generating capacity. The standard 100-kWe dynamic PCS was assumed to be composed of two

50-kWe generating units plus two totally redundant 50-kWe units. For some near-term technologies, the size of the power-producing units is limited to less than 50 kWe. In these cases, the indicated size of the unit shown in Table V-8 was divided into the 200 kWe requirement to determine the total number of units required for each flight system. For more advanced technologies, some dynamic units are projected to have capabilities in excess of 100 kWe. For these cases, two units were priced for each.

The open Brayton cycle PCS cost estimates are based on two (one operating and one redundant) 500-kWe generating units for the basic subsystem unit costs. This difference is primarily due to the application of this technology to semicontinuous power generation in space.

F. HEAT REJECTION SUBSYSTEM

Table V-9 summarizes the development, qualification, and acceptance-tested hardware cost estimates for heat rejection subsystems. The cost estimates are based on a subsystem heat rejection capacity of 1500 kWt. Four different concepts for heat rejection subsystems are shown in this table.

The direct-conducting (or direct-radiating) concept has been employed in previous space flight vehicles (e.g., SNAP-10A) and associated with RTG SiGe thermoelectric PCSs. The SNAP-10A design employed a titanium support structure with an aluminum fin bonded to the cold shoes of the SiGe TE converters. Improved versions of this concept running at heat rejection temperatures of ~800 K and employing stainless steel in the vehicle structure have been studied, but none have been built. The most recent concept for direct radiating proposed for the SP-100 system is the use of composite materials for both fin and integral structural support. In this concept, the radiating surface material provides both heat rejection for the PCS and structural support for all the subsystems with a savings in overall mass. These materials have a higher operating temperature potential than indicated in Table V-9.

The pumped-loop concept was developed extensively during the SNAP program's advanced radiator development work. The low-temperature concepts are directed at gas-Brayton cycles and organic fluids. At higher temperatures, pumped-liquid-metal radiator concepts (e.g., Na or NaK) have been developed. For liquid metal loops, stainless steel has generally been used for containment. Structural support was provided by titanium for medium heat rejection temperature subsystems and stainless steel for higher temperature subsystems. The liquid-metal pumped loop concept requires the development of electromagnetic dc conduction pumps, TE-driven pump power modules, and liquid metal volume accumulators.

Direct vapor-condensing radiators were developed for the SNAP-2 mercury Rankine PCSs and the potassium Rankine PCSs. A full ground demonstration of

TABLE V-9
HEAT REJECTION SUBSYSTEM COSTS
(cost in millions of 1983 dollars)

Heat Rejection Concept	Key Operating Conditions	Development and Qualification Costs	Unit Flight Hardware and Acceptance Test Costs	Total Subsystem Costs ^a	Development, Qualification, and Acceptance Time (years)
1. DIRECT CONDUCTING					
Low-temperature metallic fin structure	$T_R \sim 600 \text{ K}$	10-15	1.5-3	15-30	3-4
High-temperature metallic fin structure	$T_R \sim 800 \text{ K}$	15-20	1-2	20-30	4-5
High-temperature composite fin and structure	$T_R \sim 800 \text{ K}$	40-45	6-8	70-80	7-10
2. PUMPED LOOP					
a. <u>Noncondensing Fin and Structure</u>					
Low-temperature aluminum structure	$T_{IN} \leq 600 \text{ K}$	16-20	2-4	25-40	3-4
Medium-temperature SST tubes, titanium structure — Al fins	$T_{IN} \leq 800 \text{ K}$	18-22	4-6	40-50	3-4
High-temperature all superalloy structure	$T_{IN} \leq 1200 \text{ K}$	30-35	1-2	35-45	4-5
b. <u>Noncondensing Components and EM Pumps</u>					
SST body 1 or 2 throat	$T_{IN} < 922 \text{ K}$ Na, NaK	18-22	0.1-0.2	20-25	3-4
Superalloy body 1 or 2 throat	$T_{IN} < 1145 \text{ K}$ Na, Li	40-50	0.4-0.6	40-55	6-8
Pump power modules (PbTe)	$T_N < 860 \text{ K}$ $T_C = 530 \text{ K}$	18-22	0.6-0.8	20-25	3-4
Pump power modules (SiGe)	$T_H < 1250 \text{ K}$ $T_C = 700 \text{ K}$	45-55	2-3	55-70	6-8
Liquid metal expansion compensators	$T \approx 650 \text{ K}$ NaK or Na	10-14	0.1-0.2	10-15	3-4
c. <u>SST Body, Condensing Fin and Structure</u>					
Mercury SST structure	$T_{IN} \approx 620 \text{ K}$	30-35	3-5	40-60	4-6
Potassium SST structure	$T_{IN} \approx 1000 \text{ K}$	40-50	1-1.5	50-55	5-7
3. HEAT PIPE					
Mercury SST pipe	$T_E < 725 \text{ K}$ $1 \leq 1.5 \text{ kWt/cm}^2$	25-30	8-10	60-75	4-6
Potassium Nb-1 Zr pipe	$T_E < 900 \text{ K}$ $1 \leq 1.5 \text{ kWt/cm}^2$	25-30	8-10	60-75	4-6
Sodium SST pipe	$T_E \sim 1000 \text{ K}$ $2 \leq 2.5 \text{ kWt/cm}^2$	25-30	8-10	60-75	4-6
4. ADVANCED					
Liquid droplet, titanium structure	$T_{IN} \leq 800 \text{ K}$	125-175	8-15	160-240 ^b	12-16

^aIncludes costs for four and one-half flight hardware units

^bOrder of magnitude estimate

these radiator subsystems under simulated zero-g conditions was performed; however, the units were never flight qualified or tested in space. Although the radiators are generally smaller in size than other pumped-loop concepts, the direct condensing radiator design is complicated by the necessity to maintain a stable liquid-vapor interface under a zero-g environment.

The costs and schedules presented in Table V-9 are based on heat rejection subsystems in which the working fluid remains in the liquid form. If stop and restart are required, maintaining the working fluid above freezing (with higher freezing point fluids) may not be practical. Techniques for designing heat rejection loops to transition from the solid to the liquid state have not been well established. In addition, requirements for restart and dormancy periods have not been firmly established. Consequently, costs for heat rejection subsystems in which freezing occurs are not included in Table V-9. It is anticipated that to incorporate the restart capability would significantly increase development cost and schedules.

The third concept is generally known as the heat pipe radiator concept. It has been proposed as the heat rejection subsystem for TE and out-of-core thermionic PCSs. The primary advantages to this concept are its nearly isothermal heat rejection capabilities and its simplicity compared to the pumped liquid metal loop concept (e.g., no EM pumps, pump power modules, or volume accumulators). The costs shown in Table V-9 under the title "Heat Pipes" include the development, qualification, and production costs for only the heat pipes. In addition to these cost estimates, it is necessary to add the cost estimates associated with the "direct-conducting" fin and structure concepts discussed previously to obtain the total cost estimates for a heat pipe radiator.

Advanced heat rejection subsystems in their various forms (e.g., liquid droplet, moving belt, electrostatic) propose to inject the heat rejection fluid in a directed pattern into the space environment, where it would radiate

the heat energy in the fluid into space; collect this fluid; and pump it back to a PCS/HRS heat exchanger. This concept decouples the classic mass and cost penalties per 1 kWt rejected to space associated with the other heat rejection subsystem concepts discussed above. At the 1500-kWt heat rejection power level, advanced radiator concepts are not economically competitive, but at 10 to 100 times this heat rejection power level they could be the only economically feasible concept. No cost estimates could be obtained for the advanced radiator concepts, and the values shown in Table V-9 are simply order of magnitude cost estimates for reference only.

All heat rejection subsystems require micrometeoroid protection for the fluid containment boundary and emissivity coatings for the radiator surface. Both of these technologies are important to a good HRS design but are not significant in the total HRS cost estimates.

The cost estimates for the heat rejection/structural support assembly are provided to indicate the relatively small costs that would be incurred in developing, qualifying, and manufacturing these subsystems compared to the cost estimates for the reactor and PCS. Because the configuration of the heat rejection system is generally dictated by the reactor-PCS decisions, the heat rejection subsystem costs have been allocated in past programs to the system level development, qualification, and hardware costs.

VI. DEVELOPMENT COMPATIBLE SUBSYSTEM TECHNOLOGIES

The subsystem technology assessment provides the status and potentials for subsystem technologies and the key technology issues that must be resolved to achieve these potentials. The results are detailed in Section IV. Table VI-1 summarizes the key features and performance parameters for the two dominant subsystems (reactor and power conversion*) for the three generic levels of technology defined for this assessment — current technologies for near-term needs (late 1980s to early 1990s), improved technologies for mid-term needs (mid 1990s), and advanced technologies for far-term needs (late 1990s). The table presents the development compatibility — on technical and technical plus schedule bases — for the nine generic steady-state reactor subsystems with the seven generic steady-state power conversion subsystems (PCSs) defined in the technology projections.

Development compatibilities are identified for (1) strictly technical compatibility and for (2) technical plus schedule compatibility as defined in the legend on Table VI-1. Technical compatibility is defined as the ability of the two subsystems to be operated together in a space reactor electric system. This compatibility does not require the two subsystems to exactly match in terms of operational conditions — all that is required is that both be within a temperature range (defined such that neither subsystem is operated with more than a 170 K ($\sim 300^{\circ}\text{F}$) derating). For example, a 920 K (1200°F) liquid-metal-cooled fast reactor is compatible with a TAGS thermoelectric PCS which is operational at temperatures of 800 to 820 K ($\sim 1000^{\circ}\text{F}$). The matchup, although not exact, is considered reasonable, and therefore the two subsystems are designated as being technically compatible. On the other hand, a 920 K

*The remaining three subsystems — primary heat transport, shielding, and heat rejection — comprise less than 20% of the subsystem development program cost for steady-state electric systems of the multi-kWe class. The development schedule for these three subsystems supports the development program schedules for the reactor and power conversion subsystems. Therefore, the development compatibility assessment has been focused on the two dominant subsystems.

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										REACTOR SUBSYSTEMS		POWER CONVERSION SUBSYSTEMS		DEVELOPMENT COMPATIBILITY ASSESSMENT				
										TECHNOLOGY LEVEL	FUEL	CLADDING	TEMPERATURE (K (°F))	LIFE (year)				
										C ^a	P ^b	A ^c						
										LIQUID METAL COOLED THERMAL REACTORS								
										NaK COOLED ZH REACTOR								
											U-2H	HASTELOY N	520 (1200)	5				
											U-2H	HASTELOY N WITH IMPROVED BARRIER	1000 (1340)	7				
											U-2H	SEPARATE FUEL/MODERATOR COOLING SYSTEMS	1100 (1260)	7				
										HTGR REACTORS								
											GRAPHITE PARTICLE FUEL	NA	1200 (1700)	3				
											ZrC PARTICLE FUEL IN GRAPHITE	NA	1300 (1800)	5				
										GAS-COOLED FAST REACTORS								
											UO ₂ -H ₂ O	HASTELOY X	600 (1200)	1				
											UO ₂	REFRACTORY	1300 (1800)	2				
											UC UN OR CERMET	REFRACTORY	1500 (2240)	3				
										FIXED BED REACTOR								
											PARTICLE FUEL	INCONEL	1300 (1920)	3				
											PARTICLE FUEL	HIGH TEMP PRT	1300 (1880)	5				
										Na OR K-COOLED FAST REACTORS								
											UO ₂	STAINLESS STEEL	500 (1200)	7				
											UO ₂	SUPERALLOY	1100 (1920)	7				
										LI-COOLED FAST REACTORS								
											UN	REFRACTORY	1350 (1910)	2				
											UN	REFRACTORY	1400 (2060)	2				
											UN ADDITIVES	REFRACTORY	1800 (2240)	3				
										BOILING LIQUID METAL FAST REACTORS								
											UO ₂	STAINLESS STEEL	1110 (1540)	3				
											UN	REFRACTORY	1385 (2000)	7				
										THERMIONIC REACTORS								
											UO ₂	REFRACTORY	1800 (2780)	1				
											UO ₂ Mg FOIL	REFRACTORY	1800 (2780)	2				
											UO ₂ -Mg UC PARTICLE	REFRACTORY	1800 (2780)	5, 7				
										HEAT PIPE REACTORS								
											UO ₂	Na/SUPERALLOY HEAT PIPE	1100 (1920)	2				
											UO ₂	LiMg-He HEAT PIPE	1500 (2240)	2				
											IMPROVED UO ₂ OR PARTICLE FUEL	LiMg-He OR LiMg-He HEAT PIPE	1500 (2240)	4				
										THERMOELECTRICS								
													1250 (1780)	15-16	7	BINARY DOPED Si-Ge		
													858 (1085)	15-16	5	LEAD TELLURIDE COMPACT CONVERTER		
													800-820 (980-1015)	17-18	7	MODULAR TAGS CONVERTER		
													1350 (1910)	16-17	2	BINARY DOPED Si-Ge (TERNARY DOPED Si-Ge IS AN ALTERNATE)		
													1550 (2230) OR 1350 (1910)	20-22 OR 18-19	2 OR 7	HIGH PERFORMANCE CONVERTER (LIX/Sy OR EX/Sy) OR TERNARY DOPED Si-Ge		
										THERMIONICS								
													1900 (2780)	14-20	1	IN-CORE UO ₂ WITH WEEMITTER		
													1800 (2780)	20-24	2	IN-CORE VENT TO GAP OXY-D CONV		
													1600 (1800) (2420-2780)	25-27	2	OUT-OF-CORE OXY-D CONV		
														25-30	2, 7	IN-CORE ADV CONVERTER		
														28-34	2, 5	OUT-OF-CORE ADV MAT LS		
										CLOSED-CYCLE BRAYTON								
													1100 (1520)	38-42	2	30		
													1200 (1700)	43-45	3	60		
													1300-1400 (1880-2060)	48-56	4	100-200		
										ORGANIC RANKINE CYCLE								
													630 (875)	42-45	1	20		
													640 (890)	47-51	2	50		
													960 (1279) (745)	52-58	3-4	100-200		
										MERCURY RANKINE CYCLE								
													980 (1300)	22-25	1	35		
													980 (1300)	28-32	2	90		
													1000 (1340)	36-40	3	120		
										POTASSIUM RANKINE CYCLE								
													1110 (1540)	55-56	1	20-30		
													1280 (1645)	45-50	1	20-30		
													1330-1360 (1935-2025)	52-58	2, 3	50-100		
										STIRLING CYCLE								
													970 (1285)	60-64	1	16		
													1010 (1360)	67-70	4	50		
													1060 (1430)	70-75	7	100		

LEGEND

C^a - CURRENT TECHNOLOGY (LATE 80's TO EARLY 90's)

P^b - IMPROVED TECHNOLOGY (MID 90's)

A^c - ADVANCED TECHNOLOGY (LATE 90's)

Table VI-1. Development Compatibility Matrix for the Two Key Subsystems (Reactor and power conversion subsystems)

LEGEND

C^a - CURRENT TECHNOLOGY (LATE 80s TO EARLY 90s)
T^b - IMPROVED TECHNOLOGY (MID 90s)
A^c - ADVANCED TECHNOLOGY (LATE 90s)

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(1200°F) gas-cooled reactor is judged development incompatible for operation with a closed-cycle Brayton PCS to be designed for operation at 1200 K (1700°F). The rationale is that an 1100 K (1520°F) Brayton machine of earlier development vintage would suffice and be of reasonable technical compatibility. Therefore, derating an improved-technology 1200 K (1700°F) PCS would be far inferior to using a current-technology, lower-temperature, Brayton (which might be much more of an "off-the-shelf" PCS) for application with the 900 K (1200°F) gas-cooled reactor.

Two exceptions exist to the 170 K (300°F) temperature compatibility range: (1) the in-core thermionic PCS, which is judged compatible with – and only with – the thermionic reactor (and vice versa), and (2) the Rankine (i.e., boiling) PCSs. For the Rankine cycles, the reactor coolant outlet temperature must at least equal the potassium boiling temperature of the cycle.

Technical compatibility does not include a judgment as to the desirability of matching up a given pair of reactor and power conversion subsystems. For example, a gas-cooled reactor is ideally suited for matchup to a closed-cycle Brayton PCS. It is less suited for use with a Rankine cycle PCS, even though (for present purposes) it is theoretically technically compatible. Compatibility refinements of this type will be made during the "top down" modification of the development compatibility table during our upcoming system-level assessment.

In the sections that follow, development compatibility means technical and scheduler compatibility. Technical and schedule compatibilities are defined by the congruence of technical compatibility (as defined above) and scheduler compatibility – the intersection of current/current technology subsystems, improved/improved technology subsystems, or advanced/advanced technology subsystems.

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A. CURRENT TECHNOLOGIES FOR NEAR-TERM NEEDS

This section summarizes the development-compatible current-technology subsystems that could be assembled into a reactor electric system for near-term needs (late 1980s to early 1990s). This is best understood by starting with the technology assessments performed in Section IV. The future potentials of each generic type of subsystem are presented pictorially in that section. Circle diagrams are used to depict steps in technology versus time. Section IV discusses the key technology issues and risks that go along with each technology step.

Figure VI-1 now uses these same technology steps (the same circles with the same technical data) to tie various sets of the five subsystems together. Each vertical column of circles represents five current-technology subsystems that are development compatible (i.e., they are all current technology and are technically compatible). The columns do not represent what should be done, but rather what could be done and on what time frame. The figure is a useful tool for visualizing how the various subsystems fit together; it also is a simpler representation of the near-term data presented in Table VI-1.

The various columns (i.e., systems) in Figure VI-1 differ as to technical risks, but they all fall within the same risk range. For these near-term systems, the estimated availability dates (i.e., when they could first be ready for operation in space) could only be met by eliminating system ground demonstration testing from the overall development program. Thus, to meet the schedule, the program would proceed directly from subsystem ground testing to system flight testing. Subsystem qualification would need to be performed during the subsystem ground tests, and acceptance testing in parallel with the flight system tests. This is judged to be an acceptable risk since most of the current technology subsystems have a good residual data base from prior space reactor electric system programs or from the LMFBR and HTGR power reactor programs. In addition, this testing philosophy is the same as that

employed for the SNAP-10A reactor electric system, and that philosophy was considered to have reasonable risk. A somewhat higher risk would be associated with subsystems requiring more development (e.g., the Brayton and Rankine turboelectrics, the heat pipe reactor). The higher risk associated with these subsystems is also judged acceptable because the performance of those subsystems appears to be within the bounds of current materials technologies, the design lifetimes appear to be realistic, and the operating temperatures are reasonable.

Perhaps the highest risk element associated with the estimated availability dates is the reestablishment of the government-industry infrastructure necessary to accomplish the work. This infrastructure has been decimated by nearly a decade of inactivity. "Picking up the pieces" — a difficult, time-consuming process — may not be fully factored into the availability dates. In this regard, the availability dates for near-term systems are probably somewhat optimistic.

REACTOR SUBSYSTEM	NaK COOLED ZrH REACTOR			HTGR REACTOR			GAS COOLED FAST REACTOR			Na OR K COOLED FAST REACTOR			BOILING K FAST REACTOR	THERMIONIC REACTOR	HEAT PIPE REACTOR			
POWER CONVERSION SUBSYSTEM	LOW TEMPERATURE THERMOELECTRICS		FREE PISTON STIRLING	HIGH TEMPERATURE THERMOELECTRICS	CLOSED CYCLE BRAYTON TURBOELECTRIC	POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	LOW TEMPERATURE THERMOELECTRICS		FREE PISTON STIRLING	LOW TEMPERATURE THERMOELECTRICS		FREE PISTON STIRLING	POTASSIUM RANKINE TURBOELECTRIC	THERMIONIC CONVERTER	HI TEMPERATURE THERMOELECTRICS (RADIATIVE COUPLING)	CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	FREE PISTON STIRLING (HEAT EXCHANGER REQUIRED)
SHIELD SUBSYSTEM	LiH NEUTRON SHIELD PASSIVE COOLING			LiH NEUTRON SHIELD ACTIVE COOLING			LiH NEUTRON SHIELD PASSIVE COOLING			LiH NEUTRON SHIELD PASSIVE COOLING			Li NEUTRON SHIELD - ACTIVE COOLING	LiH NEUTRON SHIELD - PASSIVE COOLING	LiH NEUTRON SHIELD ACTIVE COOLING			
HEAT TRANSPORT SUBSYSTEM	PUMPED LIQUID METAL			PUMPED GAS			PUMPED GAS			PUMPED LIQUID METAL			PUMPED LIQUID METAL	PUMPED LIQUID METAL	Na/SUPERALLOY HEAT PIPES			
HEAT REJECTION SUBSYSTEM	DIRECT CONDUCTING RADIATOR		NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR	GAS RADIATOR	CONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR		GAS RADIATOR	DIRECT CONDUCTING RADIATOR		NONCONDENSING RADIATOR	CONDENSING RADIATOR	NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR	GAS RADIATOR	CONDENSING RADIATOR	NONCONDENSING RADIATOR
ESTIMATED AVAILABILITY DATE	1987 1989	1989 1990	1989 1991	1991 1992	1991 1992	1991 1992	1987 1989	1987 1989	1989 1991	1990 1991	1990 1991	1990 1991	1991 1992	1988-1990	1991 1992	1991 1992	1991 1992	1991 1992

Figure VI-1. Technologies for Near-Term Needs - Development Compatible Subsystems

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B. IMPROVED TECHNOLOGIES FOR MID-TERM NEEDS

This section summarizes the development-compatible improved-technology subsystems that could be assembled into a reactor electric system for mid-term needs (mid 1990s). As in the previous section, the subsystem circle diagrams from Section IV are arranged in technically compatible vertical columns in Figure VI-2.

In general, improved-technology subsystems employ higher operating temperatures and have longer lifetimes while operating at higher power levels than current-technology subsystems. They may require superalloy or refractory materials, improved fuels, and hotter heat rejection subsystems. Higher efficiency static and dynamic PCSs should be available for these mid-term applications.

To maintain an acceptable level of risk with these improved-technology subsystems, it was deemed prudent to pursue the total generic development program shown in Figure V-2. There would, therefore, be subsystem ground tests followed by a system ground test and by flight system flight and ground tests.

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REACTOR SUBSYSTEM	NaK COOLED ZrH REACTOR		HTGR REACTOR			GAS COOLED FAST REACTOR			FIXED BED REACTOR			Na OR K COOLED FAST REACTOR			Li COOLED FAST REACTOR			ThERMIonic REACTOR	HEAT PIPE REACTOR
	1991 1993	1991 1993	1993 1995	1993 1995	1993 1995	1993 1995	1993 1995	1993 1995	1992 1994	1992 1994	1992 1994	1992 1994	1992 1994	1992 1994	1993 1995	1993 1995	1993 1995	1992 1993	1994 1996
POWER CONVERSION SUBSYSTEM	MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	FREE PISTON STIRLING	HIGH TEMPERATURE THERMOELECTRICS	CLOSED CYCLE BRAYTON TURBOELECTRIC	POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	HIGH TEMPERATURE THERMOELECTRICS	CLOSED CYCLE BRAYTON TURBOELECTRIC	POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	CLOSED CYCLE BRAYTON TURBOELECTRIC	MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	FREE PISTON STIRLING	CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	FREE PISTON STIRLING	HIGH TEMPERATURE THERMOELECTRICS	CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	ThERMIonic CONVERTER	HIGH TEMPERATURE THERMOELECTRICS
	1991 1993	1992 1994	1992 1993	1991 1993	1993 1994	1992 1993	1991 1993	1993 1994	1991 1993	1991 1993	1992 1994	1991 1993	1991 1993	1992 1994	1992 1993	1991 1993	1993 1994	1992 1993	1991 1993
SHIELD SUBSYSTEM	LiH NEUTRON SHIELD PASSIVE COOLING		Li NEUTRON SHIELD ACTIVE COOLING			Li NEUTRON SHIELD ACTIVE COOLING			LiH NEUTRON SHIELD ACTIVE COOLING			LiH NEUTRON SHIELD ACTIVE COOLING			LiH NEUTRON SHIELD ACTIVE COOLING			LiH NEUTRON SHIELD PASSIVE COOLING	LiH NEUTRON SHIELD ACTIVE COOLING
	1986-1988	1986 1988	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1991 1993	1986-1988	1991 1993
HEAT TRANSPORT SUBSYSTEM	PUMPED LIQUID METAL		PUMPED GAS			PUMPED GAS			PUMPED GAS			PUMPED LIQUID METAL			PUMPED LIQUID METAL			PUMPED LIQUID METAL	Li/Mo Re HEAT PIPE
	1985 1987	1985-1987	1988-1990	1988 1990	1988 1990	1988 1990	1998 1990	1998 1990	1985 1987	1985 1987	1985 1987	1991 1993	1991 1993	1991 1993	1993 1995	1993 1995	1993 1995	1985-1987	1991 1993
HEAT REJECTION SUBSYSTEM	CONDENSING RADIATOR	NONCONDENSING RADIATOR	DIRECT CONDUCTING (DEPLOYABLE)	DEPLOYABLE GAS RADIATOR	CONDENSING RADIATOR	DIRECT CONDUCTING	GAS RADIATOR	CONDENSING RADIATOR	DEPLOYABLE GAS RADIATOR	CONDENSING RADIATOR	DEPLOYABLE GAS RADIATOR	GAS RADIATOR	CONDENSING RADIATOR	NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR	GAS RADIATOR	CONDENSING RADIATOR	NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR
	1987 1989	1986 1987	1990-1992	1988 1990	1987 1989	1990 1992	1986 1988	1987 1989	1988 1990	1987 1989	1988 1990	1986 1988	1987 1989	1986 1987	1985 1986	1986 1988	1987 1989	1986 1987	1990 1992
ESTIMATED AVAILABILITY DATE	1991 1993	1992 1994	1993 1995	1993 1995	1993 1995	1993 1995	1993 1995	1993 1995	1992 1994	1992 1994	1992 1994	1992 1994	1992 1994	1992 1994	1993 1995	1993 1995	1993 1995	1991 1993	1994 1996

Figure VI-2. Improved Technology for Mid-Term Needs - Development Compatible Subsystems

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C. ADVANCED TECHNOLOGIES FOR FAR-TERM NEEDS

This section summarizes the development-compatible advanced-technology subsystems that could be assembled into a reactor electric system for far-term needs (late 1990s). Again, the subsystem circle diagrams from Section IV are arranged in technically compatible vertical columns (i.e., systems) in Figure VI-3.

The number of potential sets of development-compatible subsystems is approximately twice that for either near- or mid-term applications. This proliferation is due to the "branching" of some subsystem technologies into two (or more) paths that might be possible for far-term applications.

The continuing increases in operating temperatures and power levels could require using molten LiH neutron shields, deployable (gas-cooled) radiators, and even advanced radiator concepts. Advanced fuels and refractory materials would no doubt see widespread use in such systems. To maintain acceptable risk in developing the needed subsystems, significant critical technology development and subsystem/system ground testing would be required, and the time to perform this development and testing would pace the availability dates.

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REACTOR SUBSYSTEM	NaK COOLED ZrH REACTOR		GAS COOLED FAST REACTOR						FIXED BED REACTOR		Li COOLED FAST REACTOR											
	SEPARATE FUEL/MODERATOR COOLING 1100 K (1520°F) FUEL OUTLET 2000 kWt 7 year LIFE 1997 1999	SEPARATE FUEL/MODERATOR COOLING 1100 K (1520°F) FUEL OUTLET 2000 kWt 7 year LIFE 1997 1999	IMPROVED UC UN OR CERMET FUELS 10,000 kWt 1500 K (2240°F) OUTLET 3 year LIFE 1996 1998	IMPROVED UC UN OR CERMET FUELS 10,000 kWt 1500 K (2240°F) OUTLET 3 year LIFE 1996 1998	IMPROVED UC UN OR CERMET FUELS 10,000 kWt 1500 K (2240°F) OUTLET 3 year LIFE 1996 1998	IMPROVED UC UN OR CERMET FUELS 10,000 kWt 1500 K (2240°F) OUTLET 3 year LIFE 1996 1998	IMPROVED UC UN OR CERMET FUELS 10,000 kWt 1500 K (2240°F) OUTLET 3 year LIFE 1996 1998	IMPROVED UC UN OR CERMET FUELS 10,000 kWt 1500 K (2240°F) OUTLET 3 year LIFE 1996 1998	FBR HIGH TEMPERATURE FRIT 50,000 kWt 1300 K (1880°F) OUTLET 5 year LIFE 1998 1999	FBR HIGH TEMPERATURE FRIT 50,000 kWt 1300 K (1880°F) OUTLET 5 year LIFE 1998 1999	Li COOLED REFRACTORY CLAD UN 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE 1995-1996	Li COOLED UN ADDITIVES 1500 K (2240°F) OUTLET 3000 kWt 3 year LIFE 1997 1999	Li-COOLED REFRACTORY CLAD UN 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE 1995 1996	Li COOLED UN ADDITIVES 1500 K (2240°F) OUTLET 3000 kWt 3 year LIFE 1997 1999	Li-COOLED REFRACTORY CLAD UN 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE 1995 1996	Li COOLED UN ADDITIVES 1500 K (2240°F) OUTLET 3000 kWt 3 year LIFE 1997 1999	Li-COOLED REFRACTORY CLAD UN 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE 1995 1996	Li COOLED UN ADDITIVES 1500 K (2240°F) OUTLET 3000 kWt 3 year LIFE 1997 1999	Li-COOLED REFRACTORY CLAD UN 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE 1995 1996	Li COOLED UN ADDITIVES 1500 K (2240°F) OUTLET 3000 kWt 3 year LIFE 1997 1999	Li-COOLED REFRACTORY CLAD UN 1400 K (2060°F) OUTLET MORE COMPACT 2000 kWt 2 year LIFE 1995 1996	Li COOLED UN ADDITIVES 1500 K (2240°F) OUTLET 3000 kWt 3 year LIFE 1997 1999
POWER CONVERSION SUBSYSTEM	MERCURY RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)	FREE PISTON STIRLING	HIGH TEMPERATURE THERMOELECTRICS		CLOSED CYCLE BRAYTON TURBOELECTRICS		POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)		CLOSED CYCLE BRAYTON TURBOELECTRIC		HIGH TEMPERATURE THERMOELECTRICS				CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)				POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)			
	T = 1000 K (1340°F) η _D = 36-40% LIFE = 3 YEARS Pe = 120 kWt 1995-1997	T = 1050 K (1430°F) η _D = 70-75% 7 YEAR LIFE Pe = 100 kWt 1995 1997	RARE EARTH/SULFIDES OR BORON CARBONS T _H = 1550 K (2330°F) 2 year LIFE η _D = 20-22% 1998 2000	SiGe (GaP) T _H = 1350 K (1970°F) 7 year LIFE η _D = 18-19% 1998 2000	REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η _D = 48-52% 1995-1997	CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η _D = 53-56% 1999 2001	T = 1330 K (1935°F) η _D = 52-55% LIFE = 2 years Pe = 50 kWt 1996-1997	T = 1380 K (2025°F) η _D = 56-58% LIFE = 3 years Pe = 100 kWt 1999 2001	REFRACTORY MATERIALS T = 1300 K (1800°F) LIFE = 4 years Pe 100 kWt η _D = 48-52% 1995 1997	CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η _D = 53-56% 1999 2001	SiGe (GaP) T _H = 1350 K (1970°F) 7 year LIFE η _D = 18-19% 1998 2000	SiGe (GaP) T _H = 1350 K (1970°F) 7 year LIFE η _D = 18-19% 1998 2000	REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η _D = 48-52% 1998 2000	REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η _D = 48-52% 1998 2000	REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 4 years Pe 100 kWt η _D = 48-52% 1995 1997	CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η _D = 53-56% 1999 2001	CERAMIC MATERIALS T = 1400 K (2060°F) LIFE = 4 years Pe 200 kWt η _D = 53-56% 1999 2001	REFRACTORY MATERIALS T = 1300 K (1880°F) LIFE = 2 years Pe = 50 kWt η _D = 48-52% 1995 1997	T = 1330 K (1935°F) η _D = 52-55% LIFE = 2 years Pe = 50 kWt 1996 1997	T = 1380 K (2025°F) η _D = 56-58% LIFE = 3 years Pe = 100 kWt 1999 2001	T = 1380 K (2025°F) η _D = 56-58% LIFE = 3 years Pe = 100 kWt 1999 2001	T = 1330 K (1935°F) η _D = 52-55% LIFE = 2 years Pe = 50 kWt 1996-1997
SHIELD SUBSYSTEM	LiH NEUTRON SHIELD ACTIVE COOLING		MOLTEN LiH NEUTRON SHIELD						MOLTEN LiH NEUTRON SHIELD		MOLTEN LiH NEUTRON SHIELD											
	LiH SHIELD T _{MAX} < 930 K ACTIVE COOLING 1991 1993	LiH SHIELD T _{MAX} < 930 K ACTIVE COOLING 1991 1993	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998	MOLTEN NEUTRON SHIELD 1996 1998
HEAT TRANSPORT SUBSYSTEM	PUMPED LIQUID METAL		PUMPED GAS						PUMPED GAS		PUMPED LIQUID METAL											
	OUTLET TEMP 1145 K (1600°F) SUPER ALLOY NOBLE GAS BLOWER 1989-1991	OUTLET TEMP 1145 K (1600°F) SUPER ALLOY NOBLE GAS BLOWER 1989-1991	OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER 1991 1993	OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER 1991 1993	OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER 1991 1993	OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER 1991 1993	OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER 1991 1993	OUTLET TEMP 1480 K (2200°F) REFRACTORY METAL NOBLE GAS BLOWER 1991 1993	OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER 1988 1990	OUTLET TEMP 1310 K (1900°F) REFRACTORY METAL NOBLE GAS BLOWER 1988 1990	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li, EM PUMP (NaK COOLED) 1993 1995	
HEAT REJECTION SUBSYSTEM	CONDENSING RADIATOR	NONCONDENSING RADIATOR	DIRECT CONDUCTING RADIATOR		DEPLOYABLE GAS RADIATOR		CONDENSING RADIATOR		ADVANCED RADIATOR		DIRECT CONDUCTING RADIATOR				DEPLOYABLE GAS RADIATOR				CONDENSING RADIATOR			
	CON DENSING RADIATOR 1987 1989	NONCON DENSING RADIATOR 1986 1987	DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR 1990 1991	DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR 1990 1991	GAS-COOLED DEPLOYABLE RADIATOR 1988-1989	GAS-COOLED DEPLOYABLE RADIATOR 1988-1989	CON DENSING RADIATOR 1987 1989	CON DENSING RADIATOR 1987 1989	ADVANCED CONCEPTS LIQUID DROPLET MOVING BELT ELECTROSTATIC 1995 1997	ADVANCED CONCEPTS LIQUID DROPLET MOVING BELT ELECTROSTATIC 1995 1997	DIRECT CONDUCTING RADIATOR 1985 1986	DIRECT CONDUCTING RADIATOR 1985 1986	DIRECT CONDUCTING RADIATOR 1985 1986	DIRECT CONDUCTING RADIATOR 1985 1986	GAS-COOLED DEPLOYABLE RADIATOR 1988 1989	GAS-COOLED DEPLOYABLE RADIATOR 1988 1989	GAS-COOLED DEPLOYABLE RADIATOR 1988 1989	GAS-COOLED DEPLOYABLE RADIATOR 1988 1989	CON DENSING RADIATOR 1987 1989	CON DENSING RADIATOR 1987 1989	CON DENSING RADIATOR 1987 1989	CON DENSING RADIATOR 1987 1989
ESTIMATED AVAILABILITY DATE	1997 1999	1997 1999	1998 2000	1998-2000	1996 1998	1999 2001	1996-1998	1999 2001	1998-1999	1999 2001	1998 2000	1998 2000	1998 2000	1998 2000	1996 1998	1999 2001	1999 2001	1997 1999	1997 1999	1999 2001	1999 2001	1997 1999

Figure VI-3. Advanced Technology for Far-Term Needs - Development Compatible Subsystems
(Sheet 1 of 2)

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REACTOR SUBSYSTEM	BOILING K FAST REACTOR		THERMIONIC REACTOR		HEAT PIPE REACTOR											
	<div>BOILING K COOLED REFRACTORY CLAD UN 1365 K (2000°F) OUTLET 5000 kWt 7 year LIFE</div> <div>1995-1997</div>	<div>BOILING K COOLED REFRACTORY CLAD UN 1365 K (2000°F) OUTLET 5000 kWt 7 year LIFE</div> <div>1995-1997</div>	<div>UO₂ Mo CERMET FUEL SUBLIMED Mo COLLECTOR IMPROVED INSULATOR 24 year LIFE $\eta_D = 25.27\%$</div> <div>1994-1996</div>	<div>PARTICLE FUEL SUBLIMED Mo COLLECTOR, IMPROVED INSULATOR 4-7 year LIFE $\eta_D = 28.30\%$</div> <div>1997-1999</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	<div>UC ZrC PARTICLE FUEL Li/WRe HEAT PIPES 3000 kWt 1500 K (2240°F) EVAPORATOR 4-year LIFE</div> <div>1998-2000</div>	<div>Li/MoRe HEAT PIPES IMPROVED UO₂ FUEL SWELLING 1500 kWt 1500 K (2240°F) EVAPORATOR 4 year LIFE</div> <div>1998-2000</div>	
POWER CONVERSION SUBSYSTEM	POTASSIUM RANKINE TURBOELECTRIC		THERMIONIC CONVERTER		HIGH TEMPERATURE THERMOELECTRICS				CLOSED CYCLE BRAYTON TURBOELECTRIC (HEAT EXCHANGER REQUIRED)				POTASSIUM RANKINE TURBOELECTRIC (HEAT EXCHANGER REQUIRED)			
	<div>T - 1330 K (1935°F) $\eta_D = 52.55\%$ LIFE - 2 years Pe - 50 kWt</div> <div>1996-1997</div>	<div>T - 1380 K (2025°F) $\eta_D = 56.58\%$ LIFE - 3 years Pe - 100 kWt</div> <div>1999-2001</div>	<div>UO₂-Mo CERMET FUEL SUBLIMED Mo COLLECTOR, IMPROVED INSULATOR 24 year LIFE $\eta_D = 25.27\%$</div> <div>1994-1996</div>	<div>PARTICLE FUEL SUBLIMED Mo COLLECTOR, IMPROVED INSULATOR 4-7 year LIFE $\eta_D = 28.30\%$</div> <div>1997-1999</div>	<div>RARE EARTH SULFIDES OR BORON CARBONS T_H - 1550 K (2330°F) 2 year LIFE $\eta_D = 20.22\%$</div> <div>1998-2000</div>	<div>SiGe (GaP) T_H - 1350 K (1970°F) 7 year LIFE $\eta_D = 18.19\%$</div> <div>1998-2000</div>	<div>SiGe (GaP) T_H - 1350 K (1970°F) 7 year LIFE $\eta_D = 18.19\%$</div> <div>1998-2000</div>	<div>RARE EARTH SULFIDES OR BORON CARBONS T_H - 1550 K (2330°F) 2 year LIFE $\eta_D = 20.22\%$</div> <div>1998-2000</div>	<div>REFRACTORY MATERIALS T - 1300 K (1800°F) LIFE - 4 years Pe 100 kWt $\eta_D = 48.52\%$</div> <div>1995-1997</div>	<div>CERAMIC MATERIALS T - 1400 K (2060°F) LIFE - 4 years Pe 200 kWt $\eta_D = 53.56\%$</div> <div>1995-1997</div>	<div>CERAMIC MATERIALS T - 1400 K (2060°F) LIFE - 4 years Pe 200 kWt $\eta_D = 53.56\%$</div> <div>1999-2001</div>	<div>REFRACTORY MATERIALS T - 1300 K (1800°F) LIFE - 4 years Pe 100 kWt $\eta_D = 48.52\%$</div> <div>1997</div>	<div>T - 1330 K (1935°F) $\eta_D = 52.55\%$ LIFE - 2 years Pe - 50 kWt</div> <div>1996-1997</div>	<div>T - 1380 K (2025°F) $\eta_D = 56.58\%$ LIFE - 3 years Pe - 100 kWt</div> <div>1999-2001</div>	<div>T - 1380 K (2025°F) $\eta_D = 56.58\%$ LIFE - 3 years Pe - 100 kWt</div> <div>1999-2001</div>	<div>T - 1330 K (1935°F) $\eta_D = 52.55\%$ LIFE - 2 years Pe - 50 kWt</div> <div>1996-1997</div>
SHIELD SUBSYSTEM	MOLTEN LiH NEUTRON SHIELD		LiH NEUTRON SHIELD PASSIVE COOLING		MOLTEN LiH NEUTRON SHIELD											
	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>LiH CAST IN SS VESSEL T_{MAX} < 930 K PASSIVE COOLING</div> <div>1986-1988</div>	<div>LiH CAST IN SS VESSEL T_{MAX} < 930 K PASSIVE COOLING</div> <div>1986-1988</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>	<div>MOLTEN NEUTRON SHIELD</div> <div>1996-1998</div>
HEAT TRANSPORT SUBSYSTEM	PUMPED LIQUID METAL		PUMPED LIQUID METAL		HEAT PIPES											
	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 1420 K (2100°F) REFRACTORY METALS Li EM PUMP (NaK COOLED)</div> <div>1993-1995</div>	<div>OUTLET TEMPERATURE 922 K (1200°F) STAINLESS STEEL NaK EM PUMP</div> <div>1985-1987</div>	<div>OUTLET TEMPERATURE 922 K (1200°F) STAINLESS STEEL NaK EM PUMP</div> <div>1985-1987</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>	<div>EVAPORATOR TEMPERATURE 1900 K (2960°F) W Re Li 21 kW/cm²</div> <div>1995-1997</div>
HEAT REJECTION SUBSYSTEM	CONDENSING RADIATOR				DIRECT CONDUCTING RADIATOR				DEPLOYABLE GAS RADIATOR				CONDENSING RADIATOR			
	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>NONCONDENSING RADIATOR</div> <div>1986-1987</div>	<div>NONCONDENSING RADIATOR</div> <div>1986-1987</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>DIRECT CONDUCTING CARBON/CARBON COMPOSITE RADIATOR</div> <div>1990-1991</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>GAS-COOLED DEPLOYABLE RADIATOR</div> <div>1988-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>	<div>CONDENSING RADIATOR</div> <div>1987-1989</div>
ESTIMATED AVAILABILITY DATE	1996-1998	1999-2001	1994-1996	1994-1996	1998-2000	1998-2000	1998-2000	1998-2000	1998-2000	1999-2001	1999-2001	1998-2000	1998-2000	1999-2001	1999-2001	1998-2000

Figure VI-3. Advanced Technology for Far-Term Needs - Development Compatible Subsystems (Sheet 2 of 2)
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